ENCLOSURES 4 THROUGH 7 CONTAIN PROPRIETARY INFORMATION – WITHHOLD FROM PUBLIC DISCLOSURE IN ACCORDANCE WITH 10CFR 2.390.

Xcel Energy®

L-PI-09-133
10 CFR 50.90

28 December 2009

U S Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

Prairie Island Nuclear Generating Plant Units 1 and 2
Dockets 50-282 and 50-306
License Nos. DPR-42 and DPR-60

<u>License Amendment Request for Measurement Uncertainty Recapture - Power Uprate</u>

Pursuant to 10 CFR 50.90, the Northern States Power Company, a Minnesota corporation (NSPM), doing business as Xcel Energy, hereby requests an amendment to the operating license and the plant Technical Specifications (TS) for Prairie Island Nuclear Generating Plant (PINGP). The proposed license amendment request (LAR) would increase the licensed rated thermal power (RTP) as a result of a measurement uncertainty recapture (MUR) power uprate (PU). The information provided in support of this request is based on Nuclear Regulatory Commission (NRC) Regulatory Issue Summary (RIS) 2002-03, "Guidance on the Content of Measurement Uncertainty Recapture Power Uprate Applications." Requests for Additional Information (RAI) regarding MUR PU applications for other nuclear Units were reviewed for applicability. This submittal incorporates the results of those reviews.

The proposed change would increase the licensed RTP level by 1.64 percent from 1650 megawatts thermal (MWt) to 1677 MWt. NSPM's request is based on reduced uncertainty in the RTP measurement achieved by installation of a Caldon Leading Edge Flow Meter (LEFM) checkplus™ System used to measure feedwater flow and temperature. The reduced power measurement uncertainty allows for a power uprate that is equivalent to the Title 10 Code of Federal Regulations (CFR) 50, Appendix K criteria of two percent minus the calculated LEFM-based power measurement uncertainty of 0.36 percent. Caldon Topical Reports ER-80P and ER-157P document the theory, design, and operating features of the checkplus™ System and its ability to achieve increased accuracy in main feedwater flow measurement. The NRC approved ER-80P and ER-157P in safety evaluations (SE) dated March 8, 1999 and December 20, 2001, respectively. These SEs have been referenced in previously submitted power uprate license applications. In addition, the NRC staff completed a re-evaluation of the Caldon LEFM ultrasonic flow meters (UFM) as part of the generic assessment of the hydraulic aspects of UFM application to increase licensed thermal power and issued

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NRC SE entitled, "Evaluation of The Hydraulic Aspects of the Caldon Leading Edge Flow Measurement (LEFM) Check and CheckPlus™ Ultrasonic Flow Meters (UFM)" dated July 5, 2006. As described in that SE, the NRC found that the Caldon Check and CheckPlus UFMs' performance is consistent with the Caldon Topical Reports ER-80P, Revision 0 and ER-157P, Revision 5, previously approved by the NRC staff.

The proposed change would also change the revision number of the analysis methodology prescribed in Technical Specifications for analyzing the Pressure and Temperature Limits Report (PTLR). The latest revision is an NRC-approved topical report.

This amendment request is supported by several enclosures. The following table summarizes each enclosure:

Enclosure	Content Description
1	A description and assessment of the MUR PU including: description, background, proposed license and TS changes, technical assessment, a no significant hazards consideration and environmental considerations.
2	Summary of the MUR PU evaluation following guidance provided in Regulatory Issue Summary 2002-03.
3	Affidavit of Withholding for Enclosures 4 – 7 Pursuant to 10 CFR 2.390, Cameron International Corporation
4	Caldon Report ER-532 Rev 1, Bounding Uncertainty Analysis For Thermal Power Determination At Prairie Island Unit 1 NPP Using the LEFM √ + System **(Proprietary)**
5	Caldon Report ER-533, Rev 2, Bounding Uncertainty Analysis For Thermal Power Determination At Prairie Island Unit 2 NPP Using the LEFM √ + System **(Proprietary)**
6	Caldon Report ER-583, Rev. 0, LEFM√+ Meter Factor Calculation and Accuracy Assessment for Prairie Island Unit 1 Nuclear Power Station (Alden Reports No. 2007-001/1229) **(Proprietary)**
7	Caldon Report ER-553, Rev. 2, LEFM√+ Meter Factor Calculation and Accuracy Assessment for Prairie Island Unit 2 Nuclear Power Station (Alden Reports No. 2006-163/C730) **(Proprietary)**
8	ESC Report No. 2005-11125-2.R02, "Generation Interconnection Study – Projects # G433 and G434; 38 MW Expansion of Prairie Island Units 1 and 2", dated March 24, 2006, Delyn Electrical Engineering, LLC, "G433-G434 Transient Stability Study – Supplement", Dated November 13, 2009, and Xcel letter documenting Midwest ISO acceptance of PINGP 38 MW Increase from Randall Oye to Jim Hill dated November 13, 2009.
9	Facility Operating License, TS pages marked up to show the proposed changes.
10	Revised (clean) Facility Operating License, TS pages.
11	List of regulatory commitments associated with proposed amendment
12	Revised Bases page (for information only)
13	Revised PTLR page (for information only)

As Enclosures 4 through 7 contain information proprietary to Cameron International Corporation, they are supported by an affidavit (Enclosure 3) signed by a duly sworn authority for Cameron, the owner of the information. The affidavit sets forth the basis on which the information may be withheld from public disclosure by the Commission and addresses, with specificity, the considerations listed in paragraph (b)(4) of 10 CFR 2.390 of the Commission's regulations. Accordingly, it is respectfully requested that the information, which is proprietary to Cameron, be withheld from public disclosure in accordance with 10 CFR 2.390. Correspondence with respect to the copyright or proprietary aspects of the items listed above or the supporting Cameron Affidavit, should reference the appropriate authorization letter and be addressed to Mr. Ernie Hauser, Director of Sales, Cameron's Measurement System Division, 1000 McClaren Woods Drive, Coraopolis, PA 15108.

NSPM requests approval of this LAR within six months of the acceptance date of the LAR for NRC review. Upon NRC approval, NSPM requests that the amendment be made effective on the date of issuance, but allow an implementation period of 180 days to provide sufficient time for associated administrative activities.

NSPM has determined that the information for the proposed amendment does not involve a significant hazards consideration, authorize a significant change in the types or total amounts of effluent release, or result in any significant increase in individual or cumulative occupational radiation exposure. Therefore, the proposed amendment meets the categorical exclusion requirements of 10 CFR 51.22(c)(9) and an environmental impact assessment need not be prepared.

In accordance with 10 CFR 50.91, NSPM is notifying the State of Minnesota of this LAR by transmitting a copy of this letter with non-proprietary enclosures to the designated State Official.

If there are any questions or if additional information is needed, please contact Ms. Lynne Gunderson at 612-330-6588.

Summary of Commitments

Regulatory commitments associated with this LAR are listed in Enclosure 11.

I declare under penalty of perjury that the foregoing is true and correct.
Executed on December 28, 2009

For M. Schimmel

Mark A. Schimmel, Site Vice President
Prairie Island Nuclear Generating Plant Units 1 and 2
Northern States Power Company - Minnesota

Enclosures (13)

cc: Administrator, Region III, USNRC (without Enclosures 4 – 7)
Project Manager, Prairie Island, USNRC
Resident Inspector, Prairie Island, USNRC
State of Minnesota (without Enclosures 4 – 7)

Enclosure 1

To

Letter from Mark A. Schimmel (NSPM)

To

Document Control Desk (NRC)

Evaluation of the Proposed Change

13 pages follow

ENCLOSURE 1

Evaluation of the Proposed Change

Subject: License Amendment Request for Measurement Uncertainty Recapture - Power Uprate

1.0 SUMMARY DESCRIPTION

Northern States Power Company, a Minnesota corporation (NSPM), proposes to amend the Facility Operating License and Technical Specifications to increase the licensed Maximum Power Level for the Prairie Island Nuclear Generating Plant (PINGP) Units 1 and 2. The PINGP Units 1 and 2 are currently licensed to operate at a maximum rated thermal power (RTP) of 1650 megawatts thermal (MWt). Approval is requested to increase the licensed RTP by 1.64 percent to 1677 MWt. This power increase will be accomplished by using a more accurate main feedwater flow and temperature measurement system to calculate the RTP of the Units. The system being used is the Caldon Leading Edge Flow Meter (LEFM) CheckPlus™ system. Increasing RTP by reducing measurement uncertainty is called a measurement uncertainty recapture power uprate (MUR PU). NSPM has evaluated the impact of a 1.64 percent uprate to 1677 MWt for the applicable systems, structures, components, and safety analyses at the PINGP. The results of this evaluation and the new main feedwater flow and temperature measurement system are described in Enclosure 2 of this letter, "Summary of Measurement Uncertainty Recapture Power Uprate Evaluation Following Guidance Provided in NRC Regulatory Issue Summary (RIS) 2002-03."

2.0 DETAILED DESCRIPTION

2.1 Proposed Changes

The proposed license amendment request (LAR) will revise the PINGP Facility Operating Licenses and the Technical Specifications (TS) to increase the licensed RTP by 1.64 percent from 1650 MWt to 1677 MWt. The proposed changes are described below and are also included in the marked-up Operating License Technical Specifications (TS) pages in Enclosure 9 with a clean copy of the Operating License and TS pages included in Enclosure 10.

Although Bases changes are not a part of this LAR, Enclosure 12 includes a marked-up Bases page provided for information only. The change proposed in the Bases is directly related to the changes proposed by this license amendment.

a. Revise paragraph 2.C.(1) of the Unit 1 Operating License, DPR-42, and the Unit 2 operating License, DPR-60, to authorize operation at reactor core power levels not in excess of 1677 MWt.

b. Revise TS 1.1, Definitions, RATED THERMAL POWER, to reflect the increase from 1650 MWt to 1677 MWt.

c. Revise TS 5.6.5, Core Operating Limits Report, to add Caldon Engineering Reports 80P and 157P.

d. Revise TS 5.6.6, Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR), to modify the revision number of WCAP-14040 from Revision 2 to Revision 4.

2.2 Background

The purpose of this LAR is to request Nuclear Regulatory Commission (NRC) approval of a 1.64 percent Measurement Uncertainty Recapture Power Uprate (MUR PU) for PINGP Units 1 and 2. The PINGP Units 1 and 2 are presently licensed for a RTP of 1,650 MWt each. The 1.64-percent MUR PU, which is enabled through the use of more accurate feedwater flow and temperature measurement techniques, will result in an increase of the PINGP licensed core thermal power to 1,677 MWt. The analyses comprising the current licensing basis (CLB) of the plant can accommodate the increased power because they are based on a core power of 1,683 MWt which includes 2-percent uncertainty; or, were previously analyzed at a bounding power level for other reasons.

Enclosure 2 summarizes the various evaluations and analyses of the effects of the 1.64-percent MUR PU on plant systems, components, and analyses. The supporting analyses of record bound all currently-licensed fuel designs including the 422 Vantage Plus (422V+) design approved by PINGP Units 1 and 2 Amendments 192/181. Whereas many of the MUR PU analyses were performed explicitly to model the conditions associated with the contemporary Optimized Fuel Assembly (OFA) 400V+ fuel design, other evaluations and analyses have demonstrated conformance of the new 422V+ design to operation at MUR PU conditions. These analyses of 422V+ fuel were performed and approved in the course of license amendments 192/181, and are enumerated in Table II-1.

Some accidents and transients that are described in Table II-1 of Enclosure 2 do not match the descriptions in the currently issued USAR. This is due to a time lag between the last issuance of the USAR to the NRC and the PINGP USAR revision issuance cycle. In accordance with 10 CFR 50.71(e)4, USAR revisions are issued to the NRC on a frequency not to exceed 24 months. Since the last issuance of the USAR to the NRC, updates to the USAR have been made that are not included in the NRC copy of the PINGP USAR. To facilitate the NRC's review, NSPM will provide updated USAR information to the NRC upon request.

3.0 TECHNICAL EVALUATION

3.1 Licensing Methodologies for Uprate

The analytical and licensing work supporting the PINGP MUR PU is consistent with the methodology established by Westinghouse in WCAP-10263, "A Review Plan for Uprating the Licensed Power of a Pressurized Water Reactors Power Plant," (Reference 6.1). The methodology in WCAP-10263 establishes the general approach and criteria for uprate projects including the broad categories that must be addressed. These categories include the Nuclear Steam Supply System (NSSS) parameters, systems, components, design transients, accidents, and the interfaces between the NSSS systems and the Balance of Plant (BOP) systems. This methodology includes the use of well-defined analyses input assumptions and parameter values, the use of currently approved analytical techniques, and the use of currently applicable licensing criteria and standards. This methodology has been successfully used as the basis for power uprate projects on pressurized water reactors, including measurement uncertainty recapture uprates.

Caldon Engineering Reports ER-80P, "Improving Thermal Power Accuracy and Plant Safety While Increasing Operating Power Level Using the LEFM√™ System" (Reference 6.2), and ER-157P, "Supplement to Topical Report ER-80P: Basis for a Power Uprate with the LEFM√™ Check or CheckPlus™ System" (Reference 6.3) have been reviewed by the NRC. The NRC has issued Safety Evaluations (SE) approving these topical reports for referencing in MUR PU submittals (References 6.4 and 6.5). In addition, the NRC staff completed a re-evaluation of the Caldon LEFM ultrasonic flow meters (UFMs) as part of the generic assessment of the hydraulic aspects of UFM application to increase licensed thermal power and issued a SE entitled, "Evaluation of The Hydraulic Aspects of the Caldon Leading Edge Flow Measurement (LEFM) Check and CheckPlus™ Ultrasonic Flow Meters (UFMs)" (Reference 6.6). As described in that SE, the NRC found that the Caldon Check and CheckPlus UFM's performance is consistent with the Caldon Topical Reports ER-80P, Revision 0 and ER-157P, Revision 5, previously approved by the NRC staff. NSPM is specifically applying these topical reports, and the criteria listed in the NRC SEs for them, for a requested 1.64 percent RTP increase.

In addition to the above methodologies, NSPM has taken into account the specific guidance developed by the NRC for the content of MUR PU applications. This guidance was published on January 31, 2002, as NRC RIS 2002-03, "Guidance on the Content of Measurement Uncertainty Recapture Power Uprate Applications" (Reference 6.7). Enclosure 2 of this LAR provides an evaluation of the proposed MUR PU structured to be consistent with this NRC guidance. Finally, NRC requests for additional information (RAI) regarding MUR PU applications for other nuclear Units were reviewed for applicability. This submittal incorporates the results of that review.

3.2 Licensing Approach to Plant Safety, Component, and System Analyses

The reactor core power and the NSSS thermal power are used as inputs to most plant safety, component, and system analyses. Generally, the PINGP analyses model the core and the NSSS thermal power in one of three ways:

1. Some of the analyses apply a 2-percent uncertainty to the licensed power level of 1650 MWt to account solely for the power measurement uncertainty. This results in an assumed core power level of 1683 MWt in the analyses. These analyses have not been re-performed for the 1.64 percent uprate since the sum of the requested core power level (1677 MWt) plus the decreased power measurement uncertainty (0.36 percent) results in an assumed core power level of less than 1683 MWt. Therefore, these analyses are bounded by the previously analyzed conditions.

2. Some of the analyses already employ an assumed core power level bounding the requested 1677 MWt plus the new power level measurement uncertainty of 0.36 percent. These analyses were performed at 1677 MWt core power or greater during previous plant projects. For these analyses, the available margin envelops the 1.64 percent power increase of the MUR PU. Consequently, these analyses have not been re-performed and continue to retain sufficient margin.

3. The remaining analyses are performed at zero percent power conditions or do not actually model the core power level. These analyses have not been re-performed since they are unaffected by the core power level.

3.3 Technical Assessment of the Change in Rated Thermal Power

NSPM has evaluated the impact of the proposed power uprate on safety analyses, NSSS systems and components, and BOP systems. Enclosure 2 summarizes the results of the comprehensive engineering review performed to evaluate the increase in the licensed core rated thermal power. Results of this evaluation are provided in a format consistent with the guidance provided in NRC RIS 2002-03. Additionally, the PINGP MUR PU evaluation was developed consistent with the methodology established in WCAP-10263. The results of the NSPM evaluation demonstrate that applicable acceptance criteria will continue to be met following the implementation of the proposed 1.64 percent MUR PU.

3.4 Technical Assessment of PTLR Methodology Change

PINGP's current heatup and cooldown curves are calculated per WCAP-14040 Revision 2 (Reference 6.15) as described in TS 5.6.6.b. The existing curves cover the period out to 35 effective full power years (EFPY) of operation, which was the current operating license at the time of development. In 2008, PINGP requested extension of the operating license for an additional 20 years per Reference 6.17, so revised neutron fluence calculations were performed to cover the proposed license extension period. These revised fluence calculations also addressed the 1.64 percent increase in rated thermal power proposed herein. In order to develop PTLR curves that address these proposed amendments, NSPM requests to use the methodology in WCAP-14040 Revision 4 (Reference 6.16), which has incorporated ASME Code Cases 588, 640 and 641. The analytical method described in WCAP-14040 Revision 4 has been previously reviewed and approved by the NRC as indicated in the associated Safety Evaluation dated February 27, 2004, which is included in WCAP-14040 Revision 4. NSPM has reviewed WCAP-14040 Revision 4 and determined that it is applicable to this facility, and that use of the new methodologies will provide the appropriate operating curves and limits for operation to the end of proposed licensed plant life at the proposed licensed power level. Therefore, no further technical analysis is required to validate the use of this methodology at PINGP.

Proposed revision to the PTLR is provided as Enclosure 13; the only effect being the change to WCAP-14040 revision to 4. As discussed in Enclosure 2, Section IV, the application of this new methodology has no other effect on the PTLR through its effective period of 35 EFPY.

3.5 Conclusion

NSPM is requesting a 1.64 percent increase in RTP for PINGP Units 1 and 2 from 1650 MWt to 1677 MWt. This power increase will be accomplished by using a more accurate main feedwater flow and temperature measurement system to calculate the RTP. This higher accuracy measurement will be achieved with the use of a Caldon CheckPlus™ LEFM. This LAR has taken into account industry and NRC accepted methodologies and guidelines for power uprates.

This LAR is made pursuant to 10 CFR 50.90 to modify the Operating Licenses and the Technical Specification requirements associated with RTP and the use of the power measurement uncertainty recapture in safety analyses.

4.0 REGULATORY EVALUATION

4.1 Applicable Regulatory Requirements/Criteria

On June 1, 2000 the NRC modified the Code of Federal Regulations (CFR) 10 CFR 50, Appendix K (Federal Register 65 FR 34913, June 1, 2000) to allow licensees to use a power level uncertainty of less than 2 percent in loss-of-coolant-accident (LOCA) analyses. This rulemaking provided licensees with the option of either maintaining the 2-percent power allowance between the licensed core power level and the core power level assumed in the plant licensing basis LOCA analyses, or applying a reduced allowance that accounts for more accurate feedwater flow and temperature measurement techniques.

Consistent with this change to Appendix K, NSPM proposes to reduce the power measurement uncertainty for PINGP Units 1 and 2 to 0.36 percent. Improvement in core power measurement accuracy is possible through the reduction of the uncertainty associated with feedwater flow and temperature measurement used in the power calorimetric calculation. The feedwater flow and temperature measurement uncertainty is reduced through the use of improved measurement instrumentation. This uncertainty reduction is achieved through the use of the Caldon Leading Edge Flow Meter (LEFM) CheckPlus™ System. The existing 2-percent uncertainty margin used in some of the current PINGP licensing bases analyses is re-allocated with 1.64 percent applied to increase the licensed core power level (1,677 MWt) and 0.36 percent retained to account for power measurement uncertainty. The total core power (including uncertainties) assumed in the analyses is 1,683 MWt.

4.2 Precedent

Between May 22, 2006 and October 22, 2009, the NRC issued SEs for MUR PUs for the following plants using the Caldon LEFM CheckPlus™ ultrasonic flow and temperature measurement systems:

1. North Anna Power Station, Unit Nos. 1 and 2, 1.6% MUR PU, approved on October 22, 2009 (Reference 6.14)

2. Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2, 1.38% MUR PU, approved on July 22, 2009 (Reference 6.13)

3. Davis-Besse Nuclear Power Station, Unit 1, 1.63% MUR PU, approved on June 30, 2008 (Reference 6.8)

4. Cooper Nuclear Station, 1.62% MUR PU, approved on June 30, 2008 (Reference 6.9)

5. Vogtle Electric Generating Plant, Units 1 and 2, 1.7% MUR PU, approved on February 27, 2008 (Reference 6.10)

6. Crystal River Unit 3, 1.6% MUR PU, approved on December 26, 2007 (Reference 6.11)

7. Seabrook Station Unit 1, 1.7% MUR PU, approved on May 22, 2006 (Reference 6.12).

With respect to the proposed PTLR analysis methodology change, the following license amendment provides precedent:

- Ginna Nuclear Power Plant, Amendment RE: Revised Methodology for Determining Reactor Coolant System Pressure and Low Temperature Over-Pressure Limits, dated February 23, 2009 (Reference 6.18).

4.3 No Significant Hazards Consideration

Pursuant to 10 CFR 50.90, the Northern States Power Company, a Minnesota corporation (NSPM), doing business as Xcel Energy, hereby requests an amendment to the operating license and the plant Technical Specifications (TS) for Prairie Island Nuclear Generating Plant (PINGP). The proposed amendment would increase the licensed rated thermal power (RTP) as a result of a measurement uncertainty recapture (MUR) power uprate (PU). The information provided in support of this request is based on Nuclear Regulatory Commission (NRC) Regulatory Issue Summary (RIS) 2002-03, "Guidance on the Content of Measurement Uncertainty Recapture Power Uprate Applications."

NSPM has evaluated the License Amendment Request (LAR) against the criteria stated in10 CFR 50.92 to determine if any significant: hazards consideration is involved. NSPM has concluded that this proposed LAR does not involve a significant hazards consideration. The following is a discussion of how each of the 10 CFR 50.92(c) criteria is satisfied.

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

Response: No

There are no changes as a result of the MUR PU to the design or operation of the plant that could affect system, component, or accident mitigative functions. All systems and components will function as designed and the applicable performance requirements have been evaluated and found to be acceptable.

The reduction in power measurement uncertainty allows for the accident and transient safety analyses to continue to be used without modification. This is because the preceding safety analyses were performed or evaluated at either 102 percent of 1650 MWt or higher. Those accidents or transients that were reanalyzed for MUR concluded that the existing analyses remain bounding and the conclusions presented in the Updated Safety Analysis Report (USAR) remain valid.

Analyses at these power levels support a core power level of 1677 MWt with a measurement uncertainty of 0.36 percent. Radiological consequences were performed at 102 percent of 1650 MWt (or higher) and continue to be bounding.

The primary loop components were evaluated for the effects of MUR PU conditions. These analyses also demonstrate the components will continue to perform their intended design functions.
All of the Nuclear Steam Supply System (NSSS) systems will continue to perform their intended design functions during normal and accident conditions. The auxiliary systems and components continue to comply with the applicable structural limits and will continue to perform their intended design functions. The NSSS / Balance of Plant (BOP) interface systems were evaluated and will continue to perform their intended design functions. Plant electrical equipment was also evaluated and will continue to perform within their design ratings.

Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Response: No

No new accident scenarios, failure mechanisms, or single failures are introduced as a result of the proposed change. The LEFM has been analyzed, and system failures will not adversely affect any safety-related system or any structures, systems or components required for transient mitigation. Structures, systems, and components previously required for the mitigation of an event remain capable of fulfilling their intended design function at the uprated power level. The proposed change has no adverse effects on any safety related systems or components and does not challenge the performance or integrity of any safety related system.

The proposed changes do not adversely affect any current system interfaces or create any new interfaces that could result in an accident or malfunction of a different kind than previously evaluated. Operating at the proposed RTP does not create any new accident initiators or precursors. Credible malfunctions are bounded by the current accident analyses of record or recent evaluations demonstrating that applicable criteria are still met with the proposed changes.

Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Response: No

Operation at the 1677 MWt core power does not involve a significant reduction in the margin of safety. The current accident analyses have been previously performed with a 2-percent power measurement uncertainty or at a core power bounding the 1677 MWt. System and component analyses have been completed at operating conditions that envelop the MUR uprated operating conditions. Analyses of the primary fission product barriers at uprated core powers have concluded that all relevant plant operating conditions remain satisfied in regard to integrity and compliance with the regulatory acceptance criteria. Evaluations have been reviewed and approved by the NRC or are in compliance with applicable regulatory review guidance and standards.

Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, the proposed amendment presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and accordingly, a finding of no significant hazards consideration is justified.

4.4 Conclusions

In conclusion, based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

5.0 Environmental Considerations

A review has determined that the proposed amendment would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or would change an inspection or surveillance requirement. However, the proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or a significant increase in the amounts of any effluents that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

6.0 References

6.1 WCAP-10263, "A Review Plan for Uprating the Licensed Power of a Pressurized Water Reactor Power Plant," 1983

6.2 Caldon, Inc. Engineering Report-80P, "Improving Thermal Power Accuracy and Plant Safety While Increasing Operating Power Level Using the LEFM√™ System," Revision 0, March 1997

6.3 Caldon, Inc. Engineering Report-157P, "Supplement to Topical Report ER-80P: Basis for a Power Uprate With the LEFM√™ Check or CheckPlus™ System," Revision 5, October 2001

6.4 Nuclear Regulatory Commission Safety Evaluation, Caldon, Inc., Engineering Topical Report ER-80P, "Improving Thermal Power Accuracy and Plant Safety While Increasing Power Level Using the LEFM√™ System," March 8, 1999 (ADAMS Accession No. 9903190065 - legacy library)

6.5 Nuclear Regulatory Commission Safety Evaluation, Caldon, Inc., Topical Report ER-157P, "Supplement to Topical Report ER-80P: Basis for a Power Uprate with the LEFM√™ or CheckPlus™ System," December 20, 2001 (ADAMS Accession Number ML013540256),

6.6 NRC Safety Evaluation, "Evaluation of The Hydraulic Aspects of the Caldon Leading Edge Flow Measurement (LEFM) Check and CheckPlus™ Ultrasonic Flow Meters (UFMs)", July 5, 2006 (ADAMS Accession Number ML061700222)

6.7 NRC Regulatory Issue Summary 2002-03: "Guidance on the Content of Measurement Uncertainty Recapture Power Uprate Applications," January 31, 2002 (ADAMS Accession Number ML013530183)

6.8 Davis-Besse Nuclear Power Station, Unit 1, Issuance of Amendment RE: Measurement Uncertainty Recapture Power Uprate (TAC No. MD8326), approved June 30, 2008 (ADAMS Accession Number ML081410652)

6.9 Cooper Nuclear Station, Issuance of Amendment RE: Measurement Uncertainty Recapture Power Uprate (TAC No. MD7385), approved June 30, 2008 (ADAMS Accession Number ML081540278)

6.10 Vogtle Electric Generating Plant, Units 1 and 2, Issuance of Amendments Regarding Measurement Uncertainty Recapture Power Uprate (TAC Nos. MD6625 and MD6626), approved February 27, 2008 (ADAMS Accession Number ML073200227)

6.11 Crystal River Unit 3, Issuance of Amendment Regarding Measurement Uncertainty Recapture Power Uprate (TAC No. MD5500), approved December 26, 2007 (ADAMS Accession Number ML073600419)

6.12 Seabrook Station Unit 1, Issuance of Amendment RE: Measurement Uncertainty Recapture Power Uprate (TAC No. MC8434), approved May 22, 2006 (ADAMS Accession Number ML061360034)

6.13 Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2, Issuance of Amendments RE: Measurement Uncertainty Recapture Power Uprate (TAC Nos. MD9554 and MD9555), approved July 22, 2009 (ADAMS Accession Number ML091820366)

6.14 North Anna Power Station, Unit Nos. 1 and 2, Issuance of Amendments Regarding Measurement Uncertainty Recapture Power Uprate (TAC Nos. ME0965 and ME0966), approved October 22, 2009. (ADAMS Accession Number ML092250616)

6.15 WCAP-14040-A, Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves, Revision 2, January 2006.

6.16 WCAP-14040-A, Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves, Revision 4, May 2004.

6.17 NSPM Letter to NRC, L-PI-08-024, Application for Renewed Operating License, dated April 11, 2008 (ADAMS Accession Number ML081130666).

6.18 NRC Letter to Ginna Nuclear Power Plant, Amendment RE: Revised Methodology for Determining Reactor Coolant System Pressure and Temperature and Low Temperature Over-Pressure Limits, dated February 23, 2009 (ADAMS Accession Number ML083530806)

Enclosure 2

To

Letter from Mark A. Schimmel (NSPM)

To

Document Control Desk (NRC)

Summary of Measurement Uncertainty Recapture Power Uprate Evaluation Following the Guidance Provided in Nuclear Regulatory Commission (NRC) Regulatory Issue Summary (RIS) 2002-003

107 pages follow

TABLE OF CONTENTS

LIST OF ACRONYMS

A

AC	Alternating Current
AAC	Alternate AC
ADV	Atmospheric Dump Valve
AF	Auxiliary Feedwater System
ALARA	As-Low-As-Reasonably-Achievable
AMSAC	ATWS Mitigation System Actuation Circuitry
ANS	American Nuclear Society
ANSI	American National Standards Institute
AoR	Analyses of Record
ART	Adjusted Reference Temperature
ARV	Atmospheric Relief Valve
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
ATWS	Anticipated Transient Without Scram

B

BFRV	Bypass Feedwater Regulating Valve
BOL	Beginning of Life
BOP	Balance of Plant
BTU	British Thermal Unit
BWR	Boiling Water Reactor

C

CALM	Secondary Calorimetric Program
CAP	Corrective Action Program
CC	Component Cooling Water System
CF	Correction Factor
CFR	Code of Federal Regulations
CIB	Customer Information Bulletins
CL	Cooling Water
CLB	Current Licensing Basis
CR	Control Room
CS	Containment Spray System
CST	Condensate Storage Tank
CW	Circulating Water System

D

DBA	Design Basis Accident
DC	Direct Current
DNB	Departure from Nucleate Boiling
DNBR	Departure from Nucleate Boiling Ratio
DSS	Diverse Scram System

E

EAB	Exclusion Area Boundary
ECCS	Emergency Core Cooling System
EDG	Emergency Diesel Generators
EFPY	Effective Full Power Year
EC	Engineering Change
EOL	End of Life
EOP	Emergency Operating Procedure
EPA	Environmental Protection Agency
EPRI	Electric Power Research Institute
EQ	Equipment Qualification
ERCS	Emergency Response Computer System
ERG	Emergency Response Guideline
ESF	Engineered Safety Features
ESFAS	Engineered Safety Feature Actuation System

F

FAC	Flow-Accelerated Corrosion
FBCV	Feedwater Bypass Control Valve
FCV	Feedwater Control Valve
FES	Final Environmental Statement
FIV	Flow Induced Vibration
FF	Fluence Factor
FR	Federal Register
FW	Feedwater System

G

GDC	General Design Criteria
GPM	Gallons Per Minute

H

HELB	High Energy Line Break
HFP	Hot Full Power
HI	Heat Input
HVAC	Heating, Ventilation, and Air Conditioning
HZP	Hot Zero Power

I

IASCC	Irradiation Assisted Stress Corrosion Cracking
ISI	In-Service Inspection
IST	In-Service Testing
I&C	Instrumentation and Control
IEEE	Institute of Electrical and Electronics Engineers

L

LAR	License Amendment Request
LCALM	LEFM Secondary Calorimetric
LEFM	Leading Edge Flow Meter
LOCA	Loss-of-Coolant Accident
LOOP	Loss of Offsite Power
LPZ	Low Population Zone
LTOP	Low-Temperature Overpressurization Protection

M

MCO	Moisture Carry Over
MDF	Mechanical Design Flow
MI	Mass Input
MISO	Midwest Independent System Operator
MMF	Minimum Measured Flow
MOV	Motor Operated Valve
Mpph	Million pounds per hour
MRP	Maintenance Reliability Program
MSIV	Main Steam Isolation Valve
MS	Main Steam System
MSIV	Main Steam Isolation Valve
MSLB	Main Steam Line Break
MSSV	Main Steam Safety Valve
MUR PU	Measurement Uncertainty Recapture - Power Uprate
MWe	Megawatt electrical
MWt	Megawatt thermal

N

NEI	Nuclear Energy Institute
NI	Nuclear Instrumentation
NIS	Nuclear Instrumentation System
NMC	Nuclear Management Company
NPDES	National Pollution Discharge Elimination System
NPSH	Net Positive Suction Head
NRC	Nuclear Regulatory Commission
NSPM	Northern States Power Company, Minnesota
NSSS	Nuclear Steam Supply System

O

ODCM	Offsite Dose Calculation Manual
OEA	Operating Experience Assessment
OFA	Optimized Fuel Assembly
OLTP	Original Licensed Thermal Power
OSG	Original Steam Generator

P

PCT	Peak Centerline Temperature
PINGP	Prairie Island Nuclear Generating Plant
PORV	Power-Operated Relief Valve
PAC	Pre-filter Absolute Charcoal
P-T	Pressure-Temperature
PTC	Pressure Test Code
PTLR	Pressure and Temperature Limits Report
PTS	Pressurized Thermal Shock
PWR	Pressurized Water Reactor

R

RCCA	Rod Control Cluster Assembly
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RFO	Refueling Outage
RH	Residual Heat Removal System
RIS	Regulatory Issue Summary
RPV	Reactor Pressure Vessel
RSS	Root Sum Square
RSG	Replacement Steam Generator
RTD	Resistance Temperature Detector
RTDP	Revised Thermal Design Procedure
RTP	Rated Thermal Power
RTS	Reactor Trip System
RT_{NDT}	Reference Temperature Nil-Ductility Temperature
RT_{PTS}	Reference Temperature Pressurized Thermal Shock
RV	Reactor Vessel
RWST	Refueling Water Storage Tank

S

SATP	Safety Analysis Transition Program
SBO	Station Blackout
SE	Safety Evaluation
SG	Steam Generator
SB	Stream Generator Blowdown System
SI	Safety Injection System
SRP	Standard Review Plan
SRSS	Square Root Sum of the Squares
SSC	Systems, Structures and Component

T

TDF	Thermal Design Flow
TEDE	Total Effective Dose Equivalent
T_{FW}	Feedwater Temperature
TPC	Pressure Test Code
TPM	Thermal Power Monitor
TRM	Technical Requirements Manual
TS	Technical Specifications

U

UFM	Ultrasonic Flow Meter
UFSAR	Updated Final Safety Analysis Report
USAR	Updated Safety Analysis Report
USE	Upper Shelf Energy

V

VC	Chemical and Volume Control System
VCALM	Venturi Secondary Calorimetric
V&V	Verification and Validation

W

WCAP	Westinghouse Commercial Atomic Power
WOG	Westinghouse Owners Group

Z

ZA	Auxiliary Building Special Ventilation ZA System
ZH	Safeguards Chilled Water System
ZN	Control Room Area Ventilation System

INTRODUCTION

Prairie Island Nuclear Generating Plant (PINGP) Units 1 and 2 are licensed for a Rated Thermal Power (RTP) of 1650 MWt. Through the use of more accurate feedwater flow measurement equipment, approval is sought to increase this core power by 1.64% to 1677 MWt. The impact of a 1.64% core power uprate for applicable systems, components, and safety analyses has been evaluated.

The evaluation in this amendment request follows the format in the Nuclear Regulatory Commission (NRC) Regulatory Issue Summary (RIS) 2002-03, "Guidance on the Content of Measurement Uncertainty Recapture Power Uprate Applications" dated January 31, 2002. Requests for Additional Information (RAI) regarding MUR PU LARs submitted by other licensees were reviewed for applicability to the PINGP submittal. This enclosure incorporates the results of that review.

I. Feedwater flow measurement technique and power measurement uncertainty

1. A detailed description of the plant-specific implementation of the feedwater flow measurement technique and the power increase gained as a result of implementing this technique.

Cameron (formally Caldon) manufactures the Caldon Leading Edge Flow Meter (LEFM) CheckPlus™ System used to measure feedwater flow and temperature. As described below, the PINGP Unit 1 and Unit 2 LEFM systems contain an individual LEFM metering spool piece on each of two feedwater lines for each Unit. Each meter is comprised of eight chordal paths which are separated into two independent flow planes, each plane being capable of measuring full feedwater flow and feedwater temperature. Feedwater flow and temperature are the main inputs for determining the plant secondary calorimetric power, which is used in turn to verify the core thermal power output. The LEFM uses the transit times of ultrasonic pulses traveling upstream and downstream to calculate the fluid velocity along each of four chords in a plane of the circular cross-section of the feedwater pipe. When both planes in each meter are operating, the LEFM system status is identified as “Normal” and the PINGP power measurement uncertainty is calculated to be 0.36%. When only one flow plane from a meter is operating, LEFM system status is identified as “Alert” and the power measurement uncertainty is increased slightly from 0.36% to 0.54% (References I.2.1 and I.2.2). These uncertainties include the uncertainty associated with transducer replacement described in Section I.1.D.3 below.

The four velocities in each plane are numerically integrated to determine the volumetric flow, which is then combined with pressure and temperature conditions to determine mass flow through the feedwater pipe. This flow measurement method yields highly accurate flow readings and has been approved by the Nuclear Regulatory Commission (NRC) for power uprate applications as documented in Caldon Topical Reports ER-80P and ER-157P (References I.2.3 and I.2.4).

The PINGP Unit 1 and Unit 2 LEFM CheckPlus™ Systems are each comprised of two metering section spool pieces and an Electronics Unit Cabinet. 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 spool pieces are installed in the Loop A and B feedwater lines downstream of the existing feedwater flow venturis. The Unit 2 spool pieces are installed in the Loop A and B feedwater lines upstream of the existing feedwater flow venturis. 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 Plant 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 as described in paragraph I.1.C., below.

The installation location of the Unit 1 and Unit 2 spool locations was reviewed by Cameron and determined to meet the requirements of Topical Reports ER-80P and ER-157P as part of the setup for Unit- specific calibration testing performed and documented by Cameron at Alden Labs (References I.2.5 through I.2.8). Cameron performed testing at Alden Labs of the Unit 1 and Unit 2 spool pieces in their Unit-specific piping configuration to determine the individual meter factor and individual path normalized velocities for each spool piece.

During final commissioning following installation and startup, the actual in-plant data were determined to be consistent with the data obtained at Alden Labs and that the individual system components are operating as designed within pre-established limits (Reference I.2.39). This ensures that "as-installed" measurement uncertainties are within the bounding values used in the Unit-specific power measurement uncertainty analyses. The systems are designed with internal monitoring and checking devices to ensure the system parameters are within design limits during system operation. In the event system parameters exceed pre-established limits, a system alarm occurs which results in a Control Room critical ERCS alarm indicating 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.

A. Identification (by document title, number, and date) of the approved topical report on the feedwater flow measurement technique

The referenced topical reports applicable to the proposed Measurement Uncertainty Recapture Power Uprate (MUR PU) flow measurement equipment are as follows:

1. Caldon, Inc., Engineering Topical Report ER-80P, "Improving Thermal Power Accuracy and Plant Safety While Increasing Power Level Using the LEFM√™ System," Revision 0, March 1999.
2. Caldon, Inc., Topical Report ER-157P, "Supplement to Topical Report ER-80P: Basis for a Power Uprate with the LEFM√™ or CheckPlus™ System," Revision 5, October 2001.

B. A reference to the NRC's approval of the proposed feedwater flow measurement technique

1. The NRC approved Caldon Topical Report ER-80P, Revision 0, per NRC Safety Evaluation (9903190065 - legacy library), March 8, 1999

2. The NRC approved Caldon Topical Report ER-157P, Revision 5, per NRC Safety Evaluation (ML013540256), December 20, 2001

   In addition, the NRC staff completed a re-evaluation of the Caldon LEFM ultrasonic flow meters (UFM) as part of the generic assessment of the hydraulic aspects of UFM application to increase licensed thermal power and issued NRC Safety Evaluation "Evaluation of The Hydraulic Aspects of the Caldon Leading Edge Flow Measurement (LEFM) Check and CheckPlusTM Ultrasonic Flow Meters (UFMs)," NRC Safety Evaluation (ML061700222). As described in that SE, the NRC found that the Caldon Check and CheckPlus UFMs' performance is consistent with the Caldon Topical Reports ER-80P, Revision 0 and ER-157P, Revision 5, previously approved by the NRC staff.

C. A discussion of the plant-specific implementation of the guidelines in the topical report and the staff's letter/safety evaluation approving the topical report for the feedwater flow measurement technique

Installation of the LEFM feedwater flow and temperature measurement system on Unit 1 and Unit 2 provides precision feedwater flow and temperature inputs to the ERCS Secondary Calorimetric Program used to calculate reactor thermal power (References I.2.10 and I.2.11). While there are minute differences between the Unit 1 and Unit 2 system uncertainty components as shown in Table I.1.E.1 of this enclosure, these differences are not large enough to cause a variance in the final calculated thermal power uncertainty values for Unit 1 and Unit 2. The combined uncertainties of the LEFM feedwater flow and temperature measurements are approximately 0.3% which reduces the total reactor power measurement uncertainty to 0.36% or less in normal mode when the other calorimetric input parameter uncertainties are considered. In the maintenance (ALERT) Mode, the total power uncertainty is limited to 0.54% or less, per Column 3 of Table I.1.E.1.

LEFM status and calorimetric data are supplied to the PINGP plant computer via a server-based interface program. This program makes the LEFM status and calorimetric data available for use for displays, trending, and various program uses as necessary. As part of the LEFM installation, the Control Room Thermal Power Monitor displays were revised to display the current LEFM status, allowable power based on LEFM status, and the current power levels determined by the secondary calorimetrics calculated using the LEFM and venturi inputs, and the Nuclear Instrumentation System (NIS). An ERCS alarm was provided to alert the Operators in the event the selected calorimetric source is above the allowable power. In addition, the TPM display was revised to allow the Operators to select between the secondary calorimetric using the LEFM or venturi inputs as described below and the TPM display was revised to provide alarm indication when power is greater than the allowable power.

The changes to the CALM program incorporate the LEFM feedwater flow and temperature inputs to calculate an additional independent secondary calorimetric (LCALM) in addition to the existing venturi calorimetric (VCALM). The CALM and TPM programs simultaneously display the results of each calorimetric for comparison. Other than feedwater flow and temperature, the LCALM and VCALM calorimetric programs use the same following inputs:

- Main Steam Pressure
- Main Steam and Feedwater Enthalpy
- Blowdown Flow
- RCP Net Heat Input
- Feedwater Header Pressure

In the event the LEFM internal checking system detects an alarm condition or other condition impacting a single flow plane on either LEFM meter, the LEFM status will change to "ALERT" (maintenance mode). The change in LEFM status will result in the Control Room ERCS "LEFM Change of Status" alarm sounding and the Control Room TPM display will display the LEFM status and allowable power based on the system status (1674 MWt for Alert – 1650 MWt for Fail). If current power is greater than the allowable power for the system status, the TPM display will indicate an alarm condition (red power indication).

With the LEFM in the "Alert" mode, the LCALM will continue to be used as the selected calorimetric source. If the LEFM is not returned to normal status before the next scheduled daily Power Range Nuclear Instrumentation calibration, power must be reduced equal to or less than the allowable power (1674 MWt – 99.82%).

If the LEFM status changes to "Fail", the CALM program automatically reverts to using the VCALM calorimetric using feedwater venturi flow and inline temperature inputs as the calorimetric source. A LEFM failure is displayed if 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 CALM and TPM programs provide critical computer alarms and indications to the Operators in the Control Room to display the selected calorimetric source (LEFM or Venturi), LEFM status, current power, and maximum allowable power based on LEFM status. Operator actions required to respond to the LEFM change of status are provided in a critical computer Alarm Response Procedure (Reference I.2.12) for LEFM Change of Status. In the planned revision to the TRM, if the LEFM is not returned to normal status before the next scheduled daily Power Range Nuclear Instrumentation calibration, power must be reduced equal to or less than the allowable power (1650 MWt – 98.38%).

In addition to providing feedwater flow and temperature inputs to the LCALM secondary calorimetric program, individual time-averaged correction factors based on the LEFM feedwater flow and temperature inputs are generated for each of the venturi feedwater flow and inline temperature inputs to the CALM program. These correction factors are used to normalize the VCALM mass flow and inline temperature inputs to the LEFM flow and temperature values (e.g., LEFM mass flow ÷ Venturi mass flow). The correction factors minimize the deviation between the LCALM and the VCALM calorimetric calculations and provide a mechanism for monitoring venturi fouling and instrument channel drift. In the event the correction factor quality becomes "Bad" (not normal), based on insufficient input data, VCALM will use the last known good calculated correction factor.

In the event LEFM status changes to "Fail" the secondary calorimetric program reverts to using the VCALM as the calorimetric source. VCALM will use the last known "good" calculated correction factors for the feedwater flow and temperature. Anytime the CALM program is using VCALM as the selected calorimetric source, manual Operator action is required to initiate using LCALM as the selected calorimetric source. The capability exists to manually shift to the VCALM calorimetric as well as override the LEFM generated venturi and inline temperature correction factors.

D. The dispositions of the criteria that the NRC staff stated should be addressed (i.e., the criteria included in the staff's approval of the technique) when implementing the feedwater flow measurement technique

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

Maintenance of the unit-specific LEFM system is performed each refueling using site-specific procedures (References I.2.13 and I.2.14) developed in accordance with the guidelines established in the Vendor Maintenance and Troubleshooting Manual. Proper maintenance is assured through both automatic and manual checks of the system. Calibration of the LEFM pressure inputs is performed each refueling as part of the normal calibration program for the pressure transmitters (References I.2.15 through I.2.20).

Ultrasonic signal verification and alignment is performed automatically with the LEFM system. Signal verification is possible by review of signal quality measurements performed and displayed by the LEFM system. In addition, signal verification status is also provided to the plant computer for monitoring by plant staff on a quarterly basis using plant-specific procedures (References I.2.21 and I.2.22).

Calibration and maintenance are performed by qualified personnel using site procedures. The site procedures were developed using the Caldon technical manuals. All work is performed in accordance with the site work management process (Reference I.2.23). Formal on-site vendor training in the operation and maintenance of the LEFM has been provided to PINGP Instrumentation and Control (I&C) personnel.

In addition to the calibration and maintenance of the LEFM, all other instrument components that provide fluid condition data for calculation of rated thermal power are controlled, calibrated, and monitored to the conditions represented in the overall calorimetric uncertainty evaluation done for the PINGP 1.64% power uprate.

Corrective actions involving maintenance are performed by qualified personnel. At PINGP, the LEFM systems are included in the preventive maintenance program. As a plant system, all equipment problems fall under the site work control process. All conditions that are adverse to quality are documented under the Corrective Action Program.

As described in Topical Reports ER-80P and ER-157P, the LEFM contains self-diagnostics that detect all possible system failures and changes in hydraulic velocity profiles that affect the accuracy of ultrasonic flow measurement devices. Alarm thresholds are set to provide notification prior to a condition that may lead to operation outside its design-basis accuracy. The LEFM does not perform any safety function, and is not used to directly control any plant systems. Therefore, LEFM inoperability has no immediate effect on plant operation.

During operation with an LEFM "ALERT" or "FAIL" status, the total power measurement uncertainty using the flow measurement calculated by the LEFM is increased. An "ALERT" alarm is initiated by conditions such as loss of redundant component or a process parameter is outside a predetermined range which only affects a single plane in either or both system meters. With an "ALERT" alarm, the affected loop(s) is operating with a single plane which results in the reactor power measurement uncertainty being increased slightly (from 0.36% to 0.54%). "FAIL" alarms are initiated when redundant component or process parameters are outside a predetermined range which affects both planes in a meter or the data link between the LEFM cabinet and the plant ERCS computer is lost. With a "FAIL" alarm the uncertainty increases back to the uncertainty associated with using the feedwater venturis (2%).

With an LEFM "ALERT" or "FAIL" status, operation at MUR PU conditions of 1677 MWt may continue until the next scheduled daily power calorimetric. Operation at licensed core power until the next scheduled calorimetric is permissible since the power range Nuclear Instruments (NI) will still be operating within their Technical Specification (TS) required daily calibration window. If the LEFM status is not returned to "Normal" before the next scheduled daily power calorimetric, core power must be reduced consistent with the increase in power measurement uncertainty before performance of the calorimetric. For an "ALERT" status, a reduction of 0.18% to 99.82% (1674 MWt) will be required. For a "FAIL" status, a reduction to 98.38% (original licensed power of 1650 MWt) will be required. With the LEFM in an "ALERT" or "FAIL" status, the TPM monitors will display the allowable core power associated with the current LEFM status. If the calorimetric power displayed on the TPM is greater than the displayed allowable core power value, TPM alarm indication will alert the Operators that power is above the allowable power level and provide visual indication of the required power level at the next scheduled power calorimetric.

With the LEFM operating in an "ALERT" status, once power is reduced to the allowable power displayed on the TPM monitors (99.82% - 1674 MWt), the LCALM calorimetric may be used as the source of the power calorimetric and provide correction factors for the VCALM feedwater flow and inline temperature inputs indefinitely.

Operation with the LEFM operating with a "FAIL" status requires the calorimetric source be provided from the venturi VCALM calorimetric and power reduced to the allowable value displayed on the TPM Monitors (98.38% - 1650 MWt). Operation with the last good correction factors derived from the LEFM will be limited to a maximum of seven days. Based on plant specific trending of the venturi feedwater flow and temperature correction factors, the potential drift of the feedwater venturi flow and temperature RTD instrumentation within seven days is considered quite small (<0.1%), therefore limiting the use of the last known good correction factors to seven days or less is considered acceptable. If the LEFM system status is not returned to service in the allotted time, correction factors for the VCALM feedwater flow and inline temperature inputs will be set to 1.0.

Based on the Unit 1 and Unit 2 flow errors identified in section I.1.D.2 below, if the feedwater flow and temperature correction factors are greater than 1.0, reactor thermal power calculated by VCALM must be reduced an additional amount equivalent to the combined average of the feedwater flow correction factors. (e.g. if Loop A CF = 1.0025 and Loop B CF = 1.0020, additional required power reduction would be (.0025 + .0020)/2 = .0045/2 = .00225 = .225%). No additional power reduction is required if the CF is ≤1.0.

As part of the MUR PU implementation, the above responses to a change in LEFM status will be captured in a new Technical Requirements Manual (TRM) section. In addition, the current ERCS critical Alarm Response Procedure will be updated to reflect the power limits associated with the LEFM status.

2. **For plants that currently have LEFM installed, the licensee should provide an evaluation of the operational and maintenance history of the installation and confirm that the installed instrumentation is representative of the LEFM system, and bounds the analysis and assumptions set forth in ER-80P.**

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

Following commissioning of the Unit 1 LEFM and prior to turnover for operation, monitoring of the Unit 1 LEFM showed the calorimetric calculated using the LEFM feedwater flow and temperature inputs (LCALM) indicated a higher power than the calorimetric calculation calculated using the feedwater venturis and inline feedwater temperature (VCALM) by approximately 1.0% with the largest source of the deviation being associated with the flow indication for Loop A feedwater flow. Corrective action was initiated. As part of the corrective actions, Unit 1 reactor power was reduced to maintain reactor power less than 1650 MWt based on the LCALM calorimetric until system turnover to Operations. Following commissioning and turnover of the LEFM, reactor power has been controlled based on the LCALM calorimetric using the LEFM feedwater flow and temperature inputs. In addition, the time-based correction factors calculated for the venturi feedwater flow and inline temperature indications were initiated in the VCALM calorimetric normalizing the VCALM with the LCALM. Since turnover of the Unit 1 LEFM, the system has operated normally with no maintenance being required. Normal system inspections have been completed 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. Following the return to full power, the venturi feedwater flow and temperature correction factors returned to the values (+/- 0.001) that existed prior to the trip.

Following commissioning of the Unit 2 LEFM and prior to turnover for operation, monitoring of the Unit 2 LEFM showed the calorimetric calculated using the LEFM feedwater flow and temperature inputs (LCALM) indicated a higher power than the calorimetric calculation calculated using the feedwater venturis and inline feedwater temperature (VCALM) by approximately 0.5% with the largest source of the deviation being associated with the flow indication for Loop A feedwater flow. Corrective action was initiated. As part of the corrective actions, Unit 2 reactor power was reduced to maintain reactor power less than 1650 MWt based on the LCALM calorimetric until system turnover to Operations. Following turnover of the LEFM, reactor power has been controlled based on the LCALM calorimetric using the LEFM feedwater flow and temperature inputs. In addition, the time-based correction factors calculated for the venturi feedwater flow and inline temperature indications were initiated in the VCALM calorimetric normalizing the VCALM with the LCALM. Since turnover of the Unit 2 LEFM, the system has operated normally with no maintenance being required. Normal system inspections have been completed in accordance with approved station procedures.

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

The methodology used to calculate the LEFM uncertainties is consistent with ASME Pressure Test Code (PTC) 19.1 and Instrument Society of America (ISA) 67.04 as approved in Topical Reports ER-80P and ER-157P.

The errors associated with the ERCS computer calculation of plant thermal power based on the installed venturis were determined in the PINGP Unit 1 and Unit 2 Secondary Power Calorimetric Uncertainty Calculations (Reference I.2.37 and I.2.38). These calculations use a sensitivity analysis to determine the relative impact of channel errors associated with each defined ERCS input value. Based on the sensitivity analysis, errors with a negligible contribution to the total calorimetric error were not considered. Generally, errors that were less than 10 percent of the largest errors were considered to be insignificant. The remaining errors were combined using the Square Root Sum of the Squares (SRSS) method, since the errors are evaluated as random and near normally distributed.

The LEFM equipment vendor power measurement uncertainty calculations for the Unit 1 and Unit 2 LEFM installations used the PINGP calculations and replaced the errors associated with the venturis and the inline feedwater temperature instruments with the errors for the LEFM. 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 SRSS methodology discussed above. 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.

In response to questions concerning the impact of changing or replacing flow transducers on the LEFM Check and CheckPlus™ System uncertainty analysis raised by the NRC during the NRC evaluation of the LEFM Check and CheckPlus™ System, Cameron performed testing to evaluate the affect on the system uncertainty when transducers were changed or replaced. Results of these tests are reported in Cameron Engineering Report ER-551 (Reference I.2.28) which was submitted to the NRC in April 2007 (ML071500358 and ML071500360). Cameron issued Customer Information Bulletin (CIB) 125 (Reference I.2.29) as a summary of the results of these test with proposed customer actions necessary to account for the potential changes in site-specific system uncertainty analysis. The total power measurement uncertainty calculations performed by Cameron for PINGP Units 1 and 2 include the uncertainty associated with transducer replacement.

PINGP operating and maintenance procedures for the LEFM have been developed to ensure that the assumptions and requirements of the uncertainty calculation remain valid. (References I.2.13 through I.2.22, I.2.30 and I.2.31).

4. **Licensees for plant installations where the LEFM was not installed with flow elements calibrated to a site-specific piping configuration (flow profiles and meter factors not representative of the plant-specific installation), should provide additional justification for use. This justification should show either that the meter installation is 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 the plant configuration for the specific installation, including the propagation of flow profile effects at higher Reynolds numbers. Additionally, for previously installed and calibrated LEFM, the licensee should confirm that the piping configuration remains bounding for the original LEFM installation and calibration assumptions.**

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, due to 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 eight inches closer to the inlet of the feedwater venturis than during the testing at Alden Labs. Post-installation commission testing of the Unit 2 LEFMs verified the actual plant installation remained bounded by the original LEFM installation and calibration assumptions.

E. A calculation of the total power measurement uncertainty at the plant, explicitly identifying all parameters and their individual contribution to the power uncertainty

Table I.1.E.1 below shows the uncertainty results for both Units. The overall thermal power calorimetric uncertainty for Unit 2 of 0.36% is the limiting value. For consistency in Unit operations, this limiting value will be used for both Units.

The uncertainties shown in Table I.1.E.1 were developed utilizing the calculation methodology in Caldon Topical Reports ER-80P and ER-157P.

Table I.1.E.1

PINGP Unit 1 and Unit 2 Core Thermal Power Uncertainty from ER 532 and ER 533 (References I.2.1 and I.2.2)		
Caldon Uncertainty Calculation		
	Normal	Maintenance
1. Hydraulics		
Profile Factor	0.19%	0.44%
2. Geometry		
Spool Piece Dimensions	0.11%	0.11%
Spool Piece Alignment	0.00%	0.00%
Transducer Location/Replacement	0.09%	0.13%
Spool Piece Expansion, Material Properties	0.08%	0.08%
Spool Piece Thermal Expansion, Temperature	0.00%	0.00%
3. Time Measurements		
Time of Flight Measurements	0.18% (0.19%)	0.18% (0.19%)
Non Fluid Delay	0.02%	0.02%
4. Sub Total, Volumetric Flow Uncertainty (RSS of items 1, 2 and 3)	0.31%	0.51%
5. Temperature/Correlation	0.5°F	0.5°F
6. Temperature/Spool Piece Dimensions	0.03°F	0.03°F
7. Temperature/Time of Flight, Non Fluid Delay	0.17°F	0.17°F
8. Temperature/Pressure	-0.22°F	-0.22°F
9. Feedwater Density		
Feedwater Density/ASME Correlation	0.04%	0.04%
Feedwater Density/Temperature	-0.05%	-0.05%
Feedwater Density/Pressure	0.03%	0.03%
10. Feedwater Enthalpy (Pressure & Temp)		
Derivative of Feed Enthalpy to Temperature	-0.139%	%/ °F
Derivative of Feed Enthalpy to Pressure	0.000%	%/psi
Feedwater Enthalpy/Temperature	-0.07%	-0.07%
Feedwater Enthalpy/Pressure	0.03%	0.03%
Power Uncertainty, Thermal Expansion	0.04%	0.04%
11. Steam Enthalpy (Pressure & Moisture)		
Steam Enthalpy/Moisture	-0.006% (-0.008%)	-0.006% (-0.008%)
Steam Enthalpy/Pressure	-0.083% (-0.056%)	-0.08% (-0.06%)
12. Gains/Losses		
Gain/Losses	0.052%	0.024%
13. Total Thermal Power Uncertainty		
Root Sum Square (RSS)	0.36%	0.54%

Notes:
1. Values in parentheses for Unit 2
2. The total uncertainty in temperature, as determined by the LEFM, is the Root Sum Square of the individual random contributors delineated in items 5, 6, 7 and 8

F. **The information to specifically address the following aspects of calibration and maintenance procedures related to all instruments that affect the power calorimetric:**

i. **Maintaining calibration**

Maintenance of the LEFM is performed in accordance with the guidelines established in the Vendor Maintenance and Troubleshooting Manual. Proper maintenance is assured 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.

Ultrasonic signal verification and alignment are performed automatically with the LEFM. Signal verification is possible by review of signal quality measurements performed and displayed by the LEFM. In addition, signal verification status is also provided to the plant computer for monitoring and trending by plant staff using plant specific procedures.

All other instrument components that provide fluid condition data for calculation of rated thermal power were unaffected by the addition of the LEFM and will be maintained according to existing calibration and maintenance procedures.

ii. **Controlling software and hardware configuration**

The LEFMs are designed and manufactured in accordance with Cameron's 10 CFR 50, Appendix B-compliant Quality Assurance Program. Cameron maintains procedures that provide for user notification of deficiencies that could affect the accuracy and reliability of mass flow and temperature measurements. Cameron's Verification and Validation (V&V) Program fulfills the requirements of American National Standards Institute/Institute of Electrical and Electronics Engineers – American Nuclear Society (ANSI/IEEE-ANS) Std. 7-4.3.2, "IEEE Standard Criteria for Digital Computers in Safety Systems of Nuclear Power Generating Stations," Annex E, and American Society of Mechanical Engineers (ASME) NQA-2a-1990, "Quality Assurance Requirements for Nuclear Facility Applications", Subpart 2.7 (Reference I.2.33). In addition, the program is consistent with guidance for software V&V in the Electric Power Research Institute (EPRI) TR-103291, "Handbook for Verification and Validation of Digital Systems" (Reference I.2.34).

Specific examples of quality measures undertaken in the design, manufacture, and testing of the LEFM CheckPlus™

Systems are provided in Section 6.4 and Table 6-1 of Topical Report ER-80P.

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 (Reference I.2.35).

iii. Performing corrective actions

Corrective actions are performed by qualified Maintenance personnel utilizing controlled plant procedures, and controlled by the site work control process. Any conditions that are adverse to quality are documented and evaluated under the site Corrective Action Program.

iv. 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. Conditions adverse to quality are documented and evaluated under the Corrective Action Program and subsequently transmitted to the vendor as appropriate.

v. Receiving and addressing manufacturer deficiency reports

The PINGP LEFMs are included in Cameron's QA program. Procedures are maintained for user notification of significant deficiencies and are processed through Cameron's Customer Information Bulletins (CIB). Any deficiency reports coming to PINGP for the LEFM via a Cameron CIB or industry operating experience would be screened and addressed under the Operating Experience Assessment (OEA) process.

G. A proposed outage time for the instrument, along with the technical basis for the time selected

As part of the MUR PU implementation, the proposed outage times and reactor thermal power limitations for a change in LEFM status will be captured in a new TRM section. In addition, the current ERCS critical Alarm Response Procedure will be updated to reflect the proposed outage time and power limits associated with the LEFM status.

With an LEFM "ALERT" or "FAIL" status, operation at licensed core thermal power (1677 MWt) may continue until the next scheduled daily power calorimetric. Operation at licensed core power until the next scheduled calorimetric is permissible since the nuclear instruments will still be operating within their Technical Specification (Reference I.2.36) required daily calibration window. If the calorimetric power displayed on the TPM is greater than the displayed allowable core power value, TPM alarm indication will alert the Operators that power is above the allowable power level and provides visual indication of the required power level at the next scheduled power calorimetric.

With the LEFM operating in an "ALERT" status, once power is reduced to the allowable power displayed on the TPM monitors (99.82% - 1674 MWt), the LCALM calorimetric may be used as the source of the power calorimetric and provide correction factors for the VCALM feedwater flow and inline temperature inputs indefinitely.

Operation with the LEFM operating with a "FAIL" status requires the calorimetric source be provided from the venturi VCALM calorimetric and power reduced to the allowable value displayed on the TPM Monitors (98.38% - 1650 MWt). Operation with the last good correction factors derived from the LEFM will be limited to a maximum of seven days. Based on plant specific trending of the venturi feedwater flow and temperature correction factors, the potential drift of the feedwater venturi flow and temperature RTD instrumentation within seven days is considered quite small (< 0.1%). Therefore limiting the use of the last known good correction factors to seven days or less is considered acceptable.

H. Proposed actions to reduce power level if the allowed outage time is exceeded, including a discussion of the technical basis for the proposed reduced power level

As part of the MUR PU implementation, the proposed outage times and reactor thermal power limitations for a change in LEFM status will be captured in a new TRM section. In addition, the current ERCS critical Alarm Response Procedure will be updated to reflect the proposed outage time and power limits associated with the LEFM status.

If the LEFM status is not returned to "Normal" before the next scheduled daily power calorimetric, core power must be reduced consistent with the increase in power measurement uncertainty before performance of the calorimetric. For an "ALERT" status, a reduction of approximately 0.18% to 99.82% (1674 MWt) will be required. For a "FAIL" status, a reduction to 98.38% (1650 MWt) will be required. With the LEFM in an "ALERT" or "FAIL" status, the TPM monitors display the allowable core power based on the current LEFM status. If the calorimetric power displayed on the TPM is greater than the displayed allowable core power value, TPM alarm indication will alert the Operators that power is above the allowable power level and provide visual indication of the required power level at the next scheduled power calorimetric.

With the LEFM operating in an "ALERT" status, once power is reduced to the allowable power displayed on the TPM monitors (99.82% - 1674 MWt), the LEFM LCALM calorimetric may be used as the source of the power calorimetric and provide correction factors for the VCALM feedwater flow and inline temperature inputs indefinitely.

Operation with the LEFM operating with a "FAIL" status requires the calorimetric source be provided from the venturi VCALM calorimetric and power reduced to the allowable value displayed on the TPM Monitors (98.38% - 1650 MWt). The revised TRM will require that, if the LEFM is not returned to service within seven days, the feedwater flow and temperature correction factors must be set to 1.0.

Based on the Unit 1 and Unit 2 flow errors identified in Sections 1.D.2 and 1.D.3 above, If the feedwater flow and temperature correction factors are greater than 1.0, reactor thermal power calculated by VCALM must be reduced an additional amount equivalent to the combined average of the feedwater flow correction factors (e.g., if Loop A CF = 1.0025 and Loop B CF = 1.0020, additional required power reduction would be (.0025 + .0020)/2 = .0045/2 = .00225 = .225%). No additional power reduction is required if the CFs are ≤1.0.

2. References for Section I

I.2.1 Caldon Engineering Report ER 532, "Uncertainty Analysis For Thermal Power Determination At Prairie Island Unit 1 NPP Using the LEFM √ + System", Rev 1, February 2008

I.2.2 Caldon Engineering Report ER 533, "Uncertainty Analysis For Thermal Power Determination At Prairie Island Unit 2 Using the LEFM √ + System", Rev 2, July 2008

I.2.3 Caldon Engineering Topical Report-80P, "Improving Thermal Power Accuracy and Plant Safety While Increasing Power Level Using the LEFM System," Rev 0, March 1999

I.2.4 Caldon Engineering Report-157P, "Supplement to Topical Report ER 80P: Basis for a Power Uprate with the LEFM√TM Check or CheckPlusTM System," Rev 5, October 2001

I.2.5 Cameron Engineering Report ER-583, "LEFM√+, Meter Factor Calculation and Accuracy Assessment for Prairie Island Unit 1 Nuclear Power Station", Rev 0, February 2007

I.2.6 Cameron Engineering Report ER-553, "LEFM√+, Meter Factor Calculation and Accuracy Assessment for Prairie Island Unit 2 Nuclear Power Station", Rev 2, July 2008

I.2.7 Caldon Report ALD-1093, "Hydraulic Calibration Plan For Prairie Island Unit 1 LEFM√+ 16" Chordal Spool Pieces", Rev 0, January 16, 2007

I.2.8 Caldon Report ALD-1085, "Hydraulic Calibration Plan For Prairie Island Unit 2 LEFM√+ 16" Chordal Spool Pieces", Rev 0, July 25, 2006

I.2.9 Nuclear Regulatory Commission Safety Evaluation, "Evaluation of The Hydraulic Aspects of the Caldon Leading Edge Flow Measurement (LEFM) Check and CheckPlus™ Ultrasonic Flow Meters", July 5, 2006 (ADAMS Accession Number ML061700222)

I.2.10 Prairie Island Nuclear Generating Plant, Engineering Calculation EC 1009, "Install and Commission PINGP Unit 1 Main Feedwater Ultrasonic Leading Edge Flow Meters", September 24, 2007

I.2.11 Prairie Island Nuclear Generating Plant, Engineering Calculation EC 390, "Installation and Commissioning the Prairie Island Nuclear Generating Plant (PINGP) Unit 2 Main Feedwater Ultrasonic Leading Edge Flow Meters", Rev 0, September 12, 2007

I.2.12 Prairie Island Nuclear Generating Plant, Alarm Response Procedure C47041-36, Rev 6

I.2.13 Prairie Island Nuclear Generating Plant, Preventive Maintenance Procedure ICPM 1-325, "LEFM System Refueling Outage Checks and Inspection Procedure", Rev 0, August 26, 2009

I.2.14 Prairie Island Nuclear Generating Plant, Preventive Maintenance Procedure ICPM 2-325, "Unit 2 LEFM System Refueling Outage Checks and Inspections", Rev 0, April 10, 2009

I.2.15 Prairie Island Nuclear Generating Plant Unit 1, Surveillance Procedure SP 1790, "Feedwater Control System Transmitter Calibration", Rev 14, August 25, 2008

I.2.16 Prairie Island Nuclear Generating Plant Unit 2, Surveillance Procedure SP 2790, "Feedwater Control System Transmitter Calibration", Rev 12, November 21, 2008

I.2.17 Prairie Island Nuclear Generating Plant Unit 1, Surveillance Procedure SP 1002A, "Analog Protection System Calibration", Rev 38, October 13, 2008

I.2.18 Prairie Island Nuclear Generating Plant Unit 2, Surveillance Procedure SP 2002A, "Analog Protection System Calibration", Rev 32, October 14, 2008

I.2.19 Prairie Island Nuclear Generating Plant Unit 1, Surveillance Procedure SP 1002B, "Reactor Protection and Control Transmitters Calibration/Inspection", Rev 33, July 25, 2006

I.2.20 Prairie Island Nuclear Generating Plant Unit 2, Surveillance Procedure SP 2002B, "Reactor Protection and Control Transmitters Calibration/Inspection", Rev 30, February 4, 2005

I.2.21 Prairie Island Nuclear Generating Plant Unit 1, Maintenance Procedure TP 1494, "Unit 1 LEFM System Inspection", Rev 0, October 30, 2008

I.2.22 Prairie Island Nuclear Generating Plant Unit 2, Maintenance Procedure TP 2494, "Unit 2 LEFM System Inspection", Rev 0, January 15, 2009

I.2.23 Prairie Island Nuclear Generating Plant, FP-WM-OVW-01, "Work Management Process Overview", Rev 1, November 20, 2007

I.2.24 Deleted

I.2.25 Deleted

I.2.26 American Society of Mechanical Engineers, ASME Pressure Test Code (PTC) 19.1, "Test Uncertainty", 1990

I.2.27 Instrument Society of America, (ISA) 67.04, "Setpoints for Nuclear Safety-Related Instrumentation", September 1994

I.2.28 Cameron Engineering Report ER-551P, "LEFM CheckPlus Transducer Installation Sensitivity", Rev 3, April 2008

I.2.29 Cameron Customer Information Bulletin (CIB) 125, "Transducer (Re) Placement Uncertainty", Rev 0, April 23, 2007

I.2.30 Prairie Island Nuclear Generating Plant, Maintenance Procedure 1D110, "Unit 1 LEFM System Startup and Shutdown", Rev 1, November 23, 2008

I.2.31 Prairie Island Nuclear Generating Plant, Maintenance Procedure 2D110, "Unit 2 LEFM System Startup and Shutdown", Rev 0, November 21, 2008

I.2.32 Deleted

I.2.33 Deleted

I.2.34 Electric Power Research Institute (EPRI) TR-103291, "Handbook for Verification and Validation of Digital Systems," December 1994

I.2.35 Prairie Island Nuclear Generating Plant, Fleet Procedure FP-IT-SQA-01, "Software Quality Assurance (SQA) Program", Rev 5, October 10, 2008

I.2.36 Prairie Island Nuclear Generating Plant, Unit 1 and Unit 2, Technical Specification 3.3.1.2

I.2.37 Prairie Island Nuclear Generating Plant, Calculation SPC-NI-017, "Unit 1 Secondary Calorimetry Uncertainty", Rev 1, November 11, 2004

I.2.38 Prairie Island Nuclear Generating Plant, Calculation SPC-NI-018, "Unit 2 Calorimetric Uncertainty", Rev 0, September 27, 1995

I.2.39 Caldon Vendor Manual NX-41441-5, "LEFM +2000 FC FLOW MEASUREMENT SYSTEM INSTALLATION AND COMMISSIONING MANUAL AND FIELD COMMISSIONING DATA PACKAGE", Rev 2, November 26, 2008

II. Accidents and transients for which the existing analyses of record bound plant operation at the proposed uprated power level

1. A matrix that includes information for each analysis in this category and addresses the transients and accidents included in the plant's updated final safety analysis report (UFSAR) (typically Chapter 14 or 15) and other analyses that licensees are required to perform to support licensing of their plants (i.e., radiological consequences, natural circulation cooldown, containment performance, anticipated transient without scram, station blackout, analyses to determine environmental qualification parameters, safe shutdown fire analysis, spent fuel pool cooling, flooding):

A. Identify the transient or accident that is the subject of the analysis

B. Confirm and explicitly state that:

i. the requested uprate in power level continues to be bounded by the existing analyses of record for the plant

ii. the analyses of record either have been previously approved by the NRC or were conducted using methods or processes that were previously approved by the NRC

C. Confirm that bounding event determinations continue to be valid

D. Provide a reference to the NRC's previous approvals discussed in Item B, above

An evaluation of the transients and accidents included and defined in the PINGP USAR was conducted. Each event was evaluated for MUR PU conditions using the currently approved methodology. The evaluations verify that the specific event acceptance criteria remain satisfied at the MUR PU operating conditions (1677 MWt) assuming a power measurement uncertainty consistent with the methodology identified in the USAR for the respective event.

Table II-1 below contains a list of the current USAR Section 14 analyses and other licensing basis events such as Station Blackout and the Plant Fire Protection Program, Appendix R. For transient and accident analyses, USAR Section 14 typically contains specific subsections which provide a description of the event, approved analysis methodology, key input parameters, acceptance criteria, and a brief description of how the acceptance criteria are satisfied.

The analyses of record for each of the events listed in Table II-1 below were determined to be bounding for the MUR PU; therefore, no further analysis was required.

Table II-1

USAR Section/Analysis		Analysis of Record Power Assumptions				
		Nominal Core Power	Uncer-tainty	Total Core Power	Bounds MUR PU	Reference (50.59, AoR, etc.)
USAR	**Analysis**	**B, C**	**B, C**	**B, C**	**B, C**	**D**
14.4.1	Uncontrolled RCCA Withdrawal From a Sub-critical Condition	1683	N/A	1683	Yes	NRC approval in Reference II.3.4 and II.3.22. Evaluation has shown that MUR is bounded by currently approved analyses (Reference II.3.5).
14.4.2	Uncontrolled RCCA Withdrawal at Power	1683	RTDP	1683	Yes	NRC approval in Reference II.3.4. Revised analysis under 10 CFR 50.59 demonstrated acceptable results for OFA-only cores for MUR PU conditions (Reference II.3.5). Subsequent evaluation with 422V+ fuel (and mixed cores) in Reference II.3.22.
14.4.3	RCCA Misalignment	1683	RTDP	1683	Yes	NRC approval in Reference II.3.4. Revised analysis under 10 CFR 50.59 demonstrated acceptable results for OFA-only cores for MUR PU conditions (Reference II.3.5). Subsequent evaluation with 422V+ fuel (and mixed cores) in Reference II.3.22.
14.4.4	Chemical and Volume Control System Malfunction	N/A	N/A	N/A	Yes	NRC approval in Reference II.3.4. Evaluation has shown that current approved analyses are bounding for OFA-only cores at MUR PU conditions (Reference II.3.5). Subsequent evaluation with 422V+ fuel (and mixed cores) in Reference II.3.22.
14.4.5	Start-Up of an Inactive Reactor Coolant Loop	N/A	N/A	N/A	Yes	Precluded by Technical Specifications
14.4.6	Excessive Heat Removal Due to Feedwater System Malfunction	1683	RTDP	1683	Yes	NRC approval in Reference II.3.4. Evaluation has also shown that current approved analyses are bounding for OFA-only cores at MUR PU conditions (Reference II.3.5). Revised analysis under 10 CFR 50.59 demonstrated acceptable results for OFA-only cores at MUR PU conditions (Reference II.3.5). Subsequent evaluation with 422V+ fuel (and mixed cores) in Reference II.3.22.
14.4.7	Excessive Load Increase Incident	1683	RTDP	1683	Yes	NRC approval in Reference II.3.4. Revised analysis under 10 CFR 50.59 demonstrated acceptable results for OFA-only cores at MUR PU conditions (Reference II.3.5). Subsequent

						evaluation with 422V+ fuel (and mixed cores) in Reference II.3.22.
14.4.8	Loss of Reactor Coolant Flow	1683	RTDP (DNB)	1683	Yes	NRC approval in Reference II.3.4. Revised analysis under 10 CFR 50.59 demonstrated acceptable results for OFA-only cores at MUR PU conditions (Reference II.3.5). Subsequent reanalysis with 422V+ fuel (and mixed cores) was reviewed in Reference II.3.22.
14.4.8.2	Locked Pump Rotor	1683	RTDP (DNB)	1683	Yes	NRC approval in Reference II.3.4. Revised analysis under 10 CFR 50.59 demonstrated acceptable results for OFA-only cores at MUR PU conditions (Reference II.3.5). Subsequent reanalysis with 422V+ fuel (and mixed cores) was reviewed in Reference II.3.22.
14.4.9	Loss of External Electrical Load	1650 1683 (DNB)	± 2% RTDP	1683 1683	Yes Yes	NRC approval in Reference II.3.4. Revised analysis under 10 CFR 50.59 demonstrated acceptable results for OFA-only cores at MUR PU conditions (Reference II.3.5). Subsequent evaluation with 422V+ fuel (and mixed cores) in Reference II.3.22.
14.4.10	Loss of Normal Feedwater	1650	± 2%	1683	Yes	NRC approval in Reference II.3.4. Evaluation has shown that current approved analyses are bounding for OFA-only cores at MUR PU conditions (Reference II.3.5). Subsequent evaluation with 422V+ fuel (and mixed cores) in Reference II.3.22.
14.4.11	Loss of All AC Power to the Station Auxiliaries (LOOP)	1650	± 2%	1683	Yes	NRC approval in Reference II.3.4. Evaluation has shown that current approved analyses are bounding for OFA-only cores at MUR PU conditions (Reference II.3.5). Subsequent evaluation with 422V+ fuel (and mixed cores) in Reference II.3.22.
14.5.1	Fuel Handling	NA	NA	NA	Yes	NRC approval in Reference II.3.10 at analytical power level of 1683 MWt. Evaluation of 422V+ fuel in Reference II.3.22.
14.5.2	Accidental Release of Radioactive Liquids	N/A	N/A	N/A	Yes	Not dependent on power level
14.5.3	Accidental Release-Waste Gas	1721.4	N/A	1721.4	Yes	Approval in Reference II.3.23, where the analyzed core power level bounds

						that for MUR PU conditions.
14.5.4	Steam Generator Tube Rupture	1721.4	N/A	1721.4	Yes	Approval in Reference II.3.23, where the analyzed core power level bounds that for MUR PU conditions.
14.5.5.3.1	Rupture of a Steam Pipe (Containment Response)	0	N/A	0	Yes	Approval in Reference II.3.22.
14.5.5.3.2	Rupture of a Steam Pipe (Core Response)	1683 0	RTDP N/A	1683 0	Yes Yes	NRC approval in Reference II.3.4. Revised analysis under 10 CFR 50.59 demonstrated acceptable results for OFA-only cores at MUR PU conditions (Reference II.3.5). Subsequent evaluation with 422V+ fuel (and mixed cores) in Reference II.3.22.
14.5.5.6	Dose Analyses for MSLB Outside of Containment	1650	± 2%	1683	Yes	NRC approval in Reference II.3.1, which considered the analytical power level of 1683 MWt.
14.5.6	Rupture of a Control Rod Drive Mechanism Housing (RCCA Ejection)	0 1683	N/A 0	0 1683	Yes Yes	CN-TA-03-81 Rev 0 - OFA Only Core (Reference II.3.5) CN-TA-04-89 Rev 0 – OFA Only Core w/ Gadolinia (Reference II.3.5) CN-TA-07-114 Rev 0 – Mixed OFA/422V+ and 422V+ Only Cores (Reference II.3.15) Reference II.3.2
14.6	Large Break LOCA Analysis	1683	0	1683	Yes	NRC approval in Reference II.3.27. Revised analysis at MUR conditions with the limiting fuel design approved in Reference II.3.22.
14.7	Small Break LOCA Analysis	1683	0	1683	Yes	Analysis methodology approved by 50.59 evaluation (Reference II.3.26). Revised analysis at MUR conditions reviewed and approved by NRC (Reference II.3.22).
14.8	Anticipated Transient Without Scram (ATWS)	1683	0	1683	Yes	NRC approval in Reference II.3.4. Revised analysis under 10 CFR 50.59 demonstrated acceptable results for OFA-only cores at MUR PU conditions (Reference II.3.5). Subsequent evaluation with 422V+ fuel (and mixed cores) in Reference II.3.22.
14.9	Environmental Consequences of Loss-of-Coolant Accident	1683	0	1683	Yes	NRC approval in Reference II.3.1.
14.10	Long Term Cooling Following a LOCA	1683	0	1683	Yes	NRC approval in Reference II.3.22.
8.4.4	Station Blackout	1650	± 2%	1683	Yes	NRC approval in References II.3.6 and

						II.3.7. Evaluations concluded that MUR PU has insignificant impact on the event (Reference II.3.5).
10.3.1	Safe Shutdown Systems Analysis (Plant Fire Protection Program, Appendix R)	1650	± 2%	1683	Yes	Analysis is described safe shutdown analysis documents. 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 (Reference II.3.5).

2. Event Descriptions

II.2 Table II-1 - Event Descriptions

This Section provides a brief description of those events listed in Table II-1. All the USAR analyses in Table II-1 were evaluated for an increase in RTP due to the MUR PU. The analyses include the NSSS responses with both the Replacement Steam Generator (Unit 1) and the Original Steam Generators (Unit 2).

LOCA analysis methodologies have also been previously approved for use at Prairie Island and recent review has concluded that the LOCA analyses-of-record bound the MUR power level. The results of this review are provided later in Section II.

The evaluation of the USAR events concludes that the existing analyses remain bounding and the conclusions presented in the USAR remain valid. Therefore, all applicable acceptance criteria are met for the MUR PU. The number (i.e., 14.4.1) in the far left column denotes the applicable PINGP USAR Section.

II.2.1 14.4.1 - Uncontrolled RCCA Withdrawal From Subcritical Condition

This event is defined as an uncontrolled addition of reactivity to the reactor core by withdrawal of rod cluster control assemblies resulting in a power excursion. A RCCA withdrawal incident has an extremely low probability of occurrence but could be caused by a malfunction of the reactor control or control rod drive system.

The analysis methodology for Uncontrolled RCCA Withdrawal from a Subcritical Condition was approved in 2004 with license amendments 162/153 (Reference II.3.4). Evaluation has shown that currently approved analyses are bounding for OFA (Optimized Fuel Assembly)-only cores at the nominal core power of 1677 MWt (Reference II.3.5). Subsequent reanalysis with 422V+ fuel (and mixed cores) using approved methods was performed in support of License Amendments 192/181 (Reference II.3.22).

II.2.2 14.4.2 - Uncontrolled RCCA Withdrawal at Power

This event is defined as an uncontrolled addition of reactivity to the reactor core by withdrawal of rod cluster control assemblies resulting in a power excursion. This transient could be caused by a malfunction of the reactor control or control rod drive system.

The analysis methodology for Uncontrolled RCCA Withdrawal at Power was approved in 2004 with license amendments 162/153 (Reference II.3.4). Revised analysis under 10 CFR 50.59 using current approved methods demonstrated acceptable results for OFA-only cores at the analytical core power of 1683 MWt associated with the subject MUR PU (Reference II.3.5). Subsequent evaluation with 422V+ fuel (and mixed cores) was performed in support of License Amendments 192/181 (Reference II.3.22).

II.2.3 14.4.3 - RCCA Misalignment

Two separate RCCA misalignment conditions are considered for this event. The first is statically misaligned RCCAs and the second is dropped RCCAs or an RCCA bank. In the misalignment transient, one or more RCCAs is assumed to be statically misplaced from the normal or allowed position. This situation might occur if a rod were left behind when inserting or withdrawing banks, or if a single rod were to be withdrawn. In the dropped rod or assembly transient, one or more full length RCCAs or an RCCA bank is assumed to be released by the stationary gripper coils and falls to a fully inserted position in the core.

The analysis methodology for RCCA Misalignment was approved in 2004 with license amendments 162/153 (Reference II.3.4). Revised analysis under 10 CFR 50.59 using current approved methods demonstrated acceptable results for OFA-only cores at the nominal core power of 1677 MWt associated with the subject MUR PU (Reference II.3.5). Subsequent evaluation with 422V+ fuel (and mixed cores) was performed in support of License Amendments 192/181 (Reference II.3.22).

II.2.4 14.4.4 - Chemical and Volume Control System Malfunction

A malfunction of the Chemical and Volume Control (VC) System that causes an inadvertent dilution of the Reactor Coolant System could occur at any plant operating mode. For the purposes of this analysis, all operating Modes are reviewed.

The major hazard associated with an unmitigated VC malfunction is a reduction in the DNB ratio and/or a complete loss of shutdown margin. The events, therefore, are analyzed in order to determine the minimum

DNB ratio or the response time that exists prior to a complete loss of shutdown margin.

The analysis methodology for Chemical and Volume Control System Malfunction was approved in 2004 with license amendments 162/153 (Reference II.3.4). Evaluation has shown that current approved analyses are bounding for OFA-only cores at the nominal core power of 1677 MWt associated with the subject MUR PU (Reference II.3.5). Subsequent evaluation with 422V+ fuel (and mixed cores) was performed in support of License Amendments 192/181 (Reference II.3.22).

II.2.5 14.4.5 - Start-Up of an Inactive Reactor Coolant Loop

Since there are no isolation valves or check valves in the RC System, operation of the plant with an inactive loop causes reversed flow through that loop. If there is a thermal load on the steam generator in the inactive loop, the hot leg coolant in that loop will be at a lower temperature than the core inlet temperature. The startup of the pump in the idle loop results in a core flow increase and the injection of colder water into the core. This could cause a rapid reactivity insertion and power increase.

The PINGP Technical Specifications require that both RCPs be operating when the reactor is in Mode 1 or Mode 2. One pump operation is not permitted except for startup and physics tests when the core power is less than the P-7 reactor trip interlock. In the event that one RCP trips in Mode 1 or 2, the Technical Specifications require the plant to be in Mode 3 within six hours. If an RCP trips above P-7, an automatic reactor trip will be initiated. As the Technical Specifications require both RCPs to be operating in Modes 1 or 2 when not performing tests, an analysis of this event is not necessary.

II.2.6 14.4.6 - Excessive Heat Removal Due to Feedwater System Malfunction

This event is identified as a change in steam generator feedwater conditions resulting in an increase in feedwater flow or a decrease in feedwater temperature that could result in excessive heat removal from the plant primary coolant system. Such changes in feedwater flow or feedwater temperature are a result of a failure of a feedwater control valve or feedwater bypass valve, failure in the feedwater control system, or Operator error.

The occurrences of these failures that result in an excessive heat removal from the plant primary coolant system cause the primary-side temperature and pressure to decrease significantly. The existence of a negative moderator and fuel temperature reactivity coefficients, and the actions initiated by the reactor rod control sub-system can cause core reactivity to rise, as the primary-side temperature decreases. In the absence of a reactor trip or other protective action, this increase in core power, coupled with the decrease in primary-side pressure, can challenge the core thermal limits.

The analysis methodology for Feedwater Malfunctions was approved in 2004 with license amendments 162/153 (Reference II.3.4). Revised analysis under 10 CFR 50.59 using current approved methods demonstrated acceptable results for OFA-only cores at the nominal core power of 1677 MWt associated with the subject MUR PU (Reference II.3.5). Subsequent evaluation with 422V+ fuel (and mixed cores) was performed in support of License Amendments 192/181 (Reference II.3.22).

II.2.7 14.4.7 - Excessive Load Increase Incident

An excessive load increase incident is defined as a rapid increase in steam generator steam flow that causes a power mismatch between the reactor core power and the steam generator load demand. It could result from either an administrative violation such as excessive loading by the Operator or an equipment malfunction in the steam dump control or turbine control system.

The analysis methodology for Excessive Load Increase Incidents was approved in 2004 with license amendments 162/153 (Reference II.3.4). Revised analysis under 10 CFR 50.59 using current approved methods demonstrated acceptable results for OFA-only cores at the nominal core power of 1677 MWt associated with the subject MUR PU (Reference II.3.5). Subsequent evaluation with 422V+ fuel (and mixed cores) was performed in support of License Amendments 192/181 (Reference II.3.22).

II.2.8 14.4.8.1 - Loss of Reactor Coolant Flow

A loss of coolant flow incident can result from a mechanical or electrical failure in one or more reactor coolant pumps or from a fault in the power supply to these pumps.

Simultaneous loss of electrical power to all reactor coolant pumps at full power is the most severe credible loss-of-coolant flow condition. For this condition, the reactor trip together with flow sustained by the inertia of the coolant and RCPs is sufficient to prevent fuel failures and RC System over pressurization.

With respect to the overpressure evaluation, the Loss of Flow events are bounded by the Loss of Load/Turbine Trip events, in which assumptions are made to conservatively calculate the RC and MS System pressure transients. For the Loss of Flow events, turbine trip occurs following reactor trip, whereas for the Loss of Load/Turbine Trip event, the turbine trip is the initiating fault. Therefore, the primary to secondary power mismatch and resultant RC and MS System heatup and pressurization transients are always more severe for the Loss of Load/Turbine Trip event. For this reason, no attempt is made to calculate the maximum RC or MS System pressure for the Loss of Flow events.

The analysis methodology for Loss of Reactor Coolant Flow – Flow Coastdown was approved in 2004 with license amendments 162/153 (Reference II.3.4). Revised analysis under 10 CFR 50.59 using current approved methods demonstrated acceptable results for OFA-only cores at the nominal core power of 1677 MWt associated with the subject MUR PU (Reference II.3.5). Subsequent reanalysis with 422V+ fuel (and mixed cores) using approved methods was performed in support of License Amendments 192/181 (Reference II.3.22).

II.2.9 14.4.8.2 - Locked Rotor

The accident postulated is the instantaneous seizure of the rotor of a single reactor coolant pump. This transient is due to the hypothetical instantaneous seizure of a RCP rotor. Flow through the RC System is rapidly reduced, leading to a reactor trip on a low-flow signal.

The Locked Rotor transient is initiated from full power by abruptly seizing one of the RCP shafts. The rotor is assumed to be locked for forward flow and free-spinning for reverse flow. This represents the most limiting condition for the Locked Rotor/Shaft Break accidents. The analysis assumes that the other RCP continues to operate throughout the event.

The analysis methodology for Loss of Reactor Coolant Flow – Locked Pump Rotor was approved in 2004 with license amendments 162/153 (Reference II.3.4). Revised analysis under 10 CFR 50.59 using current approved methods demonstrated acceptable results for OFA-only cores at the nominal core power of 1677 MWt associated with the subject MUR PU (Reference II.3.5). Subsequent reanalysis with 422V+ fuel (and mixed cores) using approved methods was performed in support of License Amendments 192/181 (Reference II.3.22).

II.2.10 14.4.9 - Loss of External Electrical Load

The loss of external electrical load may result from an abnormal increase in network frequency, opening the main breakers from the generator which causes a rapid large NSSS load reduction by action of the turbine control, or from a trip of the turbine-generator.

In this analysis, the behavior of the plant is evaluated for a complete loss of steam load (i.e., turbine trip) from full power without direct reactor trip. This is done to show the adequacy of the pressure relieving devices, and also to demonstrate core protection margins. The reactor is not tripped until conditions in the RC System result in a trip. The turbine is assumed to trip without actuating all the turbine stop valve limit switches. This assumption delays reactor trip until conditions in the RC System result in a trip due to other signals. Thus, the analysis assumes a worst-case transient. In addition, no credit is taken for steam dump. Main feedwater flow is terminated at the time of turbine trip, with no credit taken for auxiliary feedwater (except for long-term recovery) to mitigate the consequences of the transient.

The analysis methodology for Loss of External Electrical Load was approved in 2004 with license amendments 162/153 (Reference II.3.4). Revised analysis under 10 CFR 50.59 using current approved methods demonstrated acceptable results for OFA-only cores at the nominal core power of 1677 MWt associated with the subject MUR PU (Reference II.3.5). Subsequent evaluation with 422V+ fuel (and mixed cores) was performed in support of License Amendments 192/181 (Reference II.3.22).

II.2.11 14.4.10 - Loss of Normal Feedwater

A loss of normal feedwater (from a pipe break that can be isolated from the SGs, pump failures, valve malfunctions, or loss of outside AC power) results in a reduction in the capability of the secondary system to remove the heat generated in the reactor core.

A loss of normal feedwater transient is characterized by a rapid reduction in steam generator water level which results in a reactor trip, a turbine trip, and auxiliary feedwater actuation by the protection system logic. Following the reactor trip the power quickly falls to decay heat levels.

The analysis methodology for Loss of Normal Feedwater was approved in 2004 with license amendments 162/153 (Reference II.3.4). Evaluation has shown that current approved analyses are bounding for OFA-only cores at the nominal core power of 1677 MWt associated with the subject MUR PU (Reference II.3.5). Subsequent evaluation with 422V+ fuel (and mixed cores) was performed in support of License Amendments 192/181 (Reference II.3.22).

II.2.12 14.4.11 - Loss of All AC Power to the Station Auxiliaries (LOOP)

A loss of offsite power can result from a number of external or internal causes. The specific cause is not of concern as part of the analysis of this transient.

The analysis does not assume that power is lost as the initiating event. Rather, the analysis conservatively models a loss of normal feedwater with a subsequent loss of offsite power following the reactor trip on low-low steam generator water level. This bounds the case of an immediate loss of all AC power as the initiating event, which would result in an immediate reactor trip.

The analysis methodology for Loss of All AC Power was approved in 2004 with license amendments 162/153 (Reference II.3.4). Evaluation has shown that current approved analyses are bounding for OFA-only cores at the nominal core power of 1677 MWt associated with the subject MUR PU (Reference II.3.5). Subsequent evaluation with 422V+ fuel (and mixed cores) was performed in support of License Amendments 192/181 (Reference II.3.22).

II.2.13 14.5.1 - Fuel Handling

The following fuel handling accidents are evaluated:

1. A fuel assembly becomes stuck inside the reactor vessel;
2. A fuel assembly or RCCA is dropped onto the floor of the refueling cavity or spent fuel pool;
3. A fuel assembly becomes stuck in the penetration valve;
4. A fuel assembly becomes stuck in the transfer carriage or the carriage becomes stuck.

The possibility of a fuel handling accident is very remote because of the many administrative controls and physical limitations imposed on fuel handling operations. Also, before any refueling operations begin, verification of complete RCCA insertion is obtained by tripping all rods to obtain indication of rod drop. Boron concentration in the coolant is raised to the refueling concentration level and verified by sampling. Refueling boron concentration is sufficient to maintain the clean, cold, fully loaded core subcritical by at least 5% with all RCCAs withdrawn. The refueling cavity is filled with water meeting the same boric acid specifications.

Although, safety features make the probability of a fuel handling accident very low, it is possible that a fuel assembly could be dropped during the handling operations. Therefore, this accident is analyzed both from the standpoint of radiation exposure and accidental criticality.

Under the accident methodology for the FHA, a fuel assembly is assumed to be dropped and damaged during fuel handling. The dose analysis is performed to determine the radiological consequences of the accident.

For the FHA, PINGP has implemented the alternate source term (AST) in accordance with 10 CFR 50.67.

The Alternative Source Term (AST) analysis methodology for the Fuel Handling Accident was approved in 2004 with license amendments 166/156 (Reference II.3.10), which considered the analytical power level of 1683 MWt. Subsequent evaluation of 422V+ fuel was performed in support of License Amendments 192/181 (Reference II.3.28).

II.2.14 14.5.2 - Accidental Release of Radioactive Liquids

Vessels in the waste disposal system which are used for waste storage are housed in a Class I portion of the Auxiliary Building or the Class I* portion of the Radwaste Building. All such vessels are located inside Class I structural enclosures such as sumps, dikes, or walls or specially constructed areas which will retain spilled liquids. This ensures that the structures are capable of containing the liquid wastes during seismic events.

Thus, there are no credible accidents which would result in the release of radioactive wastes to the river in excess of the limits given in the Offsite Dose Calculation Manual (ODCM).

II.2.15 14.5.3 - Accidental Release - Waste Gas

The waste gas accident is defined as an unexpected and uncontrolled release to the atmosphere of the radioactive xenon and krypton fission gases that are stored in the waste gas storage system or the release of radioactive iodine that is present in the Volume Control Tank (VCT). Failure of a gas decay tank or the VCT or associated piping could result in a release of this gaseous activity. This analysis shows that even with the worst expected conditions, the offsite doses following release of this gaseous activity would be very low.

The analysis methodology for Accidental Release of Waste Gas is described in the PINGP USAR, which is derived from the description in original plant Final Safety Analysis Report (FSAR) Section 14.2.3 (Reference II.3.24). In this respect, the accident methodology is a legacy of original analysis and may therefore be considered approved by the original NRC SER (Reference II.3.23). Where core power level has impact on the results of this accident (e.g., the waste gas tank activity), the USAR states a core power (1721.4 MWt) that is conservatively larger than the analytical core power of 1683 MWt associated with the subject MUR PU (References VI.2.1.L and II.3.5). Thus, the current analysis-of-record represented in the USAR bounds the MUR PU for this accident.

In the course of reviewing the legacy Analysis-of-Record (AoR) for this event to determine whether it bounds MUR PU conditions, the accuracy of the AoR itself was challenged by a confirmatory evaluation. As evaluated in the NSPM Corrective Action Program (CAP), the methods and results of this evaluation did not converge with the AoR as described in the USAR; concluding that there was no basis to invalidate the AoR described in the USAR. Although the CAP may ultimately lead to corrections or revision to the AoR, the respective condition evaluation determined that continued plant operation was justified. Based on the evaluation in NSPM's CAP program and the margin inherent in the AoR's value of analytical core power (1721.4 MWt) compared to the analytical core power of the MUR PU (1683 MWt), it is evident that the conditions of the MUR PU are bounded by the AoR.

II.2.16 14.5.4 - Steam Generator Tube Rupture

The accident examined is the complete severance of a single steam generator tube with the reactor at power which leads to an increase in contamination of the secondary system due to leakage of radioactive coolant from the RC System. In the event of a coincident loss of offsite power, or failure of the condenser steam dump system, discharge of activity to the atmosphere takes place via the steam generator safety and/or power operated relief valves.

The steam generator tube material is Inconel 600 and 690 and as the material is highly ductile, it is considered that the assumption of a complete severance is conservative. Activity in the Steam and Power Conversion System is subject to continuous surveillance and an accumulation of minor leaks which cause the activity to exceed the limits established in the Technical Specifications is not permitted during operation.

The analysis methodology for Steam Generator Tube Rupture is described in the PINGP USAR, which is derived from the description in original plant Final Safety Analysis Report (FSAR) Section 14.2.4 (Reference II.3.24). In this respect, the accident methodology is a legacy of original analysis and may therefore be considered approved by the original NRC SER (Reference II.3.23). Where core power level has impact on the results of this accident, the USAR states a core power (1721.4 MWt) that is conservatively larger than the analytical core power of 1683 MWt associated with the subject MUR PU (References VI.2.1.L and M and II.3.5). Thus, the current analysis-of-record represented in the USAR bounds this accident.

In the course of reviewing the legacy Analysis-of-Record (AoR) for this event to determine whether it bounds MUR PU conditions, the accuracy of the AoR itself was challenged by a confirmatory evaluation. As evaluated in the NSPM Corrective Action Program (CAP), the methods and results of this evaluation did not converge with the AoR as described in the USAR; concluding that there was no basis to invalidate the AoR described in the USAR. Although the CAP may ultimately lead to corrections or revision to the AoR, the respective condition evaluation determined that continued plant operation was justified. Based on the evaluation in NSPM's CAP program and the margin inherent in the AoR's value of analytical core power (1721.4 MWt) compared to the analytical core power of the MUR PU (1683 MWt), it is evident that the conditions of the MUR PU are bounded by the AoR.

II.2.17 14.5.5 - Rupture of a Steam Pipe (Addresses both USAR Sections 14.5.5.3.1 and 14.5.5.3.2 in Table II-1 of this enclosure)

A Rupture of a Steam Pipe could be caused by a failure of the pipe itself, or the inadvertent opening and sticking of a valve, e.g., safety or PORV. The analyses evaluates, 1) the containment's response to main steam line break (MSLB) inside of containment, 2) the core's response to a MSLB inside containment, 3) a small steam line break, and 4) the dose analysis for a MSLB outside of containment. For an outside of containment MSLB, the effects to the core would be similar and are bounded by this analysis.

Several different break sizes are analyzed for the Steamline Rupture - Core Response transient. Depending upon the break size, the event is considered to be either a Condition III or IV event. However, some Condition II events are indistinguishable from a minor steamline break with respect to the primary system response and must satisfy Condition II criteria. Examples of such events include an excessive load increase and steam system valve malfunction events. Therefore, a subset of the Condition II criteria are applied for all break sizes analyzed for ease of interpretation.

The analysis methodology for Rupture of a Steam Pipe (Containment Response) was approved in 2009 with License Amendment 192/181 (Reference II.3.22), with consideration of the analytical core power of 1683 MWt plus reactor coolant pump heat for a total Nuclear Steam Supply System power of 1690 MWt. Thus, the analysis of record bounds the value of analytical core power associated with the subject MUR PU (Reference II.3.5).

The analysis methodology for Rupture of a Steam Pipe (Core Response) was approved in 2004 with license amendments 162/153 (Reference II.3.4). Revised analysis under 10 CFR 50.59 using current approved methods demonstrated acceptable results for OFA-only cores at the nominal core power of 1677 MWt associated with the subject MUR PU (Reference II.3.5). Subsequent evaluation with 422V+ fuel (and mixed cores) was performed in support of License Amendments 192/181 (Reference II.3.22).

II.2.18 14.5.5.6 - Dose Analyses for MSLB Outside of Containment

The analysis methodology and revised results for the Radiological Dose Consequences of Main Steam Line Break (MSLB) Outside Containment were approved by NRC in 2009 with license amendments 191/180 (Reference II.3.1), which considered the analytical power level of 1683 MWt.

A proprietary computer program (PERC2) is used to calculate the Control Room and Offsite dose due to airborne radioactivity releases following a MSLB. The MSLB dose assessment supports the implementation of Alternate Repair Criteria (ARC) as defined in USNRC Generic Letter 95-05 and approved in PINGP License Amendments 133/125. The MSLB dose assessment uses the maximum allowable accident-induced leakage that results in dose consequences that are just within the most limiting of the regulatory limits associated with the Exclusion Area Boundary (EAB), Low Population Zone (LPZ), and the Control Room.

II.2.19 14.5.6 - Rupture of a Control Rod Drive Mechanism Housing (RCCA Ejection)

This accident is a result of an extremely unlikely mechanical failure of a control rod mechanism pressure housing such that the RC System pressure would then eject the RCCA and drive shaft. The consequences of this mechanical failure, in addition to being a minor loss of coolant accident, may also be a rapid reactivity insertion together with an adverse core power distribution, possibly leading to localized fuel rod damage for severe cases. The resultant core thermal power excursion is limited by the Doppler reactivity effect of the increased fuel temperature and terminated by reactor trip actuated by high neutron flux signals.

The ultimate acceptance criteria for this event is that any consequential damage to either the core or the RC System must not prevent long-term core cooling, and that any offsite dose consequences must be within the guidelines of 10 CFR 100. To demonstrate compliance with these requirements, it is sufficient to show that the RC System pressure boundary remains intact, and that no fuel dispersal in the coolant, gross lattice distortions, or severe shock waves will occur in the core.

The analysis methodology for Rupture of a Control Rod Drive Mechanism Housing (RCCA Ejection) was approved in 2004 with license amendments 162/153 (Reference II.3.4). Evaluation has shown that currently approved analyses are bounding for OFA-only cores at the nominal core power of 1677 MWt or the analytical core power of 1683 MWt associated with the subject MUR PU (Reference II.3.5). Subsequent reanalysis with 422V+ fuel (and mixed cores) using approved methods was performed in support of License Amendments 192/181 (Reference II.3.22).

II.2.20 14.6 - Large Break LOCA Analysis

A LOCA may result from a rupture of the RC System or of any line connected to that system up to the first closed valve. Ruptures of a very small cross section will cause expulsion of the coolant at a rate which would maintain an operational water level in the pressurizer permitting the Operator to execute an orderly shutdown. A small quantity of the coolant containing fission products normally present in the coolant would be released to the containment.

Should a major break occur, depressurization of the RC System results in a pressure decrease in the pressurizer. Reactor trip signal occurs and an SI signal occurs when the respective pressurizer low pressure trip setpoint is reached (including allowances for uncertainties, etc.). The large break LOCA analysis does not model control rod insertion and thus does not specifically model a reactor trip setpoint. The injection of the borated water limits the consequences of the accident in two ways:

1. Borated water injection complements void formation in causing rapid reduction of power to a residual level corresponding to fission product decay heat.

2. Injection of borated water provides heat transfer from the core and prevents excessive clad temperatures.

A major pipe break (large break), is defined as a rupture of the reactor coolant pressure boundary with a total cross-sectional area greater than 1.0 ft^2. This is considered a Condition IV event, a limiting fault.

The analysis methodology for Large Break Loss of Coolant Accident (LBLOCA) analysis methodology was approved in 2007 with license amendments 179/169 (Reference II.3.27). Revised analysis using approved methodology at analytical power level of 1683 MWt and considering the limiting fuel design was approved in 2009 by license amendments 192/181 (Reference II.3.22).

II.2.21 14.7 - Small Break LOCA Analysis

A minor pipe break (small break) is defined as a rupture of the RC System pressure boundary with a total cross-sectional area less than 1.0 ft^2 in which the normally operating charging system flow is not sufficient to sustain pressurizer level and pressure. This is considered a Condition III event, an infrequent fault.

In 2006, NSPM approved a 50.59 evaluation (Reference II.3.26) to change analysis methodology (NOTRUMP-EM) for Small Break LOCA (SBLOCA). In 2009, NRC reviewed and approved a revised analysis at analytical core power of 1683 MWt using the approved methodology (Reference II.3.22).

II.2.22 14.8 - Anticipated Transient Without Scram (ATWS)

As defined in 10 CFR 50.62, an ATWS is an expected operational transient (such as loss of feedwater, loss of load or loss of offsite power) which is accompanied by a failure of the Reactor Protection System to shut down the reactor. All USAR Condition II transient events were evaluated with consideration towards explicitly analyzing each under ATWS conditions. For many of the Condition II events, explicit analyses were performed. Events were not explicitly analyzed for ATWS conditions if the transient either (1), does not require a reactor trip to mitigate the consequences of the event in the analysis, or (2), results in consequences bounded by either another analyzed transient or an ATWS event transient selected for analysis.

The analysis methodology for Anticipated Transient Without Scram (ATWS) was approved in 2004 with license amendments 162/153 (Reference II.3.4). Revised analysis under 10 CFR 50.59 using current approved methods demonstrated acceptable results for OFA-only cores at the nominal core power of 1677 MWt associated with the subject MUR PU (Reference II.3.5). Subsequent evaluation with 422V+ fuel (and mixed cores) was performed in support of License Amendments 192/181 (Reference II.3.22).

II.2.23 14.9 - Environmental Consequences of Loss-of-Coolant Accident

As stated in the USAR, the NRC has established guidelines in 10 CFR Part 100 for radiation doses resulting from accidental releases of radioactivity from a reactor plant. PINGP stays within the dose criteria set forth in 10 CFR Part 100 following the design basis accident and releases consistent with NRC Regulatory Guide 1.4 assumptions.

The PINGP containment system has one feature of particular importance to the environmental consequences of a loss-of-coolant accident and that is the presence of two barriers in series to fission product leakage: the Reactor Containment Vessel and the Shield Building.

Long term uncontrolled leakage of radioactivity to the external atmosphere prior to filtration or decay is prevented by fans which establish a slight negative pressure with respect to the atmosphere in the annulus within approximately three minutes after the accident. The amount of long term filtered exhaust released to the environment is sufficient to maintain the negative annulus pressure and compensate for inleakage. In general, all exhaust from the Shield Building will have experienced several passes of filtration as a result of the recirculation feature. A measurable negative pressure with respect to atmosphere will be drawn in the Auxiliary Building Special Ventilation Zone within six minutes after initiation.

The analysis of the Shield Building Ventilation sub-system filter envelops the Auxiliary Building Special Ventilation sub-system filters as well. The analysis assumes the maximum allowable containment leakage is all loaded on the two Shield Building Ventilation filters. In an accident, some iodine may bypass the Shield Building and be adsorbed on the Auxiliary Building Special Ventilation filters. Since the maximum allowable leakage to the Auxiliary Building Special Ventilation Zone is 0.10%/day at 46 psig per the Containment Leakage rate testing program, the loading on the two Auxiliary Building Special Ventilation filters would be bounded by the loading on the Shield Building Ventilation filters. Therefore, the Auxiliary Building Special Ventilation filter temperature could not exceed 165°F.

The analysis methodology and revised results for the Radiological Dose Consequences of LOCA were approved by NRC in 2009 with license amendments 191/180 (Reference II.3.1), which considered the analytical power level of 1683 MWt

II.2.24 14.10 - Long Term Cooling Following a LOCA The analysis methodology and revised results for Long Term Cooling Following a LOCA were approved by NRC in 2009 with license amendments 192/181 (Reference II.3.22), which considered the analytical power level of 1683 MWt. That analysis used the WCOBRA/TRAC thermal-hydraulic computer code and evaluated core cooling for nearly 3,000 seconds. A decay heat model was used that is in accordance with 10 CFR 50, Appendix K, and additional assumptions are used to correct for empirically-observed over-prediction of two-phase level swell. The analysis demonstrated that, although a second heatup occurs during an assumed safety injection interruption, this heatup is limited to approximately 800°F, and stable core coverage can be maintained for the long-term period following the postulated LOCA.

II.2.25 8.4.4 - Station Blackout

A Station Blackout (SBO) exists when there is a Loss of Offsite Power (LOOP) and concurrent loss of both of a Unit's Emergency Diesel Generator sources. PINGP meets the SBO rule of 10 CFR 50.63 (June

21, 1988) and the related guidance of Reg. Guide 1.155 (August, 1988). An SBO is assumed to occur on only one Unit of a two Unit site, in accordance with Reg. Guide 1.155. After either Emergency Diesel Generator in the non-SBO Unit has completed load sequencing and has provided power to the designated safeguards equipment, the Operator will manually close two series bus tie breakers to the SBO Unit's associated safeguards bus. The involved bus tie breaker pairs are 15-8 and 25-17 (interconnecting buses 15 and 25), and 16-10 and 26-1 (interconnecting buses 16 and 26). These breakers are normally open during plant operation. Tests and analysis have shown that the non-SBO Unit's Emergency Diesel Generator is available and the interconnecting bus ties can be closed within ten minutes of the realization that an SBO condition exists. Under assumptions used by Reg. Guide 1.155 and NUMARC-8700 for a plant of PINGP's configuration, AC electrical power will be restored to at least one safeguards bus on the SBO Unit from offsite or from one of its own Emergency Diesel Generators within four hours of the onset of an SBO.

The analysis of Station Blackout (SBO) was approved in 1997 with License Amendments 103/96 (References II.3.6 and II.3.7). The power level assumed in the SBO analyses was 1683 MWt. The evaluation performed in conjunction with the above referenced License Amendment Request concluded that current SBO conditions remain bounding at MUR PU conditions at the analytical core power of 1683 MWt (Reference II.3.5).

II.2.26 10.3.1 - Plant Fire Protection Program (Appendix R)

The USAR states that to comply with 10 CFR 50, Appendix R, Section III.G, a Safe Shutdown analysis was performed which was subsequently included in Operations Manual F5, Appendix E. This procedure describes the safe shutdown methodology, including the identification of safe shutdown systems, safe shutdown components, associated circuit concerns (common power supply, common enclosure, and spurious operation concerns), and a summary of approved exemptions from Appendix R requirements. This procedure also summarizes the analysis of safe shutdown equipment and circuits, and is maintained current with plant design.

The USAR also states that for a fire in the Control Room and Relay Room, which requires Control Room evacuation, alternative shutdown capability is implemented from the hot shutdown panel location. The equipment protected from a fire in the Control Room and Relay Room was analyzed in the Safe Shutdown analysis. The alternate shutdown methodology is described in Operations Manual F5, Appendix E.

The USAR further states that Operations Manual F5, Appendix E also includes a summary of compliance with 10 CFR 50, Appendix R, Sections III.J (Emergency Lighting) and III.O (Reactor Coolant Pump Lube Oil Collection Systems).

The analysis of Appendix R safe shutdown is described in PINGP safe shutdown analysis documents which constitute the Current Licensing Basis. 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. (Reference II.3.5)

3. References for Section II

II.3.1 Nuclear Regulatory Commission, Safety Evaluation, Revision to Loss-Of-Coolant Accident (LOCA) and Main Steam Line Break (MSLB) Accident Radiological Dose Consequences Analyses and Affected Technical Specifications, Amendments No. 191 and 180 (TAC NOS. MD9140 AND MD9141), June 19, 2009 (ADAMS Accession Number ML091490611)

II.3.2 Nuclear Regulatory Commission, Safety Evaluation, Correction to Safety Evaluation Supporting Amendment Nos. 192 and 181 RE: Technical Specification Changes to Allow Use of Westinghouse 0.422-Inch OD 14x14 Vantage+ Fuel, (TAC Nos. MD9142 AND MD9143), August 31, 2009 (ADAMS Accession Number ML092240332)

II.3.3 Prairie Island Nuclear Generating Plant "Reload Safety And Licensing Checklist" (Form EPF-301-4), Revision 4, July 1, 2005.

II.3.4 Nuclear Regulatory Commission, Safety Evaluation, Issuance of Amendments RE: License Amendment Request dated March 25, 2003, For Safety Analyses Transition, Amendments No. 162 and No. 153 (TAC Nos. M8128 and M8129), April 28, 2004 (ADAMS Accession Number ML040900209)

II.3.5 Prairie Island Nuclear Generating Plant, Engineering Calculation EC 12713, Revised Analytical Core Power and Power Measurement Uncertainty For Transient, Safety, and Accident Analysis to 1683 MWt Including Required Uncertainties.

II.3.6 Nuclear Regulatory Commission, Safety Evaluation, Issuance of Amendments RE: Auxiliary Electrical System Changes, Amendments No. 110 and No. 103 (TAC Nos. M88856 and M88857), May 17, 1994

II.3.7 Nuclear Regulatory Commission, Safety Evaluation, Amendments Nos. 103 and 96 to Facility Operating License Nos. DPR-42 and DPR-60 (TAC Nos. M83070 and M883071), December 17, 1992

II.3.8 Westinghouse, Calculation CN-TA-07-104, Rod Withdrawal From Subcritical Analysis for the 422V+ (Heavy Bundle) Fuel Transition Program, Rev 0, November 1, 2007

II.3.9 Westinghouse, Calculation CN-TA-07-110, Loss of Flow (LOF) Analysis for 422V+ (Heavy Bundle) Fuel Transition Program, Rev 1, January 30, 2008

II.3.10 Nuclear Regulatory Commission, Safety Evaluation, Issuance of Amendments RE: Selective Implementation of Alternate Source Term For Fuel Handling Accidents, Amendments Nos. 166 and 156 (TAC Nos. MC1843 and MC1844), September 10, 2004 (ADAMS Accession Number ML042430504)

II.3.11 Prairie Island Nuclear Generating Plant, Calculation GEN-PI-026, "Safe Shutdown Analysis for Compliance with 10 CFR 50, Appendix R, Section III.G", Rev 5, February 14, 2006

II.3.12 Prairie Island Nuclear Generating Plant, Calculation GEN-PI-051, Fuel Handling Accident Dose Analysis – Heavy Load Drop, Rev 1A, October 28, 2004

II.3.13 Fauske & Associates, Inc., Calculation FAI/07-63, Fuel Handling Accident (FHA) Dose Results Applying AST Methodologies (RG 1.183) In Support of Heavy Bundle Fuel (V422V+) Project, Rev 0, May 25, 2008

II.3.14 Westinghouse, Calculation CN-TA-07-101, Locked Rotor/Shaft Break Analysis for 422V+ (Heavy Bundle) Fuel Transition Program, Rev 0, October 2, 2007

II.3.15 Westinghouse, Calculation CN-TA-07-114, Rod Ejection Analysis for 422V+ (Heavy Bundle) Fuel Transition Program, Rev 0, October 22, 2007

II.3.16 Westinghouse, Calculation CN-LIS-07-175, PRAIRIE ISLAND 422V+ FUEL TRANSITION BELOCA ASTRUM ANALYSIS: Units 1 and 2 (NSP/NRP) Uncertainty Analysis, Rev 0

II.3.17 Westinghouse, Calculation CN-LIS-00-158, SBLOCA Analysis for Prairie Island (NSP)-Auxiliary Feedwater Flow Change Evaluation, Rev 2, February 13, 2002

II.3.18 Westinghouse, Calculation CN-LIS-07-131, Prairie Island Units 1 and 2 (NSP/NRP) 422V+ (Heavy Bundle) Fuel Upgrade Program Small Break LOCA Analysis, Rev 0, January 21, 2008

II.3.19 Westinghouse, Calculation CN-LIS-09-06, Revalidation of WCAP-11925 Minimum Recirculation Flow Requirements for the Prairie Island Units 1 and 2 Fuel Transition Project, Rev 0, February 5, 2009

II.3.20 Westinghouse, Calculation CN-LIS-03-174, Prairie Island (NSP/NRP) Boron Buildup Calculation to Support the SATP/RSG Program and Resolution of IR-03-302-M005, November 24, 2003

II.3.21 Westinghouse, Calculation CN-LIS-07-126, Prairie Island Units 1 & 2 (NSP/NRP) Post-LOCA Long Term Cooling Analysis in Support of the 422V+ Fuel Transition Program, Rev 0, January 18, 2008

II.3.22 NRC letter to NSPM, Issuance of Amendments RE: Technical Specifications Changes to Allow Use of Westinghouse 0.422-inch OD 14X14 VANTAGE+ Fuel (TAC Nos. MD9142 and MD 9143), Amendments 192/181, dated July 1, 2009 (ADAMS Accession Number ML091460809)

II.3.23 Safety Evaluation by the Directorate of Licensing US Atomic Energy Commission in the Matter of Northern States Power Company Prairie Island Nuclear Generating Plant Units 1 and 2, Goodhue County, Minnesota, Docket Nos. 50-282 and 50-306, dated September 28, 1972 (Original Plant SER)

II.3.24 Prairie Island Final Safety Analysis Report (FSAR)

II.3.25 Prairie Island Nuclear Generating Plant Calculation PSAT 3142CT.QA.04, Calculation and Comparison of Dose Potential for PINGP OFA Fuel vs. Heavy Bundle Fuel, Revision 0 (Polestar Calculation)

II.3.26 Prairie Island Nuclear Generating Plant 50.59 Evaluation 1050, Revised Small Break LOCA Analysis Using the NOTRUMP Code: SI into the Broken Loop and COSI Condensation Model (WCAP-10054-P-A Add. 2 Rev. 1)

II.3.27 NRC letter to NMC, Correction Letter for the License Amendments to Incorporate Large-Break Loss-Of-Coolant Accident Analysis Using ASTRUM for Prairie Island Nuclear Generating Plant, Units 1 and 2, (TAC Nos. MD2567 and MD2568), Amendments 179/169, dated January 15, 2008 (ADAMS Accession Number ML080040029)

II.3.28 Prairie Island Nuclear Generating Plant Engineering Change EC-13179, Refuel Unit 1 and Unit 2 with a New "Heavy Bundle" Fuel Design During 1R26 and 2R26, Rev. 0

III. Accidents and transients for which the existing analyses of record do not bound plant operation at the proposed uprated power level

1. **This section covers the transient and accident analyses that are included in the plant's USFAR (typically Chapter 14 or 15) and other analyses that are required to be performed by licensees to support licensing of their plants (i.e., radiological consequences, natural circulation cooldown, containment performance, anticipated transient without scram, station blackout, analyses for determination of environmental qualification parameters, safe shutdown fire analysis, spent fuel pool cooling, flooding).**

2. **For analyses that are covered by the NRC approved reload methodology for the plant, the licensee should:**

   A. **Identify the transient/accident that is the subject of the analysis**

   B. **Provide an explicit commitment to re-analyze the transient/accident, consistent with the reload methodology, prior to implementation of the power uprate**

   C. **Provide an explicit commitment to submit the analysis for NRC review, prior to operation at the uprated power level, if NRC review is deemed necessary by the criteria in 10 CFR 50.59**

   D. **Provide a reference to the NRC's approval of the plant's reload methodology**

3. **For analyses that are not covered by the reload methodology for the plant, the licensee should provide a detailed discussion for each analysis. The discussion should:**

   A. **Identify the transient or accident that is the subject of the analysis**

   B. **Identify the important analysis inputs and assumptions (including their values), and explicitly identify those that changed as a result of the power uprate**

   C. **Confirm that the limiting event determination is still valid for the transient or accident being analyzed**

   D. **Identify the methodologies used to perform the analyses, and describe any changes in those methodologies**

   E. **Provide references to staff approvals of the methodologies in Item D. above**

F. Confirm that the analyses were performed in accordance with all limitations and restrictions included in the NRC's approval of the methodology

G. Describe the sequence of events and explicitly identify those that would change as a result of the power uprate

H. Describe and justify the chosen single-failure assumption

I. Provide plots of important parameters and explicitly identify those that would change as a result of the power uprate

J. Discuss any change in equipment capacities (e.g., water supply volumes, valve relief capacities, pump pumping flow rates, developed head, required and available net positive suction head (NPSH), valve isolation capabilities) required to support the analysis

K. Discuss the results and acceptance criteria for the analysis, including any changes from the previous analysis

There are no analyses of record that do not bound plant operation at the proposed power level.

4. References for Section III

None

IV. Mechanical/Structural/Material Component Integrity and Design

1. A discussion of the effect of the power uprate on the structural integrity of major plant components. For components that are bounded by existing analyses of record, the discussion should cover the type of confirmatory information identified in Section II, above. For components that are not bounded by existing analyses of record, a detailed discussion should be provided.

A. This discussion should address the following components:

i. Reactor vessel, nozzles, and supports

The reactor vessel inlet temperature (527.4°F) and outlet temperature (592.6°F) under the PINGP Units 1 and 2 MUR PU are bounded by the values used in the current reactor vessel stress analysis reports (Reference IV.2.1). The impact of the NSSS design transients on the reactor vessel has been assessed as part of the reactor vessel structural evaluation. The loadings from the revised MUR PU transients produce stresses that are still within the ASME Code allowable values for this equipment. There are no changes due to seismic loading, since seismic loads remain unchanged under the MUR PU.

The piping interface loads, including loss-of-coolant-accident loads from the loop piping analysis, have been considered as part of the inlet nozzle and outlet nozzle structural evaluation. Qualification of these nozzles using these loads was demonstrated by meeting the ASME Code stress limits in the nozzles.

The reactor vessel components have been evaluated to account for the change in design transients and the interface loads at the reactor internals/vessel interface, reactor vessel supports, and reactor vessel inlet and outlet nozzles. The stresses, including fatigue usage factors in the reactor vessel components, meet the ASME Code allowable under the requirements of Section III, 1968 edition up to and including the Winter 1968 addenda. Stress levels in supports continue to meet allowable criteria specified in the AISC Sixth Edition and the PINGP USAR.

ii. **Reactor core support structures and vessel internals**

Structural evaluations of the reactor core support structures and vessel internals demonstrated that the structural integrity of these components was not adversely affected either directly by the MUR PU RC System conditions and transients, or by secondary effects on reactor thermal-hydraulic or structural performance.

iii. **Control rod drive mechanisms**

The revised design conditions resulting from the MUR PU were reviewed against the CLB for these components. Because the MUR PU conditions are bounded by the CLB, no further evaluation was required.

iv. **Nuclear Steam Supply System (NSSS) piping, pipe supports, branch nozzles**

The revised design conditions resulting from the MUR PU were reviewed against the CLB for all components under consideration and found to be bounded, with the clarification provided below for the Unit 2 Lower Lateral Supports.

The design basis of Unit 2 had previously assumed that the Unit 1 analysis and load generation calculations enveloped the Unit 2 RC System pressure boundary components and the supports for the reactor pressure vessel, reactor coolant pumps (RCPs), and steam generators. However, subsequent review in the past had determined that the Unit 2 Steam Generator Lower Lateral Supports were actually different than the Unit 1 Lower Lateral Supports; leading to further review of the condition. Contemporary review of this condition in the NSPM Corrective Action Program (CAP) has concluded that the difference in support designs does not adversely affect the operability of the Unit 2 supports considering current plant design conditions. The MUR-PU would have negligible effect on this condition; therefore, operation at MUR-PU conditions is acceptable.

v. Balance-of-plant (BOP) piping (NSSS interface systems, safety related cooling water systems, and containment systems)

The revised design conditions resulting from the MUR PU were reviewed against the CLB and the revised design conditions were determined to be bounded by the CLB, with the following clarification related to Main Steam Safety Valve thrust force analysis.

In the course of reviewing the Analysis-of-Record (AoR) for the MSSV thrust forces to determine whether it bounds MUR PU conditions, the AoR itself was found to be discrepant in some regard, and the condition was entered into the NSPM Corrective Action Program (CAP). The respective condition evaluation determined that continued plant operation was justified. Because the AoR in this case involves some discrepancy (but otherwise supports continued reactor operation), NSPM specifically evaluated the MUR PU effects on that AoR and states the bounding nature of the existing AoR in specific terms of those effects. In the instance of the revised MSSV thrust forces on the main steam pipe stress, NSPM evaluation found that the effect (loading on the piping) is bounded by the value currently assumed in the AoR and therefore, the effects of the MUR do not challenge the conclusions of the AoR for this item. In summary, the AoR embodies reactor plant operation and related system conditions up to the analytical core power of 1683 MWt, so it is concluded that the conditions of the MUR PU are bounded by the thrust load Analysis-of-Record.

vi. Steam generator tubes, secondary side internal support structures, shell, and nozzles

No changes in the RC System design or operating pressure were made as part of the power uprate and the effects of operating temperature changes (T_{hot}/T_{cold}) remain within allowable limits. Since the design transients will not change as a result of the MUR PU and no additional transients are proposed, the existing loads, stresses and fatigue values remain valid. However, with the increased flow conditions within the steam generator steam drum expected from the MUR PU conditions, material loss in the carbon steel steam drum components may be initiated or accelerated. Periodic steam generator inspections will continue to detect degradation that may occur.

vii. Reactor coolant pumps

The revised design conditions resulting from the MUR PU were reviewed against the CLB for these components. Because the MUR PU conditions are bounded by the CLB, no further evaluation was required.

viii. Pressurizer shell, nozzles, and surge line

The Unit 2 Pressurizer Surge Line and surge line nozzles experience higher thermal stresses than are accounted for in the current fatigue analysis, resulting in higher fatigue usage factors. However, there is some margin available. An analysis has demonstrated acceptable ASME Section III Equation 12 stress and fatigue usage in the surge line and reactor coolant loop nozzle for PINGP Unit 2. Since the stress and fatigue usage factors are acceptable for continued plant operation, it was therefore concluded that no structural integrity issue exists that affects current operation. The revised design conditions resulting from the MUR PU were reviewed against the current conditions and determined to be acceptable.

ix. Safety-related valves

The revised design and operating conditions resulting from the MUR PU were reviewed against the CLB for these components. Valve capacities, setpoints and operating conditions were confirmed to be valid for MUR PU conditions.

B. The discussion should identify and evaluate any changes related to the power uprate in the following areas:

i. stresses

<u>NSSS Piping and Components</u>

In addition to the components addressed above, all branch pipe lines connected to the Reactor Coolant Loop (RCL) were evaluated for the effects of the MUR PU to confirm that the branch piping and nozzles continue to meet allowable stress criteria. The MUR PU affects pipe rupture and thermal loads acting on the branch lines.

The pipe rupture analysis for the RPV and the RCL piping resulted in revised faulted anchor movements at the branch connections to the RCL. The revised MUR PU displacement for a nozzle connection was compared to the faulted anchor movement input into the current stress analysis for the branch line. Where a revised displacement was larger than the anchor movement considered in the current stress analysis, revised stress levels were calculated for the branch pipe and nozzle. The revised stress levels were then compared to the allowable stress levels for the faulted condition.

The current stress analyses for the branch lines were also reviewed to confirm that the current analysis envelops the expected operating conditions for MUR PU. The review involved a comparison of the temperatures and pressures input into the current stress analysis to the temperatures and pressures expected with MUR PU.

Stress levels in the branch lines and nozzles connected to the RCL remain less than the allowable values defined by the B31.1 code (Reference IV.2.30) and USAR Section 12.2 (Reference IV.2.31) for the revised operating parameters and revised pipe rupture loads for the MUR PU.

BOP Piping and Components

The MUR PU results in minimal changes in the piping operating pressures and temperatures and these are bounded by design pressures and temperatures. The design pressure, thermal, dead weight, and dynamic (seismic and MSIV fast valve closure) loading are either not affected or are bounded by the CLB. Additionally, all calculated thermal/hydraulic parameters are expected to remain within the analyzed ranges for operation at the MUR PU conditions with tube plugging levels of up to 15 percent for Unit 2 and up to 10 percent for Unit 1, with the exception of moisture carryover (MCO) for the PINGP Unit 2 steam generators.

Unit 1 Steam Generators were replaced in 2004. Unit 2 Steam Generators are still original operating equipment, scheduled for replacement in 2013. They are currently operating at an MCO level greater than the design specification of 0.25%. Based on best estimate projection of MCO, using previously measured MCO as a basis, the projected MCO at MUR PU conditions for Unit 2 would be 1.46% with 15% of steam generator tubes plugged. Unit 2 currently has approximately 8.2% tubes plugged.

Corrective action was initiated for further consideration of current operation under MCO conditions and the effects of operation at higher MCO conditions. The highest risk component for MCO-related erosion was considered to be the high pressure turbine. However, no such effects were seen on Unit 1, where MCO was higher than at current Unit 2 operation, nor were such effects seen on Unit 2 from a September 2008 outage inspection.

Furthermore, FAC program data have indicated no additional erosion from this condition; the current FAC program is considered to be adequate in monitoring for this effect. Other components were also evaluated for operation at higher MCO conditions. One outcome of this was to initiate a revision of the MS stress analysis to ensure MSSV thrust force is acceptable at the higher-than-design MCO condition. This will be complete prior to uprate. This is the same analysis identified as being initiated as in Section IV.1.A.v., NSSS Piping.

Finally, it is expected that the MCO condition projected will be conservative, as projections were made before it was realized from LEFM data that Unit 2 had historically been operating at approximately 0.5% over rated thermal power (see Section I.1.D.1.). Thus MCO was projected for a 1.64% uprate, but the actual uprate relative to the projected MCO level is only about 1.1%

Continued operation at the projected MCO conditions is considered acceptable based on this consideration.

ii. **cumulative usage factors**

The revised design conditions resulting from the MUR PU were reviewed against the CLB for the NSSS ASME components. The MUR PU conditions are bounded by the CLB. Therefore, the stresses and cumulative usage factors remain valid.

iii. **flow induced vibration**

Stresses and forces were calculated for FIV, in the RV Internals and are shown in the following tables. Table IV.1.B.1 compares FIV high-cycle stresses to the code endurance limit, Table IV.1.B.2 summarizes FIV forces relative to the reference Westinghouse 2-loop plant and Table IV.1.B.3 shows the calculated stresses in the thermal shield supports at the MUR PU conditions

Table IV.1.B.1

Summary of Calculated FIV High-Cycle Stresses and Code Endurance Limit (Reference IV.2.2)		
Component	**Alternating Stress (psi)**	**ASME Code Endurance Limit for High-Cycle Fatigue (psi)**
Core Barrel Upper Girth Weld at Flange	4,500	≥23,700
Core Barrel Lower Girth Weld	4,500	≥23,700
Core Barrel Outlet Nozzles	< 750	≥23,700
Thermal Shield Flexures	9,972	≥23,700
Thermal Shield Bolts	10,357	≥23,700
Lower Core Support Plate	362	≥23,700

Table IV.1.B.2

Summary of FIV Forces Relative to Reference 2-Loop Plant (Reference IV.2.3)				
Location	**NSP/NRP Force**		**Reference 2-Loop Plant Force (lb)**	
	Shear (lb)	**Moment (in-lb)**	**Shear (lb)**	**Moment (in-lb)**
Upper Support Column Base Welds	399	3,227	1,900	5,429
Upper Support Column Top welds	458	6,003	1,722	15,803

Table IV.1.B.3

Summary of Calculated RCP-Induced Vibration Thermal Shield Support Stresses[1] (Reference IV.2.32)			
Component	**Alternating Stress (psi)**	**Allowable Stress (psi)**	**Factor of Safety**
Top Support Bolts	9,307	23,700	2.5
Flexures	8,960	23,700	2.6
Flexure Bolts	5,387	23,700	4.4

Note:
1. Stresses represent total thermal shield stresses including the contribution from RCP-induced vibration.

The analysis of the Unit 2 Model 51 steam generators indicates that as a result of the MUR PU, the fluid-elastic stability ratios will remain below the allowable limit of 1.0 for the entire tube bundle assembly except for the peripheral tubes in Rows 6 through 8 plus the interior U-bend region tubes in Rows 10 and 11. Although the tubes at these locations have maximum stability ratios greater than 1.0, their turbulence-induced displacements and tube wear predictions are well below the respective allowables. Therefore, the stability ratio of the steam generators is acceptable under MUR PU conditions. With the uprated power of 1,690 MWt, the turbulence-induced displacements are well below the allowable of one-half the lateral gap between the tubes. After MUR PU, the predicted tube wear for the entirety of plant operation is below the allowable limit of 40-percent of the tube wall thickness. With regard to flow-induced tube vibration stress, the maximum peak bending stress is well below the 1.5 Sm limits of the Alloy 600 tube material of 34.95 ksi. Consequently, the contribution to fatigue is negligible, and fatigue degradation from FIV is not anticipated. The cumulative fatigue usage factor for the tubes is less than 1.0.

The analyses that support the Unit 1 RSGs were also reviewed to determine the impact of the MUR PU. It was confirmed that the MUR PU conditions have no impact on the RSGs' tube vibration because current analyses bound MUR PU conditions.

Therefore operation at the MUR PU conditions is acceptable with respect to tube stresses resulting from tube vibration, cumulative fatigue usage factors, and potential tube wear.

iv. changes in temperature (pre- and post-uprate)

See Table IV.1.B.4

v. changes in pressure (pre- and post-uprate)

See Table IV.1.B.4

vi. changes in flow rates (pre- and post-uprate)

See Table IV.1.B.4

Table IV.1 B.4

Pre-Uprate and Post-Uprate Temperature, Pressure and Flow Rate Changes		
Parameter	**Pre-MUR PU 1657 MWt**	**Post-MUR PU 1690 MWt**
T-avg	560.0°F	560.0°F
T-hot	592.1°F	592.6°F
T-cold	527.9°F	527.4°F
Feedwater Temperature	434.9°F	437.5°F
Steam Temperature	514.1/506.2°F	513.2°F /505.0°F
RC System Pressure (nominal)	2250 psia	2250 psia
Steam Pressure	772 psia/719 psia	765 psia/712 psia
RC System Flow/Loop	89,000 gpm TDF 91,250 gpm MMF 103,300 gpm MDF	89,000 gpm TDF 91,700 gpm MMF 106,000 gpm MDF
Steam Flow Total	7.20 Mpph/7.19 Mpph	7.36 Mpph

The values in Table IV.1.B.4 are for NSSS Thermal Power with 0% steam generator tube plugging. The pre-uprate values used in Table IV.1.B.4 are the 1657 MWt NSSS design values used in the PINGP Safety Analysis Transition Program (SATP) Engineering Report WCAP-16206-P (Reference IV.2.4). The Post MUR PU values are the 1690 MWt NSSS design values developed by Westinghouse for the MUR PU Program (Reference IV.2.5).

<u>Temperature</u>

T_{avg} remains the same. T_{hot} and T_{cold} are expected to increase and decrease by approximately 0.5 °F, respectively. Feedwater temperature increases by approximately 2.6 °F, and steam temperature decreases, by approximately 1.0 °F. The MUR PU conditions are bounded by the CLB, no further evaluation is required.

The post MUR PU values displayed in Table IV.1.B.4 are bounding values associated with a NSSS power of 1690 MWt (1683 MWt + 7 MWt RCP heat) with 0% steam generator tube plugging. A 1.64% MUR PU will result in a nominal power of 1677 MWt, or an NSSS power of 1684 MWt, resulting in actual operating temperatures for T_{avg}, feedwater and steam that are bounded by the analytical values. Therefore, no further evaluation is required.

Pressure

RC System pressure remains unchanged due to MUR PU. However, during the course of verification of structural integrity it was found that plant operation was consistently slightly higher than the pressure used in the current analysis of record. As a result, calculated RC System jet impingement and pipe thrust load calculations do not use a bounding RC System pressure. Corrective action was initiated. The estimated load increase due to this condition was less than 1% over calculated values. The calculated loads are currently well within allowable values, and it was therefore concluded that no structural integrity issue existed that affected current operation. Since MUR PU has no effect on RC System operating pressure, operation at MUR PU is also supported by this conclusion.

Flow

Due to the insignificant change in T_{cold}, RC System flows remain unchanged. Main steam and feedwater mass flows are expected to increase by 2.0% or less under MUR PU conditions, and have been determined to be bounded by the CLB. Thus, no further evaluation is required.

vii. high-energy line break locations

The PINGP Unit 1 and 2 HELB analysis ensures that all inside and outside containment SSCs important to safety are designed to accommodate the environmental effects of postulated accidents. These SSCs are protected against the dynamic effects of a pipe break, including the effects of missiles, pipe whip, and discharging fluids in addition to flooding and compartment temperature and pressure effects. At PINGP Units 1 and 2, high-energy piping systems are defined as those having a service temperature of 200°F and above and a design pressure above 275 psig during normal and upset conditions.

PINGP analyses confirming the plant's ability to withstand the dynamic effects of the rupture of a main steam or feedwater line inside containment and all high-energy pipe line breaks outside of containment are provided in Appendix I (Reference IV.2.6) of the USAR. Analyses confirming the plant's ability to withstand the dynamic effects of the rupture of all other high-energy pipe lines inside of containment are provided in Sections 4.6.2.2 and 12.2.2.1.9 of the USAR (Reference IV.2.7). No high-energy lines are added or modified with the exception of the addition of the LEFM spool pieces in the main feedwater lines to the steam generators. The impact of LEFM installation on the existing feedwater system piping analyses is addressed in a revised outside containment feedwater piping stress analysis for each Unit (References IV.2.8.A and B)

All main steam, steam generator blowdown, and chemical and volume control line breaks are analyzed at no load conditions and are therefore, unaffected by the MUR PU.

The MUR PU results in an increase in feedwater temperature of 2.6°F in the main feed line temperature. Due to lower density at MUR PU conditions, the higher feedwater temperature decreases the break mass flow rate below that which was assumed in the existing calculations for compartment flooding and peak pressure. The 2.6°F increase in feedwater temperature due to the MUR PU increases the enthalpy of the short-term feed line break at full power by up to 2.8 Btu/lbm, which could affect the rate of temperature rise in the compartment. These changes have been reviewed and determined to be inconsequential and therefore, the current outside-containment HELB analyses bound MUR PU conditions and remain valid for operation with an MUR PU. (Reference IV.2.1).

With particular respect to flooding that might be caused by secondary plant equipment, the MUR PU has no appreciable effect on the results because the project involves no appreciable hardware changes; the only significant hardware change is the installation of the flow instruments described previously. The MUR PU involves no changes to secondary system inventory (e.g., hotwell level setpoints) and no changes to equipment performance capabilities that might affect the flooding rate. For example, the project does not affect the main feedwater pumps, their pump curve, or the character of the associated piping. Thereby, the MUR PU does not affect the flooding rate that might be caused by a postulated break in the system.

viii. **Jet impingement and thrust forces**

An evaluation of all high-energy line piping was conducted. This evaluation included the effects of jet impingement, pipe whip and thrust loads. The MUR PU conditions are bounded by the CLB, no further evaluation is required.

C. **The discussion should also identify any effects of the power uprate on the integrity of the reactor vessel with respect to:**

i. **Pressurized Thermal Shock (PTS) calculations**

The PTS calculations were performed for PINGP Units 1 and 2 using the latest procedures specified by the NRC in 10 CFR 50.61 (Reference IV.2.10). New fluence projections accounting for the MUR PU with both OFA and 422 V+ fuel were developed. These projections are documented in References IV.2.11 and IV.2.12 for Units 1 and 2, respectively. Calculated bounding RT_{PTS} values for Units 1 and 2 are presented in Tables IV.1.C.1 and IV.1.C.2, respectively. All RT_{PTS} values remain below the NRC screening criteria values using the projected uprated fluence values for PINGP Units 1 and 2. The MUR PU conditions are bounded by the CLB, no further evaluation is required.

Table IV.1.C.1

Bounding RT_{PTS} Calculations for Prairie Island Unit 1

Reference IV.2.28

Material	RG 1.99 R2 Method	CF (°F)	Fluence (10^{19} n/cm^2)	FF(1)	ΔRT_{PTS}(2) (°F)	$RT_{NDT(U)}$(3) (°F)	Margin(4) (°F)	RT_{PTS}(5) (°F)
Intermediate Shell Forging C	Position 1.1	44	5.162	1.409	61.98	14	34.0	110
	Position 2.1	54.7	5.162	1.409	77.05	14	34.0	125
Lower Shell Forging D	Position 1.1	44	5.026	1.403	61.75	-4.0	34.0	92
Weld Seam W3	Position 1.1	69.7	4.969	1.401	97.66	-13.0	56.0	141
	Position 2.1	80.8	4.969	1.401	113.21	-13.0	56.0	156
Nozzle Shell Forging B	Position 1.1	51	1.770	1.157	59.00	-4.0	34.0	89
Weld Seam W2	Position 1.1	79.5	1.770	1.157	91.97	0.0	65.5	157

Notes:

1. FF = fluence factor = $f^{(0.28 - 0.1 \log (f))}$
2. $\Delta RT_{PTS} = CF * FF$
3. Initial RT_{NDT} values are measured values, with the exception of Weld Seam W2.
4. $M = 2 *(\sigma_i^2 + \sigma_\Delta^2)^{1/2}$
5. $RT_{PTS} = RT_{NDT(U)} + \Delta RT_{PTS}$ + Margin

Table IV.1.C.2

Bounding RT_{PTS} Calculations for Prairie Island Unit 2

Reference IV.2.29

Material	RG 1.99 R2 Method	CF (°F)	Fluence (10^{19} n/cm^2)	FF[(1)]	ΔRT_{PTS}[(2)] (°F)	$RT_{NDT(U)}$[(3)] (°F)	Margin[(4)] (°F)	RT_{PTS}[(5)] (°F)
Intermediate Shell Forging C	Position 1.1	44	5.196	1.410	62.03	14.0	34.0	110
Lower Shell Forging D	Position 1.1	51	5.112	1.407	71.74	-4.0	34.0	102
	Position 2.1	59.6	5.112	1.407	83.84	-4.0	34.0	114
Weld Seam W3	Position 1.1	51.6	5.043	1.404	72.45	-31.0	56.0	97
	Position 2.1	80.2	5.043	1.404	112.6	-31.0	28.0	110
Upper Shell Forging B	Position 1.1	44	1.743	1.153	50.72	-13.0	34.0	72
Weld Seam W2 [(6)]	Position 1.1	69.7	1.743	1.153	80.35	-13.0	56.0	123
	Position 2.1	80.8	1.743	1.153	93.14	-13.0	56.0	136

Notes:
1. FF = fluence factor = $f^{(0.28 - 0.1 \log (f))}$
2. $\Delta RT_{PTS} = CF * FF$
3. Initial RT_{NDT} values are measured values
4. $M = 2 *(\sigma_i^2 + \sigma_\Delta^2)^{1/2}$
5. $RT_{PTS} = RT_{NDT(U)} + \Delta RT_{PTS} +$ Margin
6. Surveillance data for this weld is based on use of Prairie Island Unit 1 weld surveillance data (Weld Seam W3), which was fabricated with Wire Type UM40, Heat No. 1752, Flux Type UM89. Note that different flux lots were used for the Unit 1 surveillance weld (Weld Seam W3) and Unit 2 Weld Seam W2

ii. Fluence evaluation

Revised fluence projections on the PINGP Units 1 and 2 reactor vessels for the MUR PU were calculated and used to determine the impact on the reactor vessel integrity evaluations. Fluence values are used to evaluate the end-of-license transition temperature shift (reference temperature nil-ductility temperature (ΔRT_{NDT})) for the determination of the surveillance capsule withdrawal schedules and Upper Shelf Energy (USE) values. The neutron fluence values are also used in the Adjusted Reference Temperature (ART) calculations to determine the applicability of the heatup and cooldown curves, Emergency Response Guidelines (ERG) limits, and RT_{PTS} values.

PINGP Units 1 and 2 have surveillance capsule withdrawal schedules that meet the intent of ASTM E185-82 (Reference IV.2.13) and 10 CFR Part 50, Appendix H (Reference IV.2.14) for their original end-of-license. PINGP Units 1 and 2 remain in ERG Category I through end-of-license based upon the fluence projections considering the MUR PU. All vessel beltline materials at both Units have RT_{PTS} values at end-of-license that are below the screening criteria provided in 10 CFR 50.61. All vessel beltline materials at both Units have projected USE values that are above the screening criteria provided in 10 CFR 50, Appendix G (Reference IV.2.15) at end-of-license. Core inlet temperatures after the MUR PU remain valid for the embrittlement correlations utilized in this evaluation.

The industry has established thresholds for irradiation assisted stress corrosion cracking (IASCC) among other aging issues such as loss of fracture toughness, void swelling, etc., for various PWR RPV internal components and is developing applicable inspection guidelines. PINGP will continue to participate in these industry initiatives including the Material Reliability Program (MRP) and will implement the associated criteria and inspection guidelines when approved by the NRC by factoring them into the RV internal inspections program, as appropriate. The development of a PWR Vessel Internals Program is included as part of the PINGP License Renewal implementation program.

The MUR PU conditions are bounded by the CLB; therefore, no further evaluation is required.

iii. Heatup and cooldown pressure-temperature limit curves

A review of the applicability dates of the heatup and cooldown curves was performed. These curves are currently contained in the Pressure and Temperature Limits Report (PTLR) (Reference IV.2.16) applicable to 35 EFPY. The review of their term of applicability was performed by comparing the fluence used to generate the current heatup and cooldown curves to the uprated fluence for the beltline materials in the reactor vessels. This review concluded that the revised fluence (reflecting the MUR PU) at 35 EFPY is lower than that used in developing the current P-T limit curves. Fluence is a key parameter in the ART calculation, with a lower fluence resulting in lower ART values. Therefore, the applicability dates for the 35 EFPY P-T curves remain valid. The MUR PU conditions are bounded by the CLB; therefore, no further evaluation is required.

iv. Low-temperature overpressure protection

The Low-Temperature Overpressurization Protection (LTOP) provides RC System pressure relief capability during relatively low temperature operation. Two pressurizer power operated relief valves (PORV) are used to provide the automatic relief capability during the design basis mass input (MI) and heat input (HI) transients to automatically prevent the RC pressure from exceeding the pressure and temperature limits of 10 CFR 50, Appendix G. The design basis MI and HI transients for both PINGP Units are defined in the PINGP Technical Specifications (Reference IV.2.17) and in the PTLR.

The LTOP PORV setpoint analyses are performed at reactor shutdown and RC System cold conditions. Therefore, the MUR PU does not affect the LTOP PORV setpoint determination. The critical parameters for LTOP PORV setpoint determination are 1) the design basis MI and HI transients, 2) RC System volumes, 3) pressurizer PORV characteristics, 4) wide range pressure/temperature uncertainties, and 5) pressure temperature limits of 10 CFR 50, Appendix G. These critical parameters will not change for the MUR PU. The MUR PU conditions are bounded by the CLB; therefore, no further evaluation is required.

v. **Upper shelf energy (USE)**

The integrity of the reactor vessel may be affected by changes in system temperatures and pressures resulting from the power uprate. To address this consideration, an evaluation was performed to assess the impact of the MUR PU on the USE values for all the reactor vessel beltline materials in the PINGP Units 1 and 2 reactor pressure vessels. USE decrease projections were calculated for PINGP Units 1 and 2 using procedures specified by the NRC in Regulatory Guide 1.99 (Reference IV.2.18). Based on the current analysis, all vessel materials are expected to have a USE greater than 50 ft-lb through end-of-license as required by 10 CFR 50, Appendix G. All USE values were predicted using the 1/4T neutron fluence projections. The MUR PU conditions are bounded by the CLB; therefore, no further evaluation is required.

vi. **Surveillance capsule withdrawal schedule**

The current surveillance capsule withdrawal schedule for PINGP Units 1 and 2 is based on ASTM E185-82. 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. The capsules removed from the PINGP Units 1 and 2 vessels to date meet the intent of ASTM E185-82 for the original end-of-license of both Units (References IV.2.19 and IV.2.20).

Updated neutron fluence projections were used to calculate ΔRT_{NDT} values to determine the minimum number of capsules to be withdrawn in the surveillance capsule withdrawal schedule. These fluence projections, accounting for the MUR PU, indicate the ΔRT_{NDT} of the beltline materials remains between 100°F and 200°F. Both Units, therefore, continue to maintain a "four capsule" minimum withdrawal schedule in accordance with ASTM E185-82.

D. The discussion should identify the code of record being used in the associated analysis, and any changes to the code of record.

The Original Code of Record is contained in the USAR, Table 4.1-11, reproduced below as Table IV.1.D.1. The MUR PU does not impact the Code of Record.

Table IV.1.D.1

Component	Original Code of Record
Reactor Vessel	ASME III Class A, 1968 Edition Winter 1968 Addenda
Reactor Vessel Head	ASME III Class 1, 1998 Edition 2000 Addenda
CRDM Housing	ASME III Class 1, 1998 Edition 2000 Addenda
Core Exit Thermocouple Nozzle Assemblies (CETNA)	ASME III Class 1, 1998 Edition 2000 Addenda
Steam Generators – Tube Side	ASME III: Unit 1 – Class 1,1995 Edition through 1996 Addenda; Unit 2 – Class A, 1965 Edition Winter 1966 Addenda
Steam Generators – Shell Side	ASME III: Unit 1 – Class 2, 1995 Edition through 1996 Addenda; Unit 2 – Class C, 1965 Edition Winter 1966 Addenda
Reactor Coolant Pump Casing	No Code (Design per ASME III – Article 4, 1968 Edition Winter 1969 Addenda)
Pressurizer	ASME III Class A: Unit 1 – 1965 Edition Summer 1966 Addenda; Unit 2 – 1965 Edition Winter 1966 Addenda
Pressurizer Relief Tank	ASME III Class C
Pressurizer Safety Valves	ASME III 1968 Edition
RC System Piping	Unit 1: ASA B31.1 1955; Unit 2: USAS B31.1.0 1967
Reactor Coolant Gas Vent System Piping	ASME III Class 1 & 2 1983 Edition
Reactor Vessel Level Instrumentation Piping (from head penetration to isolation Valve RC-17-3 [2RC-17-3])	ASME III Class 1 & 2 1998 Edition 2000 Addenda

E. The discussion should identify any changes related to the power uprate with regard to component inspection and testing programs and erosion/corrosion programs, and discuss the significance of these changes. If the changes are insignificant, the licensee should explicitly state so.

Component inspection and testing programs that could be affected by the MUR PU were reviewed. The review included the current In-Service Inspection (ISI) Program, the In-Service Testing (IST) Program, the Motor Operated Valve (MOV) Program, and the Flow Accelerated Corrosion (FAC) Program.

ISI Program

There are no modifications or replacement of ASME Code Class components that provide safety related functions for the MUR PU. Additionally, there are no new safety related functions required of existing equipment and no new safety related equipment will be installed to accommodate the MUR PU. Therefore, the ISI Program at PINGP is not affected by the 1.64% MUR PU.

IST Program

There are no modifications or replacement of ASME Code Class components that provide safety related functions for the MUR PU. Additionally, there are no new safety related functions required of existing equipment and no new safety related equipment will be installed to accommodate the MUR PU. Therefore, the current IST Program at PINGP remains valid after the MUR PU is implemented.

Motor-Operated Valves

The PINGP Units 1 and 2 MOV Program addresses the requirements of NRC Bulletin 85-03 (Reference IV.2.21), Generic Letter (GL) 89-10 (Reference IV.2.22), and GL 96-05 (Reference IV.2.23) for MOV thrust and torque requirement calculations. PINGP has addressed the requirements of GL 95-07 (Reference IV.2.24) regarding MOV pressure lock and thermal binding requirement calculations. The MOV Program applies to all safety related MOVs.

Valves in the NSSS and BOP included in the MOV Program were reviewed for potential impact by the MUR PU. The MUR PU conditions are bounded by the CLB; therefore, no further evaluation is required.

FAC Program

NRC Generic Letter 89-08 (Reference IV.2.25) addresses the concern of erosion/corrosion of the inner pipe wall of high-energy, carbon steel piping. NRC Generic Letter 89-13 (Reference IV.2.26) addresses concerns regarding the capability of plant open-cycle cooling water systems to fully perform their intended design functions as required by the plant design bases. This concern includes the effects of erosion/corrosion on service water piping. PINGP Units 1 and 2 have a comprehensive Flow-Accelerated Corrosion (FAC) Program based on the requirements of NSAC-202L-R3 (Reference IV.2.27). Susceptible safety related and non-safety related systems and components are modeled at PINGP Units 1 and 2 using the Electric Power Research Institute's CHECWORKS software.

The PINGP response to FAC in carbon steel piping is programmatically controlled through plant procedures which do not require change as a result of the MUR PU. A recent sensitivity study indicates that the CHECWORKS model provides a reasonable and conservative prediction of actual wear rates using best-estimate plant parameters. Based on a review of the PINGP Units 1 and 2 FAC Program, it was determined that the methods, guidelines, acceptance criteria, and procedures that make up the PINGP FAC Program will continue to adequately ensure plant and personnel safety following the MUR PU.

Since MUR PU conditions increase moisture carryover, operating temperature, pressure and flow velocities in several BOP piping systems, the MUR PU does have a minor impact on the FAC Program. As a result, the 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 study indicated a negligible impact by MUR on wear-rate predictions. The results are indicated in Tables IV.1.E.1 and IV.1.E.2.

Table IV.1.E.1
Unit 1 FAC Information
Comparison of Current Power Conditions to MUR Power Conditions

System	Component	Nominal Thickness (inches)	Predicted Thickness at RFO 26 Applying Current Power Conditions (inches)	Predicted Thickness at RFO 26 Applying MUR Power Conditions (inches)	Ratio of Current Predicted Thickness to MUR Predicted Thickness (Current/MUR)
CONDENSATE	XH-106-385-A E1	0.375	0.307	0.307	1.000
CONDENSATE	XH-106-159-B E7	0.375	0.307	0.307	1.000
CONDENSATE	XH-106-159-A E9	0.375	0.305	0.305	1.000
CONDENSATE	XH-106-159-A E10	0.375	0.301	0.301	1.000
CONDENSATE	XH-106-159-B R1	0.375	0.324	0.324	1.000
CONDENSATE	XH-106-158-B E6	0.375	0.307	0.307	1.000
FEEDWATER	XH-106-129-D E5	1.031	0.819	0.819	1.000
FEEDWATER	XH-106-129-C S1	1.031	0.740	0.740	1.000
HEATER DRAINS	XH-106-18922-A E4	0.276	0.169	0.169	1.000
HEATER DRAINS	XH-106-119-B E1	0.500	0.374	0.374	1.000
SG BLOWDOWN	NF-88727 E15	0.237	0.188	0.188	1.000
SG BLOWDOWN	NF-88728 E10	0.237	0.188	0.188	1.000
SG BLOWDOWN	NF-88728 E2	0.237	0.188	0.188	1.000
SG BLOWDOWN	NF-88728 E5	0.237	0.188	0.188	1.000
SG BLOWDOWN	NF-88727 E19	0.237	0.188	0.188	1.000
SG BLOWDOWN	NF-88728 E9	0.237	0.188	0.188	1.000
SG BLOWDOWN	NF-88728 E1	0.237	0.188	0.188	1.000
SG BLOWDOWN	NF-88728 E8	0.237	0.188	0.188	1.000
SG BLOWDOWN	NF-88727 E9	0.237	0.188	0.188	1.000
SG BLOWDOWN	NF-88727 R1	0.216	0.170	0.170	1.000

Table IV.1.E.2

Unit 2 FAC Information
Comparison of Current Power Conditions to MUR Power Conditions

System	Component	Nominal Thickness (inches)	Predicted Thickness at RFO 25 Applying Current Power Conditions (inches)	Predicted Thickness at RFO 25 Applying MUR Power Conditions (inches)	Ratio of Current Power Level Predicted Thickness to MUR Power Level Predicted Thickness (Current/MUR)
BLEED STEAM	XH-1006-5-C R5	0.375	0.260	0.259	0.996
CONDENSATE	XH-1106-430-B T1	0.375	0.321	0.321	1.000
CONDENSATE	XH-1106-433-A E5	0.375	0.315	0.315	1.000
FEEDWATER	XH-1106-245-E P3	1.031	0.864	0.864	1.000
FEEDWATER	XH-1106-245-F E1	1.031	0.784	0.784	1.000
FEEDWATER	XH-1106-245-G E4	1.281	1.001	1.001	1.000
HEATER DRAINS	XH-1106-412-A E1	0.375	0.302	0.302	1.000
HEATER DRAINS	XH-1106-412-A E4	0.375	0.296	0.296	1.000
HEATER DRAINS	XH-1106-412-A P5	0.375	0.307	0.307	1.000
HEATER DRAINS	XH-1106-412-B E1	0.375	0.302	0.302	1.000
HEATER DRAINS	XH-1106-436-A P2	0.500	0.437	0.437	1.000
HEATER DRAINS	XH-1106-437-A P7	0.432	0.349	0.349	1.000
HEATER DRAINS	XH-1106-438-B E6	0.432	0.330	0.330	1.000
HEATER DRAINS	XH-1106-438-B R4	0.432	0.363	0.363	1.000
HEATER DRAINS	XH-1106-463-B E1	0.322	0.312	0.312	1.000
HEATER DRAINS	XH-1106-464-A P6	0.322	0.231	0.231	1.000
HEATER DRAINS	XH-1106-464-C E3	0.322	0.227	0.227	1.000
HEATER DRAINS	XH-1106-7706-A E4	0.276	0.221	0.221	1.000
HEATER DRAINS	XH-1106-7706-B E6	0.276	0.268	0.268	1.000
HEATER DRAINS	XH-1106-7707-B R2	0.276	0.223	0.223	1.000

As required by existing configuration control procedures, CHECWORKS models will be revised as appropriate, as part of MUR implementation effort to incorporate actual flow and process system conditions that are determined for MUR PU conditions. Programmatically, the results of these upgraded models will be factored into future surveillance and piping repair plans, as applicable.

F. The discussion should address whether the effect of the power uprate on steam generator tube high cycle fatigue is consistent with NRC Bulletin 88-02, "Rapidly Propagating Fatigue Cracks in Steam Generator Tubes," February 5, 1988.

Operating conditions on the secondary side of the steam generators will change as a result of the MUR PU. These changes can affect previous high-cycle fatigue evaluations of the unsupported U-bend tubes in both Westinghouse Model 51 steam generators in Unit 2. The previously performed evaluations were aligned with the methods and assumptions described in NRC Bulletin 88-02 (Reference IV.2.9) regarding rapid propagation of fatigue cracks in Westinghouse Model 51 steam generator tubes.

The total fatigue usage factor of the limiting tube should remain below 1.0 at the end of the Unit 2 licensed operating life. This usage factor is calculated considering both past operation at the current power level and future operation at the MUR PU as well as potential minimum secondary side steam pressures.

An evaluation showing that the Unit 2 steam generator tubing will not rupture by high-cycle fatigue in the manner of the North Anna Unit 1 tube was performed by a relative assessment, since methods for analytical prediction of stability ratios (SR) incorporate greater uncertainties than a relative ratio. In other words, the stress amplitudes and displacements associated with a specific value of SR can be more accurately determined by tube rupture analysis of North Anna Unit 1. Details of the method of this analysis were consistent with those of the original high cycle U-bend fatigue analyses performed in accordance with NRC Bulletin 88-02.

The relative stability ratios (RSR) and associated fatigue usage factors as a result of the MUR PU versus the maximum allowable RSR under the described conditions were determined. The maximum allowable RSRs for the critical tube location for both analyzed operating conditions (thermal design and best-estimate flow) are larger than the calculated RSRs and the fatigue usage factors are less than one.

Since the maximum allowable RSRs for the critical tube U-bend fatigue location at both the thermal and design best-estimate fluid flow rates for the Unit 2 steam generators uprated operating conditions are larger than the calculated RSRs, the fatigue usage factor will be less than one. Therefore, the acceptance criterion of a fatigue usage factor less than 1.0 for the most critical tube in the bundle is met.

2. References for Section IV

IV.2.1 WCAP-16917-NP, Prairie Island Nuclear Generating Plant Units 1 & 2 Measurement Uncertainty Recapture Power Uprate Licensing Report, Section 7.1, Rev. 0, February 2009

IV.2.2 WCAP-16917-NP, Prairie Island Nuclear Generating Plant Units 1 & 2 Measurement Uncertainty Recapture Power Uprate Licensing Report, Section 7.3-3, Rev. 0, February 2009

IV.2.3 WCAP-16917-NP, Prairie Island Nuclear Generating Plant Units 1 & 2 Measurement Uncertainty Recapture Power Uprate Licensing Report, Section 7.3-4, Rev. 0, February 2009

IV.2.4 WCAP-16206-P, "Safety Analysis Transition Program Engineering Report For Prairie Island Nuclear Power Plant", Volume 1 (Tables 1-2 and 1-3), February 2004

IV.2.5 Westinghouse Letter NSP-07-24, Nuclear Management Company Prairie Island Units 1 and 2 MUR Power Uprate Program NSSS Design Parameters With Revised MMF, April 20, 2007

IV.2.6 Prairie Island Nuclear Generating Plant, Updated Safety Analysis Report, Appendix I, Postulated Pipe Failure Analysis Outside of Containment, Revision 29

IV.2.7 Prairie Island Nuclear Generating Plant, Updated Safety Analysis Report, Sections 4.6.2.2 and 12.2.2.1.9, Revision 29

IV.2.8 Prairie Island Nuclear Generating Plant, Feedwater Outside Containment Piping Analyses

A. PI-996-1-P01, Evaluation of Feedwater Piping System Outside Containment – U1, Rev. 1, August 9, 2007

B. PI-996-1-P02, Evaluation of Feedwater Piping System Outside Containment – U2, Rev. 1, August 10, 2007

IV.2.9 NRC Bulletin Number 88-02, "Rapidly Propagating Fatigue Cracks in Steam Generator Tubes," February 5, 1988.

IV.2.10 Code of Federal Regulations, 10 CFR 50.61, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events," Federal Register, 60 FR 65468, December 19, 1995

IV.2.11 WCAP-14781, "Evaluation of Pressurized Thermal Shock for Prairie Island Unit 1," S. L. Abbott, Rev. 4, September, 2007

IV.2.12 WCAP-14638-NP, "Evaluation of Pressurized Thermal Shock for Prairie Island Unit 2," Rev. 4, September, 2007

IV.2.13 ASTM E185-82, "Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels"

IV.2.14 Code of Federal Regulations, 10 CFR 50, Appendix H, "Reactor Vessel Material Surveillance Program Requirements," Federal Register, Volume 60 FR 65476, December 19, 1995

IV.2.15 Code of Federal Regulations, 10 CFR 50, Appendix G, "Fracture Toughness Requirements," Federal Register, Volume 60 FR 65474, December 19, 1995

IV.2.16 Prairie Island Nuclear Generating Plant, Pressure and Temperature Limits Report (PTLR), Rev.3, October 21, 2002

IV.2.17 Prairie Island Nuclear Generating Plant Unit 1 and Unit 2 Technical Specifications

IV.2.18 Nuclear Regulatory Commission, Regulatory Guide RG 1.99, "Radiation Embrittlement of Reactor Vessel Materials", Revision 2, May 1988

IV.2.19 WCAP-14779, "Analysis of Capsule S from the Northern States Power Company Prairie Island Unit 1 Reactor Vessel Radiation Surveillance Program," S. L. Abbott, et al., Rev. 2,January 1997

IV.2.20 WCAP-14613, "Analysis of Capsule P from the Northern States Power Company Prairie Island Unit 2 Reactor Vessel Radiation Surveillance Program," S. L. Abbott, et al., Rev. 2, February 1998

IV.2.21 NRC IE Bulletin 85-03, "Motor-Operated Valve Common Mode Failures during Plant Transients Due to Improper Switch Settings," November 15, 1985

IV.2.22 USNRC Generic Letter GL 89-10, "Safety-Related Motor-Operated Valve Testing and Surveillance", June 28, 1989

IV.2.23 USNRC Generic Letter GL 96-05, "Periodic Verification of Design-Basis Capability of Safety-Related Motor-Operated Valves," September 18, 1996

IV.2.24 USNRC Generic Letter GL 95-07, "Pressure Locking and Thermal Binding of Safety-Related Power-Operated Gate Valves", August 17, 1995

IV.2.25 USNRC Generic Letter GL 89-08, "Erosion/Corrosion – Induced Pipe Wall Thinning," May 2, 1989

IV.2.26 USNRC Generic Letter 89-13, "Service Water System Problems Affecting Safety-Related Equipment," July 18, 1989

IV.2.27 EPRI NSAC-202L-R3, "Recommendations for an Effective Flow-Accelerated Corrosion Program", May 2006

IV.2.28 Westinghouse Calculation CN-MRCDA-07-59, Prairie Island Units 1 & 2 Measurement Uncertainty Recapture: Reactor Vessel Integrity Evaluation, Rev. 1, August 10, 2007

IV.2.29 Westinghouse Calculation CN-REA-07-28, Reactor Vessel Fluence Calculations for Prairie Island MUR uprate Program, Rev. 1

IV.2.30 USAS-B31.1.0-1967 USA Standard (USAS) Code for Pressure Piping – Power Piping

IV.2.31 Prairie Island Nuclear Generating Plant, Updated Safety Analysis Report, Table 12.2-13, Revision 30

IV.2.32 WCAP-16917-NP, Prairie Island Nuclear Generating Plant Units 1 & 2 Measurement Uncertainty Recapture Power Uprate Licensing Report, Section 7.3-5, Rev. 0, June 2009

V. Electrical Equipment Design

1. A discussion of the effect of the power uprate on electrical equipment. For equipment that is bounded by the existing analyses of record, the discussion should cover the type of confirmatory information identified under Section II, above. For equipment that is not bounded by existing analyses of record, a detailed discussion should be included to identify and evaluate the changes related to the power uprate. Specifically, this discussion should address the following items:

A. emergency diesel generators

B. station blackout equipment

C. environmental qualification of electrical equipment

D. grid stability

Normal Station Electrical

Main Generator and Associated Equipment

The main generator is a synchronous type, direct driven, hydrogen cooled machine supplied by Westinghouse Electric Corp. The machine is nominally rated at 659 MVA with 60 psig H_2 pressure, 0.9 power factor, three phase, 60 hz, 20,000 volts, 1800 rpm. The generator for each Unit is connected to the generator step up (GSU) transformers via the Isolated Phase Bus Duct. The isophase bus duct, Bus duct cooling, generator step-up transformers, overhead lines to the switchyard, and switchyard equipment are all designed to support operation of the main generators within the generator capability curves. The current duty loading is based on the generator operating at design MVA, minimum PF, and minimum voltage. Following the MUR PU, the generator will continue to be operated with the original design conditions in accordance with the existing generator capability curves.

Normal Station Service

For normal operation, each Unit has a 40 MVA station auxiliary transformer with dual low side windings that supply power to the four non-safety related 4160 VAC and associated 480 VAC buses for each Unit. Each station auxiliary transformer low-side winding typically feeds two of the four non-safety related 4160 VAC buses. The total station service load for each Unit is typically within the range of 28 – 31 MVA. There are no major equipment modifications associated with the implementation of the MUR PU, therefore, the only increase in station service loading on each Unit is associated with a minor increase in the load required by the main feedwater and condensate pumps which corresponds to an increase in the station service for each Unit of 0.095 MVA.

When a Unit is not on line, power to the four non-safety related 4160 VAC and associated 480 VAC buses for each Unit can be fed from either of two (1R or 2R) reserve auxiliary transformers.

4160/480 VAC Emergency Power System

During normal operation, the 4160/480 VAC emergency power for each Unit (Bus 15 and 16 for Unit 1, Bus 25 and 26 for Unit 2) is supplied from offsite power through either of two reserve auxiliary transformers (1R from 161 KV bus, or 2R from 345 KV bus) or the cooling tower transformers 11 and 12. Unit 1 normal alignment of the 4160 VAC emergency buses is from reserve auxiliary transformer 1R (Train A) and Cooling Tower Transformer CT 11 (Train B) while the Unit 2 normal alignment of the 4160 VAC emergency buses is from reserve auxiliary transformer 2R (Train A) and Cooling Tower Transformer CT 12 (Train B). There are no additional loads added to the emergency power buses to support the MUR power uprate therefore there are no changes to the reserve auxiliary transformer loadings to support MUR PU.

A. Emergency diesel generators

Each Emergency Diesel Generator, as a backup to the normal standby AC power supply, is capable of sequentially starting and supplying the power requirements of one of the redundant sets of engineered safety features for its reactor Unit. In addition, in the event of a station blackout (SBO) condition, each Emergency Diesel Generator is capable of sequentially starting and supplying the power requirements of the hot shutdown (Mode 3, Hot Standby in ITS) loads for its Unit, as well as the essential loads of the blacked out Unit, through the use of manual bus tie breakers interconnecting the 4,160V AC buses.

The emergency onsite power system provides a reliable onsite source of 4,160V AC power to the engineered safeguards electrical buses. The emergency onsite power system consists of two emergency diesel generators per Unit that supply the necessary loads to shut down and maintain the plant in a safe shutdown condition to mitigate the consequences of postulated design basis accidents.

The emergency onsite power system was reviewed for potential impact by the MUR PU and found to be unaffected. The MUR PU conditions are bounded by the CLB; therefore, no further evaluation is required.

B. Station blackout equipment

The power level assumed in the Station Blackout (SBO), 10 CFR 50.63, Loss of All AC Power (Reference V.2.1) submittal analyses was 102% of rated power. PINGP USAR, Section 14.4.11 (Reference V.2.2) performed a bounding, worst-case analysis for the loss of all AC power to the station auxiliaries. This analysis was verified as being unchanged for MUR PU conditions.

At multi-Unit sites, such as PINGP, where the combination of emergency AC power sources exceeds the minimum redundancy requirements for safe shutdown (non-design-basis accident (DBA)) of all Units, the remaining emergency AC power sources may be used as alternate AC power sources provided they meet the applicable requirements. If these criteria are not met, station blackout must be assumed on all the Units, per 10 CFR 50.2 (Reference V.2.3).

PINGP Units 1 and 2 implemented design features to create a four-hour SBO-coping plant. However, most of the typical considerations for a plant with a four-hour coping time are not important for PINGP Units 1 and 2 because one of the emergency diesel generators (EDG) on the non-affected Unit is configured to act as an Alternate AC (AAC) source in accordance with Regulatory Guide 1.155, (Reference V.2.4). AAC can be connected from the unaffected PINGP Unit to the one experiencing SBO within 10 minutes. During that time, the direct current (DC) battery system at the station will provide instrumentation and control power so that safe plant operation can continue until the AAC is connected.

The diesel generators have sufficient capacity and capability to provide power to the shutdown buses within 10 minutes of the realization of the SBO event in accordance with USAR 8.4.4 (Reference V.2.5) for the required duration of 4 hours.

Having AAC connected to the affected Unit within 10 minutes negates issues with loss of cooling to the reactor coolant pump (RCP) seals. The RCP seals at PINGP Units 1 and 2 are durable enough to maintain their nominal function for 10 minutes without cooling during an SBO event. Other systems associated with SBO not affected by the MUR PU include HVAC, containment isolation, and heat tracing of SBO-coping equipment. There are no expected changes to any of these systems' capabilities due to the MUR PU.

The methodology and assumptions with regard to equipment operability associated with the SBO analysis are unchanged with the MUR PU. There is no change in the ability of the turbine-driven auxiliary feedwater pumps, supplied with steam from the steam generators, and other onsite resources needed to support reactor heat removal. Therefore, the equipment operability assumptions and calculation methods are not affected by the MUR PU.

An assessment of the impact of changes needed to implement the MUR PU (such as cold leg fluid temperature, core power, or setpoints) indicated the effects to be negligible. The slightly higher heat input in the primary and secondary systems will result in a small increase in the duration of equipment operation (such as steam relief), but it does not impact the continuous rating of electrical equipment.

In 2005, PINGP performed a study (Reference V.2.6) to confirm that Technical Specification 3.7.6.1 (Reference V.2.7) requirements for Condensate Storage Tank (CST) volume remained satisfied following installation of the replacement steam generators. The calculation was performed for the RSGs and the OSGs at 102% of 1,650 MWt and assuming 2-hour duration in hot standby mode followed by 6-hour cooldown to residual heat removal initiation. Calculations indicate that the RSGs are the more limiting and require 99,372 gallons of CST volume to support the 8-hour scenario.

A 2007 re-evaluation (Reference V.2.8) of the required CST volume was conducted for the MUR PU conditions. In addition to considering the RSGs, the analysis also assumes 2 hours in hot standby mode followed by a 6-hour natural circulation cooldown. The required usable storage volume was calculated at 98,800 gallons for the combination of system configurations involving the RSGs. This correlates well with the results of previous engineering calculations, given the slightly different calculation inputs and assumptions of the two analyses. Therefore, Surveillance Requirement 3.7.6.1 which requires more than 100,000 gallons in the CST on-hand at each Unit remains acceptable for the proposed MUR PU.

Loss of HVAC during SBO is considered for the containment, main Control Room, AF turbine-driven pump room, and the Battery Rooms. The pre-MUR PU analysis found that the temperatures in these rooms would remain acceptable in accordance with the criteria provided in NUMARC 87-00 Rev. 1 (Reference V.2.9). These temperature studies are not strongly dependent on the rated reactor core power. Furthermore, the plant is licensed to have AAC available within 10 minutes which would make any temperature excursion brief. Thus, the existing assessments remain applicable for the MUR PU.

The station blackout equipment was reviewed for potential impact by the MUR PU. The MUR PU conditions are bounded by the CLB; therefore, no further evaluation is required.

C. Environmental Qualification of electrical equipment

The normal temperature, pressure, and humidity environment inside containment is based on the main steam and feedwater operating temperatures, T_{cold}, T_{avg}, T_{hot}, the RCP heat load, and other electrical loads inside containment. The main steam operating temperature decreases with MUR PU. The slight increase in feedwater operating temperature, 2.6 °F (434.9 °F to 437.5 °F) has a negligible effect on the environmental temperatures. The values for T_{cold}, T_{avg}, and T_{hot}, at 102% of the original licensed thermal power (OLTP) bound operation at MUR PU. The RCP motor electrical duty (starting current, running current, etc.) and cooling loads do not change under MUR PU. Therefore, the heat load associated with the RCP does not change with MUR PU. Further, there are no hardware changes or changes to the design electrical load demand (or timing of the load sequencing) for any NSSS/ECCS pump under MUR PU. Since there is little or no change in the main steam and feedwater operating temperatures, T_{cold}, T_{avg}, T_{hot}, the RCP heat load, and other electrical loads inside containment, the normal temperature, pressure, and humidity environment inside containment is acceptable for MUR PU.

The containment integrity analysis (i.e., peak containment pressure and temperature) is bounded by the LOCA analyses which are based on an analytical power level at 102% OLTP, which is unchanged by MUR PU. Further, the containment subcompartment pressurization design basis loads are bounding for MUR PU.

The limiting environmental conditions inside the containment used for equipment qualification are based on the MSLB, and the containment design pressure. The current MSLB analysis and containment design pressure remain unchanged for the MUR PU conditions. The basis of the steam line break analysis is the full guillotine main steam line break at Hot Zero Power (HZP) for peak compartment pressure, which remains unchanged for MUR PU operating conditions and Hot Full Power (HFP) for peak compartment temperature, which is bounding for MUR PU conditions.

MUR PU does not relocate existing heat sources or disturb existing insulation outside containment. The MUR PU results in a small increase in overall heat load from the increase in fluid heat loads and motor heat loads in the Auxiliary and Turbine Buildings. Evaluations have shown that the increase in heat load as a result of MUR PU is insignificant. Therefore, the MUR PU does not impact normal pressures, temperatures, and humidity in EQ areas outside containment.

The current limiting temperature and pressure transients used for qualification due to pipe breaks outside containment are based on the effects of a potential HELB. Evaluations indicate that the effects of postulated HELB events remain bounding for MUR PU.

Containment spray does not apply to normal operations, rather it relates to accident scenarios. The chemical composition of the containment spray in the containment during an accident is based on the volume and concentrations of boron in the refueling water storage tank and NaOH in the caustic addition standpipe. The boron concentration and volume requirement in Technical Specification 3.5.4, "Refueling Water Storage Tank" (Reference V.2.10), are not impacted by MUR PU. The injection and containment spray flow requirements for the LOCA and containment analyses are not impacted by MUR PU. Further, the Containment Spray System design duty is not changed by MUR PU (delivered flow, head, delay time associated with EDG sequencing and containment spray line fill time). Therefore, these volumes and concentrations remain unchanged for the core uprate confirming that the existing containment spray EQ limits remain valid.

In accordance with 10 CFR 50.49, safety-related electrical equipment must be qualified to survive the radiation environment at their specific location during normal operation and during an accident.

For purposes of environmental qualification, PINGP is divided into various environmental areas. The radiological environmental conditions noted for these areas are the maximum conditions expected to occur and are representative of the whole area. Normal operation parameters represent 40 years of operation. Post-accident radiation exposure levels are determined for a 1-year period following a LOCA.

Radiation doses (normal + accident) will increase by ~ 1.64% after the implementation of the MUR PU. This increase in radiation is addressed as follows:

- For Westinghouse Qualified Equipment – This equipment remains as described in WCAP-8587 (Reference V.2.11) and is qualified for normal and accident source terms for an enveloping core power of 4100 MWt. This core power will continue to envelope PINGP operations at MUR PU.

- For Non-Westinghouse Qualified Equipment – The environmental levels currently used to support electrical equipment qualification reflect a core power of 1683 MWt and bound operation at MUR PU conditions inclusive of the margin for power level uncertainty.

D. Grid stability

The Midwest Independent System Operator (MISO) evaluated the collective impact of a proposed 38 MWe expansion of PINGP Units 1 and 2 (19 MWe/Unit) (References V.2.12 and V.2.13). PINGP Unit 1 is currently rated at approximately 551 MWe net capacity and PINGP Unit 2 is currently rated at approximately 545 MWe net capacity. The total station generation increase of 38 MWe used in the evaluation bounds and includes the expected load increase (approximately 9 MWe/Unit) from the PINGP 1.64% MUR PU and other anticipated system load increases (10 MWe/Unit) from efficiency and equipment changes unrelated to the MUR PU. The evaluation addressed grid stability and steady-state issues associated with the collective impact on the transmission system of a 38 MWe electrical output increase at PINGP.

The MISO evaluation consisted of a transmission interconnection study that evaluated the collective impact of a total 38 MWe increase on the transmission system and included system performance evaluation based on stability analysis and steady-state analysis. In addition, a supplemental analysis was performed to confirm the upgrades meet the transmission system study requirements of IEEE 765, IEEE Standard for Preferred Power Supply (PPS) for Nuclear Power Generating Stations (NPGS). The stability analysis was used as the basis for determining if the PINGP expansion meets regional stability requirements. The evaluation concluded that the 38 MWe from the PINGP electrical output increase would not adversely impact transmission system stability and no stability criteria violations were observed for simulated faults. The steady-state analysis indicated that the 38 MWe PINGP expansions would not require any transmission improvements to allow interconnection. The supplemental analysis demonstrated that following the PINGP expansions, the voltages at PINGP 161 and 345 kV buses were within the PINGP voltage criteria for all the disturbances studied.

The MISO generation interconnection study, supplemental transient stability study, and corresponding acceptance letter are included as Enclosure 8 to this License Amendment Request.

2. References for Section V

V.2.1 Code of Federal Regulations, 10 CFR 50.63, "Loss of All alternating current power", September 22, 1998

V.2.2 Prairie Island Nuclear Generating Plant, Updated Safety Analysis Report, Section 14.4.11, "Loss of All AC Power to the Station Auxiliaries (LOOP)", Rev.30

V.2.3 Code of Federal Regulations, 10 CFR 50.2, "Definitions", December 4, 2007

V.2.4 Nuclear Regulatory Commission, Regulatory Guide RG 1.155, "Station Blackout," August 1988

V.2.5 Prairie Island Nuclear Generating Plant, Updated Safety Analysis Report, Section 8.4.4, "Station Blackout," Revision 30

V.2.6 Prairie Island Nuclear Generating Plant, Calculation ENG-ME-443, "Condensate Storage Tank Sizing", Rev. 4, October 7, 2005

V.2.7 Prairie Island Nuclear Generating Plant, Unit 1 and Unit 2, Technical Specifications, DPR-42/60, Amendment 158/149, Section 3.7.6

V.2.8 Westinghouse Calculation Note, CN-SEE-07-10, "Prairie Island 1&2 MUR Program Condensate Storage Tank Required Storage Volume", Rev. 1

V.2.9 Nuclear Energy Institute (Nuclear Utility Management and Resource Council), NUMARC 87-00, "Guidelines and Technical Bases for NUMARC Initiatives Addressing Station Blackout at Light Water Reactors," Rev. 1, August 1991

V.2.10 Prairie Island Nuclear Generating Plant, Unit 1 and Unit 2, Technical Specification 3.3.4.

V.2.11 WCAP-8587 "Methodology", "Methodology for Qualifying Westinghouse WRD Supplied NSSS Safety Related Electrical Equipment," Revision 6-A, March 1983.

V.2.12 ABB Inc., ESC Report No. 2005-11125-2.R02, "Generation Interconnection Study – Projects # G433 and G434; 38 MW Expansion of Prairie Island Units 1 and 2", March 24, 2006

V.2.13 Delyn Electrical Engineering, LLC, "G433-G434 Transient Stability Study – Supplement", November 13, 2009

VI. System Design

1. A discussion of the effect of the power uprate on major plant systems. For systems that are bounded by existing analyses of record, the discussion should cover the type of confirmatory information identified under Section II, above. For systems that are not bounded by existing analyses of record, a detailed discussion should be included to identify and evaluate the changes related to the power uprate. Specifically, this discussion should address the following systems:

A. NSSS interface systems for pressurized-water reactors (PWR) (e.g., main steam, steam dump, condensate, feedwater, auxiliary/emergency feedwater) or boiling-water reactors (BWR) (e.g., suppression pool cooling), as applicable

As part of the PINGP Units 1 and 2 MUR PU, the following BOP fluid systems were reviewed to assess compliance with Westinghouse NSSS/BOP interface requirements identified in the PINGP USAR:

- Main Steam System
- Steam Dump Sub-system
- Condensate and Feedwater Systems
- Auxiliary Feedwater System
- Steam Generator Blowdown System

The review was performed based on the range of NSSS design parameters developed to support a NSSS power level of 1,690 MWt. The various interface systems were reviewed for the purpose of providing interface information that could be used in the more detailed BOP analyses. The results are summarized below.

<u>Main Steam System</u>

The major components of the Main Steam (MS) System are the steam generator steam safety valves, the steam generator power-operated atmospheric relief valves (ARV), the main steam isolation valves (MSIV) and associated check and bypass valves (Reference VI.2.1.A).

The plant safety analyses for the MUR PU confirm that the safety valve capacity is adequate for overpressure protection. The design requirements for steam generator safety valves (as well as the ARVs and condenser/atmospheric steam dump valves) include a maximum flow capacity of 890,000 lb/hr at 1,085 psig per valve in the event a valve opens and fails to close (References VI.2.1.B and VI.2.2). The maximum safety valve capacity, which defines the small steam line break transient, was reviewed for potential impact by the MUR PU. The MUR PU conditions are bounded by the CLB, no further evaluation is required.

The steam generator ARVs automatically modulate and exhaust to atmosphere whenever the steam line pressure exceeds a predetermined setpoint to minimize safety valve lifting during steam pressure transients. The steam generator ARV set pressure for these operations is between zero-load steam pressure and the setpoint of the lowest-set steam generator safety valve. Since neither of these pressures change for the proposed range of MUR PU, there is no need to change the ARV setpoint. The steam generator ARVs are sized to have a minimum relieving capacity equal to 10 percent of the maximum calculated steam flow at no-load pressure (Reference VI.2.1.B). The steam generator ARVs were reviewed for potential impact by the MUR PU. The MUR PU conditions are bounded by the CLB; therefore, no further evaluation is required.

The MSIVs, in conjunction with the check valves, are located outside the containment and downstream of the main steam safety valves (MSSV) and ARVs. The MSIVs function to prevent the uncontrolled blowdown of more than one steam generator and to minimize the RC System cooldown and containment pressure to within acceptable limits following a main steam line break (MSLB). The check valves prevent flow of steam into the broken loop from the unbroken loop. Since the MUR PU conditions are bounded by the current analyses, the design loads and associated stresses resulting from rapid closure of the MSIVs and check valves during a MSLB remain the same. The MSIVs and check valves were reviewed for potential impact by the MUR PU. The MUR PU conditions are bounded by the CLB; therefore, no further evaluation is required.

The MSIV bypass valves are used to warm the main steam lines and equalize pressure across the MSIVs prior to their opening. The MSIV bypass valves perform their function at no-load and low power conditions where MUR PU has no impact on main steam conditions (i.e., steam flow and steam pressure). Consequently, the MUR PU has no impact on the interface requirements for the MSIV bypass valves.

Steam Dump Sub-system

The plant operability analysis for the MUR PU verified that the current load rejection capability of 40 percent of plant rated electrical load without a reactor trip remains at MUR PU conditions with no changes to the NSSS control systems setpoints and time constants (References VI.2.1.B and VI.2.3).

PINGP Units 1 and 2 each have four atmospheric dump valves and one condenser dump valve. The current steam dump capacity analysis demonstrates that the steam dump sub-system has a minimum capacity of 35.6 percent total steam flow at the proposed range of NSSS design parameters for the MUR PU with no changes to the NSSS control systems setpoints and time constants. (Reference VI.2.3). Therefore the current load rejection capability of 40 percent of plant rated electrical load without a reactor trip could still be achieved at MUR PU conditions.

Condensate and Feedwater Systems

The major components of the Condensate (CD) and Feedwater (FW) Systems that were evaluated were the feedwater control valves (FCV), feedwater bypass control valves (FBCV), and the condensate and feedwater pumps (Reference VI.2.1.C). These components were evaluated using the NSSS design parameters for the MUR PU.

Redundant feedwater isolation is accomplished by closing the FCVs and the bypass FCVs in conjunction with the main feedwater pump discharge isolation valves and tripping the main feedwater pumps. Isolation of feedwater flow is required to prevent containment overpressurization and excessive RC System cooldowns. To accomplish this function the FCVs, the bypass FCVs and the main feedwater pump discharge isolation valves must be capable of quick closure, following the receipt of any feedwater isolation signal. The most severe conditions occur following a steam line break from no-load conditions with the conservative assumption that all feedwater pumps are in service providing maximum flow following the break. The design loads and associated stresses resulting from rapid closure of these valves were reviewed for potential impact by the MUR PU. The MUR PU conditions (feedwater flow and density) are bounded by the CLB; therefore, no further evaluation is required.

The condensate and feedwater pumps' available head, in conjunction with the FCV characteristics, must provide sufficient margin for feedwater control to ensure adequate flow to the steam generators during steady-state and transient operation. Based on field data, the PINGP Units 1 and 2 full-load FCV valve average lift is 78 percent and 75.2 percent, respectively, at the current non-uprated full-load conditions. Westinghouse recommends that FCV lift be less than 85 percent at full load. Accordingly, the current lift of the FCVs at full load is within the Westinghouse guideline (Reference VI.2.2).

For the range of NSSS design parameters approved for MUR PU, the average FCV position on Units 1 and 2 is expected to increase about three percent which results in full-load lift on both Units of less than 85 percent (References VI.2.4 and VI.2.6). Accordingly, the FCV lift is judged to be acceptable at full load for both steady-state and transient feedwater control.

To provide effective control of flow during normal operation, the FCVs are required to stroke open or closed within 20 seconds over the anticipated inlet pressure control range of approximately 0 to 1,500 psig (Reference VI.2.2). The MUR PU conditions are bounded by the CLB; therefore, no further evaluation is required.

Auxiliary Feedwater System

The Auxiliary Feedwater (AF) System (Reference VI.2.1.C) supplies feedwater to the secondary side of the steam generators when the normal feedwater system is not available, thereby maintaining the heat sink of the steam generators. The system provides feedwater to the steam generators during normal Unit startup, hot standby, and cooldown operations, and also functions as an engineered safeguards system. In the latter function, the AF System is directly relied upon to prevent core damage and system overpressurization in the event of transients and accidents such as a loss of normal feedwater or a secondary system pipe break.

The minimum flow requirements of the AF System at MUR PU conditions are bounded by current safety analyses, and the results of the plant safety analyses confirm that the current AF System performance is acceptable for the MUR PU. The MUR PU conditions are bounded by the CLB; therefore, no further evaluation is required.

The auxiliary feedwater pumps are normally aligned to take suction from the CSTs. To fulfill the engineered safety features design functions, sufficient condensate must be available during transient or accident conditions to enable the plant to be placed in a safe shutdown condition. An analysis concluded that the current minimum usable CST inventory of 100,000 gallons meets the plant licensing bases for the range of NSSS design parameters approved for MUR PU (References VI.2.4 and VI.2.6). The CSTs were reviewed for potential impact on the auxiliary feedwater pumps by the MUR PU. The MUR PU conditions are bounded by the CLB; therefore, no further evaluation is required.

Steam Generator Blowdown System

The Steam Generator Blowdown (SB) System controls the chemical composition of the steam generator secondary-side water within the specified limits (Reference VI.2.1.D). The SB System also controls the buildup of solids in the steam generator secondary-side tubesheet.

The blowdown flow rates required during plant operation are based on chemistry control and tubesheet sweep requirements to control the buildup of solids. The blowdown flow rate required to control chemistry and the buildup of solids in the steam generators is tied to allowable condenser in-leakage, total dissolved solids in the plant Circulating Water (CW) System, and allowable primary to secondary leakage. The blowdown required to control secondary chemistry and steam generator solids was reviewed for potential impact by the MUR PU. The MUR PU conditions are bounded by the CLB; therefore, no further evaluation is required.

Based on the no-load steam pressure and the minimum full-load steam pressure, the MUR PU will not impact blowdown flow control.

B. Containment systems

Containment Structure and Containment Isolation

No changes to the containment structure or containment isolation are being made as part of the MUR PU. The containment structure and containment isolation components are periodically tested for containment design integrity. There are no changes in the test programs based on a 1.64% power uprate. The containment response for a main steam line break was performed at 1650 MWt with 2% uncertainty. The current loss–of-coolant accident (LOCA) containment integrity analysis is based on 102% of the current licensed power (1650 MWt). Both of these analyses were reviewed for potential impact by the MUR PU. The MUR PU conditions are bounded by the CLB; therefore, no further evaluation is required.

Primary Containment Ventilation System

The Primary Containment Ventilation (ZC) System is designed to remove heat from containment during normal plant operation and during and following design basis events by removing heat from the reactor containment following a loss-of-coolant (LOCA) or main steam line break (MSLB) accident. The ZC System, working in parallel with the Containment Spray (CS) System, is designed to remove sufficient heat from the reactor containment to keep the containment from exceeding its design pressure and temperature. The ZC System removes the normal heat loss from equipment and piping in the reactor containment during normal plant operation including the reactor coolant pumps, control rod drive mechanisms, reactor vessel support pads and reactor cavity cooling sub-systems. None of these sub-systems are required to operate during post accident conditions. The containment dome recirculation sub-system is designed to circulate and mix gases following a LOCA to prevent hydrogen accumulation. The vacuum relief sub-system is provided to protect the reactor containment against excessive differential pressures. Differential pressure conditions (vacuum) may exist inside the containment if containment heat removal capability exceeds the heat inputs during normal or post accident operations (References VI.2.1.D and VI.2.1.E).

The ZC System consists of four fan coil units located in the Reactor Containment. The heat sink for the fan coils is provided by the Auxiliary Building chilled water sub-system or by the CL System. During emergency situations the heat sink for the fan coils is provided by the CL System. The fan coil units are sized such that any three fan coil units will provide adequate heat removal capacity from the Reactor Containment during normal and full-power operation, to maintain interior air temperatures below the maximum allowable temperature of 120°F, and to obtain temperatures below 104°F in accessible areas during hot standby operation. Additional circulating fans provide a positive flow of air to the areas around the CRDMs, the reactor cavity, and reactor coolant pump motors.

The fan coil units are also utilized for emergency cooling under post-accident conditions to depressurize the containment atmosphere to the order of 3 psig and 150°F in the long term post accident. Two fan coil units and one containment spray pump provide sufficient heat removal capability to maintain the post-accident containment pressure and temperature below the design value of 46 psig at 268°F (100% relative humidity). Analysis has shown that the operation of one containment spray pump during the injection phase and the heat removal capability equivalent to a single fan coil unit at maximum fouling conditions in conjunction with passive heat sinks, is sufficient to maintain containment pressure less than design (References VI.2.D and VI.2.E).

Since the current licensing basis LOCA or MSLB inside containment analyses are based on 102% of the current licensed power (1650 MWt), MUR PU conditions are bounded by the CLB; therefore, no further evaluation is required.

C. Safety related cooling water systems

Safety Injection System / Containment Spray System – USAR Section 6

The required volume, duration, and heat rejection capability of the Safety Injection (SI) System and Containment Spray (CS) System flows in the event of a break is determined based on analytical and empirical models that simulate reactor and containment conditions subsequent to the postulated RC System and MS System breaks. As a result of these analyses, the system and component criteria necessary to demonstrate compliance with regulatory requirements at the MUR PU conditions are established. The SI and CS Systems were reviewed for potential impact by the MUR PU. The MUR PU conditions are bounded by the CLB; therefore, no further evaluation is required

Residual Heat Removal System

The MUR PU increases the residual heat generated in the core during normal cooldown, refueling operations, and accident conditions. This provides a higher heat load on the RH System heat exchangers during cooldown and also during refueling outages. The increased heat loads are transferred to the Component Cooling (CC) System and ultimately to the Cooling Water (CL) System.

The current licensing basis analyses (References VI.2.4 and VI.2.5) associated with demonstrating the Technical Specification cooldown time limits are satisfied and are performed based on 102% of the current licensed power level of 1650 MWt. The plant Technical Specifications require that the plant be in cold shutdown (Mode 5) within 30 hours with required equipment for power operation out of service. With both trains of the RH and CC Systems' equipment, cold shutdown can be achieved in 7.8 hours at MUR PU conditions if RH System operation is initiated 4 hours after reactor shutdown. For the worst-case scenario, that is, loss of RCPs coupled with the loss of one residual heat removal pump and one CC pump, cold shutdown will be achieved in 21.2 hours after reactor shutdown if residual heat removal operation is initiated no later than 12 hours after reactor shutdown. (Reference VI.2.5). Therefore, the current analysis of record demonstrates continued compliance with the Technical Specification cooldown time requirements at the MUR PU conditions.

The current analysis of record (Reference VI.2.8) which verifies the Appendix R safe shutdown cooldown requirement to be in cold shutdown within 72 hours after reactor shutdown assumes 102% of the current licensed power of 1650 MWt. The current calculation assumes the cooldown is started within 60 hours after shutdown. The cooldown calculation (Reference VI.2.5) confirms the current Appendix R cooldown analysis (Reference VI.2.8) will continue to bound the power level and plant conditions at MUR PU conditions. Therefore, the current analysis of record demonstrates continued compliance that the plant can be cooled down to cold shutdown conditions within 72 hours provided the cooldown is started within 60 hours after shutdown at MUR PU conditions. At PINGP, there are no time-related repairs that would be required to effect cold shutdown and the associated safe-shutdown equipment would remain in compliance with 10 CFR Part 50, Appendix R requirements.

As shown above, the review of the RH System (Reference VI.2.1.F) for potential impact by MUR PU demonstrates that the RH System will continue to satisfy the plant cooldown requirements at MUR PU conditions and no further evaluation is required.

<u>Component Cooling System – USAR Section 10.4.2</u>

The Component Cooling (CC) System was reviewed for potential impact by the MUR PU. The changes to normal plant operating parameters which impact the CC System are negligible. The accident analyses relevant to the CC System are performed assuming a core thermal power level of at least 102% of the current rated value which bound the MUR PU operating conditions. Since the changes to the CC System resulting from the MUR PU are negligible, and the current accident analyses impacting the CC System are bounding for the post MUR PU accident condition, the CC System design remains bounding for operation under MUR PU conditions and no further evaluation is required.

<u>Cooling Water System – USAR Section 10.4.1</u>

The CL System was reviewed for potential impact by the MUR PU. The review determined that no equipment modifications are required and no new normal or safety related loads are added to the system. MUR PU has a negligible impact, if any, on the normal CL System and component duties and the changes in flow rates and operating limits are within the existing system design. Following a loss-of-offsite-power and/or loss-of-coolant-accident, the CL System provides cooling water directly to the following essential loads:

- Component cooling heat exchangers
- Containment fan coils
- Control room chiller
- Auxiliary feed pumps (suction supply)
- Safeguards traveling screens
- Diesel-driven cooling water pump jacket cooler and pump gear oil cooler
- Unit 1 emergency diesel generators cooling
- Containment cooling system

The safety related heat loads on the CL System are based on 102% of the current licensed core power. Since there is sufficient margin in this system to meet various non-safety related duties and no new loads are imposed under MUR PU conditions, MUR PU conditions will remain bounded by the CLB; therefore, no further evaluation is required.

D. Spent fuel pool storage and cooling

The power level used in the Spent Fuel Pool Cooling (SF) System (Reference VI.2.1G) calculation (Reference VI.2.7) already assumes the decay heat associated with 102% of the current operating power (1650 MWt) and the maximum expected core operating time (30,000 hours). Since the SF System analyses already assume conditions that bound the MUR PU conditions, no further evaluation is required.

E. Radioactive waste systems

Normal annual radiological effluents were evaluated for an uprate to 1,677 MWt. Based on the evaluations performed, the liquid and gaseous radwaste system design will be capable of maintaining normal operational offsite releases and doses within the requirements of 10 CFR 20 and 10 CFR 50, Appendix I (References VI.2.9 and VI.2.10). The effluents will remain bounded by the Final Environmental Study estimates.

The volume of solid waste would not be expected to increase proportionally to the increase in core power. This is because the MUR PU neither appreciably affects installed equipment performance, nor requires changes in system operation. The volume of solid radioactive waste generated by a plant is better correlated to outages rather than operating power levels. The radioactive waste systems (Reference VI.2.1.H) were reviewed for potential impact by the MUR PU. The MUR PU conditions are bounded by the CLB; therefore, no further evaluation is required.

This topic is discussed further in Section VII.5.A.

F. Engineered Safety Features (ESF) heating, ventilation, and air conditioning systems

Safeguards Chilled Water Sub-system

The function of the safeguards chilled water sub-system (Reference VI.2.1.I) is to remove heat from the Control Room, the Auxiliary Building and Turbine Building switchgear, the residual heat removal pits, the Relay Room, the Computer Room, and Event Monitoring Rooms (Train A and Train B). The sub-system functions during both normal plant operations and accident conditions. The sub-system performs an essential function in that it cools critical equipment and is, therefore, Design Class III and safety related.

The safeguards chilled water sub-system was reviewed for potential impact by the MUR PU. The Event Monitoring Rooms experience a minor increased heat load from introduction of the LEFM CheckPlus cabinets (Reference VI.2.11). The safeguards chilled water sub-system has adequate capacity for the increase in the duty for this room. Otherwise, design heat loads for this sub-system are not impacted by MUR PU conditions and therefore, MUR PU conditions are bounded by the CLB and no further evaluation is required.

Control Room Area Ventilation System

The Control Room Area Ventilation (ZN) System (Reference VI.2.1.J) was reviewed for potential impact by the MUR PU. The ZN System isolates the outside atmosphere and recirculates a portion of the Control Room atmosphere through pre-filter absolute charcoal (PAC) filters to maintain the dose to the Control Room Operator less than the limits specified in 10 CFR 50, Appendix A, GDC 19 (Reference VI.2.12) for a design basis accident. The ZN System also maintains the Control Room at suitable temperature conditions for personnel habitability and equipment operability post-accident. Since the accidents affecting the Control Room habitability are based on 102% of the current licensed power of 1650 MWt which bounds the post MUR PU conditions, no further evaluation of the ZN System is required.

Auxiliary Building Special Ventilation System – USAR 10.3.4

The Auxiliary Building Special Ventilation (ZA) System was reviewed for potential impact by the MUR PU. The ZA System is designed to reliably collect significant portions of any potential containment system leakage that might bypass the Shield Building annulus and to cause it to pass through charcoal filters before reaching the environment. The system is also provided to filter any leakage from systems that could recirculate primary coolant during LOCA mitigation.

Component parameters post MUR PU are bounded by the original design equipment ratings, and for accident response, are designed for heat loads associated with power operation at 102% of the current licensed power. Therefore the MUR PU conditions are bounded by the CLB and no further evaluation is required.

Spent Fuel Pool Special Ventilation Sub-system

The spent fuel pool special ventilation sub-system Reference VI.2.1.K) is designed to provide specialized ventilation of the spent fuel pool area in the event that high radiation is detected. This is a safeguards system and complies with the requirements of Proposed Criteria for Nuclear Power Plant Protection Systems IEEE 279-68 (Reference VI.2.13), as well as accepted industry standards for power plant equipment and all applicable state and local codes and regulations.

The completely enclosed spent fuel pool area is normally ventilated and exhausted through roughing and HEPA filters. In the event of high radiation in the pool area, signals from radiation monitors in the normal ventilation exhaust duct isolate and shut down the normal ventilation system and initiate the spent fuel pool special ventilation sub-system. Ventilation is then accomplished via the spent fuel pool special ventilation sub-system which shares the exhaust portion of the containment in-service purge sub-system. The air flow is therefore directed through the redundant roughing, HEPA, and charcoal filters in this system.

The spent fuel pool special ventilation sub-system is actuated automatically by a high radiation signal from one of the radiation monitors (R-25 for train A and R-31 for train B) located in the exhaust ducts of the system. This high radiation signal also automatically shuts down the spent fuel pool normal ventilation sub-system. Since this sub-system is completely redundant, either of the two trains can be tripped manually from the Control Room.

The spent fuel pool special ventilation sub-system was designed to maintain negative pressure with all spent fuel pool enclosure doors closed. During spent fuel handling when the spent fuel pool normal ventilation sub-system operability is required, all enclosure doors are maintained closed except personnel doors. Opening of these doors is acceptable for personnel use providing the doors are not blocked open.

The spent fuel pool special ventilation sub-system was reviewed for potential impact by the MUR PU. An evaluation of the spent fuel pool special ventilation sub-system has determined that there is no power dependent piping or equipment contained in this area. The heat released by the fuel pool is dependent on the pool temperature. The temperature limits on the pool will not change with the MUR PU and as indicated in Section VI.1.D above, there will be no increase in the design heat load released from the pool. Therefore, the MUR PU conditions are bounded by the CLB and no further evaluation is required.

2. References for Section VI

VI.2.1 Prairie Island Nuclear Generating Plant, Update Safety Analysis Report, Rev. 30:

A. Main Steam System, Section 11.7

B. Steam Safety, Relief and Dump Systems, Section 11.4

C. Condensate, Feedwater and Auxiliary Feedwater Systems, Section 11.9

D. Containment Vessel Air Handling, Section 5.2.1.4

E. Containment Air Cooling System, Section 6.3

F. Engineered Safety Features Systems Section 6

G. Spent Fuel Pool Cooling System, Section 10.2.2

H. Plant Radioactive Waste Control Systems, Section 9

I. Plant Cooling System, Section 10.4

J. Control Room Ventilation System, Section 10.3.3

K. Spent Fuel Pool Ventilation Systems, Section 10.3.7

L. Appendix D

M. Steam Generator Tube Rupture, Section 14.5.4

VI.2.2 WCAP-7451, "Steam Systems Design Manual," February 1970.

VI.2.3 Westinghouse Calculation CN-FSE-07-19, Rev 0, "Prairie Island 1 and 2 Steam Dump System Capacity to Support the MUR Uprate Program",

VI.2.4 Prairie Island Nuclear Generating Plant, Calculation ENG-ME-443, "Condensate Storage Tank Sizing" Rev 4, October 7, 2005

VI.2.5 Westinghouse Calculation, CN-FSE-07-27, "Prairie Island 1 & 2 MUR Cooldown Analysis", Rev 1, June 27, 2007.

VI.2.6 Westinghouse Calculation, CN-SEE-07-10, "Prairie Island 1 & 2 MUR Program Condensate Storage Tank Required Storage Volume, Rev. 1

VI.2.7 Prairie Island Nuclear Generating Plant, Calculation ENG-ME-476, "Spent Fuel Pool Thermal Heat Load With Core Off-load Starting at 50 hours" Rev 0, Addenda 1, July 27, 2004

VI.2.8 Prairie Island Nuclear Generating Plant, Calculation CF.PX.00.OPS.009, "Appendix R, RHR Cooldown Calculations", March 19, 2001

VI.2.9 Code of Federal Regulations, 10 CFR 20, "Standards for Protection Against Radiation", November 16, 2005

VI.2.10 Code of Federal Regulations, 10 CFR 50, Appendix I, "Numerical Guides for Design Objectives and Limiting Conditions for Operation to Meet the Criterion 'As Low as is Reasonable Achievable' for Radioactive Material in Light-Water-Cooled Nuclear Power Reactor Effluents", August 28, 2007

VI.2.11 Prairie Island Nuclear Generating Plant, Calculation ENG-ME-278, "Loss of Safeguards Chilled Water Room Heat-Up Calculation", Rev 2, July 25, 2007

VI.2.12 General Design Criterion GDC-19, "Control Room," of Appendix A, "General Design Criteria for Nuclear Power Plants, August 28, 2007

VI.2.13 Institute of Electrical and Electronics Engineers, IEEE 279-68, "Proposed Criteria for Nuclear Power Plant Protection Systems", 1968

VII. Other

1. **A statement confirming that the licensee has identified and evaluated operator actions that are sensitive to the power uprate, including any effects of the power uprate on the time available for operator actions.**

   A review of the Operator actions and time available for Operator actions has been performed. The review determined engineered safety features system design, setpoints, and Emergency and Abnormal procedural requirements are based on 1683 MWt (102% of the current licensed power of 1650 MWt) which bounds the proposed MUR PU. Therefore, Operator actions and the time available for Operator actions to be completed will be unaffected by the MUR PU.

2. **A statement confirming that the licensee has identified all modifications associated with the proposed power uprate, with respect to the following aspects of plant operations that are necessary to ensure that changes in operator actions do not adversely affect defense in depth or safety margins:**

A. Emergency and abnormal operating procedures

The modifications (Reference VII.6.1) associated with the implementation of the MUR PU will be limited to re-scaling the plant control and protection instrumentation consistent with the increase in 100% nominal core power from 1650 MWt to 1677 MWt. The current Reactor Protection or ESFAS system setpoints do not require change for MUR. Emergency and Abnormal Operating Procedures that are power dependent will be reviewed, as appropriate, as part of MUR implementation effort.

A Control Room Alarm Response Procedure currently exists (Reference VII.6.2) that contains the specific actions to take in the event the LEFM status changes from Normal to either ALERT or FAIL status. This procedure will be revised to incorporate the administrative restrictions for allowable plant operating power level based on the actual LEFM status discussed earlier in Sections I.1.D, I.1.G, and I.1.H of this enclosure.

B. Control room controls, displays (including the safety parameter display system) and alarms

The PINGP ERCS serves as the plant computer system as well as the safety parameter display system (SPDS). The LEFM is not relied upon for any emergency procedure actions; therefore, there are no changes to the SPDS displays. The LEFM is connected to the plant ERCS through a server-based LEFM/ERCS interface program to allow the data from the LEFM to be stored in the ERCS data base. Once in the ERCS data base, the LEFM information is available for use for displays, trending, and various program uses as necessary. ERCS displays were added to allow monitoring of LEFM status.

As described earlier in Section I.1.C and I.1.D of this enclosure, the ERCS Thermal Power Monitor and CALM programs and displays were revised as part of the LEFM installation on each Unit (References VII.6.3 and VII.6.4). The CALM program was revised to perform and display parallel calorimetric calculations VCALM and LCALM. The results of the VCALM and LCALM are both displayed on the ERCS Calorimetric display screens as well as the calorimetric source selected by the Operator.

The TPM program displays were revised to display the current LEFM status, allowable power based on LEFM status, and the current power levels determined by both the VCALM and LCALM calorimetric calculations. In addition, the TPM display was revised to allow the Operator to select the calorimetric source for display on the TPM monitor and provide alarm indication (red allowable power trend) when power is above the allowable power based on LEFM status. As part of the MUR PU implementation modification, the changes to these programs will be limited to adjusting the allowable licensed thermal power values used in the programs. Also changes to power dependent programs such as Xenon and Samarium will be revised with the allowable thermal power.

As part of the MUR PU implementation modification, current alarms will be evaluated and recalibrated as necessary to reflect small process condition changes in some BOP systems. However, it is not anticipated that any existing alarms will be modified or deleted and no setpoint changes are anticipated. Also, the Operator response time to existing alarms will remain the same.

C. The control room plant reference simulator

As part of the MUR PU implementation modification, the Simulator Calorimetric and TPM programs will be revised with the new administrative power limits based on LEFM status.

D. The operator training program

As part of the MUR PU implementation modification, required changes to plant procedures and alarm responses will be included in the Operator Training Program. In addition, training will be provided covering the implementation of the allowable at-power administrative limits and new TRM governing LEFM out-of-service time.

3. A statement confirming licensee intent to complete the modifications identified in Item 2 above (including the training of operators), prior to implementation of the power uprate

As part of the MUR PU implementation modification, PINGP plans to complete the following actions prior to implementation of the MUR PU:

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

ii. 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 discussed earlier in Sections I.1.D, I.1.G, and I.1.H of this enclosure.

iii. Revision of the ERCS Thermal Power Monitor and CALM Programs to adjust the allowable licensed thermal power values used in these programs. Alarms will require evaluation and re-calibration as necessary to reflect small process changes in some BOP systems. In addition, other core power dependent ERCS programs such as Xenon, NIS Power, and Boron Concentration will be revised to include the revised core power of 1677 MWt.

iv. Revision of the Simulator Calorimetric and TPM 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 include the revised core power of 1677 MWt.

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

vi. Operator training to address the implementation of the allowable at-power administrative limits and new TRM governing LEFM out-of-service time.

4. A statement confirming licensee intent to revise existing plant operating procedures related to temporary operation above "full steady-state licensed power levels" to reduce the magnitude of the allowed deviation from the licensed power level. The magnitude should be reduced from the pre-power uprate value of 2 percent to a lower value corresponding to the uncertainty in power level credited by the proposed power uprate application.

PINGP will revise the existing plant operating procedures to limit plant power to less than or equal to the new rated power level of 1677 MWt. Additionally, plant operating procedures will be revised to specify power reductions from the licensed power level based on LEFM status as previously stated in Section VII.3, for these modifications. As part of the TPM program changes, the time greater than 100% power incremental monitoring levels will be reduced with a maximum power based on a power measurement uncertainty of 0.36%.

5. A discussion of the 10 CFR 51.22 criteria for categorical exclusion for environmental review:

10 CFR 51.22(c)(9) provides criteria for, and identification of, licensing and regulatory actions eligible for categorical exclusion from performing an environmental assessment. A proposed amendment to an operating license for a facility requires no environmental assessment if operation of the facility in accordance with the proposed amendment would not: (i) involve a significant hazards consideration, (ii) result in a significant change in the types or significant increase in the amounts of any effluents that may be released offsite, or (iii) result in a significant increase in individual or cumulative occupational radiation exposure.

The response to these criteria is as follows:

(i) The proposed license amendment does not involve a significant hazards consideration as previously evaluated in Section 4.3 of Enclosure 1.

(ii) Section VII.5.A addresses this criterion as follows:

A. A discussion of the effect of the power uprate on the types or amounts of any effluents that may be released offsite and whether or not this effect is bounded by the final environmental statement and previous Environmental Assessments for the plant.

A review considering the operating license, the requirements of the National Pollution Discharge Elimination System (NPDES) and State Disposal System Permit MN0004006 (Reference VII.6.5), and the information contained in the Final Environmental Statement (FES) (Reference VII.6.6) was performed. The FES assessed a design power level of 1,721 MWt, which envelopes the proposed MUR PU increase. Effluents from the plant that could change as a result of the MUR PU are thermal discharges and radiological effluents. Although increases in discharge amounts associated with the proposed power uprate are possible, they will remain within acceptable limits. Annual radiological discharges will continue to be a small percentage of the allowable limits and the FES estimates. The effluents are described below.

<u>Thermal Discharge</u>

The MUR PU is expected to have only a slight impact on thermal discharge to the river system. Historically, the NPDES permit thermal limits have been reached infrequently during extreme summer conditions, requiring down-powering. The number of days during which NPDES permit limits are reached is not expected to increase due to the MUR PU.

In the thermal performance modeling performed, the additional thermal discharge as a result of the MUR PU did not result in any additional days where the downstream river daily average temperature limit of 86°F was exceeded. The additional thermal discharge as a result of the MUR PU did not result in any months where the 5°F monthly average river temperature rise limit was exceeded. Furthermore, there is no projected increase in monthly average river temperature rise as a result of the MUR PU.

The thermal performance modeling results did indicate minor increases in discharge canal temperatures. However, based on the model, the daily average temperature increases only resulted in one additional day where the discharge canal temperature guideline of 95°F might be exceeded by less than 1°F. In any event, plant procedures would continue to ensure compliance with discharge limits as they direct circulating water system operation to prevent exceeding the 95°F discharge canal temperature.

Based on this small predicted temperature rise, the overall impact on the discharge canal exit temperature is expected to be negligible due to the MUR PU.

<u>River Intake Limits and Circulating Water/Cooling Tower Flow Capability</u>

The thermal performance modeling was based on current permit flow limits and circulating water and cooling tower system capabilities. The river intake limits given in the current NDPES permit will allow for sustained MUR PU operations without the need for significant down powering in excess of current operational limitations.

<u>Liquid, Gaseous, and Solid Radiological Waste</u>

Normal annual radiological effluents were evaluated for an uprate to 1,677 MWt. Based on the evaluations performed, the liquid and gaseous radwaste system design will be capable of maintaining normal operational offsite releases and doses within the requirements of 10 CFR 20 (Reference VII.6.7) and 10 CFR 50, Appendix I (Reference VII.6.8). The effluents will remain bounded by the FES estimates.

For an existing facility that is undergoing a power uprate, the volume of solid waste would not be expected to increase proportionally to the increase in core power. This is because the MUR PU neither appreciably affects installed equipment performance, nor requires changes in system operation. The volume of solid radioactive waste generated by a plant is better correlated to outages rather than operating power levels. Therefore, the implementation of the MUR PU will not have a significant impact on solid radioactive waste generation.

The liquid, gaseous, and solid radiological waste effluents that may be released offsite are bounded by the Final Environmental Statement and previous Environmental Assessments for the plant. Since these MUR PU conditions are bounded by the CLB, no further evaluation is required.

(iii) Section VII.5.B addresses this criterion as follows:

B. A discussion of the effect of the power uprate on individual or cumulative occupational radiation exposure

The current licensed core thermal power for PINGP Units 1 and 2 is 1,650 MWt. Analyses were performed for operation of the Units up to a core power of 1,683 MWt (2% above the current 1,650 MWt core power). This value bounds operation under MUR PU conditions and includes an allowance for calorimetric uncertainty.

A bounding 2% increase will not require modification of existing radiation shielding. Operational radiation exposure is currently significantly less than the regulatory limits. In part, this is an indication of the conservatism in the radiation shielding design. Furthermore, operational radiation exposure is controlled through the plant as-low-as-reasonably-achievable (Radiation Protection ALARA Program), which includes provisions for controlling access to areas with elevated radiation fields and the use of supplemental shielding on a case-by-case basis.

As shown in NUREG-0713 (Reference VII.6.9), in 2004, the average dose per exposed worker at PINGP Units 1 and 2 was approximately 120 mrem (normal operations and outages). This is < 3% of the limit (5 rem) allowed by 10 CFR 20. Therefore, any increase in occupational exposure associated with MUR PU will continue to remain well below the 10 CFR 20 limit. Individual worker exposure will continue to be maintained within acceptable limits by the PINGP ALARA program, which controls access to radiation areas.

The individual or cumulative occupational radiation exposure was reviewed for potential impact by the MUR PU. The MUR PU conditions are bounded by the CLB; therefore, no further evaluation is required.

Additionally, the MUR PU will not impact the existing vital area access missions, and does not impose new access requirements. Operator exposure while performing vital access functions at MUR-PU conditions will remain within the dose guidance of NUREG-0737, II.B.2.

Based on the above, it has been determined that this license amendment request meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment needs to be prepared in conjunction with the issuance of the proposed license amendment.

6. References for Section VII:

VII.6.1 Prairie Island Nuclear Generating Plant, Engineering Change, EC 547, "Implementation of the PINGP Unit 1 and Unit 2 Measurement Uncertainty Recapture (MUR) Uprate (05FW02, Part B)"

VII.6.2 Prairie Island Nuclear Generating Plant Alarm Response Procedure C47041, Rev 7, October 14, 2008

VII.6.3 Prairie Island Nuclear Generating Plant, Engineering Change, EC 1009, "Install and Commission the PINGP- Unit 1 Main Feedwater Ultrasonic Leading Edge Flow Meters"

VII.6.4 Prairie Island Nuclear Generating Plant, Engineering Change, EC 390, "Install and Commission the PINGP-Unit 2 Main Feedwater Ultrasonic Leading Edge Flow Meters"

VII.6.5 NPDES Permit Number MN 0004006

VII.6.6 Final Environmental Statement By The United States Atomic Energy Commission Directorate of Licensing Related To The Proposed Issuance of An Operating License For The Prairie Island Nuclear Generating Plant By The Northern States Power Company Docket Nos. 50-282 and 50 306, May 1973.

VII.6.7 Code of Federal Regulations 10 CFR 20, Standards for Protection Against Radiation

VII.6.8 Code of Federal Regulations, 10 CFR 50, Appendix I, Numerical Guides for Design Objectives and Limiting Conditions for Operation to Meet the Criterion "As Low as is Reasonable Achievable" for Radioactive Material in Light-Water-Cooled Nuclear Power Reactor Effluents

VII.6.9 Nuclear Regulatory Commission, NUREG-0713, "Occupational Radiation Exposure at Commercial Nuclear Power Reactors and Other Facilities", 2006

VIII. Changes to technical specifications, protection system settings, and emergency system settings

1. **A detailed discussion of each change to the plant's technical specifications, protection system settings, and/or emergency system settings needed to support the power uprate:**

   A. **A description of the change**

   B. **Identification of analyses affected by and/or supporting the change**

   C. **Justification for the change, including the type of information discussed in Section III above, for any analyses that support and/or are affected by the change**

   <u>Technical Specification Changes</u>

   All Technical Specification changes are described in detail in Enclosure 1, Section 2.1.

   <u>Reactor Trip System (RTS) and Engineered Safety Feature Actuation System (ESFAS) Setting Changes</u>

   Review of the RTS and ESFAS functions due to an increase in nominal core power and fluid condition changes associated with a 1.64% MUR PU determined the errors associated with the RTS and ESFAS functions remain unchanged by the MUR PU process. The review also determined that all RTS and ESFAS nominal trip setpoints and allowable values remain acceptable for operation at post MUR PU conditions. Therefore, it is not necessary to change RTS or ESFAS nominal trip setpoints and allowable values in the operating plant or in Technical Specifications due to the MUR PU.

   The plant's Technical Specifications, protection system settings, and/or emergency system settings were reviewed for potential impact by the MUR PU. The MUR PU conditions are bounded by the CLB; therefore, no further evaluation is required.

2. **References for Section VIII:**

   None

Enclosure 3

To

Letter from Mark A. Schimmel (NSPM)
To

Document Control Desk (NRC)

Affidavits of Withholding Pursuant to 10 CFR 2.390, Cameron International Corporation

8 pages follow

Measurement Systems

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CAMERON

September 10, 2009
CAW 09-05

Document Control Desk
U. S. Nuclear Regulatory Commission
Washington, DC 20555

APPLICATION FOR WITHHOLDING PROPRIETARY INFORMATION FROM PUBLIC DISCLOSURE

Subject:
1. Caldon® Ultrasonics Engineering Report ER-532 Rev. 1 "Bounding Uncertainty Analysis for Thermal Power Determination at Prairie Island Unit 1 Using the LEFM✓ + System"

2. Caldon® Ultrasonics Engineering Report No. ER-583 Rev. 0, "LEFM✓ + Meter Factor Calculation and Accuracy Assessment for Prairie Island Unit 1 Nuclear Power Station"

3. Caldon® Ultrasonics Engineering Report ER-533 Rev. 2 "Bounding Uncertainty Analysis for Thermal Power Determination at Prairie Island Unit 2 Using the LEFM✓ + System"

4. Caldon® Ultrasonics Engineering Report No. ER-553 Rev. 2, "LEFM✓ + Meter Factor Calculation and Accuracy Assessment for Prairie Island Unit 2 Nuclear Power Station"

Gentlemen:

This application for withholding is submitted by Cameron International Corporation, a Delaware Corporation (herein called "Cameron") on behalf of its operating unit, Caldon Ultrasonics Technology Center, pursuant to the provisions of paragraph (b)(1) of Section 2.390 of the Commission's regulations. It contains trade secrets and/or commercial information proprietary to Cameron and customarily held in confidence.

The proprietary information for which withholding is being requested is identified in the subject submittal. In conformance with 10 CFR Section 2.390, Affidavit CAW 09-05 accompanies this application for withholding setting forth the basis on which the identified proprietary information may be withheld from public disclosure.

Accordingly, it is respectfully requested that the subject information, which is proprietary to Cameron, be withheld from public disclosure in accordance with 10 CFR Section 2.390 of the Commission's regulations.

Correspondence with respect to this application for withholding or the accompanying affidavit should reference CAW 09-05 and should be addressed to the undersigned.

Very truly yours,

CR Hastings

Calvin R. Hastings
General Manager

Enclosures (Only upon separation of the enclosed confidential material should this letter and affidavit be released.)

September 10, 2009
CAW 09-05

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

ss

COUNTY OF ALLEGHENY:

Before me, the undersigned authority, personally appeared Calvin R. Hastings, who, being by me duly sworn according to law, deposes and says that he is authorized to execute this Affidavit on behalf of Cameron International Corporation, a Delaware Corporation (herein called "Cameron") on behalf of its operating unit, Caldon Ultrasonics Technology Center, and that the averments of fact set forth in this Affidavit are true and correct to the best of his knowledge, information, and belief.

CR Hastings
Calvin R. Hastings
General Manager

Sworn to and subscribed before me

this 10th day of

September, 2009

Joann B Thomas
Notary Public

COMMONWEALTH OF PENNSYLVANIA
Notarial Seal
Joann B. Thomas, Notary Public
Findlay Twp., Allegheny County
My Commission Expires July 28, 2011
Member, Pennsylvania Association of Notaries

1. I am the General Manager of Caldon Ultrasonics Technology Center, and as such, I have been specifically delegated the function of reviewing the proprietary information sought to be withheld from public disclosure in connection with nuclear power plant licensing and rulemaking proceedings, and am authorized to apply for its withholding on behalf of Cameron.

2. I am making this Affidavit in conformance with the provisions of 10 CFR Section 2.390 of the Commission's regulations and in conjunction with the Cameron application for withholding accompanying this Affidavit.

3. I have personal knowledge of the criteria and procedures utilized by Cameron in designating information as a trade secret, privileged or as confidential commercial or financial information. The material and information provided herewith is so designated by Cameron, in accordance with those criteria and procedures, for the reasons set forth below.

4. Pursuant to the provisions of paragraph (b) (4) of Section 2.390 of the Commission's regulations, the following is furnished for consideration by the Commission in determining whether the information sought to be withheld from public disclosure should be withheld.

   (i) The information sought to be withheld from public disclosure is owned and has been held in confidence by Cameron.

   (ii) The information is of a type customarily held in confidence by Cameron and not customarily disclosed to the public. Cameron has a rational basis for determining the types of information customarily held in confidence by it and, in that connection utilizes a system to determine when and whether to hold certain types of information in confidence. The application of that system and the substance of that system constitutes Cameron policy and provides the rational basis required. Furthermore, the information is submitted voluntarily and need not rely on the evaluation of any rational basis.

Under that system, information is held in confidence if it falls in one or more of several types, the release of which might result in the loss of an existing or potential advantage, as follows:

(a) The information reveals the distinguishing aspects of a process (or component, structure, tool, method, etc.) where prevention of its use by any of Cameron's competitors without license from Cameron constitutes a competitive economic advantage over other companies.

(b) It consists of supporting data, including test data, relative to a process (or component, structure, tool, method, etc.), the application of which data secures a competitive economic advantage, e.g., by optimization or improved marketability.

(c) Its use by a competitor would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, and assurance of quality, or licensing a similar product.

(d) It reveals cost or price information, production capacities, budget levels, or commercial strategies of Cameron, its customer or suppliers.

(e) It reveals aspects of past, present or future Cameron or customer funded development plans and programs of potential customer value to Cameron.

(f) It contains patentable ideas, for which patent protection may be desirable.

There are sound policy reasons behind the Cameron system, which include the following:

(a) The use of such information by Cameron gives Cameron a competitive advantage over its competitors. It is, therefore, withheld from disclosure to protect the Cameron competitive position.

(b) It is information that is marketable in many ways. The extent to which such information is available to competitors diminishes the Cameron ability to sell products or services involving the use of the information.

(c) Use by our competitor would put Cameron at a competitive disadvantage by reducing his expenditure of resources at our expense.

(d) Each component of proprietary information pertinent to a particular competitive advantage is potentially as valuable as the total competitive advantage. If competitors acquire components of proprietary information, any one component may be the key to the entire puzzle, thereby depriving Cameron of a competitive advantage.

(e) Unrestricted disclosure would jeopardize the position of prominence of Cameron in the world market, and thereby give a market advantage to the competition of those countries.

(f) The Cameron capacity to invest corporate assets in research and development depends upon the success in obtaining and maintaining a competitive advantage.

(iii) The information is being transmitted to the Commission in confidence, and, under the provisions of 10 CFR §§ 2. 390, it is to be received in confidence by the Commission.

(iv) The information sought to be protected is not available in public sources or available information has not been previously employed in the same manner or method to the best of our knowledge and belief.

(v) The proprietary information sought to be withheld are the submittals titled:

- Caldon® Ultrasonics Engineering Report ER-532 Rev. 1 "Bounding Uncertainty Analysis for Thermal Power Determination at Prairie Island Unit 1 Using the LEFM✓ + System"

- Caldon® Ultrasonics Engineering Report No. ER-583 Rev. 0, "LEFM✓ + Meter Factor Calculation and Accuracy Assessment for Prairie Island Unit 1 Nuclear Power Station"

- Caldon® Ultrasonics Engineering Report ER-533 Rev. 2 "Bounding Uncertainty Analysis for Thermal Power Determination at Prairie Island Unit 2 Using the LEFM✓ + System"

- Caldon® Ultrasonics Engineering Report No. ER-553 Rev. 2, "LEFM✓ + Meter Factor Calculation and Accuracy Assessment for Prairie Island Unit 2 Nuclear Power Station"

It is designated therein in accordance with 10 CFR §§ 2.390(b)(1)(i)(A,B), with the reason(s) for confidential treatment noted in the submittal and further described in this affidavit. This information is voluntarily submitted for use by the NRC Staff in their review of the accuracy assessment of the proposed methodology for LEFM CheckPlus Systems used by Prairie Island Nuclear Power Stations 1 and 2 for an MUR UPRATE.

Public disclosure of this proprietary information is likely to cause substantial harm to the competitive position of Cameron because it would enhance the ability of competitors to provide similar flow and temperature measurement systems and licensing defense services for commercial power reactors without commensurate expenses. Also, public disclosure of the information would enable others to use the information to meet NRC requirements for licensing documentation without the right to use the information.

The development of the technology described in part by the information is the result of applying the results of many years of experience in an intensive Cameron effort and the expenditure of a considerable sum of money.

In order for competitors of Cameron to duplicate this information, similar products would have to be developed, similar technical programs would have to be performed, and a significant manpower effort, having the requisite talent and experience, would have to be expended for developing analytical methods and receiving NRC approval for those methods.

Further the deponent sayeth not.