June 23, 2004

Mr. Daniel J. Malone
Site Vice President
Palisades Nuclear Plant
27780 Blue Star Memorial Highway
Covert, MI 49043

SUBJECT: PALISADES PLANT - ISSUANCE OF AMENDMENT REGARDING MEASUREMENT UNCERTAINTY RECAPTURE POWER UPRATE (TAC NO. MB9469)

Dear Mr. Malone:

The U.S. Nuclear Regulatory Commission (Commission) has issued the enclosed Amendment No. 216 to Facility Operating License No. DPR-20 for the Palisades Plant. This amendment revises the operating license and technical specifications (TSs) in response to your application dated June 3, 2003, as supplemented by letters dated October 6, 2003, January 15, and February 13, 2004.

The amendment revises the operating license and TSs to increase the licensed rated thermal power by 1.4 percent from 2530 megawatts thermal (MWt) to 2565.4 MWt using measurement uncertainty recapture.

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

Sincerely,

/RA/

John F. Stang, Senior Project Manager
Project Directorate III, Section 1
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-255

Enclosures: 1. Amendment No. 216 to License No. DPR-20
2. Safety Evaluation

cc w/encls: See next page

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ADAMS Accession No. ML040970622 (Cover Letter and Amendment)
ADAMS Accession No. ML041550116 (License and TS Pages)
ADAMS Accession No. ML040970623 (Package)

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NUCLEAR MANAGEMENT COMPANY, LLC

DOCKET NO. 50-255

PALISADES PLANT

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 216
License No. DPR-20

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

   A. The application for amendment by Nuclear Management Company, LLC (NMC or the licensee), dated June 3, 2003, as supplemented by letters dated October 6, 2003, January 15, and February 13, 2004, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-20 is hereby amended to read as follows:

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 216, and the Environmental Protection Plan contained in Appendix B are hereby incorporated in the license. NMC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

In addition, the license is amended to revise paragraph 2.C.(1) to reflect the increase in the reactor core power level. Paragraph 2.C.(1) is hereby amended to read as follows:

> NMC is authorized to operate the facility at steady-state reactor core power levels not in excess of 2565.4 Megawatts thermal (100 percent rated power) in accordance with the conditions specified herein.

3. This license amendment is effective as of the date of its issuance, and is to be implemented within 90 days of the date of issuance. Prior to implementation of the license amendment, NMC shall:

   A. Conduct operator training on the proposed power uprate.

   B. Revise plant procedures to address operation with the Crossflow ultrasonic flow measurement system out of service.

   C. Revise plant procedures to address operation with the plant process computer (PPC) feedwater flow indication or a PPC feedwater temperature indication out of service.

   D. NMC will revise plant procedures to add at least 0.1% power conservative margin to the calculated UFM correction factors (the ratio of UFM measured feedwater flow to venturi measured flow) to establish the final UFM correction factors (values entered into the plant heat balance calculation). This procedure revision will occur prior to implementation of the proposed power uprate.

FOR THE NUCLEAR REGULATORY COMMISSION

William H. Ruland, Director
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Operating License and Technical Specifications

Date of Issuance:

ATTACHMENT TO LICENSE AMENDMENT NO. 216

FACILITY OPERATING LICENSE NO. DPR-20

DOCKET NO. 50-255

Replace the following page of Facility Operating License No. DPR-20 with the attached revised page. The revised page is identified by amendment number and contains marginal lines indicating the areas of change.

REMOVE	INSERT
3	3

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

REMOVE	INSERT
1.1-5	1.1-5
3.3.1-6	3.3.1-6

Palisades Plant

Safety Evaluation for Amendment No. 216

Measurement Uncertainty Recapture Power Uprate

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RELATING TO AMENDMENT NO. 216 TO FACILITY OPERATING LICENSE NO. DPR-20

NUCLEAR MANAGEMENT COMPANY, LLC

PALISADES PLANT

DOCKET NO. 50-255

1.0 INTRODUCTION

By application to the U.S. Nuclear Regulatory Commission (NRC) dated June 3, 2003, as supplemented by letters dated October 6, 2003, and January 15, 2004, the Nuclear Management Company, LLC (NMC or the licensee) requested an amendment to the Facility Operating License and the Technical Specifications (TSs) for the Palisades Plant (Palisades or the facility). The proposed amendment would increase the licensed reactor core power level by 1.4 percent from 2530 megawatts thermal (MWt) to 2565.4 MWt. The proposed increase is considered a measurement uncertainty recapture (MUR) power uprate. The licensee's request is based on the reduced reactor thermal power measurement uncertainty provided by the installation and use of an Ultrasonic Flow Measurement Device (UFMD) consisting of an Ultrasonic Flow Measurement (UFM) system called "Crossflow[1]."

Specifically, the proposed changes would revise:

1. Paragraph 2.C.(1) of the operating license, DPR-20, to authorize operation at steady-state reactor core power levels not in excess of 2565.4 MWt.

2. TS 1.1, "Definitions," to change RATED THERMAL POWER (RTP) to reflect the increase from 2530 MWt to 2565.4 MWt.

3. TS Table 3.3.1-1, "Reactor Protective System Instrumentation," Item 1, "Variable High Power Trip," to change the maximum allowable value from 111 percent to 109.4 percent.

The licensee's June 3, 2003, application for license amendment relies, in part, upon the approval of a separate application, dated October 17, 2002 (ADAMS Accession No. ML023020583), addressing a related change in a constant in the variable thermal

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[1] The UMF system at Palisades is described in the Combustion Engineering topical report CENPD-397-P, "Improved Flow Measurement Accuracy Using Crossflow Ultrasonic Flow Measurement Technology," Revision 01, dated May 2000. By letter dated March 20, 2000, the NRC staff approved this report for referencing, subject to certain requirements in addition to those specified in the topical report.

margin/low pressure trip equation. This submittal is addressed separately by Amendment No. 214, dated January 8, 2004 (ADAMS Accession No. ML032541030). The October 6, 2003, January 15, and February 13, 2004, supplemental letters, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the Nuclear Regulatory Commission (NRC) staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on July 8, 2003 (68 FR 40714).

2.0 BACKGROUND

Nuclear power plants are licensed to operate at a specified core thermal power. Title 10 of the *Code of Federal Regulations* (10 CFR), Part 50, Appendix K, requires licensees to assume that the reactor has been operating continuously at a power level at least 1.02 times the licensed power level when performing loss-of-coolant accident (LOCA) and emergency core cooling system (ECCS) analyses. This requirement is included to ensure that instrumentation uncertainties are adequately accounted for in the analyses. Appendix K to 10 CFR Part 50 allows licensees to assume a power level lower than 1.02 times the licensed power level (but not less than the licensed power level), provided the licensee has demonstrated that the proposed value adequately accounts for instrumentation uncertainties. The licensee has proposed to use a value of 0.5925 percent. To achieve this level of accuracy, the licensee previously installed a Combustion Engineering Crossflow ultrasonic flow measurement system (Crossflow system) for measuring the main feedwater (FW) flow at Palisades. The Crossflow system provides a more accurate measurement of FW flow than the FW flow measurement accuracy assumed during the development of the original Appendix K requirements and that of the FW flow venturis currently used to calculate reactor thermal output. The Crossflow system will measure FW mass flow to within plus or minus (±)0.5 percent for Palisades. This bounding FW mass flow uncertainty was used to calculate a total power measurement uncertainty of ±0.5925 percent. On the basis of this, NMC proposes to reduce the power measurement uncertainty required by Appendix K to 0.5925 percent. The improved power measurement uncertain obviates the need for the 2 percent power margin originally required by Appendix K, thereby allowing an increase in the reactor power available for electrical generation by 1.4 percent.

3.0 EVALUATION

The NRC staff's evaluation of the proposed Palisades MUR power uprate is based on the guidance provided by Regulatory Issue Summary (RIS) 2002-03, "Guidance on the Content of Measurement Uncertainty Recapture Power Applications." RIS 2002-03 delineates the appropriate scope and level of detail for the review and approval of an MUR power uprate application. For every technical area where the proposed MUR power uprate conditions are bounded by existing design and licensing bases analyses, the NRC staff has confirmed that the proposed conditions continue to be bounded.

For situations where the proposed MUR power uprate conditions are not bounded by existing design and licensing bases, the licensee has performed new analyses, and the NRC staff has conducted an independent evaluation.

In several places in this safety evaluation (SE), the NRC staff refers to NUREG-0800, "Standard Review Plan [SRP] for the Review of Safety Analysis Reports for Nuclear Power

Plants LWR Edition," as guidance used during the review. The NRC staff notes that the Standard Review Plan (SRP) was used solely for general technical guidance. The NRC staff reviewed the licensee's application, as supplemented, for compliance with the Palisades licensing basis, not NUREG-0800.

3.1 Instrumentation and Controls

The NRC staff's review in the area of instrumentation and controls covers (1) the proposed plant-specific implementation of the FW flow measurement device, and (2) the power uncertainty calculations (NRC RIS 2002-03, Attachment 1, Section I). The NRC staff's review is conducted to confirm that the licensee's application of CENPD-397-P-A, "Improved Flow Measurement Accuracy Using Crossflow Ultrasonic Flow Measurement Technology," is consistent with the NRC staff's approval of this topical report. The NRC approved topical report CENPD-397 in its SE dated March 20, 2000. This topical report covered the use of the Crossflow UFM system for reducing the uncertainty associated with feewater flow measurement. The NRC staff also reviews the power uncertainty calculations to ensure that (1) the proposed uncertainty value of 0.6 percent correctly accounts for the uncertainties due to power level instrumentation error, and (2) the calculations meet the relevant requirements of Appendix K to 10 CFR Part 50. The NRC staff has also reviewed the licensee's response to a recent concern addressed in a Westinghouse technical bulletin and advisory letter regarding a potential signal interference issue that can adversely affect FW flow measurement.

3.1.1 Regulatory Evaluation

Nuclear power plants are licensed to operate at a specified core thermal power and the uncertainty of the calculated values of this thermal power determines the probability of exceeding the power level assumed in the design-basis transient and accident analyses. In this regard, 10 CFR 50, Appendix K, requires LOCA accident and ECCS analyses to assume that the reactor had operated continuously at a power level at least 102 percent of the licensed thermal power to allow for uncertainties, such as instrument error. The 2 percent power margin was intended to address uncertainties related to heat sources in addition to instrument measurement uncertainties. Later, the NRC concluded that, at the time of the original ECCS rulemaking, the 2 percent power margin requirement appeared to be based solely on considerations associated with power measurement uncertainty. This development could justify a reduced margin between the licensed power level and the power level assumed in the ECCS analysis and, therefore, a power uprate.

In order to reduce an unnecessarily burdensome regulatory requirement and to avoid unnecessary exemption requests, the Commission published the final rule in the June 1, 2000, *Federal Register*, (Volume 65, Number 106, Rules and Regulations, pages 34913-34921). This final rule allows licensees the option of justifying a smaller margin for power measurement uncertainty by using more accurate instrumentation to calculate the reactor thermal power or maintaining the current margin of 2 percent power. Licensees may apply the reduced margin to request a license amendment from the NRC staff authorizing plant operation at a higher power level or use the margin for a license amendment to relax the ECCS-related TS. A license amendment request for power uprate should include a justification for the reduced power measurement uncertainty to support that proposed power uprate.

3.1.2 Technical Evaluation

Neutron flux instrumentation is calibrated to the core thermal power, which is determined by an automatic or manual calculation of the energy balance around the plant nuclear steam supply system (NSSS). This calculation is called a "Secondary Calorimetric" for a pressurized-water reactor (PWR) and a "Heat Balance" for a boiling-water reactor. NMC's submittal named this calculation "Secondary Calorimetric Heat Balance." The accuracy of this calculation depends primarily upon the accuracy of FW flow and main steam (MS) and FW temperature and pressure measurements. FW flow is the most significant contributor to the core thermal power uncertainty. A more accurate measurement of this parameter will result in a more accurate determination of core thermal power and, thereby, a more accurate calibration of the nuclear instrumentation.

The instrumentation used for measuring FW flow is typically an orifice plate, a venturi meter, or a flow nozzle. These devices generate a differential pressure proportional to the FW velocity in the pipe. Of the three differential pressure devices, a venturi meter is most widely used for FW measurement in nuclear power plants. The major advantage of a venturi meter is a relatively low head loss as the fluid passes through the device. The major disadvantage of the device is that the calibration of the flow element shifts when the flow element is fouled, which causes the meter to indicate a higher differential pressure and hence a higher than actual flow rate. This leads the plant operator to calibrate nuclear instrumentation high. Calibrating the nuclear instrumentation high is conservative with respect to the reactor safety, but causes the electrical output to be proportionally low when the plant is operated at its thermal power rating. To eliminate the fouling effects, the flow device has to be removed, cleaned, and recalibrated. Due to the high cost of recalibration and the need to improve flow instrumentation uncertainty, the industry assessed other flow measurement techniques and found the Crossflow UFM to be a viable alternative. The measurement uncertainties due to venturi fouling and instrumentation drift and calibration shifts are essentially eliminated when a Crossflow UFM is used. The crossflow UFM does not replace the currently installed plant venturi, but provides the licensee an in-plant capability for periodically recalibrating the FW venturi to adjust for the effect of fouling. A unique advantage of the Crossflow UFM system is that it is installed external to the pipe in which flow is to be measured, thereby eliminating any possibility of compromising pressure boundary integrity.

The crossflow UFM consists of four ultrasonic transducers mounted on a metal support frame which is clamped on the FW piping. There is one upstream and one downstream transducer station, and each station includes one transmitting and one receiving transducer. The operation of a cross-correlation UFM is based on the fact that an ultrasonic beam traveling across fluid flowing in a pipe is affected (modulated) by the turbulence (eddies) present in the flowing liquid. When this modulated signal is processed, a random signal that is a signature of the flowing eddies can be obtained. The Crossflow UFM calculates the time a unique pattern of eddies take to pass between two sets of ultrasonic transducers and divides the known distance between the two sets of transducers by the calculated time to obtain the flow velocity. This measured velocity is not an average velocity (highest velocity is at the center of the pipe), and therefore, is multiplied by the "Velocity Profile Correction Factor" (VPCF) to obtain the average velocity of the fluid flowing in the pipe.

The Crossflow UFM system consists of a Mounting/Transducer Support Frame with ultrasonic transducers, a signal conditioning unit (SCU), and a data processing computer (DPC). The

DPC receives a FW flow signal from the SCU and FW pressure and temperature input from the plant computer. Using a built-in signal processing algorithm, the Crossflow DPC calculates fluid velocity and converts the fluid velocity to a mass flow using flow, temperature, and pressure as calculation inputs. The Crossflow FW mass flow is periodically compared to the FW venturi mass flow to determine the correction factor that must be applied to the venturi mass flow to obtain the corrected mass flow. This corrected mass flow is used in calculating core thermal power and thereby calibrating nuclear instrumentation in accordance with the plant TS requirements.

The licensee's submittal referenced Westinghouse, formerly ABB Combustion Engineering (ABB-CE), Topical Report CENPD-397-P-A, "Improved Flow Measurement Accuracy Using Crossflow Ultrasonic Flow Measurement Technology" which describes the Crossflow UFM system for the measurement of FW flow and provides a generic basis for the proposed 1.4 percent power uprate. This topical report was approved by the NRC staff in March 2000. The topical report indicates, using Crossflow UFM for FW flow measurement, the UFM is able to achieve an uncertainty of 0.5 percent or better with a 95 percent confidence interval. The topical report provides specific guidelines and equations for determining uncertainty values of the Crossflow input parameters (VPCF, inside diameter, transducer spacing, FW density, and Crossflow time delay). The plant-specific uncertainties are determined when the meter is installed, using the guidelines and equations provided in the topical report. The topical report states that a trained ABB-CE (now Westinghouse) representative installs the hardware and software of the Crossflow UFM.

NMC's submittal included a plant-specific heat balance uncertainty calculation for Palisades. NMC used an NRC staff approved methodology to statistically combine the power measurement uncertainty components to determine the Secondary Side Power Calorimetric uncertainty. The secondary side power calorimetric uncertainties are in four principal areas: FW flow, FW enthalpy, steam enthalpy, and blowdown flow. FW flow measurement uncertainty is the largest contributor to power calorimetric measurement uncertainty. Using the methodology described in Section 5 of Topical Report CENPD-397-P-A, the plant-specific calculation established a value of 0.44 percent measurement uncertainty of actual flow for the installed Crossflow UFM at Palisades. This flow measurement uncertainty was combined with uncertainties associated with the secondary calorimetric parameters to calculate the power measurement uncertainty using Crossflow UFM. At Palisades, the ratio between the UFM measurement and the plant venturi measurement provides a correction factor to the venturi measured FW flow. The licensee calculated the power measurement uncertainties with the corrected and uncorrected FW flow. This calculation indicated that when UFM corrected indicated power is 100 percent of RTP (2565.4 MWt) the actual power could be a maximum of 100.49 percent (2578 MWt) or a minimum of 99.45 percent (2550 MWt). Also, for the uncorrected FW flow (UFM out of service), when the indicated power is 100 percent of the RTP, the actual power could be a maximum of 101.13 percent or a minimum of 98.79 percent. The Crossflow uncertainty calculation supports an uncertainty in the reactor power measurement of less than 0.6 percent (actual calculated 0.49 percent), and thus provides sufficient justification for the proposed 1.4 percent power uprate using Crossflow UFM for FW flow measurement.

By Amendment No. 214 (ADAMS Accession No. ML032541030), dated January 8, 2004, the NRC staff has approved a change to Palisades TS Table 3.3.1-2 to modify a constant in the equation of the setpoint for the variable thermal margin/low pressure (TM/LP) trip. This change reflects the proposed MUR and the decreased uncertainty of the new digital thermal margin monitors. Amendment No. 214 is based upon the licensee's separate application, dated October 17, 2002 (ADAMS Accession No. ML023020583), as supplemented by letter dated December 10, 2003 (ADAMS Accession No. ML033570393).

In its application dated June 3, 2003, the licensee proposes to change the variable high-Power trip (VHPT) setpoint allowable value (AV). Specifically, the AV for the VHPT in TS Table 3.3.1-1 would be changed from the current ≤111 percent RTP to ≤ 109.4 percent RTP. This change reflects the proposed 1.4 percent power uprate from the current 2530 MWt to 2565.4 MWt. The NRC staff requested the licensee to confirm that, except for TM/LP and VHPT, no other trip function AV is affected by the proposed power uprate. The licensee's response stated that none of the other reactor protective system trips were affected by the proposed increase in RTP. The NRC staff also requested the licensee to provide assurance of the adequacy of the proposed AV of VHPT to account for all instrumentation loop uncertainties including those that are not measured during Channel Operational Test (COT). This is to provide sufficient margin to assure that the Analytical Limit is not violated. The licensee responded that the plant VHPT setpoint AV was determined by conservatively combining the COT and non-COT instrumentation loop uncertainties to assure sufficient margin between the AV and AL. The NRC staff found the licensee's response acceptable.

The NRC staff's safety evaluation report on Westinghouse Topical Report CENPD-397-P-A included four additional requirements to be addressed by a licensee referencing this topical report for power uprate. NMC's submittal addressed each of the four requirements as follows:

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

   The licensee stated that maintenance and calibration of the UFM system components are performed using NMC's site control processes. The licensee described its program for the calibration and maintenance of all other instrumentation, in addition to the Crossflow UFM, whose measurement uncertainties affect the plant power calorimetric uncertainties. The program includes controlling software and hardware configuration, reporting deficiencies to the manufacturers, receiving and addressing manufacturer deficiency reports, and performing corrective actions.

   The licensee stated that, at Palisades, the Crossflow system is not connected to the plant process computer (PPC) and does not perform any automatic safety-related or plant control functions. It is used as an offline calibration tool to calibrate the venturi FW flow indication on a monthly interval. Each month the ratio of UFM flow to venturi flow is determined to establish a conservative "UFM correction factor" which is manually entered into the PPC to adjust the venturi FW flow measurement. The licensee stated that the procedure for completing the evaluation is treated like a TS surveillance and, therefore, includes a 25 percent grace period to the monthly interval. If the UFM system is inoperable, then the system is either repaired to operable status in the allowed outage time (AOT) of 31 days, or power is reduced and the PPC flow correction factors are manually reset (i.e., no FW flow correction or credit for UFM calculations). Prior to exceeding the AOT, the reactor power level will be reduced to 2550 MWt, or 99.4 percent of the uprated thermal power. This power level is consistent with the power measurement uncertainty analysis based on FW flow measurement with the venturi instrumentation. Reactor power, plus the uncertainty in reactor power, remains less than the analyzed power level of 2580.6 MWt.

To assure that the UFM provides an accurate measurement of FW flow and input into the plants heat balance calculation, the licensee has revised its procedures. Once the MUR power uprate license amendment has been implemented and the new rated thermal power level of 2565.4 MWt has been established, the new procedure will be in effect. The procedure will require all correction factors associated with UFM to be removed if the reactor power goes below 95 percent of rated thermal power. In addition, a new 100 percent power level of 2550 MWt will then be established. Once the reactor is stablized, the UFM correction factors can be reapplied and 100 percent can be reestablished as 2565.4 MWt. If the PPC becomes inoperable, daily verifications of the UFM FW flow is performed to verify UFM correction factor applicability. In the event that the PPC FW flow indication or PPC FW temperature indication is out of service, then a manual heat balance calculation would be required. The larger uncertainties associated with any of these two conditions will require a 100 percent thermal power value of 2530 MWt in the power calorimetric. NMC will revise plant procedures to address operation with PPC FW flow and temperature indications out of service prior to implementation of the proposed power uprate.

The UFM correction factors have been used at Palisades since 1997 with a bi-weekly surveillance frequency. Since May 2001, the surveillance frequency was changed from bi-weekly to monthly. During this period, only twice was the calculated UFM correction factor found non-conservative with respect to the correction factor applied in the heat balance calculation (approximately 0.01 percent of 2530 MWt which is 0.25 MWt). Prior to implementation of the proposed power uprate, NMC will revise plant procedures to include at least 0.1 percent power conservatism when the UFM correction factors are established for use in the plant heat balance calculation. Based on information in NMC's submittals, the NRC staff concludes that the licensee's plant procedures will reasonably assure instrumentation capability to provide acceptable power calorimetric uncertainty for the proposed power uprate.

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

   At Palisades, the Crossflow system has been in use since 1997 and has been reliably used to provide correction for the venturi fouling to allow operation at 100 percent RTP. During 1999, the UFM transducers and brackets were replaced along with an upgrade of computers and software. NMC stated that the currently installed Crossflow system is representative of the Crossflow UFM described in Topical Report CENPD-397-P-A, Revision 01, and is bounded by the requirements set forth in the topical report. The NRC staff finds this acceptable.

3. The licensee should confirm that the methodology used to calculate the uncertainty of the Crossflow UFM in comparison to the current FW flow instrumentation is based on accepted plant setpoint methodology (with regard to the development of instrument

uncertainty). If an alternate methodology is used, the application should be justified and applied to both the venturi and the UFM for comparison.

The licensee stated that the Crossflow UFM measurement uncertainty calculations are consistent with the methodology described in Topical Report CENPD-397-P-A, Revision 01. These calculations are based on accepted plant instrument uncertainty methodology, which incorporates the aspects of ANSI/ISA-S67.04.01-2000, ”Methodologies for the Determination of Setpoints for Nuclear Safety-Related Instrumentation.” The licensee stated that Crossflow system implementing procedures at Palisades ensure the assumptions and requirements of the uncertainty calculation remain valid. The NRC staff has reviewed the plant-specific calculations and finds them acceptable.

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

   The licensee stated that the Crossflow installation at Palisades is equivalent to known calibration and plant configurations for the specific installation, including the propagation of flow profile effects at higher Reynolds numbers. The velocity profile correction factor is calculated as described in Section 5.6 of Topical Report CENPD-397-P-A, Revision 01. The transducers are installed on straight pipe runs and are far enough from disturbances to conform to the installation requirements of the topical report. The NRC staff finds this acceptable.

The NRC staff finds that NMC’s response adequately addressed and resolved the four additional plant-specific requirements about Crossflow UFM maintenance and calibration, hydraulic configuration, and procedures and contingency plans for an inoperable Crossflow UFM. The licensee used an approved methodology to calculate the plant-specific Crossflow and power calorimetric measurement uncertainties.

Additionally, in a Technical Bulletin, TB-03-6, “Crossflow Ultrasonic Flow Measurement System Signal Issues,” dated September 5, 2003, and in a subsequent Nuclear Safety Advisory Letter, NSAL-03-12, “Crossflow Ultrasonic Flow Measurement System Flow Signal Interference Issues,” dated December 5, 2003 (Accession No. ML033421289), Westinghouse and its Crossflow partner, the Advanced Measurement Analysis Group, Inc. (AMAG), discussed a flow signal interference issue that has the potential to adversely affect the FW flow measurement. Specifically, the bulletin and letter identified a potential concern that:

> Plant mechanical equipment in the FW system in combination with the unique plant-specific acoustic response characteristics of the piping system has the potential to cause flow signal interference that can lead to an incorrect and potentially

non-conservative determination of the venturi flow correction factor. The presence of flow signal interference or correlated noise can result in a bias (shift) in the Crossflow time-delay measurement, which is the time it takes for the eddies within the fluid to pass between the two ultrasonic beams. When the time-delay is biased high, the flow measurement is biased low (non-conservative direction with respect to assessment of plant power level).

Westinghouse/AMAG forwarded the bulletin and letter to all Crossflow users and included a number of recommendations to maintain system uncertainty certification. By letter dated February 13, 2004, NMC addressed its plans to conform to the recommendations in the Westinghouse/AMAG bulletin and letter for the Palisades as follows:

1. Revise plant procedures to add precaution to appropriately evaluate Crossflow system performance if a modification is performed in the proximity of the Crossflow installation and obtain ultrasonic flow measurement (UFM) vendor technical support as needed to assist in performance evaluations for UFM related modifications or observed atypical system performance prior to the implementation of this MUR power uprate license amendment.

2. Revise plant procedures to add limits for correction factor variation and appropriate actions to be taken as recommended in Nuclear Safety Advisory Signal Interference Issues," dated December 5, 2003.

3. Revise plant procedures to add trending of UFM FW flow and the fluctuation in the UFM FW flow buffered value as recommended in NSAL-03-12, prior to the implementation of this MUR power uprate license amendment.

4. Conduct post-uprate UFM frequency spectrum analysis within 60 days of implementation of this MUR power uprate license amendment.

On the basis of its review of NMC's response to the bulletin and advisory letter, including the above four commitments, the NRC staff concludes that NMC has adequately addressed the potential signal interference concern and that reasonable assurance exists that applicable regulatory criteria, such as maximum allowable power level, will continue to be met once the MUR is implemented at Palisades.

3.1.3 Summary

Based on a review of the licensee's submittals, including the plant-specific calculations of the plant power calorimetric measurement uncertainty, the NRC staff finds that the thermal power measurement uncertainty for the Palisades using the Crossflow UFM is limited to 0.6 percent of actual reactor thermal power, which supports the proposed 1.4 percent thermal power uprate. The NRC staff also finds that the proposed TS change to VHPT setpoint AV is acceptable. The licensee has adequately addressed the four additional requirements outlined in the NRC staff SER on the Crossflow Topical Report CENPD-397-P-A and potential concerns for signal interference identified in a recent Westinghouse/AMAG technical bulletin and advisory letter.

3.2 Reactor Systems

3.2.1 Regulatory Evaluation

The NRC staff review in the area of reactor systems covers the impact of the proposed MUR power uprate on (1) fuel design, (2) nuclear design, (3) thermal-hydraulic design, (4) performance of control and safety systems connected to the reactor and reactor coolant system (RCS), and (5) LOCA and non-LOCA transient analyses (NRC RIS 2002-03, Attachment 1, Sections II, III, and VI). The review is conducted to verify that the licensee's analyses bound plant operation at the MUR power level and that the results of the licensee's analyses related to the areas under review continue to meet the applicable acceptance criteria following implementation of the proposed MUR power uprate.

Nuclear power plants are licensed to operate at a specified core thermal power. Part 50 of Title 10 of the *Code of Federal Regulations*, Appendix K, requires licensees to assume that the reactor has been operating continuously at a power level at least 1.02 times the licensed power level when performing LOCA and ECCS analyses. The *Code of Federal Regulations* (CFR) include this requirement to ensure that the analyses adequately account for instrumentation uncertainties. Appendix K to 10 CFR Part 50 allows licensees to assume a power level lower than 1.02 times the licensed power level (but not less than the licensed power level), provided licensees have demonstrated that the proposed value adequately accounts for instrumentation uncertainties. In its application, the licensee proposed to use a value of 1.006. To achieve this level of accuracy, NMC will install the more accurate FW flow measurement meter described in NRC-approved Topical Report CENPD-397-P-A for licensing applications. The currently installed venturi flow meter will remain in place.

NMC proposes to increase the power output of the Palisades by the difference between the original 1.02 multiplier of 10 CFR Part 50, Appendix K and its proposed value of 1.006, because of the more accurate flow meter. Since the analyses of record for LOCA and ECCS performance assumed a power level of 1.02 times the licensed power level, a 1.4 percent increase in power could be achieved without necessitating reanalyses of these events. Additionally, NMC evaluated other design-basis analyses to ensure appropriate accounting of the power level uncertainties.

3.2.2 Technical Evaluation

3.2.2.1 Uncontrolled Control Rod Bank Withdrawal from a Subcritical or Low Power Start-up Condition

An uncontrolled control rod bank withdrawal transient may be caused by a malfunction of the reactor control or rod control systems. This withdrawal will uncontrollably add positive reactivity to the reactor core, resulting in a power excursion. However, the variable overpower trip will terminate the accident.

The NRC staff found that the licensee performed this analysis assuming a core power level of 2580.6 MWt using an acceptable methodology. Additionally, in its letter dated October 6, 2003, the licensee stated that it performed the minimum departure from nucleate boiling ratio (DNBR) analyses at a nominal core power level of 2565.4 MWt. The analyses indicate that the SRP acceptance criteria continue to be met, i.e., the minimum DNBR remains above the limit value, and the fuel centerline temperatures do not exceed the melting point.

Since the licensee performed the analyses using bounding core power levels and acceptable methodologies, the NRC staff finds them acceptable for the 1.4 percent MUR power uprate to 2565.4 MWt.

3.2.2.2 Uncontrolled Control Rod Bank Withdrawal at Power

Similar to the control rod bank withdrawal from subcritical conditions, an uncontrolled withdrawal at power accident can be caused by a malfunction of the reactor control or rod control systems. This withdrawal uncontrollably adds positive reactivity to the reactor core, resulting in a power excursion, until the thermal margin/low pressure trip terminates the accident.

The NRC staff found that the licensee performed this analysis assuming a core power level of 2580.6 MWt using an acceptable methodology. Additionally, in its letter dated October 6, 2003, the licensee stated that it performed the minimum DNBR analyses at a nominal core power level of 2565.4 MWt. The results indicate that the SRP acceptance criteria continue to be met, i.e., the minimum DNBR remains above the limit value, and the fuel centerline temperatures do not exceed the melting point.

Since the licensee performed the analyses using bounding core power levels and acceptable methodologies, the NRC staff finds them acceptable for the 1.4 percent MUR power uprate to 2565.4 MWt.

3.2.2.3 Single Control Rod Withdrawal

An electrical or mechanical failure in the rod control system could cause the inadvertent withdrawal of a single control rod. Like the control rod bank withdrawal, this transient also adds positive reactivity to the reactor core, resulting in a power excursion.

The NRC staff found that the licensee performed this analysis assuming a core power level of 2580.6 MWt using an acceptable methodology. Additionally, in its letter dated October 6, 2003, the licensee stated that it performed the minimum DNBR analyses at a nominal core power level of 2565.4 MWt. The results indicate that the SRP acceptance criteria continue to be met, i.e., the minimum DNBR remains above the limit value, and the fuel centerline temperatures do not exceed the melting point.

Since the licensee performed the analyses using bounding core power levels and acceptable methodologies, the NRC staff finds them acceptable for the 1.4 percent MUR power uprate to 2565.4 MWt.

3.2.2.4 Boron Dilution Event

The chemical and volume control system (CVCS) can be used to add unborated water to the RCS. This addition may happen inadvertently because of operator error or a system malfunction, and cause an unwanted increase in reactivity and a decrease in shutdown margin. The operator must stop this unplanned dilution before the shutdown margin is eliminated.

The licensee evaluated the boron dilution event for the uprated power conditions over the spectrum of plant operations, from power operation with a core power level of 2580.6 MWt to refueling. The NRC staff found that the results indicate that the SRP acceptance criteria continue to be met, i.e., the minimum DNBR remains above the limit value, the peak RCS and MS system pressures remain below 110 percent of their design values, and the minimum operator action time to eliminate dilution exceeds 30 minutes for refueling and 15 minutes for all other operating conditions.

Since the SRP acceptance criteria continue to be met for this accident over the full range of operating conditions, up to a core power level of 2580.6 MWt, and the analyses bound the requested core power level of 2565.4 MWt, the analyses are acceptable for the 1.4 percent MUR power uprate.

3.2.2.5 Dropped Rod/Bank Event

The dropped rod/bank transients result in a negative reactivity insertion, which causes a shift in the power distribution of the core. The power redistribution increases peaking factors among certain fuel assemblies and could lead to localized fuel damage.

The NRC staff found that the licensee performed these analyses assuming a core power level of 2580.6 MWt using an acceptable methodology. Additionally, in its letter dated October 6, 2003, the licensee stated that it performed the minimum DNBR analyses at a nominal core power level of 2565.4 MWt. The results indicate that the SRP acceptance criteria continue to be met, i.e., the minimum DNBR remains above the limit value, and the fuel centerline temperatures do not exceed the melting point.

Since the licensee performed the analyses using bounding core power levels and acceptable methodologies, the NRC staff finds them acceptable for the 1.4 percent MUR power uprate.

3.2.2.6 Core Barrel Failure

A circumferential rupture of the core support barrel could cause the core to shift until it is stopped by the core stop supports. However, the motion of the core relative to the inserted rod banks will induce a small reactivity transient. The licensee evaluated this transient and determined that it remains bounded by the control rod ejection event.

Since the control rod ejection event bounds this transient, and since the NRC staff found the control rod ejection event acceptable for the power uprate, the NRC staff also finds the core barrel failure analysis acceptable.

3.2.2.7 Malposition of the Part-Length Control Rod Group

Palisades has four part-length control rods originally intended for controlling the axial power distribution in the core. These rods are not connected to any reactor trip circuit nor will they drop into the core on a reactor trip or loss of power. Additionally, during power operation, the current TSs preclude the use of these control rods, and the licensee maintains them in the fully withdrawn position.

Because these control rods are maintained in a fully withdrawn position and are not used during power operation, the NRC staff determined that the malposition of the rod group is not a credible event for the MUR power uprate.

3.2.2.8 Statically Misaligned Control Rod/Bank

Statically misaligned control rods and banks also cause adverse power distributions in the core. The power redistribution increases peaking factors among certain fuel assemblies and could lead to localized fuel damage.

The NRC staff found that the licensee performed this analysis assuming a core power level of 2580.6 MWt using an acceptable methodology. Additionally, in its letter dated October 6, 2003, the licensee stated that it performed the minimum DNBR analyses at a nominal core power level of 2565.4 MWt. The results indicate that the SRP acceptance criteria continue to be met, i.e., the minimum DNBR remains above the limit value, and the fuel centerline temperatures do not exceed the melting point.

Since the licensee performed the analyses using bounding core power levels and acceptable methodologies, the NRC staff finds them acceptable for the 1.4 percent MUR power uprate.

3.2.2.9 Loss of Reactor Coolant Flow

A mechanical or electrical failure in one or more reactor coolant pumps or a fault in the power supply to these pumps may cause a partial or complete loss of forced coolant flow. If the reactor is powered at the time of the incident, the loss of coolant flow causes a rapid increase in coolant temperature. This increase could result in departure from nucleate boiling (DNB).

The NRC staff found that the licensee performed this analysis assuming a core power level of 2580.6 MWt using an acceptable methodology. Additionally, in its letter dated October 6, 2003, the licensee stated that it performed the minimum DNBR analyses at a nominal core power level of 2565.4 MWt. The results indicate that the SRP acceptance criteria continue to be met, i.e., the minimum DNBR remains above the limit value and the RCS and MS system pressures remain below 110 percent of their design values.

Since the licensee performed the analyses using bounding core power levels and acceptable methodologies, the NRC staff finds them acceptable for the 1.4 percent MUR power uprate to 2565.4 MWt.

3.2.2.10 Reactor Coolant Pump Rotor Seizure

A reactor coolant pump rotor seizure causes the flow through the affected reactor coolant loop (RCL) to rapidly decrease, and the reactor trips on a low reactor coolant flow signal. The sudden reduction in core coolant flow while the reactor is powered results in decreased core heat transfer, which may cause fuel damage.

The NRC staff found that the licensee performed this analysis assuming a core power level of 2580.6 MWt using an acceptable methodology. Additionally, in its letter dated October 6, 2003, the licensee stated that it performed the minimum DNBR analyses at a nominal core power level of 2565.4 MWt. The results indicate that the minimum DNBR remains above the limit value and the RCS and MS system pressures remain below 110 percent of their design values.

Since the licensee performed the analyses using bounding core power levels and acceptable methodologies, the NRC staff finds them acceptable for the 1.4 percent MUR power uprate to 2565.4 MWt.

3.2.2.11 Startup of an Inactive Reactor Coolant Loop

The transient involving the startup of an inactive loop at the incorrect temperature occurs when one reactor coolant pump is out of service. With the hot-leg temperature of the inactive loop lower than the reactor core inlet temperature, this startup results in the injection of cold water into the core. The injection causes a reactivity insertion and subsequent power increase.

The licensee's TSs prohibit power operation with less than all four reactor coolant pumps in operation. Therefore, the licensee determined that this accident is not credible for Palisades. Because the TSs preclude operation with less than four reactor coolant pumps in operation, the NRC staff agrees with the licensee's assessment.

3.2.2.12 Excessive Feedwater Incident

An excessive FW incident occurs when relatively cool FW or too much FW is supplied to the steam generators (SGs). This action causes excess heat removal by the secondary side, which increases core power above full power. This transient could occur through the accidental opening of the FW regulating valves or the accidental opening of a FW bypass valve.

The NRC staff found that the licensee's analysis is bounded by the increase in steam flow event, and the results of this event indicate that the SRP acceptance criteria continue to be met, i.e., the minimum DNBR remains above the limit value and the RCS and MS system pressures remain below 110 percent of their design values. Therefore, the NRC staff finds the incident acceptable for the 1.4 percent MUR power uprate.

3.2.2.13 Increase in Steam Flow (Excess Load)

An excess load incident occurs when a rapid increase in steam flow causes a power mismatch between the reactor core power and the SG load demand. The increased heat removal from the primary side coinciding with a negative moderator temperature coefficient will cause the power in the core to increase. Excessive loading rates may result in a reactor trip initiated by the reactor protection system.

The NRC staff found that the licensee performed the analyses assuming a reactor core power level of 2580.6 MWt, and the results indicate that the SRP acceptance criteria continue to be met, i.e., the minimum DNBR remains above the limit value and the RCS and MS system pressures remain below 110 percent of their design values.

Since the licensee performed the analyses based upon a core power level of 2580.6 MWt using an acceptable methodology, and the analysis bounds the requested core power level of 2565.4 MWt, the NRC staff finds the analysis acceptable for the 1.4 percent MUR power uprate.

3.2.2.14 Loss of External Load

A loss of external electrical load event occurs when an electrical disturbance causes the loss of a significant portion of the generator load or when the turbine trips. This loss of load causes both the primary and secondary pressures and temperatures to increase.

The NRC staff found that the licensee performed the analyses assuming a reactor core power level of 2565.4 MWt for the DNB case and 2530 MWt with a 2 percent uncertainty (2580.6 MWt) for the overpressure case. The results indicate that the SRP acceptance criteria continue to be met, i.e., the minimum DNBR remains above the limit value, and the RCS and MS system pressures remain below 110 percent of the design values.

Since the licensee performed the analyses based upon a core power level of 2565.4 MWt for the DNB case and 2580.6 MWt for the pressurization case, using an acceptable methodology, and the analyses bound the requested core power level of 2565.4 MWt, the NRC staff finds the analyses acceptable for the 1.4 percent MUR power uprate.

3.2.2.15 Loss of Normal Feedwater

A loss of normal FW event reduces the capability of the secondary system to remove the heat generated in the reactor core. If the reactor were not tripped or if an alternate supply of FW were not supplied to the plant, core damage could occur. Currently, the Palisades loss of normal FW analysis models a core power level of 102 percent of 2530 MWt (2580.6 MWt). This power level bounds the requested uprate core power level of 2565.4 MWt with an 0.6 percent uncertainty. Since the current analysis bounds the power uprate, the NRC staff finds it acceptable.

3.2.2.16 Steam-line Rupture Incident

The steam-line rupture incident models an uncontrolled steam release from the secondary system because of a break of the main steamline. The most limiting steam pipe accidents occur when the reactor is at no load conditions. With the reactor in this condition, the steam release will cool the RCS. Since the RCS has a negative moderator temperature coefficient, this cooling may cause the core to become critical and return to power, possibly causing fuel damage.

Because the most limiting case of this accident occurs at no load conditions, the core response portion of the steam pipe rupture accident remains independent of power level. Since the core response portion of this accident is not influenced by power level, the NRC staff finds the core response acceptable for the licensee’s proposed power uprate to 2565.4 MWt.

3.2.2.17 Steam Generator Tube Rupture with Loss of Offsite Power-Thermal/Hydraulic

For a SG tube rupture (SGTR), the thermal-hydraulic analysis calculates the primary to secondary break flow and the steam released to the environment. The input parameters that could change as a result of the uprate include: power, hot-leg temperature, cold-leg temperature, MS temperature, and MS pressure. An increase in reactor power is expected to increase the primary to secondary side break flow potentially resulting in larger radiological consequences. However, the methodology used in the current licensing basis analysis includes a 2.0 percent margin in reactor power for the calculation of the break flow. The analyzed 2.0 percent margin for reactor power bounds the power uprate of 1.4 percent with a 0.6 percent uncertainty. The NRC staff finds the analyses acceptable for the 1.4 percent MUR power uprate.

3.2.2.18 Control Rod Ejection

The consequences of a control rod ejection include a rapid positive reactivity insertion together with an adverse core power distribution, which could lead to localized fuel rod damage. The NRC staff found that the licensee performed the analyses assuming a reactor core power level of 0 percent power for the Hot Zero Power case and up to 2530 MWt with a 2 percent uncertainty (2580.6 MWt) for the Hot Full Power case. The results indicate that the Palisades licensing basis acceptance criteria continue to be met, since the results indicate no fuel failure due to DNB, and the peak RCS pressure remains below 110 percent of design limits.

Since the licensee performed the analyses based upon a core power levels of both 0 MWt and 2580.6 MWt using an acceptable methodology, and the analyses bound the requested core power level of 2565.4 MWt, the NRC staff finds the analyses acceptable for the 1.4 percent MUR power uprate.

3.2.2.19 Large-Break LOCA

For Palisades, a large-break LOCA (LBLOCA) includes a rupture of the RCS piping up to and including a double-ended guillotine break on the pump discharge side of a cold-leg pipe. Should such a major break occur, the RCS rapidly depressurizes until the pressure nearly equals the containment pressure. Safety injection (SI) initiates upon receipt of a high containment pressure setpoint. After the end of the blowdown, the ECCS will reflood the reactor.

The NRC staff found that the licensee performed the analyses assuming a reactor core power level of 2530 MWt with a 2.0 percent uncertainty (2580.6 MWt). The results indicate that the acceptance criteria of 10 CFR 50.46 continue to be met, i.e., the peak cladding temperature remains below 2200 ^{o}F, the maximum cladding oxidation remains below 17 percent of thickness before oxidation, the maximum hydrogen generation remains below 1 percent of the hypothetical amount, the core remains in a coolable geometry, and the long-term core coolability is maintained.

Since the licensee performed the analyses based upon a core power level of 2580.6 MWt using an acceptable methodology, and the analyses bound the requested core power level of 2565.4 MWt, the NRC staff finds the analyses acceptable for the 1.4 percent MUR power uprate.

3.2.2.20 Small-Break LOCA

Some small-break LOCAs (SBLOCAs) cause the expulsion of reactor coolant at a rate which can be accommodated by the charging pumps. The charging pumps then would maintain pressurizer water level, permitting the operator to execute an orderly shutdown. However, for larger breaks, the fluid exiting the break causes a depressurization of the RCS. SI will occur when an appropriate SI initiation setpoint is reached.

The NRC staff found that the licensee performed the analyses assuming a reactor core power level of 2530 MWt with a 2.0 percent uncertainty (2580.6 MWt). The results indicate that the 10 CFR 50.46 acceptance criteria continue to be met, i.e., the peak cladding temperature remains below 2200 ^{o}F, the maximum cladding oxidation remains below 17 percent of thickness before oxidation, the maximum hydrogen generation remains below 1 percent of the hypothetical amount, the core remains in a coolable geometry, and the long-term core coolability is maintained.

Since the licensee performed the analyses based upon a core power level of 2580.6 MWt using an acceptable methodology, and the analyses bound the requested core power level of 2565.4 MWt, the NRC staff finds the analyses acceptable for the 1.4 percent MUR power uprate.

3.2.2.21 Reactor Internals Structural Behavior Following a LOCA

To evaluate the internal forces generated by a LOCA, NMC analyzed the case of a double-ended rupture of a 42-inch pipe. In the analyses, the licensee demonstrated that the combined loadings were less than the allowable limits, thus ensuring maintenance of a coolable core geometry. Furthermore, in its letter dated June 3, 2003, NMC indicated that the structural loading analysis is not dependent upon power level and, therefore, the current analyses bound those for the power uprate. Because the current analyses are bounding, the NRC staff finds them acceptable for the power uprate to 2565.4 MWt.

3.2.2.22 Natural Circulation Cooldown

Upon loss of power to the reactor coolant pumps, natural circulation within the RCS provides the necessary coolant flow for core cooling and residual heat removal (RHR). The goal of coping with a natural circulation cooldown event is to prevent voiding in the upper head of the RCS pressure vessel to avoid interruption of natural circulation flow.

The NRC staff found that the licensee performed the analysis assuming a reactor core power level of 2570 MWt. The results indicate that Palisades has adequate RCS flow and auxiliary FW for decay heat removal.

Since the licensee performed the analysis based upon a core power level of 2570 MWt using an acceptable methodology, and the analysis bounds the requested core power level of 2565.4 MWt, the NRC staff finds the analysis acceptable for the 1.4 percent MUR power uprate.

3.2.2.23 Anticipated Transient Without Scram

The licensee installed an anticipated transient without scram (ATWS) mitigation system actuation circuitry (AMSAC) and a diverse scram system (DSS) at Palisades, thereby satisfying the requirements of 10 CFR 50.62(b). After the implementation of the power uprate, the AMSAC and the DSS will continue to operate at Palisades in compliance with the requirements of the ATWS rule. Because of the safety afforded by both the AMSAC and the DSS, no plant-specific ATWS analyses are required to support the MUR power uprate at Palisades.

3.2.2.24 Station Blackout

In the coping analysis for a station blackout (SBO) event, the licensee performed calculations assuming a core power level of 2530 MWt with a 2.0 percent uncertainty, which equates to 2580.6 MWt. Since this analysis continues to bound the requested core power level of 2565.4 MWt with a 0.6 percent uncertainty (2580.6 MWt), the NRC staff finds it acceptable for the requested power uprate.

3.2.2.25 Long-Term Post-LOCA Core Cooling

10 CFR 50.46(b)(5) establishes the long-term cooling requirements following a loss-of-coolant accident. One issue with long-term cooling is ensuring that boric acid (H_3BO_3) accumulation will not prevent core cooling. Because of boron precipitation, the NRC found plants that require changes to the operating procedures to ensure adequate hot-leg switch-over times.

NMC evaluated the effects of post-LOCA long-term cooling using the NRC-approved methodology described in Combustion Engineering topical report CENPD-254-P-A, “Post-LOCA Long Term Cooling Evaluation Model,” dated June 1997. Furthermore, the licensee determined that for a power level of 2580.6 MWt the hot and cold-leg injection of the long-term core cooling occurs between 5.5 and 6.5 hours after the LOCA starts. This occurs more than 22 hours prior to the time at which boric acid precipitation would occur.

Since the power level for the current hot-leg switchover analysis bounds the uprated conditions, and since boron precipitation is not predicted to occur, the NRC staff finds the analysis to be acceptable for the power uprate to 2565.4 MWt.

3.2.2.26 Low Temperature Overpressure Protection

During low temperature operation, the low-temperature overpressure protection system controls RCS pressure so the integrity of the reactor coolant pressure boundary is not compromised by violating the pressure-temperature limit curves. The reactor vessel is the limiting reactor coolant pressure boundary component for demonstrating such protection.

NMC determined that the pressure-temperature limit curves remain valid for the uprated conditions. Because the low-temperature overpressure protection system protects against these curves and are not directly affected by the power uprate, the NRC staff finds the system acceptable for the uprated power to 2565.4 MWt.

3.2.2.27 Low Pressure Safety Injection

The low pressure SI system provides water inventory and cooling to the RCS in the event of a design-basis accident (DBA). Because of the associated decay heat increase, a power uprate causes a greater demand on the SI system for response time, flow rate, and flow duration. The licensee evaluated the SI system up to a power level of 2580.6 MWt and determined that the system performance remains acceptable for the 1.4 percent power uprate. Since the system remains adequate up to a power level of 2580.6 MWt, the NRC staff finds it acceptable for the uprated power level of 2565.4 MWt with a 0.6 percent uncertainty.

3.2.2.28 Changes to Technical Specifications, Protection System Settings, and Emergency System Settings

NMC determined that because of the power uprate, the VHPT TS value would need to be updated. The licensee proposed changing the TS value from 111 percent of reactor power to 109.4 percent of reactor power. This change in effect would keep the post-uprate trip setpoint allowable value at the same reactor power level as the original allowable value.

Because of an increase in rated power, not changing this TS value would cause the trip to activate later in an accident, i.e. at a higher power level. Conversely, maintaining the current power level for the trip would be conservative. Since it is conservative to change the TS value to 109.4 percent power, thus keeping the actual setpoint at the same reactor power level, the NRC staff finds the licensee’s proposal acceptable.

3.2.2.29 NSSS Design Parameters

The NSSS design parameters provide the RCS and secondary system conditions for use in the NSSS analyses and evaluations. NMC presented parameters for the power levels of 2530 MWt and 2565.4 MWt. The key parameters included core power, RCS pressure, cold-leg temperature, hot-leg temperature, SG pressure, main FW temperature, mains steam flow, main feed flow, SG liquid inventory, SG vapor inventory, T_{ave} range, and steam temperature. The differences between the parameters at 2530 MWt and 2565.4 MWt included an increased core power level, decreased cold-leg temperature, increased hot-leg temperature, lower steam pressure, increased FW temperature, higher steam and feed flow rates, and lower SG liquid and vapor inventories. The NRC staff evaluated these changes to the plant conditions and found them to adequately represent the plant behavior at the specified power levels; therefore, the NRC staff finds the NSSS design parameters acceptable.

3.2.3 Summary

The NRC staff reviewed the licensee’s analyses to support operations of Palisades at a maximum core power level of 2565.4 MWt. Based on this review, the NRC staff finds that the supporting safety analyses were performed using acceptable methods; the input parameters of the analyses adequately represent the plant conditions at the uprated power level; and the analytical results meet the applicable acceptance criteria. Therefore, the NRC staff concludes that the supporting analyses are acceptable for the power uprate to 2565.4 MWt.

3.3 <u>Electrical Systems</u>

The NRC staff review in the area of electrical systems covers the impact of the proposed MUR power uprate on (1) environmental qualification of electrical equipment; (2) offsite power systems with respect to grid stability, including performance of the main generator, main transformer, isolated phase bus, station power transformers (SPTs), startup transformers (STs), and safeguard transformer; (3) emergency diesel generator loading; (4) direct current (dc) distribution system; and (4) SBO.

3.3.1 Environmental Qualification of Electrical Equipment

3.3.1.1 Regulatory Evaluation

The term “environmental qualification” applies to equipment important to safety to assure this equipment remains functional during and following design-basis events. The NRC staff’s review covers the environmental conditions that could affect the design and safety functions of electrical equipment including instrumentation and control. The NRC staff’s review is to ensure compliance with the acceptance criteria thus ensuring that the equipment continues to be capable of performing its design safety functions under all normal environmental conditions, anticipated operational occurrences, and accident and post accident environmental conditions. Acceptance criteria are based on 10 CFR 50.49 as it relates to specific requirements regarding the qualification of electrical equipment important to safety that is located in a harsh environment. Specific Review criteria are contained in SRP Section 3.11.

3.3.1.2 Technical Evaluation

The proposed power uprate has no effect on the Palisades environmental qualification (EQ) program. The EQ evaluation parameters assume reactor power of at least 2580.6 MWt, 102 percent of the current RTP of 2530 MWt. Therefore, the programs, and activities that are currently in place are not affected by the proposed 1.4 percent power uprate. No physical change to the facility is necessary; therefore, no equipment reviews were performed.

3.3.1.3 Summary

The NRC staff has reviewed the licensee’s submittal of the effects of the proposed power uprate on the environmental qualification of the electrical equipment and concludes that the analyses performed at 102 percent bounds the 1.4 percent proposed power uprate and, therefore, the design is acceptable.

3.3.2 Offsite Power System

3.3.2.1 Regulatory Evaluation

Prior to the introduction of general design criteria (GDC) 17 of Appendix A to 10 CFR Part 50, AEC Criterion 39 was used to evaluate the adequacy of the electric power systems. Criterion 39 requires that sufficient offsite and redundant, independent, and testable standby auxiliary sources of electrical power are provided to attain a prompt shutdown and continued maintenance of the plant in a safe condition under all credible circumstances. The capacity of the power sources is adequate to accomplish all required engineered safety features functions under all postulated DBA conditions. Acceptance criteria are based on Criterion 39.

3.3.2.2 Technical Evaluation of Grid Stability

The licensee performed the analysis using a power flow computer simulation of the operating system including interconnections to the other utilities. A review of the results indicated that the power uprate does not have any adverse impact on the grid stability.

The NRC staff reviewed the licensee's submittal and concluded that the impact of the power uprate on the grid stability is insignificant. Therefore, the plant continues to meet the requirements of Criterion 39 for grid stability with this power uprate.

3.3.2.3 Technical Evaluation of Main Generator

The main generator is rated at 955 megavars ampere (MVA) at a 0.85 power factor. At the anticipated value of 1.4 percent power uprate, the generator output will be 834.5 MWe at 0.87 power factor (959 MVA).

The NRC staff reviewed the licensee's submittal and determined that the generators output will approximately remain the same as the design rating and therefore, operating the main power transformers at the uprated power condition is acceptable.

3.3.2.4 Technical Evaluation of Main Transformer

The main transformer is rated at 975 MVA. The maximum MVA capability of the main generator is at 955 MVA which is within the rating of the main transformer.

The NRC staff reviewed the licensee's submittal and concluded that the anticipated power uprate of 1.4 percent is below the maximum main transformer design rating and, therefore, operating the main power transformers at the uprated power condition is acceptable.

3.3.2.5 Technical Evaluation of Isolated Phase Bus

The isolated phase bus connects the main generator to the primary windings of the main transformer and the SPTs. The isolated phase bus is rated at 22 kV, 26,400 amperes. At a power factor of 0.87, the generator gross electrical output would require an isolated phase bus rating of 25,062 amperes which is within the rating of the isolated phase bus.

The NRC staff reviewed the licensee's submittal and concluded that the impact of power uprate of 1.4 percent is below the design rating of the isolated phase bus and, therefore, operating the isolated phase bus at the uprated power condition is acceptable.

3.3.2.6 Technical Evaluation of Station Power Transformers

The SPTs 1-1, and 1-3 have dual secondary outputs and are rated at 12.6 MVA each. The SPT 1-2 is rated at 8.96 MVA. The loading on each of the SPTs has not changed with the power uprate.

The NRC staff reviewed the licensee's submittal and concluded that the SPTs loading resulting from the 1.4 percent power uprate has not changed and is below the maximum design rating and, therefore, operating these transformers at the uprated power condition is acceptable.

3.3.2.7 Technical Evaluation of Startup Transformers

The startup transformers (STs) 1-1, and 1-3 have dual secondary outputs and are rated at 12.6 MVA each. The ST 1-2 is rated at 10.6 MVA. The loading on each of the STs has not changed with the power uprate.

The NRC staff reviewed the licensee's submittal and concluded that the STs loading resulting from the 1.4 percent power uprate has not changed and is below the maximum design rating and, therefore, operating these transformers at the uprated power condition is acceptable.

3.3.2.8 Technical Evaluation of Safeguard Transformer

The safeguard transformer is rated at 10.5 MVA. The loading on the safeguard transformer has not changed with the MUR power uprate.

The NRC staff reviewed the licensee's submittal and concluded that the safeguard transformer loading resulting from the 1.4 percent power uprate has not changed and is below its maximum design rating and, therefore, operating the safeguard transformer at the uprated power condition is acceptable.

3.3.2.9 Summary

The NRC staff has reviewed the licensee's submittal for the effect of the proposed power uprate on the offsite power system and concludes that the offsite power system will continue to meet the requirements of Criterion 39 following implementation of the proposed power uprate. The NRC staff further concludes that the impact of the proposed power uprate on grid stability is insignificant. Therefore, the NRC staff finds the proposed power uprate acceptable with respect to the offsite power system.

3.3.3 Emergency Diesel Generators

3.3.3.1 Regulatory Evaluation

The alternating current (ac) onsite power system includes those standby power sources, distribution systems, and auxiliary supporting systems provided to supply power to the safety-related equipment. The NRC staff's review covers the descriptive information, analyses, and referenced documents for the ac onsite power system. Acceptance criteria are based on Criterion 39 as it relates to the capability of the ac onsite power system to perform its intended functions during all plant operating and accident conditions. Specific review criteria are contained in SRP Sections 8.1 and 8.3.1.

3.3.3.2 Technical Evaluation

The emergency diesel generators are designed to furnish reliable ac power for safe plant shutdown and for operation of engineered safeguards, when no offsite power is available. The engineered safeguard loads have not changed. The capacity of each emergency diesel generator is adequate to support the operation of required engineered safeguards under DBA conditions.

3.3.3.3 Summary

The NRC staff has reviewed the licensee's submittal for the effect of the proposed power uprate on the ac onsite power system and concludes that the licensee has adequately accounted for the effects of the proposed power uprate on the system's functional design. The NRC staff further concludes that the ac onsite power system will continue to meet the requirements of

Criterion 39 following implementation of the proposed power uprate. Therefore, the NRC staff finds the proposed power uprate acceptable with respect to the onsite ac power system.

3.3.4 DC Distribution System

3.3.4.1 Regulatory Evaluation

The dc power systems include those dc power sources and their distribution systems and auxiliary supporting systems provided to supply motive or control power to safety-related equipment. The NRC staff's review covers the information, analyses, and referenced documents for the dc onsite power system. Acceptance criteria are based on Criterion 39 and 10 CFR Part 50.63 as they relate to the capability of the dc onsite electrical power to facilitate the functioning of structures, systems, and components important to safety. Specific review criteria are contained in SRP Sections 8.1 and 8.3.2.

3.3.4.2 Technical Evaluation

The dc distribution system is designed to supply power during normal, shutdown, accident and post accident conditions. The 1.4 percent MUR power uprate does not affect the dc system.

3.3.4.3 Summary

The NRC staff has reviewed the licensee's submittal for the effect of the proposed power uprate on the dc onsite power system and concludes that the 1.4 percent MUR power uprate does not affect the dc system. The NRC staff further concludes that the dc onsite power system will continue to meet the requirements of Criterion 39 following implementation of the proposed power uprate. Therefore, the NRC staff finds the proposed power uprate acceptable with respect to the dc onsite power system.

3.3.5 Station Blackout

3.3.5.1 Regulatory Evaluation

Station blackout (SBO) refers to the complete loss of ac electric power to the essential and nonessential switchgear buses in a nuclear power plant. SBO involves the loss of offsite power concurrent with turbine trip and failure of the onsite emergency ac power system. SBO does not include the loss of available ac power to buses fed by station batteries through inverters or the loss of power from "alternate ac sources". The NRC staff's review focuses on the impact of the proposed power uprate on the plant's ability to cope with and recovery from an SBO event as based on 10 CFR 50.63. Specific review criteria are contained in SRP Sections 8.1 and Appendix B to SRP 8.2.

3.3.5.2 Technical Evaluation

The evaluation of an SBO event for Palisades was performed in accordance with the requirements of regulatory guide (RG) 1.155, "Station Blackout." This evaluation determined an acceptable SBO duration for Palisades of 4 hours. This 4-hour coping duration was based on the reliability and configuration of the offsite power system and the reliability of the diesel generators. To provide assurance that the plant could cope with a SBO of 4 hours duration, several factors were considered. These areas included the following:

Condensate Inventory
Class 1E Battery Capacity
Compressed Air
Effects of Loss of Ventilation
Containment Isolation
Reactor Vessel Inventory

The licensee has determined that the only factor potentially affected by the proposed power uprate is the condensate (CD) inventory required to provide decay heat removal for the 4-hour duration.

The SBO analysis was approved by the NRC staff dated May 20, 1991. In that SE, the NRC calculated the minimum CD inventory based on a power level of 102 percent of 2530 MWt. This minimum inventory was determined to be 57,100 gallons. Palisades TS require maintaining an inventory of 100,000 gallons. Therefore, the proposed power uprate has no effect on the SBO coping capability.

3.3.5.3 Summary

The NRC staff has reviewed the licensee's submittal on the effect of the proposed power uprate on the plant's ability to cope with and recover from an SBO event for the period of time established on the plant's licensing basis. The plant has adequate CD inventory for decay heat removal during an SBO of 4-hour duration. The NRC staff concludes that the licensee has adequately evaluated the effects of the proposed power uprate on SBO and demonstrated that the plant will continue to meet the requirements of 10 CFR 50.63 following the implementation of the proposed power uprate. Therefore, the NRC staff finds the proposed power uprate acceptable with respect to a SBO.

3.3.6 Summary

The NRC staff has evaluated the effect of power uprate on the necessary electrical systems and environmental qualification of electrical equipment and components. Results of these evaluations show that the increase in the core thermal power would have negligible impact on the grid stability, SBO, or the environmental qualification of electrical equipment. This is consistent with Criterion 39, 10 CFR 50.63, and 10 CFR 50.49 and the proposed change is, therefore, acceptable.

3.4 <u>Civil and Engineering Mechanics</u>

3.4.1 Regulatory Evaluation

The NRC staff review in the area of mechanical and civil engineering covers structural and functional integrity of piping systems, components and their supports, including core support structures, which are designed in accordance with the rules of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section III, and ASA/USAS/ANSI B31.1. The NRC staff's evaluation considered GDC 1, 2, 4, 10, 14, and 15.

The NRC staff review focused on verifying that the licensee has provided reasonable assurance of the structural and functional integrity of piping systems, components, component internals, and their supports under normal and vibratory loadings, including those due to fluid flow, postulated accidents, and natural phenomena such as earthquakes.

The acceptance criteria are based on continued conformance with the requirements of the following regulations: (1) 10 CFR Part 50, 50.55a, and GDC 1 as they relate to structures and components being designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety function to be performed; (2) GDC 2 as it relates to structures and components important to safety being designed to withstand the effects of earthquakes combined with the effects of normal or accident conditions; (3) GDC 4 as it relates to structures and components important to safety being designed to accommodate the effects of, and to be compatible with, the environmental conditions of normal and accident conditions; (4) GDC 10 as it relates to reactor internals being designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences; (5) GDC 14 as it relates to the reactor coolant pressure boundary being designed, fabricated, erected, and tested to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture; and (6) GDC 15 as it relates to the RCS being designed with sufficient margin to ensure that the design conditions are not exceeded.

The specific review areas are contained in the NRC SRP Section 3.9. The review also includes the plant-specific provisions of General Letter (GL) 89-10 and GL 96-05, as related to plant-specific program for motor-operated valves, GL 95-07, as related to the pressure locking and thermal binding for safety-related gate valves, and the plant-specific evaluation of the GL 96-06 program regarding the over-pressurization of isolated piping segments.

3.4.2 Technical Evaluation

The NRC staff reviewed the Palisades power uprate amendment request dated June 3, 2003. The review focused on the effects of power uprate on the structural and pressure boundary integrity of piping systems and components, their supports, and reactor vessel and internal components, the control rod drive mechanisms (CRDMs), and the balance-of-plant (BOP) piping systems.

The proposed 1.4 percent power uprate will increase the RTP level from 2530 MWt to 2565.4 MWt. The power uprate will be achieved by additional opening of the turbine throttle valves, which will result in an increase in steam flow, a decrease in SG pressure, and an increase in the temperature difference across the core. The primary coolant system (PCS) pressure, flow rate, and average temperature will remain the same.

The table on Page 13 of Attachment 4 to the licensee's June 3, 2003, submittal shows the pertinent temperatures, pressures, and flow rates for the current conditions and the uprated conditions. At full power, the hot-leg temperature increases from 582.7 to 583.0 degrees Fahrenheit, the cold-leg temperature decreases from 573.3 to 573.0 degrees Fahrenheit, the SG pressure decreases from 770 to 765.8 psia, the steam flow increases from 11.114 to 11.297 million pounds per hour (Mlbm/hr), the FW temperature increases from 439.5 to 440.7 degrees Fahrenheit, and the FW flow increases from 11.174 to 11.357 Mlbm/hr. The proposed uprate does not change heatup or cooldown rates or the number of cycles assumed in the design analyses. In addition, there are no changes in the design transients since the safety analyses were performed at 102 percent of RTP. Thus, the limiting analyses are still bounding.

The design parameters for the PCS and SGs are found in Tables 4-1 and 4-4, respectively, of the Palisades final safety analysis report (FSAR). The PCS components, including the reactor vessel, core support structures, and SGs, were designed to operate at a core power level of 2650 MWt. The PCS components are designed to 650 degrees Fahrenheit (except the pressurizer, which is designed to 700 degrees Fahrenheit) and 2,500 psia. The SGs are designed for a combined steam flow of 11.572 Mlbm/hr. The FW system design temperature is 450 degrees Fahrenheit.

3.4.2.1 Reactor Pressure Vessel (RPV) and Internals

The licensee indicated that the code of record for the reactor vessel, nozzles, and supports is ASME Section III, 1965 Edition, including all addenda through the Winter 1965 Addenda. The reactor vessel internals were designed prior to the introduction of specific criteria for these components in the ASME Code; however, the internals were designed in accordance with the ASME Code, Section III, 1965 Edition where required.

The licensee compared the expected temperatures and pressures for the proposed power uprate condition against the analysis of record. The licensee confirmed that the design and operating temperatures and pressures used in the analysis of record continue to bound the conditions expected for the proposed uprated power level. In addition, the current design-basis transients, including LOCA structural analysis, remain valid for the proposed power uprate. The licensee concluded that the analysis of record bounds the uprated power conditions. As such, the current design-basis stresses and cumulative usage factor (CUF) analyses for the reactor vessel and internal components will continue to meet the code allowable limits and are, therefore, acceptable for the proposed uprated power conditions. Since the operating temperatures, operating pressures, and design transients in the analysis of record remain bounding, the NRC staff concurs with the licensee’s assessment that the RPV and internals are acceptable for operation at the uprated power level.

3.4.2.2 Control Rod Drive Mechanisms

The licensee stated that the code of record for the pressure retaining components of the CRDMs is the ASME Code, Section III, Subsection NB, 1989 Edition. The licensee confirmed in its submittal that the temperatures and pressures used in the CRDM design analyses continue to bound the conditions at the proposed uprated power level. In addition, the current design-basis transients, including LOCA structural analysis, remain valid for the proposed power uprate. The CRDM components will continue to meet the code limits and are, therefore, acceptable for the proposed uprated power conditions. Since the operating temperatures and pressures and the design transients in the analysis of record remain bounding, the NRC staff concurs with the licensee’s assessment that the CRDMs are acceptable for operation at the uprated power level.

3.4.2.3 Reactor Coolant Piping and Components

The PCS piping was designed to USAS B31.1, 1955. However, the licensee has also evaluated some portions of the PCS piping based on ASME Code, Section III, 1965 Edition. As a result of its evaluation, the licensee concluded that all stresses meet the appropriate Code allowables. On the basis of its review, since the design-basis temperatures and pressures bound the ranges of conditions expected at the proposed power uprate, the NRC staff concurs with the licensee's conclusion that the PCS piping is acceptable for the proposed power uprate.

The SGs were designed to the ASME Code, Section III, 1977 Edition. There is no change in the primary system flow rate, and the primary coolant system temperatures and pressures used in the design continue to bound the uprate conditions. There is an increase in the steam flow and FW flow, and there is a decrease in the secondary side pressure. The steam and FW pressures and flow rates used in the design of the SGs continue to bound the expected uprate conditions. The pressure difference between the primary coolant system and the secondary side during full power operation increases by approximately 4.2 psi (approximately 0.3 percent). The licensee stated that this increase in pressure differential is less than variation during the plant normal operation. Also, since the design of the SGs included modeling of flow-induced vibration and the steam and FW flow rates remain bounded by the design flow rates, the licensee concluded that the power uprate will have no effect on flow-induced vibration.

The pressure retaining parts of the reactor coolant pumps (RCPs) were designed in accordance with the ASME Code, Section III, 1968 Edition. On the basis of its review, the licensee confirmed that the design-basis temperature of the RCPs is bounding for the uprated conditions.

The code of record for the pressurizer, including the nozzles, is the ASME Code, Section III, 1965 Edition, with addenda through Winter 1966. The code of record for the pressurizer surge line is ANSI B31.1, 1973 Edition, including the Summer 1973 Addenda. The licensee had evaluated the slightly lower cold-leg temperature (entering the spray nozzle) and the slightly higher hot-leg temperature (entering the surge nozzle) during full power operation at the uprate conditions. The licensee determined that the temperatures and pressures used in the analyses of record continue to bound the uprate conditions.

The code of record for the pressurizer safety valves is ASME Code, Section III, 1965 Edition with addenda through Winter 1965. The power level, temperatures, and pressures used in the design continue to bound the uprate conditions. In addition, the analysis of record for the loss of external load event, which is used to assess the capability of the valves to prevent overpressure, assumed 102 percent reactor power and remains valid for the uprated conditions.

Based on the above, the NRC staff agrees with the licensee's conclusion that the design of piping, components, including the SGs, RCPs, and pressurizer, and their supports, is adequate to maintain the structural and pressure boundary integrity of the reactor coolant loop because the analyses of record parameters are bounding for the proposed 1.4 percent power uprate condition.

3.4.2.4 BOP Piping and Safety Related Valves

The licensee evaluated the BOP piping systems by comparing the conditions for the proposed power uprate with the analysis of record conditions and the current operating conditions. The BOP piping, including MS, CD, FW, auxiliary FW, and SG blowdown, were designed to the Code for Pressure Piping, USAS B31.1, 1955 Edition, or the Power Piping Code, USAS B31.1, 1967 Edition. All safety related BOP piping has been reanalyzed to ANSI B31.1, 1973 Edition, with addenda through the Summer 1973.

The licensee determined that the temperatures and pressures used in the design-basis analysis and the analysis of record for the BOP piping continue to bound the uprate conditions. The FW temperature at full power increases from 439.5 to 440.7 degrees Fahrenheit, but remains below the system design temperature of 450 degrees Fahrenheit. With respect to high-energy line breaks (HELB), the licensee's evaluation demonstrated that the break locations are not effected on any system and that the jet impingement forces do not increase for those systems that are analyzed for jet impingement forces. The structural integrity of the spent fuel pool cooling system is not effected by the proposed power uprate because the licensee will use administrative controls (e.g., cool time) to ensure that the decay heat load is less than the cooling capacity of the spent fuel pool cooling system.

The licensee did not identify any changes to the plant-specific provisions of GL 89-10 and GL 96-05, related to motor operated valves, GL 95-07, related to pressure locking and thermal binding of safety-related gate valves, or GL 96-06, related to over-pressurization of isolated piping segments. The licensee determined that BOP valves are not affected by the uprate because the slight changes in operating parameters (temperatures, pressures, and flow rates) are within the design limits and do not effect maximum pressure or temperature differentials. The NRC staff does not anticipate any changes to the analysis of overpressurization of isolated piping segments because the analysis of record for containment temperature and pressure was performed at 102 percent of current RTP and remains bounding for the uprate conditions. Therefore, the NRC staff does not expect any changes to the plant-specific provisions of GL 89-10, GL 96-05, GL 95-07, or GL 96-06.

The licensee concluded that the Palisades BOP piping systems remain acceptable for operation at the uprated conditions. Based on the above, the NRC staff agrees with the licensee's conclusion that the proposed 1.4 percent power uprate will not have adverse effects on BOP systems or including safety-related valves.

3.4.3 Summary

On the basis of its review described above, the NRC staff finds that the proposed power uprate will not have an adverse impact on the structural integrity of the piping systems, components, their supports, reactor internals, core support structures, CRDMs, BOP piping, or safety-related valves.

3.5 Dose Consequences Analysis

3.5.1 Regulatory Evaluation

The NRC staff review covers the impact of the proposed MUR power uprate on the results of dose consequence analyses (NRC RIS 2002-03, Attachment 1, Sections II and III). The review is conducted to verify that the results of the licensee's dose consequence analyses continue to meet the acceptance criteria in 10 CFR Part 100, and/or 10 CFR Part 50, Appendix A, GDC 19, as applicable, following implementation of the proposed MUR power uprate.

RIS 2002-03, "Guidance on the Content of Measurement Uncertainty Recapture Power Uprate Applications," recommends that to improve efficiency of the NRC staff's review, licensees requesting an MUR uprate should identify existing DBA analyses of record which bound plant operation at the proposed uprated power level. For any DBA for which the existing analyses of record do not bound the proposed uprated power level, the licensee should provide a detailed discussion of the reanalysis.

This SE section addresses the impact of the proposed changes on previously analyzed DBA radiological consequences. As with any license amendment, the licensee must show, and the NRC staff must find acceptance, that the plant continues to meet dose limit criteria given in 10 CFR Part 100.11 for offsite doses, 10 CFR Part 50, Appendix A, GDC 19 (or equivalent for plants licensed before the GDC were in existence) with respect to control room habitability. Except where the licensee proposed a suitable alternative, in performing this review, the NRC staff utilized the regulatory guidance provided in applicable sections of NUREG-0800, SRP, Chapter 15, for DBAs and SRP Chapter 6.4 for control room habitability.

3.5.2 Technical Evaluation

The NRC staff reviewed the regulatory and technical analyses, as related to the radiological consequences of DBAs, performed by NMC in support of its proposed license amendment. Information regarding these analyses was provided in Section II of the licensee's June 3, 2003, submittal.

The NRC staff reviewed the impact of the proposed 1.4 percent power uprate on DBA radiological analyses, as documented in Chapter 14 of the Palisades FSAR. In its submittal, NMC stated that the current radiological analyses of record for Palisades were unaffected by the requested power uprate, because they were performed assuming a nominal core power of 2580.6 MWt. Analyses performed at this power bound analyses performed assuming the requested uprated power of 2565.4 MWt with a 0.6-power measurement uncertainty. Using the current Palisades FSAR documentation in addition to information in the June 3, 2003, submittal, the NRC staff verified that the existing Palisades FSAR Chapter 14 radiological analyses source term and steam release assumptions, as appropriate, bound the conditions for the proposed 1.4 percent power uprate to 2565.4 MWt.

3.5.3 Summary

Based on the above discussion, the NRC staff finds that the existing Palisades FSAR Chapter 14 radiological analyses, which were analyzed assuming a core thermal power of 2580.6 MWt, remain bounding for the proposed 1.4 percent power uprate to 2565.4 MWt, considering the higher accuracy of the FW measurement instrumentation. These analyses of record show that the radiological consequences of postulated DBAs meet the dose limits given in 10 CFR 100.11 and 10 CFR Part 50, Appendix A, GDC 19, as well as applicable dose acceptance criteria given in NUREG-0800, SRP Chapter 15. Therefore, the MUR increasing the RTP by 1.4 percent from 2530 MWt to 2565.4 MWt is acceptable with regard to the radiological consequences of postulated DBAs.

3.6 Materials and Chemical Engineering

3.6.1 Regulatory Evaluation

The NRC staff's review in the area of materials and chemical engineering covers the effects that the proposed MUR power uprate would have on licensee programs for addressing SG tube degradation mechanisms, erosion/corrosion, and other NSSS systems. This review is conducted to verify that the implementation of the proposed MUR power uprate satisfies 10 CFR 50.55 and 10 CFR Part 50. Additional guidance for the NRC staff's review of the topics within the materials and chemical engineering area include the guidance contained in Chapters 4, 5, and 6 of NUREG-0800. In this section, the NRC staff also evaluates the end-of-license peak vessel neutron fluence and its effects on pressurized thermal shock (PTS), pressure temperature limit curves, and low temperature overpressure protection. RG 1.190 delineates acceptable methods for calculating pressure vessel fluence. However, the licensee asserts that the peak vessel fluence for Palisades after the proposed power uprate is bounded by the value of record, due to an extended outage in the year 2001.

3.6.2 Technical Evaluation

3.6.2.1 Integrity of the Reactor Vessel, Internals, and Welding

In this section, the NRC staff evaluates the impact of the MUR uprate with respect to factors affecting the integrity of the reactor vessel, its internals, and welds. These factors include the pressure-temperature (P-T) limits, upper shelf energy (USE) analysis, PTS, reactor vessel (RV) internal support components and the RV surveillance program.

The NRC staff's requirements for generating P-T limit curves are given in 10 CFR Part 50, Appendix G, "Fracture Toughness Requirements." Section IV.A.2 of 10 CFR Part 50, Appendix G, requires that the P-T limits for the RV of a light-water reactor must be at least as conservative as the P-T limit generation methods of Appendix G to Section XI of the ASME Boiler and Pressure Vessel Code. For materials in the beltline region of the vessel, the rule requires the calculations of P-T limits to take into account the effects of neutron irradiation on the reference temperatures for nil ductility (i.e., RT_{NDT} values) for the materials used to fabricate the RV and to incorporate any relevant RV surveillance capsule data that are required to be reported as part of the licensee's implementation of its 10 CFR Part 50, Appendix H, RV material surveillance program. Additional guidance for the NRC staff's review of RV P-T limit curves is provided in Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," SRP Section 5.3.2, "Pressure-Temperature Limits," and Branch Technical Position MTEB 5-2, "Fracture Toughness Requirements."

The NRC staff's requirements for USE are given in 10 CFR Part 50, Appendix G. Section IV.A.2. of 10 CFR Part 50, Appendix G, requires that the RV beltline materials have a minimum

USE value of 75 ft-lb. in the unirradiated condition, and to maintain a minimum USE value above 50 ft-lb. throughout the life of the facility, unless it can be demonstrated through analytical engineering analyses (i.e., through equivalent margins analyses) that lower values of USE would provide acceptable margins of safety against fracture equivalent to those required by Appendix G of Section XI to the ASME Boiler and Pressure Vessel Code. For materials in the beltline region of the vessel, the rule requires USE calculations to account for the effects of neutron irradiation on the USE values for the materials and to incorporate any relevant RV surveillance capsule data that are required to be reported as part of the licensee's implementation of its 10 CFR Part 50, Appendix H, RV material surveillance program. Additional guidance for the NRC staff's review of USE analyses is provided in RG 1.99, Revision 2, SRP Section 5.3.2, and Branch Technical Position MTEB 5-2. Appendix K to Section XI of the ASME Code and Regulatory Guide 1.161, "Evaluation of Reactor Pressure Vessels with Charpy Upper-Shelf Energy Less Than 50 Ft-Lb." may also be used as guidance when USE equivalent margins analyses are required.

The NRC staff's requirements for protecting the RVs of PWRs against PTS events are stated in 10 CFR 50.61, "Requirements for Assuring RV Integrity Against PTS Events." The rule requires RV materials made of carbon or low-alloy steel materials to meet a maximum screening criterion for nil-ductility reference temperatures (i.e., RT_{PTS} values). The rule's screening criteria are 270 °F for axial weld materials and base metal materials (i.e., plates or forging materials) and 300 °F for circumferential weld materials. The rule provides methods for calculating these RT_{PTS} values. For materials in the beltline region of the vessel, the rule requires the calculations to take into account the effects of neutron irradiation on the RT_{PTS} values for the materials and to incorporate any relevant RV surveillance capsule data that are required to be reported as part of the licensee's implementation of its 10 CFR Part 50, Appendix H, RV material surveillance program.

The NRC's regulatory requirements related to the establishment of a facility's RV surveillance capsule program and withdrawal schedule are given in Appendix H to 10 CFR Part 50, which also references the guidance in American Society for Materials and Testing Standard Practice E 185, "Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels." SRP Section 5.3.1, "Reactor Vessel Materials," also applies.

Maintenance of the structural integrity of the RV internals is required in order to demonstrate that the functional requirements of the RV internals are met. These functional requirements include core support and ECCS performance aspects. As such, the structural integrity of the RV internals is linked to regulatory requirements in 10 CFR 50.46 regarding ECCS performance and maintaining a coolable core geometry. Additional guidance regarding the evaluation of the structural integrity of RV internals may be found in SRP Section 3.9.3, "ASME Code Class 1, 2, and 3 Components, Component Supports, and Core Support Structures," and in ASME Code Sections III and XI, or other standards which were used in the NRC's review of the original licensing basis of a particular facility.

The NRC staff has reviewed the licensee's application as related to the materials and chemical engineering areas discussed above and determined that the existing analyses of record bound the proposed operation of the plant at the uprated power level. The results of the NRC staff's review for the areas discussed above of the materials and chemical engineering scope are summarized in Table 3.6.1 below.

Table 3.6.1
Materials and Chemical Engineering - Summary of NRC Staff Review

Topic	Application Section and Page Number	UFSAR Section	Bounded by NRC-approved Analysis (Y/N and Reference)	NRC Staff Conclusion
Vessel & Internals Integrity and Welding				
10 CFR Part 50 Appendix G – P-T Limits	IV. 1. C. iii (page 23)	4.4.2	Y Reference 1	Acceptable
10 CFR Part 50 Appendix G - USE	IV. 1. C. v (page 24)	4.4.2	Y Reference 2	Acceptable
10 CFR 50.61 PTS Events	IV. 1. C. i (Page 22)	4.4.2	Y Reference 2	Acceptable
10 CFR Part 50 Appendix H RPV Surveillance Program	IV. 1. C. vi (Pages 24 - 25)	4.5.3	Y Reference 2	Acceptable
Structural Integrity of Control Rod Drive Mechanism Nozzles	IV. 1. A. iii (Page 13)	4.2.4	Y References 3 & 4	Acceptable
Structural Integrity of RV Internals	IV. 1. A. ii (Page 13)	3.2.3	Y References 3 & 5	Acceptable
Structural Integrity of the Reactor Coolant Pump Flywheels	IV. 1. E. iii (Page. 26)	4.3.5	Y Reference 6	Acceptable

Table 3.6.1 References:

1. Palisades License Amendment No. 163, dated March 2, 1995 [Approved Pressure-Temperature Limits].

2. NRC letter by Hood to Haskell, “Palisades Plant - Reactor Vessel Neutron Fluence Evaluation and Revised Schedule for Reaching Pressurized Thermal Shock Screening Criteria (TAC No. MA8250),” dated November 14, 2000.

3. ASME Boiler and Pressure Vessel Code, Section III, 1965 Edition through and including the 1965 Winter Addenda.

4. Palisades Updated Final Safety Analysis Report, Revision 23, Section 4.2.4, dated October 30, 2001.

5. Palisades Updated Final Safety Analysis Report, Revision 23, Section 3.2.3, dated October 30, 2001.

6. Palisades License Amendment No. 182, “Primary Coolant Pump Flywheel Inspection Technical Specification,” dated May 15, 1998.

With respect to P-T limit curves, the current curves for heatup and cooldown are given in Palisades TS Figures 3.4.3-1 and 3.4.3-2, respectively, as shown for License Amendment No. 189. As discussed elsewhere in this section, the neutron fluence at the end of the license with the proposed power uprate is bounded by the value established in the current fluence evaluation which, in turn, is bounded by the fluence value used in determining the heatup and cooldown PT limit curves. Therefore, the NRC staff finds that the current curves continue to be sufficient for proposed power uprate.

With respect to PTS events, the NRC staff previously approved revised neutron fluence values and the PTS assessment for Palisades by letter dated November 14, 2000. The licensee’s

MUR Power Uprate Analysis Report assesses the impact of the power uprate on the neutron fluence values for the RV materials as a function of the impact the increase in fluence values will have on the effective full power days for the unit. This assessment indicates that the fluence values used in latest PTS assessment bounds the sight increase to the fluence values assumed for the MUR power uprate. Therefore, the most up-to-date PTS evaluation for Palisades is still valid even for the uprated conditions for the plant. The NRC staff notes that, for PTS, the Palisades reactor vessel is limited by the axial welds made from heat number W5214, which have a RT_{PTS} value of 266 °F. This value is 4 °F within the screening criteria in 10 CFR 50.61 for axial welds and base metal materials, which is 270 °F. 10 CFR 50.61(b) requires licensees to update the PTS assessment whenever there is a significant (changes to the PTS values are considered significant if either the previous value or current value, or both values, exceed the screening criterion prior to the expiration of the operating license) change in the projected values of the PTS, or upon request for a change in the expiration date for operation of the facility.

Regarding the structural Integrity of the RV internals, the licensee stated that the actual increase in power to the proposed uprate level will be accomplished by the additional opening of the turbine throttle valves. The resulting increase in steam flow will cause the temperature difference across the core to increase. However, at the proposed uprated power level, the RCS pressure, RCS average temperature, and RCS flow rate will be no different from those for the current full power level. The NRC staff has reviewed the information provided by the licensee and concurs with the licensee's conclusion. Since the power uprate will not result in a change to the RCS pressure and RCS temperature, the NRC staff concludes that the proposed power will not impact the structural integrity assessments for the RV internals

3.6.2.2 Structural Integrity Requirements for CRDM and Vessel Head Penetration Nozzles

Maintenance of the leakage and structural integrity of the CRDM nozzles is directly related to requirements in Appendix A to 10 CFR Part 50, GDC 14, or, for those plants licensed prior to the development of Appendix A to 10 CFR Part 50, similar requirements which were imposed during the NRC staff's review of the facility's operating license. Guidance regarding an acceptable evaluation of the potential for primary water stress corrosion cracking (PWSCC) and the performance of appropriate inservice inspections has been provided in NRC Bulletin 2002-02, "Reactor Pressure Vessel Head and Vessel Head Penetration Nozzle Inspection Programs," and Order EA-03-009, which established interim inspection requirements for PWR reactor pressure vessel heads.

The licensee's MUR Power Uprate Analysis Report lists the principal plant parameters at the current and uprated power levels, and does not indicate any changes to the RCS temperature and pressure due to the power uprate. Without any change in these parameters, the NRC staff concludes that the proposed power uprate will have negligible effects on crack growth rates or postulated flaws in J-groove welds of nozzles. In addition, new inspection requirements of Order EA-03-009 provide for additional assurance of the structural integrity of the nozzles.

3.6.2.3 Structural Integrity Requirements for RCP Flywheels

Maintenance of the structural integrity of RCP flywheels is important to assure the RCP flywheels can maintain a continuous coastdown from 120 percent of the design rotor speed for the RCP impellers and to preclude the potential for missile generation as a result of their failure.

The evaluation of RCP flywheel integrity is related to Appendix A to 10 CFR Part 50, GDC 1 and GDC 4, or, for those plants licensed prior to the development of Appendix A to 10 CFR Part 50, similar requirements which were imposed during the NRC staff's review of the facility's operating license. The NRC staff concludes that the proposed power uprate will have negligible effects on the structural integrity of the RCP flywheel.

3.6.2.4 Reactor Pressure Vessel Neutron Fluence and Its Effects

The Palisades peak vessel fluence to the end of the current license was approved on November 14, 2000. Instead of revising fluence calculation, the licensee proposes to increase the fluence by the power uprate amount of 1.4 percent, which is a reasonable assumption. The licensee calculated the number of full power days to be added to the existing full power days to the end of the current license. However, in 2001, the plant had an extended maintenance outage. If the excess maintenance days are subtracted from the power uprate full power days, the remaining full power days are lower than those approved on November 14, 2000. Thus, the value of the vessel fluence at the end of the current license with the power uprate, will be lower than the value calculated without the power uprate. Therefore, the current value is conservative, and therefore, is acceptable.

Because the value of the fluence remains unchanged, the PTS screening temperature (RT_{PTS}), the P-T limit curves, and the low temperature overpressure protection limits remain unchanged.

3.6.2.5 Steam Generator Structural Integrity Evaluation

RG 1.121, "Bases for Plugging Degraded PWR SG Tubes," describes an acceptable method for establishing the limiting safe condition of degradation in the SG tubes, beyond which the tubes found defective by the established inservice inspection shall be removed from service. There are three factors that should be considered for a SG operational limit: (1) the minimum tube wall thickness needed in order for tubes with defects to sustain the imposed loading under normal operating conditions and postulated accident conditions, (2) an operational allowance for degradation between inspections, and (3) the crack size permitted to meet the leakage limit allowed per SG by the plant TS. RG 1.121 states that: "...Calculations and analytical procedures and the operational history of the SG are used to arrive at the minimum acceptable tube wall thickness and thus are the basis for defining the plugging criteria..." The level of acceptable degradation is referred to as the repair limit. The allowable tube repair limit, in accordance with RG 1.121, is obtained by incorporating into the resulting structural limit an allowance for continued growth of the flaw and an allowance for eddy current measurement uncertainty.

The licensee identified six active damage mechanisms in the Palisades SGs during the 2003 refueling outage which were evaluated based on the Palisades SG program. These degraded areas of the tubes were inspected and monitored each refueling outage. The NRC staff found a discrepancy in the original submittal in which NMC indicated that circumferential outside diameter stress corrosion cracking (ODSCC) at the hot-leg top of the tubesheet is not affected by the slight increase in T_{hot} until the end of the license. This part of the original application was eliminated, and in its letter dated October 6, 2003, the licensee replaced it with the following statement: "...T_{hot} at Palisades of 582.7 ^{o}F is low for Alloy 600 tubing per existing industry experience, and a 0.3 ^{o}F increase is expected to have a negligible effect on these new active mechanisms. The greatest effect will be seen on mechanical tube wear..." In its October 6, 2003, letter, the licensee also stated that ODSCC at the hot-leg top of the tubesheet and axial PWSCC within the expanded tubesheet region were identified as new active damage mechanisms during the last refueling outage. The licensee indicated that these damage mechanisms are affected by time and temperature (i.e., T_{hot}), but not by an increase in secondary side steam flow.

The licensee indicated that it will continue its SG program to monitor and maintain SG tube integrity for the proposed power level conditions. The NRC staff has reviewed the licensee's assessment of the impact of the proposed MUR power uprate on SG tube integrity. Based on the above, the NRC staff concludes that the licensee has adequately addressed these impacts and has demonstrated that the plant will continue to meet the requirements of 10 CFR Part 50 following implementation of the proposed Palisades MUR power uprate. Therefore, the NRC staff finds the proposed MUR power uprate acceptable with respect to SG tube integrity.

3.6.2.6 Steam Generator Blowdown System

Control of secondary side water chemistry is important for preventing degradation of SG tubes. The SGBS provides a means for removing SG secondary-side impurities and thus assists in maintaining acceptable secondary side water chemistry in the SGs. The design-basis of the SGBS includes consideration of expected and design flows for all modes of operation. The NRC staff's review covers the ability of the SGBS to remove particulate and dissolved impurities from the SG secondary side during normal operation and anticipated operational occurrences. The NRC's acceptance criteria for the SGBS are based on GDC-14 for secondary water chemistry control to ensure the integrity of reactor coolant pressure boundary material.

At Palisades the SG blowdown is manually controlled using a throttle valve that will keep the appropriate specified flow rate. The licensee calculated that the SG operating pressure at the proposed power uprate is slightly lower than the current operating pressure. The slightly lower pressure will result in the slight increase in the SGBS flowrate; however, the slight increase in the flow rate will have minimum impact in the system and will not affect the proposed analyzed requirements for the proposed power level increase. The design-basis of SGBS is described in the Palisades FSAR Chapter 4.

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

3.6.2.7 Flow Accelerated Corrosion

Flow accelerated corrosion (FAC) is a corrosion mechanism occurring in carbon steel components exposed to single or two phase fluid in a piping system. The components made from stainless steel are less likely to be affected by FAC, which is significantly reduced in components made from steel containing small amounts of chromium or molybdenum. The rates of material loss by FAC depend on velocity of flow, temperature, steam quality, oxygen content, and pH of the coolant. During plant operation, control of these parameters is limited and the optimum conditions for minimizing FAC effects, in most cases, cannot be achieved. Loss of material by FAC may, therefore, occur. The licensee has implemented a program to monitor FAC. The licensee's FAC program is based on NUREG-1344, GL 89-08, and the guidelines in the Electric Power Research Institute (EPRI) report NSAC-202L-R2. It consists of predicting loss of material using the CHECWORKS computer code, visual inspection, and volumetric examination of the affected components.

The NRC staff reviewed the effects of the proposed MUR power uprate on FAC and the adequacy of the licensee's FAC program to predict the rate of material loss so that repair or replacement of damaged components could be made before they reach critical pipe thickness. The NRC staff's evaluation is based on the structural evaluation of the minimum acceptable wall thickness for the components undergoing degradation by FAC specified in EPRI report, Recommendations for an Effective Flow Accelerated Corrosion Program (April 1999).

The proposed power uprate FAC analysis for Palisades involved an increase in the flow rates of the MS, main FW, CD, heater drain, and extraction systems. CHECWORKS software predicted a corrosion wear rate for the piping of the mentioned systems to be within the wear rate scope for the licensee's FAC program. The licensee has demonstrated that the wear rates under the power rate conditions are within the acceptable limits for critical pipe wall thickness as specified in FSAR, Section 10.4.1, Pipe Wall Thinning Inspection Program.

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

3.6.2.8 Other NSSS Systems

3.6.2.8.1 Chemical and Volume Control System

The CVCS and boron recovery system (BRS) provide means for (a) maintaining the required water inventory and quality in the RCS, (b) supplying seal water flow to the RCP and pressurized auxiliary spray, (c) controlling the boron neutron absorber concentration in the reactor coolant, (d) controlling the primary water chemistry and reducing coolant radioactivity level, and (e) supplying recycled coolant for demineralized water makeup for normal operation and high pressure injection flow to the ECCS in the event of postulated accidents.

The analyzed conditions for the proposed power level of 2565.4 MWt were obtained based on an analysis of 2650 MWt in the original design of the primary systems. The primary coolant system activity is less during normal operation than for the proposed power level. This causes a slight increase in the primary coolant system activity in the power uprated condition, but the increase will be controlled by the CVCS by keeping the primary coolant activity constant.

The licensee's analysis also demonstrated that the primary coolant inventory will not be affected by the proposed increase in power level. In its June 3, 2003, submittal, the licensee stated that:

“...Since neither the primary coolant system inventory, the primary coolant system pressure, the range for T_{ave}, nor the allowable rates for power changes heatup, or cooldown are changing, charging and letdown flows will remain at their current values and the ability of the chemical and volume control system to maintain primary coolant system volume will not be affected by the proposed power uprate...” The NRC staff agrees with the licensee’s assessment.

The licensee also analyzed the BRS based on a power output of 2650 MWt and stated that the CVCS is capable of adding sufficient boric acid to maintain the shutdown margin required during a primary system cooldown following a reactor trip. Based on the system design, the charging pumps will supply sufficient boric acid to ensure sufficient negative reactivity to compensate for reactivity changes. The value of T_{cold} for the proposed power uprate value is slightly lower than the current power level; therefore, the effects of the proposed higher power level on the regenerative or letdown heat exchangers are negligible and the design pressures and temperatures will continue to bound.

The NRC staff has reviewed the licensee’s evaluation of the effects of the proposed MUR power uprate on the CVCS and BRS and concludes that the licensee has adequately addressed changes in the temperature of the reactor coolant and their effects on the CVCS and BRS. The NRC staff further concludes that the licensee has demonstrated that the CVCS and BRS will continue to be acceptable because they will continue to meet the requirements of GDC-14 and bounded by Section 5.1 of the updated FSAR.

3.6.2.8.2 Low Pressure Safety Injection

The licensee analyzed the low pressure SI system based on 2580.6 MWt, a 102 percent of the current power level value of 2530 MWt. For the proposed power uprate the decay heat load is higher than for the current power level. The higher decay heat load could cause a slight delay in the initiation of shutdown cooling and a slight increase in the system’s load. However, the initiation and operation of shutdown cooling will still be controlled by the limits on primary coolant system temperature and pressure. The pressure in the primary coolant system under the power uprated condition is the same as under the current power level. An increase of 0.3 ^{o}F in primary coolant temperature (T_{hot}) is negligible. Therefore, the low pressure SI system will not be affected by the power uprate.

The NRC staff has reviewed the licensee’s evaluation of the effects of the proposed MUR power uprate on the low pressure SI and concludes that the licensee has adequately addressed changes in the primary pressure and their effects on the low pressure SI. The NRC staff further concludes that the licensee has demonstrated that the low pressure SI will continue to be acceptable and will continue to meet the requirements of GDC-14.

3.6.3 Summary

The NRC staff has reviewed the licensee’s assessment of the impact of the proposed MUR power uprate on the RV integrity, integrity of RCP flywheels, SG tube integrity, erosion/corrosion program, the effects of end-of-license vessel neutron fluence, FAC program, and other NSSS systems as presented in the licensee’s request. Based on the above evaluation, the NRC staff concludes that the licensee has adequately addressed these impacts and has demonstrated that the plant will continue to meet the applicable requirements following implementation of the proposed MUR power uprate. Therefore, the NRC staff finds the proposed MUR power uprate acceptable with respect to the materials and chemical engineering issues discussed above

3.7 Human Factors

3.7.1 Regulatory Evaluation

The area of human factors deals with programs, procedures, training, and plant design features related to operator performance during normal and accident conditions (NRC RIS 2002-03, Attachment 1, Section VII, Items 1 through 4). The NRC staff's human factors evaluation is conducted to confirm that operator performance will not be adversely affected as a result of system changes required for the proposed MUR power uprate. The NRC staff's review covers licensee's plans for addressing changes to operator actions, human-system interfaces, and procedures and training required for the proposed MUR power uprate. The NRC's acceptance criteria for human factors are based on 10 CFR 50.54(i) and (m), 10 CFR 50.120, 10 CFR 55.59, and GDC 19.

3.7.2 Technical Evaluation

The NRC staff has developed a standard set of questions for the review of the human factors area. The licensee has addressed these questions in its June 3, 2003, application and supplements. Following is a summary of the licensee's responses and the NRC staff's conclusions.

3.7.2.1 Operator Actions

The licensee indicated that the proposed MUR power uprate is not expected to have any significant affect on the manner in which the operators control the plant during normal operations or transient conditions. The licensee also indicated that all operator actions that were taken credit for in the safety analyses would still be valid following implementation of the proposed MUR power uprate. The NRC staff finds the implementation of the proposed MUR power uprate at the Palisades will not have an adverse effect either on operator actions or safe operation of the facility.

3.7.2.2 Emergency and Abnormal Operating Procedures

The licensee indicated that there are currently no Emergency Operating Procedures or Abnormal Operating Procedures that need to be changed as a result of the 1.4 percent MUR power uprate. However, the licensee stated that plant procedures will be revised to address operation with the Crossflow ultrasonic flow measurement system out of service (See "Summary of Commitments," of the licensee's June 3, 2003, application). Based on the above, the NRC staff finds that necessary procedures will be changed or updated prior to the implementation of the license and TSs changes associated with the proposed MUR power uprate. The NRC staff finds this acceptable.

3.7.2.3 Control Room Controls, Displays, and Alarms

The Crossflow system is not connected to the plant process computer (PPC). The Crossflow system is used to determine best estimate FW flow in support of calculating correction factors that are manual inputs into the PPC and used to correct the FW flow as measured by venturi. Control room controls and alarms are unaffected by the proposed uprate. The control room indications for various plant parameters may change slightly, but will remain within the range of existing instrumentation. No changes to alarms or operator response to alarms are expected. Operation at the proposed increased power conditions will have a negligible effect on the displays of the PPC. The licensee will provide operator training to the necessary plant staff on the proposed power uprate prior to implementation of the proposed power uprate (See "Summary of Commitments," of the licensee's June 3, 2003, application). The NRC staff finds this acceptable.

3.7.2.4 Control Room Plant Reference Simulator

The licensee stated that plant administrative procedures will ensure that the necessary changes to the control room simulator will be made prior to implementation of the proposed power uprate. The NRC staff finds this acceptable.

3.7.2.5 Operator Training Program

The licensee stated that plant administrative procedures will ensure that the necessary changes to the operator training program will be made prior to implementation of the proposed power uprate. The licensee committed to provide operator training to the necessary plant staff on the proposed power uprate prior to implementation of the proposed power uprate (See "Summary of Commitments," of the licensee's June 3, 2003, application). The NRC staff finds this acceptable.

3.7.3 Summary

The NRC staff has reviewed the licensee's planned actions related to the human factors area and concludes that licensee has adequately considered the impact of the proposed MUR power uprate on changes to operator actions, procedures, plant hardware, and associated training programs to ensure that operators' performance is not adversely affected by the proposed MUR power uprate. The NRC staff further concludes that the licensee will continue to meet the requirements of 10 CFR 50.54(i) and (m), 10 CFR 50.59, 10 CFR 50.120, and 10 CFR 55.59 following implementation of the proposed MUR power uprate. Therefore, the NRC staff finds the proposed MUR power uprate acceptable with respect to the human factors aspects of required system changes.

3.8 <u>Plant Systems</u>

3.8.1 Regulatory Evaluation

The NRC staff review in the area of plant systems covers the impact of the proposed MUR power uprate on (1) containment performance analyses and containment systems, (2) safe shutdown fire analyses and required systems, (3) spent fuel pool cooling analyses and systems, (4) flooding analyses, (5) NSSS interface systems, (6) radioactive waste systems,

(7) engineered safety feature (ESF) heating, ventilation, and air conditioning systems, and (8) safety-related cooling water systems (NRC RIS 2002-03, Attachment 1, Sections II, III, and VI). The review is conducted to verify that the licensee's analyses bound the proposed plant operation at the MUR power level and that the results of licensee analyses related to the areas under review continue to meet the applicable acceptance criteria following implementation of the proposed MUR power uprate. Guidance for the NRC staff's review of plant systems is contained in Chapters 3, 6, 9, 10, and 11 of NUREG-0800.

3.8.2 Technical Evaluation

The NRC staff has developed a standard set of questions for the review of the plant systems area. The licensee has addressed these questions in its June 3, 2003, application. Following is a summary of the licensee's responses and the NRC staff's conclusions.

3.8.2.1 Containment Performance Analyses and Containment Systems

The licensee is not making changes to the containment structure or containment isolation systems as part of the MUR power uprate. The containment response for a MS line break (MSLB) was performed by the licensee at its current power of 2530 MWt with 2 percent uncertainty or 2580.6 MWt. The licensee's current LOCA containment integrity analysis is based on 102 percent of the current licensed power of 2530 MWt or 2580.6 MWt. Both of these analyses bound operation at the MUR uprated power of 2565.4 MWt. The NRC staff finds this acceptable.

The licensee evaluated the containment air cooler system for a 1.4 percent power uprate or 2565.4 MWt. For normal operation with a core power of 2565.4 MWt, the licensee determined that the uprate would have an insignificant effect on the amount of heat that must be removed by the containment air coolers during normal operation. The loads on the equipment located inside containment that produces heat, such as motors or transformers, would not change, There would be a negligible change in the losses from the primary coolant system piping since the full power hot-leg temperature will increase slightly while the cold-leg temperature will decrease slightly. Likewise, the full power FW piping temperature will increase slightly while the MS temperature will decrease slightly. The licensee stated that these changes are insignificant and expected to be within the existing capacity of the coolers and the associated service water system. The NRC staff finds this acceptable. For accident conditions, the licensee's current analysis for LOCA containment integrity and the MSLB containment response has been performed at 102 percent of 2530 MWt or 2580.6 MWt. This bounds the 1.4 percent power uprate of 2565.4 MWt; therefore, the NRC staff finds containment performance analyses and containment systems acceptable for the 1.4 percent MUR power uprate.

3.8.2.2 Safe Shutdown Fire Analyses and Required Systems

The licensee's current safe shutdown fire analysis is performed at 2530 MWt with a 2 percent uncertainty added or 2580.6 MWt. This bounds the 1.4 percent power uprate of 2565.4 MWt; therefore the NRC staff finds the safe shutdown fire analyses and required systems acceptable for the 1.4 percent MUR power uprate.

3.8.2.3 Spent Fuel Pool Cooling Analyses and Systems

The licensee's current spent fuel pool cooling analysis is performed at 2530 MWt with a 2 percent uncertainty added or 2580.6 MWt. This bounds the 1.4 percent power uprate of 2565.4 MWt; therefore, the NRC staff finds the spent fuel pool cooling analyses and systems acceptable for the 1.4 percent MUR power uprate.

3.8.2.4 Flooding Analyses

The licensee's current flooding study does not depend on power level; therefore, the licensee's current flooding study, approved in NRC letter from Wambach (NRC) to VandeWalle (Consumers), "Supplement to the Integrated Plant Safety Assessment Report (IPSAR) for the Palisades Plant," dated November 7, 1983, bounds the 1.4 percent MUR power uprate. The NRC staff finds the flooding analyses acceptable for the 1.4 percent power uprate.

3.8.2.5 NSSS Interface Systems

The licensee performed a evaluation of a 1.4 percent power uprate for the NSSS interface systems. The licensee evaluated the following BOP fluid systems for NSSS/BOP interface: MS, SGBD, CD and FW system, and auxiliary feedwater (AFW) system.

The licensee performed an analysis of the MS system at the 1.4 percent power uprate conditions. The licensee concluded that the installed safety valve capacity is adequate. The licensee's evaluation of the capacity of the atmospheric dump valves concluded that the combined rated capacity of the atmospheric dump valves remains at 30 percent of steam flow with reactor at full power, as stated in the FSAR. The licensee determined that the design of the MS isolation valves, and associated pipe loads are not impacted by the power uprate. The NRC staff finds the MS system acceptable for the 1.4 percent MUR power uprate.

The licensee performed an analysis of the SGBD system at the 1.4 percent power uprate conditions. The licensee stated that SGBD is manually controlled by having an operator throttle the appropriate valves to achieve a specified flow rate, which is procedurally controlled to remain within analyzed conditions. At the proposed uprate power level, the SG operating pressure will be slightly lower than the current full power SG operating pressure. This means the operator may have to simply open the valves slightly more than at present to achieve a specified flow rate, but will not impact analyzed requirements. The licensee determined SGBD is not impacted by the power uprate. The NRC staff finds the SGBD system acceptable for the 1.4 percent MUR power uprate.

The licensee performed an analysis of the CD and FW systems at the 1.4 percent power uprate conditions. The licensee evaluated the CD and FW systems piping, pumps, valves, and pressure-retaining components to ensure their ability to operate at the increased flow rates, temperatures, and pressures associated with the power uprate. The licensee determined that the CD and FW systems are not impacted by the power uprate. The NRC staff finds the CD and FW systems acceptable for the 1.4 percent MUR power uprate.

For the AFW system, the licensee's states that the design-basis auxiliary FW flow is established by the loss of normal FW event which was performed at 2530 MWt with a 2 percent uncertainty added or 2580.6 MWt. This bounds the 1.4 percent power uprate of 2565.4 MWt.

In addition, the licensee states that the CD storage and primary makeup storage tanks are designed assuming a reactor trip from 102 percent current RTP. This also bounds the 1.4 percent power uprate of 2565.4 MWt; therefore, the NRC staff finds the AFW system acceptable for the 1.4 percent MUR power uprate.

3.8.2.6 Radioactive Waste Systems

The licensee stated that the functioning of the liquid radioactive waste system is not impacted by the power uprate and that the current gaseous radioactive waste system accident analysis is performed at 2530 MWt with a 2 percent uncertainty added or 2580.6 MWt. This bounds the 1.4 percent power uprate of 2565.4 MWt; therefore, the NRC staff finds the radioactive waste systems acceptable for the 1.4 percent MUR power uprate.

3.8.2.7 ESF Heating, Ventilation, and Air Conditioning Systems

The licensee stated that the 1.4 percent proposed power uprate results in no additional equipment and no additional loading of existing equipment in the engineered safeguards equipment rooms, control room, emergency diesel generator rooms, fuel handling area, or the electrical equipment, switchgear, cable spreading and battery rooms. The licensee states that the proposed power uprate will have no impact on the ventilation systems for these areas and therefore, the ESF heating, ventilation, and air conditioning systems (HVAC) are not affected by the proposed power uprate. The NRC staff finds the ESF HVAC systems acceptable for the 1.4 percent MUR power uprate.

3.8.2.8 Safety-Related Cooling Water Systems

The licensee performed evaluations of the following safety-related cooling water systems at the 1.4 percent power uprate conditions or 2565.4 MWt: component cooling water (CCW) system, service water (SW) system, and low pressure SI system.

At the Palisades, the CCW system provides a heat sink for the removal of process and operating heat from safety-related components following a design-basis event. The licensee states that the accident analyses for the LOCA demonstrates that the system is capable of performing this function. The licensee’s current LOCA analysis is performed at 2530 MWt with a 2 percent uncertainty added or 2580.6 MWt. This bounds the 1.4 percent power uprate of 2565.4 MWt. In addition, the system also provides a heat sink for the removal of process and operating heat from various non-safety related components during normal operation. The licensee performed an analysis of CCW operation with a 1.4 percent increase in decay heat. The licensee stated that the initiation of shutdown cooling will still be controlled by the limits on primary coolant system temperature and pressure. The licensee also states that there will be no change in the temperatures, pressures, or flow rates experienced by the low pressure SI system during plant shutdown and, therefore, no change in the temperatures, pressures, or flow rates experience by the CCW system during plant shutdown, other than a slight increase in the actual cooldown time. The NRC staff finds the CCW system acceptable for the 1.4 percent MUR power uprate.

The licensee performed an analysis of the SW system and determined that the current analysis is bounding for the power uprate. The NRC staff finds the SW cooling water system acceptable for the 1.4 percent MUR power uprate.

The licensee performed an analysis of the low pressure SI system and determined that the current analysis is bounding for the power uprate. The NRC staff finds the low pressure SI system acceptable for the 1.4 percent MUR power uprate.

3.8.3 Summary

The NRC staff has reviewed the licensee's safety analyses of the impact of the proposed MUR power uprate on (1) containment performance analyses and containment systems, (2) safe shutdown fire analyses and required systems, (3) spent fuel pool cooling analyses and systems, (4) flooding analyses, (5) NSSS interface systems, (6) radioactive waste systems, (7) ESF HVAC systems, and (8) safety-related cooling water systems. The NRC staff concludes that the results of licensee's analyses related to these areas would continue to meet the applicable acceptance criteria following implementation of the proposed MUR power uprate. Therefore, the NRC staff finds the proposed MUR power uprate acceptable with respect to plant systems.

4.0 LICENSE AND TECHNICAL SPECIFICATION CHANGES

4.1 Change to Facility Operating License No. DPR-20

The licensee proposes to revise paragraph 2.C.(1) of the operating license, DPR-20, to authorize operation at steady-state reactor core power levels not in excess of 2565.4 MWt (100-percent rated power).

On the basis of the evaluation provided in Section 3.0 above, the NRC staff finds the proposed change acceptable.

4.2 Change to TS 1.1

The licensee proposes to revise the definition of "RATED THERMAL POWER" to reflect the increase from 2530 MWt to 2565.4 MWt.

On the basis of the evaluation provided in Section 3.0 above, the NRC staff finds the proposed change acceptable.

4.3 Change to TS Table 3.3.1-1

The licensee proposes to change the maximum allowable value for the variable high power trip from 111 percent to 109.4 percent.

On the basis of the evaluation provided in Section 3.1.2 above, the NRC staff finds the proposed changes acceptable.

5.0 REGULATORY COMMITMENTS

To support the proposed 1.4 percent MUR power uprate at Palisades, the licensee made the following commitments to be completed prior to implementation of the proposed power uprate:

NMC will conduct operator training on the proposed power uprate.

NMC will revise plant procedures to address operation with the Crossflow ultrasonic flow measurement system out of service.

NMC will revise plant procedures to address operation with the PPC FW flow indication or a PPC FW temperature indication out of service.

NMC will revise plant procedures to include at least 0.1 percent power conservatism when the ultrasonic flow meter correction factors are established for use in the plant heat balance calculation.

The NRC staff considered the above commitments as part of its evaluation in Section 3.0 above and finds the commitments appropriate for the proposed MUR power uprate. The NRC staff has conditioned the implementation of the proposed MUR power uprate on completion of the above commitments.

The licensee has also committed that, within 90 days of the proposed MUR power uprate, it will:

Revise plant procedures to add precaution to appropriately evaluate Crossflow system performance if a modification is performed in the proximity of the Crossflow installation and obtain UFM vendor technical support as needed to assist in performance evaluations for UFM related modifications or observed atypical system performance.

Revise plant procedures to add limits for correction factor variation and appropriate actions to be taken as recommended in Nuclear Safety Advisory Signal Interference Issues," dated December 5, 2003.

Revise plant procedures to add trending of UFM FW flow and the fluctuation in the UFM FW flow buffered value as recommended in NSAL-03-12.

Conduct post-uprate UFM frequency spectrum analysis.

The NRC staff will monitor the licensee's progress and completion of these post-uprate commitments through normal regulatory oversight and inspection processes.

6.0 STATE CONSULTATION

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

7.0 ENVIRONMENTAL CONSIDERATION

The amendment changes the requirements with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration and there has been no public comment on such finding (68 FR 40714). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

8.0 CONCLUSION

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

Attachment: List of Acronyms

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LIST OF ACRONYMS

AC	alternating current
AEC	Atomic Energy Commission
AFW	auxiliary feedwater
AMAG	advanced measurement analysis group
AMSAC	ATWS mitigation system actuation circuitry
AOT	allowed outage time
ASME	American Society of Mechanical Engineers
ATWS	anticipated transient without scram
AV	allowable value
B&PV	Boiler and Pressure Vessel
BOP	balance-of-plant
BRS	boron recovery system
CCW	component cooling water
CD	condensate
CFR	*Code of Federal Regulations*
COT	channel operational test
CRDM	control rod drive mechanism
CUF	cumulative fatigue usage factor
CVCS	chemical and volume control system
DBA	design-basis accident
DC	direct current
DNB	departure from nucleate boiling
DNBR	DNB ratio
DSS	diverse scram system
ECCS	emergency core cooling system
EPRI	electric power research institute
EQ	environmental qualification

ESF	engineered safety feature
FAC	flow-accelerated corrosion
FSAR	final safety analysis report
FW	feedwater
GDC	general design criteria
GL	generic letter
HELB	high-energy line break
HVAC	heating ventilation and air conditioning
PALISADES PLANT	Palisades Plant
LBB	leak-before-break
LBLOCA	large-break LOCA
LOCA	loss-of-coolant accident
MOV	motor-operated valve
MS	main steam
MSLB	main steamline break
MUR	measurement uncertainty recaputure
MVA	megavars ampere
MWt	megawatts thermal
NMC	Nuclear Management Company, LLC
NRC	Nuclear Regulatory Commission
NSAL	Nuclear Safety Advisory Letters
NSSS	Nuclear Steam Supply System
ODSCC	outside diameter stress corrosion cracking
PCS	primary coolant system
P-T	pressure-temperature
PPCS	plant process computer system
PTS	pressurized thermal shock
PWR	pressurized-water reactor
PWSCC	primary water stress-corrosion cracking

RAI	request for additional information
RCL	reactor coolant loop
RCP	reactor coolant pump
RCS	reactor coolant system
RG	regulatory guide
RHR	residual heat removal
RIS	Regulatory Issue Summary
RPV	reactor pressure vessel
RTP	rated thermal power
SBLOCA	small-break loss-of-coolant accident
RV	reactor vessel
SBO	station blackout
SCU	signal conditioning unit
SE	safety evaluation
SG	steam generator
SGBD	steam generator blowdown
SGBS	steam generator blowdown system
SGTR	steam generator tube rupture
SI	safety injection
SPT	station power transformers
SRP	Standard Review Plan
ST	startup transformers
SW	service water
TS	Technical Specification
UFM	Ultrasonic Flow Measurement
UFMD	Ultrasonic Flow Measuring Device
USE	upper shelf energy
VHPT	variable high-power trip