July 8, 2003

Mr. Thomas Coutu
Site Vice President
Kewaunee Nuclear Power Plant
Nuclear Management Company, LLC
N490 Highway 42
Kewaunee, WI 54216-9511

SUBJECT: KEWAUNEE NUCLEAR POWER PLANT - ISSUANCE OF AMENDMENT (TAC NO. MB7225)

Dear Mr. Coutu:

The U.S. Nuclear Regulatory Commission (NRC) has issued the enclosed Amendment No. 168 to Facility Operating License No. DPR-43 for the Kewaunee Nuclear Power Plant (KNPP). This amendment revises the operating license and technical specifications (TSs) in response to your application dated January 13, 2003, as supplemented February 27, March 6, March 14, April 30, June 9, and June 30, 2003.

The amendment revises the KNPP operating license and TSs to increase the licensed rated power by 1.4 percent from 1650 megawatts thermal to 1673 megawatts thermal using measurement uncertainty recapture.

As requested by your staff, the NRC staff issued a draft version of the enclosed safety evaluation (SE) by letter dated June 3, 2003, and requested that you review it to verify that factual information is accurate and complete. By your June 9, 2003, supplemental letter, you provided comments on the draft SE. The NRC staff has evaluated your comments and incorporated them as appropriate. We note that your comments did not change our conclusions discussed in the draft SE.

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

Sincerely,
/RA/

John G. Lamb, Project Manager, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-305

Enclosures: 1. Amendment No. 168 to License No. DPR-43
2. Safety Evaluation

cc w/encls: See next page

July 8, 2003

Mr. Thomas Coutu
Site Vice President
Kewaunee Nuclear Power Plant
Nuclear Management Company, LLC
N490 Highway 42
Kewaunee, WI 54216-9511

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cc w/encls: See next page *See memo **See previous concurrence ***By email dated 7/2/03

OFFICE	PDIII-1/PM	PDIII-1/LA	SPLB/SC	SPLB/SC	EMCB/SC	EMCB/SC	EMCB/SC	EEIB/SC	EEIB/SC
NAME	JLamb	THarris***	SWeerakkody*	EWeiss*	SCoffin*	TChan*	ALund*	IAhmed* for EMarinos	CHolden*
DATE	07/07/03	07/02/03	05/29/03	05/29/03	04/16/03	05/19/03	05/21/03	05/29/03	05/01/03

OFFICE	IEHB/SC	EMEB/SC	SRXB/SC	SPSB/SC	OGC	PDIII-1/PM	PDIII-1/SC	PDIII/D	DLPM/D
NAME	DTrimble*	KManoly*	JUhle*	FReinhart*	AHodgdon**	MShuaibi	LRaghavan	WRuland	TMarsh
DATE	05/19/03	05/23/03	05/27/03	05/14/03	06/13/03	07/07/03	07/07/03	07/07/03	07/07/03

ADAMS Accession No. ML031530734 (Cover Letter and Amendment)
ADAMS Accession No. ML031910321 (License and TS Pages)
ADAMS Accession No. ML031910330 (Package)
OFFICIAL RECORD COPY

Kewaunee Nuclear Power Plant

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NUCLEAR MANAGEMENT COMPANY, LLC

DOCKET NO. 50-305

KEWAUNEE NUCLEAR POWER PLANT

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 168
License No. DPR-43

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 January 13, 2003, as supplemented February 27, March 6, March 14, April 30, June 9, and June 30, 2003, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations 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-43 is hereby amended to read as follows:

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

The Technical Specifications contained in Appendix A, as revised through Amendment No. 168, are hereby incorporated in the license. The licensee's shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance, and is to be implemented within 120 days of the date of issuance. Prior to implementation of the license amendment, the licensee shall:

A. Complete revisions to affected documents (i.e., procedures) and provide appropriate training to the necessary plant staff for changes associated with the installation of the Crossflow Ultrasonic Flow Measuring Device (UFMD) and the implementation of the new rated power.

B. Ensure the plant-specific analysis has been completed and that plant specific uncertainties are equal to or less than those provided to Westinghouse for the calculation of the power measurement uncertainty.

C. Complete revisions to affected operations procedures and provide appropriate training to operations for the implementation of the new rated power and the administrative restrictions for inoperable Crossflow UFMDs.

D. Update the environmental qualification plan to include the new containment exclusion areas for the pressurizer, steam generator, and reactor coolant pump vaults.

E. Complete the investigation of the reserve auxiliary transformer procedural limit and implement changes as necessary.

F. Complete modifications associated with the MUR power uprate; this includes the installation of the Crossflow UFMDs and implementation of the plant process computer system and control room alarm functions.

G. Complete rescaling and setting changes of the protection system as necessary.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

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

Attachment: Changes to the Technical Specifications

Date of Issuance: July 8, 2003

ATTACHMENT TO LICENSE AMENDMENT NO. 168

FACILITY OPERATING LICENSE NO. DPR-43

DOCKET NO. 50-305

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
Operating License, page 3	Operating License, page 3
TS iv	TS iv
TS vi	TS vi
TS 1.0-4	TS 1.0-4
TS 3.1-6	TS 3.1-6
TS B 3.1-6	TS B 3.1-6
TS B 3.1-7	TS B 3.1-7
Figure TS 3.1-1	Figure TS 3.1-1
Figure TS 3.1-2	Figure TS 3.1-2
TS 6.9-3	TS 6.9-3
TS 6.9-4	TS 6.9-4
TS 6.9-5	TS 6.9-5
----	TS 6.9-6

Kewaunee Nuclear Power Plant

Safety Evaluation for Amendment No. 168

Measurement Uncertainty Recapture Power Uprate

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATING TO AMENDMENT NO. 168 TO FACILITY OPERATING LICENSE NO. DPR-43

NUCLEAR MANAGEMENT COMPANY, LLC

KEWAUNEE NUCLEAR POWER PLANT

DOCKET NO. 50-305

1.0 INTRODUCTION

By application dated January 13, 2003, as supplemented February 27, March 6, March 14, April 30, June 9, and June 30, 2003, the Nuclear Management Company, LLC (NMC or the licensee) requested an amendment to the Facility Operating License and the Technical Specifications (TSs) for the Kewaunee Nuclear Power Plant (KNPP). The proposed amendment would increase the licensed reactor core power level by 1.4 percent from 1650 megawatts thermal (MWt) to 1673 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" and an ultrasonic temperature measurement (UTM) system called "CORRTEMP."

Specifically, the proposed changes would revise:

1. Paragraph 2.C.(1) of the operating license, DPR-43, to authorize operation at reactor core power levels not in excess of 1673 MWt.

2. The note on the following pages regarding the KNPP Pressure-Temperature (P-T) Limitation Curves: TS vi, TS 3.1-6, Figure TS 3.1-1, Figure TS 3.1-2, TS B3.1-6 and TS B3.1-7. The note will be revised to read, "[1]The curves are limited to 31.1 EFPY due to changes in vessel fluence associated with operation at uprated power." The value for effective full power years (EFPY) will change from the current value of 28 to 31.1 EFPY.

3. TS 1.0.m, RATED POWER, to reflect the increase from 1650 MWt to 1673 MWt.

4. TS 6.9.4, "Core Operating Limits Report (COLR)," as follows:

    4.a Revise the text of proposed TS 6.9.4.B of the letter from M. E. Warner to Nuclear Regulatory Commission (NRC) Document Control Desk, "License Amendment Request 185 to the Kewaunee Nuclear Power Plant Technical Specifications, 'Core Operating Limits Report Implementation'," dated July 26, 2002, to explain the use of the Crossflow system power measurement uncertainty in other topical reports listed in the COLR. To describe this change in applying the power measurement uncertainty, the licensee proposed the following text to be inserted just prior to the listing of topical reports in proposed TS 6.9.4.B:

ENCLOSURE

"The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC. When an initial assumed power level of 102 percent of the original rated power is specified in a previously approved method, 100.6 percent of uprated rated power may be used only when the main feedwater flow measurement (used as the input for reactor thermal output) is provided by the Crossflow ultrasonic flow measurement system (Crossflow system) as described in report (15) listed below. When main feedwater flow measurements from the Crossflow system are unavailable, a power measurement uncertainty consistent with the instrumentation used shall be applied."

"Future revisions of approved analytical methods listed in this Technical Specification that currently reference the original 10 CFR 50, Appendix K uncertainty of 102 percent of the original rated power should include the condition given above allowing use of 100.6 percent of uprated rated power in the safety analysis methodology when the Crossflow system is used for main feedwater flow measurement."

"The approved analytical methods are described in the following documents:"

4.b Add reference (15) to proposed TS 6.9.4.B for topical report, CENPD-397-P-A, "Improved Flow Measurement Accuracy Using Crossflow Ultrasonic Flow Measurement Technology," May 2000.

It is important to note that the licensee's January 13, 2003, submittal relied in part upon the approval of three previous submittals by KNPP. These submittals were:

(1) Alternate source term (AST) methodology for design-basis radiological analysis accident source term dated March 19, 2002 (ADAMS Accession No. ML020870565),

(2) The COLR dated July 26, 2002 (ADAMS Accession No. ML02220080), and

(3) 422 VANTAGE Plus (422V+) fuel transition dated July 26, 2002 (ADAMS Accession No. ML 022200503).

The NRC staff completed and approved these three submittals on the following dates:

(1) AST Amendment was KNPP Amendment No. 166, dated March 17, 2003 (ADAMS Accession No. ML030210062),

(2) COLR Amendment was KNPP Amendment No. 165, dated March 11, 2003 (ADAMS Accession No. ML030700456), and

(3) 422V+ fuel transition amendment was KNPP Amendment No. 167, dated April 4, 2003 (ADAMS Accession No. ML030940276).

As requested by the licensee, the NRC staff issued a draft version of the safety evaluation (SE), by letter dated June 3, 2003, and requested that the licensee review it to verify that factual information is accurate and complete. By supplemental letter dated June 9, 2003, the licensee provided comments on the draft SE. The NRC staff has evaluated the licensee's comments and incorporated them as appropriate. The NRC staff notes that the licensee's comments did not change the NRC staff findings or conclusions discussed in the draft SE.

The February 27, March 6, March 14, April 30, June 9, and June 30, 2003, supplements provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the NRC staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on February 4, 2003 (68 FR 5679).

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 0.6 percent. To achieve this level of accuracy, the licensee will install a Combustion Engineering Nuclear Power LLC (CENP) Crossflow ultrasonic flow measurement system (Crossflow system) for measuring the main feedwater flow at KNPP. The Crossflow system provides a more accurate measurement of feedwater flow than the feedwater flow measurement accuracy assumed during the development of the original Appendix K requirements and that of the feedwater flow venturis currently used to calculate reactor thermal output (RTO). The Crossflow system will measure feedwater mass flow to within plus or minus (±)0.5 percent for KNPP. This bounding feedwater mass flow uncertainty was used to calculate a total power measurement uncertainty of ±0.6 percent. Based on this, KNPP proposes to reduce the power measurement uncertainty required by Appendix K to 0.6 percent. The improved power measurement uncertainty 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 Kewaunee 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.

The U.S. Atomic Energy Commission (AEC) issued a "Safety Evaluation of the Kewaunee Nuclear Power Plant" on July 24, 1972, supplemented December 18, 1972, and May 10, 1973. The AEC performed a technical review of the KNPP against the General Design Criteria (GDC) in effect at the time and concluded that the KNPP design generally conforms to the intent of the GDC.

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 SRP was used solely for general technical guidance. The licensee's January 13, 2003, application, supplemented February 27, March 6, March 14, April 30, June 9, and June 30, 2003, was reviewed for compliance with the KNPP licensing basis, not NUREG-0800.

3.1 Instrumentation and Controls

3.1.1 Regulatory Evaluation

The NRC staff's review in the area of instrumentation and controls covers (1) the proposed plant-specific implementation of the feedwater 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.

3.1.2 Technical Evaluation

3.1.2.1 Ultrasonic Flow Measurement Device

The proposed instrumentation consists of the Crossflow UFM system and the CORRTEMP UTM system and is called an UFMD. The Crossflow UFM consists of four ultrasonic transducers, in a set of two, mounted on a metal support frame which attaches externally to the feedwater piping. The two transducer sets are a known distance apart and each set injects an ultrasonic signal perpendicular to the pipe axis. By measuring the time a unique pattern of eddies takes to pass between the two transducer sets, the velocity of the fluid is determined. A UTM is externally attached in proximity to the UFM for measuring the feedwater temperature. One UFM and one UTM in each of the feedwater loops, A and B, will be installed at KNPP. There will be one UFMD electronics cabinet receiving UFM and UTM sensor data from each of the feedwater loops. Proper operation of the UFMDs will allow KNPP to operate at the higher rated thermal powr (RTP) of 1673 MWt, a 1.4 percent increase.

3.1.2.2 System Operation at KNPP

In the licensee's April 30, 2003, letter, NMC provided the figure shown on page 10 for the feedwater alpha or "A" loop data processing and communication links; loop bravo or "B" data processing and communications are identical. This figure was provided as information to facilitate understanding of the system application at KNPP and depicts the integration and operation of the UFMD system and its interface with the plant process computer system (PPCS). The UFMD electronics cabinet takes inputs from the UTM and UFM sensors in each feedwater loop. These sensor signals are processed using the Crossflow software for the UFMs and the CORRTEMP software for the UTMs. The UFMD generated values of flow and temperature are compared with existing venturi flowmeters (venturis) and resistance temperature detector (RTD) signals provided from the PPCS to develop instantaneous and average UFMD correction factors, C_f and C_t, respectively for the two feedwater flow instrument channels (UFM flow/PPCS flow) and feedwater temperature channel (UTM temp/PPCS temp) in each loop. Quality factors, Q_f and Q_t, for each correction factor are generated in the UFMD electronics cabinet by Crossflow and CORRTEMP software and sent to the PPCS screen (PPCS SCRN). The quality factor data is determined and saved with the corresponding correction factor. The UFMD software running in the UFMD electronics cabinet ensures the correction factors are maintained within the required uncertainty. Individual correction factors for feedwater flow and temperature are provided from the UFMD to the PPCS and are used to calculate the corrected feedwater flow and temperature to be used by the PPCS RTO program. The RTO is used for monitoring reactor power such that the operator can control reactor power to less than the licensed limit.

The UFMDs must be in service and providing good quality correction factors to the PPCS RTO program prior to increasing power greater than 1650 MWt. NMC provided the following RTP limits to be imposed consistent with the feedwater instrumentation available.

Available Power Measurement Instrumentation	**Associated Uncertainty**	**Power Level Restriction**
UFMDs (UFMs and UTMs operable)	0.6 %	1673 MWt
Crossflow UFMs and feedwater RTDs (UTM inoperable)	0.8 %	1670 MWt
Feedwater venturis (UFMD inoperable)	2.0 %	1650 MWt

The operators will use the PPCS SCRN as part of the human machine interface to administratively control reactor power. If UFMD correction factors become questionable, for instance, due to sensor failures or loss of UFMD connections, the PPCS SCRN will display this information as a bad Q_f or Q_t. Audible and visual alarms are provided to warn operators of the possible need to lower RTO consistent with the abnormal instrumentation quality factor readings.

The only changes to the RTO computer program, in support of the power uprate request, will be those changes associated with receiving the UFMD correction factors for use in the RTO

calculation. The calculations performed in the RTO program are described in WCAP-15591, the KNPP instrument uncertainty methodology document contained in Attachment 7 of the licensee’s January 13, 2003, submittal.

3.1.2.3 Use of UTMs at KNPP

The NRC staff submitted a series of questions in a request for additional information (RAI) to NMC regarding the use of the UTMs and how they support the power uprate request. NMC stated the following:

> The only discussion pertaining to UTMs in the topical report is a statement that improving the accuracy of the feedwater temperature can improve the density term in feedwater flow determination. This can lead to more accurate density measurement and lower total feedwater flow measurement uncertainty. The NMC decided to implement the use of higher accuracy feedwater temperature measurement instrumentation (e.g. the UTMs).

The UTM is an ultrasonic temperature measurement system named CORRTEMP which uses clamp on ultrasonic transducers to measure feedwater temperature. An unavailable UTM sensor will require operators to reduce RTP by 0.2 percent if the UTM is not returned to service within the allowed outage time of 24 hours.

The NRC staff concludes that the licensee’s use of the UTMs, including its uncertainty and sensitivity determination, calibration and PPCS interface, is appropriate for use as proposed in the KNPP 1.4 percent MUR power uprate request.

3.1.2.4 NMC Use of NRC Staff SER on Topical Report CENPD-397

The NRC SE regarding CENPD-397, dated March 20, 2000, includes four additional requirements, in addition to the guidelines outlined in the topical report, that must be addressed by licensees referencing this topical report. NMC’s submittal addressed each of these 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 process and contingencies for an inoperable Crossflow UFM and the effect on thermal power measurement and plant operation.

In the licensee’s letters dated January 13, and April 30, 2003, NMC stated that implementation of the MUR power uprate license amendment will include developing the necessary procedures and documents required for operation, maintenance, calibration, testing and training at the uprated power level with the new UFMDs. The UFM maintenance items that are anticipated are summarized:

(A) Reboot the signal processing unit every two months.

(B) Perform signal conditioning unit (SCU) self test monthly (performed automatically by installed software).

(C) Perform the reflected signal strength indication scan after cold shutdown, startup, after a feedwater change of greater than 100 °F, and after a year of continuous operation. (This is planned to be performed on a yearly basis in addition to the above).

(D) Recalibrate the SCU every refueling outage (approximately 18 months) by returning to the vendor.

The internal time delay accuracy check will be performed automatically by the Crossflow UFMD software during normal operation. NMC stated that the maintenance for the UTMs will be to reboot the Signal Conditioning/Processing Unit (SCP) every two months, recalibrate the SCP every refueling outage by returning it to the vendor and perform annual hard disk maintenance. With regard to the remaining instrumentation used in power measurement uncertainty calculations, NMC stated that they are calibrated and maintained on specified frequencies through the use of appropriate plant instrumentation and controls procedures.

KNPP proposed operation with unavailable UFMs or UTMs is as follows:

(a) If the UFMD becomes unavailable, plant operations at a core thermal output up to rated power may continue for a maximum of 24-hours after the last valid UFMD correction factor was used in the calorimetric calculation for use in the daily nuclear power range surveillance. The 24-hour time period is based on the minimum frequency for the calibration of the power range channels found in KNPP TSs. Since the nuclear power range channel will have been adjusted using the heat balance calculated with a valid Crossflow UFMD correction factor, the nuclear power range channel calibration will be acceptable until the next performance of the surveillance.

(b) If the UFMs or UTMs become unavailable, the operators will receive a computer alarm that will generate a control board annunciator alarm. The operators will enter an operating procedure which will direct them through the actions for a UFM or UTM failure. The procedure will require the UFM or UTM to be returned to service prior to the next power range channel surveillance. If the UFMs or UTMs are not returned to service prior to the next surveillance time, reactor power will be reduced consistent with limits provided in the table discussed in Section 3.1.2.2 “System Operation at KNPP,” above. The basis for reducing the power to 1670 MWt is the relaxation of the Appendix K rule. The change in the rule allows KNPP to use the Crossflow system UFMs with the feedwater system RTDs to calculate a correction factor for the power measurement uncertainty. The power measurement uncertainty for the UFMs and the RTDs is 0.8 percent. The basis for reducing power to 1650 MWt is the calorimetric uncertainty required of the Appendix K rule.

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

This is not applicable since a Crossflow UFM has not been used at KNPP prior to the MUR power uprate.

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

Westinghouse calculated the power uncertainties using the Westinghouse Revised Thermal Design Procedure WCAP-11397 which was approved by the staff in a January 17, 1989 SER. WCAP-15591 Revision 1 contained in the licensee's January 13, 2003, submittal and the licensee's letter dated April 30, 2003, which included supporting calculations, provide the methodology and uncertainty calculations to support the 1.4 percent power uprate request. WCAP-15591 documents, in part, the determination of power and associated instrument uncertainties for plant operation with UFMs and UTMs, UFMs and RTDs and venturis and RTDs. The power uncertainties of these three conditions are used to generate the RTP power limit table provided in the previous section 3.1.2.2 discussing system operation.

According to WCAP-15591, instrument uncertainties can be described with random, normal, two-sided probability distributions. The methodology used to combine the uncertainty components for a channel is the square root sum of the squares of those groups that are statistically independent. Uncertainties that are dependent are combined arithmetically into independent groups. The feedwater flow uncertainty reported in WCAP-15591 is 0.5 percent for UFM flow and 1.1 °F for UTM temperature. The Advanced Measurement and Analysis Group (AMAG) specifically calculates the feedwater mass flow uncertainties for KNPP. The plant-specific uncertainty evaluation will provide verification that the assumed feedwater mass flow and temperature measurement uncertainties used for the measurement uncertainty calculation for the plant-specific configuration remain bounded. NMC will ensure that the plant-specific analysis has been completed and that the plant-specific uncertainties are equal to or less than those provided to Westinghouse for the calculation of the power measurement uncertainty. The NRC staff reviewed the tables for power calorimetric sensitivities, power calorimetric instrumentation uncertainties and secondary side power calorimetric measurement uncertainty for the three conditions described in this section.

The NRC staff verified that NMC has appropriately identified all sources of uncertainty for reactor power level and that the calculations have been performed correctly. Therefore, the uncertainty methodology used to calculate

KNPP power level instrument uncertainty is acceptable. WCAP-15591 Revision 1 was approved by the NRC staff SER for the KNPP 422 VANTAGE + Fuel with PERFORMANCE Features Amendment 167 dated April 4, 2003. The NRC staff finds that WCAP-15591, Revision 1 is an acceptable setpoint methodology with regard to the development of instrument uncertainty.

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

The KNPP Crossflow UFM was calibrated according to the topical report CENPD-397-P-A, Revision 1 dated May 2000. A UFM in feedwater loop “A” is installed where the flow is fully developed. As such, this UFM does not need calibration. The UFM in feedwater loop “B” is installed where the flow is not fully developed and thus needed a one-time in-situ calibration. For this calibration, a temporary stand-alone UFMD was installed on the full flow feedwater bypass line. The temporary UFM was installed at a location in the bypass line meeting the conditions for fully developed flow and was used for the UFM “B” calibration.

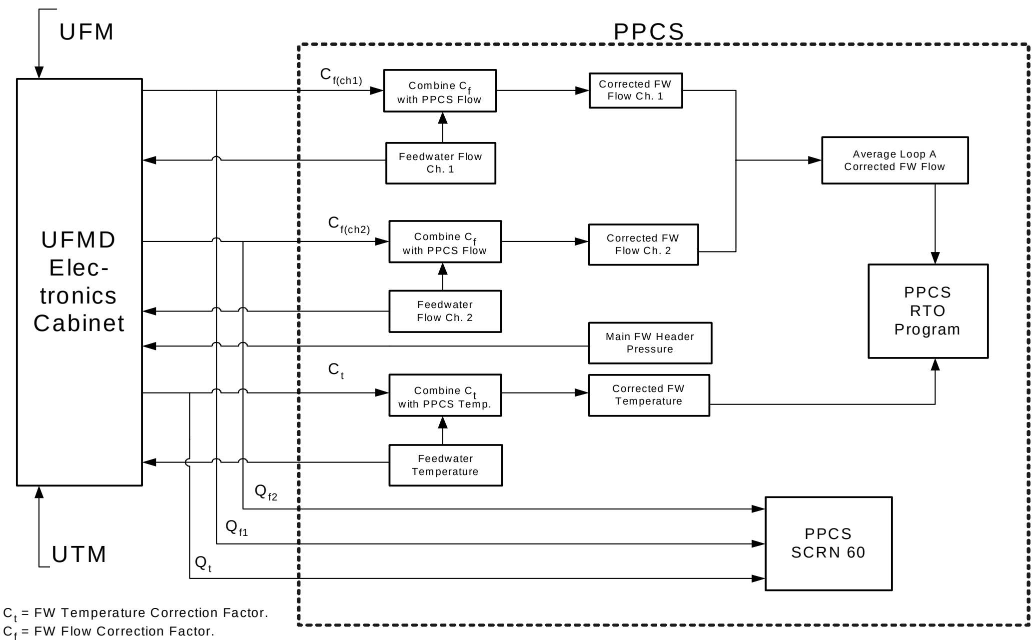
The NRC staff finds that the licensee’s response to the plant-specific requirements stated in the NRC’s safety evaluation for topical report CENPD-397 were appropriately addressed and are acceptable.

The NRC staff finds that NMC sufficiently addressed requirements and adequately resolved plant-specific issues related to the UFMDs and their use to support the power uprate request. These include maintenance and calibration, installation, hydraulic configuration, and procedures for inoperable UFMs or UTMs. NMC used an appropriate methodology which adequately accounted for the uncertainties due to power level instrumentation error.

3.1.3 Summary

The NRC staff has reviewed the licensee’s proposed plant-specific implementation of the feedwater flow measurement device and the power uncertainty calculations. The NRC staff finds that the licensee’s response to the plant-specific requirements stated in the NRC’s safety evaluation for topical report CENPD-397 were appropriately addressed and are acceptable and is consistent with the NRC staff’s approval of this topical report. The NRC staff also concludes that the licensee has adequately accounted for the uncertainties due to power level instrumentation error in their power level uncertainty calculations and demonstrated that the calculations meet the relevant requirements of 10 CFR Part 50, Appendix K. Therefore, the NRC staff finds the proposed 1.4 percent MUR power uprate acceptable with respect to instrumentation and controls.

Feedwater Loop A Data Point Processing and Communication Links



C_t = FW Temperature Correction Factor.
C_f = FW Flow Correction Factor.
Q_t = Quality Signal For UFMD Temperature.
Q_f = Quality Signal For UFMD Feedwater Flow.

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, 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. The NRC staff reviewed the KNPP to the AEC criterion issued in the "Safety Evaluation of the Kewaunee Nuclear Power Plant" on July 24, 1972, supplemented December 18, 1972, and May 10, 1973. The NRC staff used guidance and acceptance criteria for the reactor systems contained in Chapters 4, 5, 6, and 15 of NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants LWR Edition," as a reference.

3.2.2 Technical Evaluation

3.2.2.1 Accidents and Transients Bounded by the Existing Analyses of Record

3.2.2.1.1 Loss of Normal Feedwater

A loss of normal feedwater 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 feedwater were not supplied to the plant, core damage could occur. Currently, the KNPP loss of normal feedwater analysis models at a core power level of 102 percent of 1650 MWt (1683 MWt). This power level bounds the requested MUR power uprate core power level of 1673 MWt with an 0.6 percent uncertainty. Since the current analysis bounds the power uprate, the NRC staff finds it acceptable.

3.2.2.1.2 Anticipated Transient Without Scram (ATWS)

The licensee installed an ATWS mitigation system actuation circuitry (AMSAC) at KNPP, thereby satisfying the requirements of 10 CFR 50.62(b). After the implementation of the 1.4 percent MUR power uprate, the AMSAC will continue to operate at KNPP in compliance with the requirements of the ATWS rule. As a supplement to AMSAC, the licensee installed a diverse scram system (DSS). Because of the increased safety afforded by the DSS, no plant-specific ATWS analyses are required to support the 1.4 percent MUR power uprate at KNPP.

3.2.2.1.3 Steam Generator Tube Rupture - Thermal/Hydraulic

For a Steam Generator Tube Rupture (SGTR), the thermal-hydraulic analysis calculates the primary to secondary break flow and the steam released to the environment. The KNPP licensing basis SGTR analysis uses a simplified mass and energy balance method. The input parameters that could change as a result of the 1.4 percent MUR power uprate include power, hot-leg temperature, cold-leg temperature, steam temperature, and steam pressure. An increase in reactor power could slightly change these parameters, resulting in an increase in

steam release due to a small increase in system energy. However, the methodology used in the current licensing basis analysis includes a 2.0 percent margin in reactor power for the calculation of the feedwater flows and steam releases. The analyzed 2.0 percent margin for reactor power bounds the MUR power uprate of 1.4 percent with a 0.6 percent uncertainty.

3.2.2.1.4 Station Blackout (SBO)

In their coping analysis for an SBO event, the licensee performed calculations assuming a core power level of 1650 MWt with a 2.0 percent uncertainty, which equates to 1683 MWt. Since this analysis continues to bound the requested core power level of 1673 MWt with a 0.6 percent uncertainty (1683 MWt), the NRC staff finds it acceptable for the requested 1.4 percent MUR power uprate.

3.2.2.2 Other Accidents and Transients

3.2.2.2.1 Uncontrolled Rod Cluster Control Assembly (RCCA) from Subcritical

An uncontrolled RCCA bank withdrawal accident 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 power range high neutron flux reactor trip (low setting) will terminate the accident.

The licensee submitted their analysis of this accident to the NRC by letter dated July 26, 2002 (Letter from M. E. Warner, Nuclear Management Company to USNRC, “Licence Amendment Request 187 to the Kewaunee Nuclear Power Plant Technical Specifications, Conforming Technical Specification Changes for Use of Westinghouse VANTAGE + Fuel,” Docket No. 50-305, License No. DPR-43, Letter No. NRC-02–067, dated July 26, 2002). By letter dated April 4, 2003 (Letter from John Lamb, USNRC to Thomas Coutu, Site Vice President, Kewaunee Plant, “Kewaunee Nuclear Power Plant - Issuance of Amendment (TAC No. 5718),” April 4, 2003), the NRC staff found that the licensee performed this analysis assuming that a reactor trip takes place at 35 percent of an assumed core power level of 1772 MWt. The NRC staff also determined that the licensee used an approved methodology and the results of the analysis indicate that the SRP acceptance criteria continue to be met, i.e., the minimum departure from nucleate boiling ratio (DNBR) remains above the limit value, and the fuel centerline temperatures do not exceed the melting point.

Since the licensee performed this analysis based upon a core power level of 1772 MWt using an approved methodology, the NRC staff finds that it bounds the requested core power level of 1673 MWt. Therefore, the NRC staff finds the analysis acceptable for the 1.4 percent MUR power uprate.

3.2.2.2.2 Uncontrolled RCCA Withdrawal at Power

Similar to the RCCA withdrawal from subcritical, an uncontrolled RCCA withdrawal at power accident can be caused by a malfunction of the reactor control or rod control systems. This withdrawal will also uncontrollably add positive reactivity to the reactor core, resulting in a power excursion. Depending upon the reactivity insertion rate, either the power range high neutron flux reactor trip or the over temperature ΔT (OTΔT) reactor trip will terminate the accident.

The licensee submitted their analysis of this accident to the NRC by letter dated July 26, 2002. By letter dated April 4, 2003, the NRC staff found that the licensee performed this analysis assuming a core power level of 1772 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 fuel centerline temperatures do not exceed the melting point.

Since the licensee performed this analysis at 1772 MWt using an approved methodology, the NRC staff finds that it bounds the requested core power level of 1673 MWt. Therefore, the NRC staff finds the analysis acceptable for the 1.4 percent MUR power uprate.

3.2.2.2.3 RCCA Misalignment

The RCCA misalignment accidents include the cases of a dropped RCCA, a dropped RCCA bank, and a statically misaligned RCCA. The dropped RCCA transients result in a negative reactivity insertion, which cause a shift in the power distribution of the core. Similarly, statically misaligned RCCAs 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 licensee submitted their analyses of these accidents to the NRC by letter dated July 26, 2002. By letter dated April 4, 2003, the NRC staff found that the licensee performed the analyses assuming a core power level of 1772 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 fuel centerline temperatures do not exceed the melting point.

Since the licensee performed the analyses at a core power level of 1772 MWt using an approved methodology, the NRC staff finds that they bound the requested core power level of 1673 MWt. Therefore, the NRC staff finds the analyses acceptable for the 1.4 percent MUR power uprate.

3.2.2.2.4 Chemical and Volume Control System Malfunction

The chemical and volume control system (CVCS) can be used to add unborated water to the reactor coolant system (RCS). This addition may happen inadvertently because of operator error or 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 CVCS malfunction (boron dilution) event for the uprated power conditions over the spectrum of plant operations, from power operation with a core power level of 1772 MWt to refueling. The licensee submitted their analyses to the NRC by letter dated July 26, 2002. By letter dated April 4, 2003, 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 main steam 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 1772 MWt, the NRC staff finds that the

analyses bound the requested core power level of 1673 MWt. Therefore, the analyses are acceptable for the 1.4 percent MUR power uprate.

3.2.2.2.5 Startup of an Inactive Reactor Coolant Loop

The transient for the startup of an inactive loop at the incorrect temperature occurs when one reactor coolant pump (RCP) 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 submitted to the NRC, by letter dated July 26, 2002, that the TSs limit the reactor power to less than 2 percent rated thermal power when only one RCP is in operation. At that power level, the hot-leg temperature of the inactive loop would be very close to the cold-leg inlet temperature. For this reason, the licensee determined that no analysis is needed to show that the DNBR limit is satisfied for this event at KNPP. By letter dated April 4, 2003, the NRC staff agreed with the licensee’s assessment and concluded that the KNPP TSs will prevent unacceptable results from a potential transient due to startup of an inactive reactor coolant loop. Therefore, the SRP acceptance criteria continue to be met, i.e., the minimum DNBR remains above the limit value and the RCS and main steam system pressures remain below 110 percent of their design values.

Since the SRP acceptance criteria will continue to be met, the NRC staff finds this transient acceptable for the 1.4 percent MUR power uprate.

3.2.2.2.6 Excessive Heat Removal Due to Feedwater System Malfunctions

A feedwater system malfunction occurs when relatively cool feedwater or excessive feedwater 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 feedwater regulating valves or the accidental opening of a feedwater bypass valve.

The licensee submitted their analyses for this transient to the NRC by letter dated July 26, 2002. By letter dated April 4, 2003, the NRC staff found that the licensee performed the analyses assuming a reactor core power level of 1772 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 main steam system pressures remain below 110 percent of their design values.

Since the licensee performed the analyses based upon a core power level of 1772 MWt using an approved methodology, the NRC staff finds that they bound the requested core power level of 1673 MWt. Therefore, the NRC staff finds the analyses acceptable for the 1.4 percent MUR power uprate.

3.2.2.2.7 Excessive Load Increase Incident

An excessive load increase incident occurs when a rapid increase in steam flow causes a power mismatch between the reactor core power and the SG load demand. The RCS accommodates a 10-percent step load increase or a 5 percent per minute ramp load increase

between 15 and 95 percent power. However, loading rates exceeding these values may result in a reactor trip initiated by the reactor protection system.

The licensee submitted their analysis for this transient to the NRC by letter dated July 26, 2002. By letter dated April 4, 2003, the NRC staff found that the licensee performed the analysis assuming a reactor core power level of 1772 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 main steam system pressures remain below 110 percent of their design values.

Since the licensee performed the analysis based upon a core power level of 1772 MWt using an approved methodology, the NRC staff finds that it bounds the requested core power level of 1673 MWt. Therefore, the NRC staff finds the analysis acceptable for the 1.4 percent MUR power uprate.

3.2.2.2.8 Loss of Reactor Coolant Flow (Coastdown Events)

A mechanical or electrical failure in one or more RCPs 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 licensee submitted their analyses for this transient to the NRC by letter dated July 26, 2002. By letter dated April 4, 2003, the NRC staff found that they performed the analyses assuming a reactor core power level of 1772 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 main steam system pressures remain below 110 percent of their design values.

Since the licensee performed the analyses based upon a core power level of 1772 MWt using an approved methodology, the NRC staff finds that it bounds the requested core power level of 1673 MWt. Therefore, the NRC staff finds the analyses acceptable for the 1.4 percent MUR power uprate.

3.2.2.2.9 Locked Rotor Transient Analysis

A locked rotor accident results from the instantaneous seizure of a RCP rotor. The flow through the affected reactor coolant loop rapidly decreases 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 licensee submitted their analyses for this transient to the NRC by letter dated July 26, 2002. By letter dated April 4, 2003, the NRC staff found that the licensee performed the analyses assuming a reactor core power level of 1772 MWt with a 2 percent uncertainty (1807 MWt), and the results indicate that the KNPP licensing basis acceptance criteria continue to be met, i.e., the maximum RCS and main steam system pressures remain below acceptable design limits considering potential brittle and ductile fracture, the core remains coolable and intact, and the maximum clad temperature remains below 2700 ^{o}F.

Since the licensee performed the analyses based upon a core power level of 1772 MWt with a 2 percent uncertainty (1807 MWt), using an approved methodology, the NRC staff finds that it bounds the requested core power level of 1673 MWt. Therefore, the analyses are acceptable for the 1.4 percent MUR power uprate.

3.2.2.2.10 Loss of Electrical Load (Overpressure and DNB)

A loss of external electrical load event occurs when an electrical disturbance causes the loss of a significant portion of the generator load. KNPP is analyzed to accept a large load rejection at 50 percent of plant-rated power without a reactor trip. Currently, there is a discrepancy between the 2002 update of the KNPP Updated Safety Analysis Report (USAR) and the MUR power uprate submittal. The USAR states the following: “The Reactor Coolant System can accept a complete loss of external load from full power without a reactor trip.” The licensee states that the reason for the discrepancy is that the USAR change has not been processed. The licensee stated that the changes will be made in the next planned KNPP USAR revision that will occur approximately November 2003.

The licensee submitted their analyses for this transient to the NRC by letter dated July 26, 2002. By letter dated April 4, 2003, the NRC staff found that the licensee performed the analyses assuming a reactor core power level of 1772 MWt for the DNB case and 1772 MWt with a 2 percent uncertainty (1807 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 pressure and the main steam system pressure remains below 110 percent of the design values.

Since the licensee performed the analyses based upon a core power level of 1772 MWt and 1807 MWt, using an approved methodology, the NRC staff finds that it bounds the requested core power level of 1673 MWt. Therefore, the NRC staff finds the analyses acceptable for the 1.4 percent MUR power uprate.

3.2.2.2.11 Loss of Alternating Current (AC) Power to the Plant Auxiliaries

The loss of all AC power to the station auxiliaries transient results in a loss of all power to auxiliary systems including the RCPs, condensate pumps, etc. Upon the loss of power, core cooling and removal of residual heat is accomplished by natural circulation in the reactor coolant loops, aided by auxiliary feedwater and safety valves on the secondary side.

The licensee submitted their analysis for this transient to the NRC by letter dated July 26, 2002. By letter dated April 4, 2003, the NRC staff found that the licensee performed the analysis assuming a reactor core power level of 1772 MWt with a 2 percent uncertainty (1807 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 main steam system pressures remain below 110 percent of their design values.

Since the licensee performed the analysis based upon a core power level of with a 2 percent uncertainty (1807 MWt), using an approved methodology, the NRC staff finds that it bounds the requested core power level of 1673 MWt. Therefore, the NRC staff finds the analysis acceptable for the 1.4 percent MUR power uprate.

3.2.2.2.12 Rupture of a Steam Pipe (Core Response)

The rupture of a steam pipe accident models an uncontrolled steam release from an SG, which could include steam pipe breaks and valve malfunctions. 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. The safety injection (SI) system eventually terminates this accident by supplying boric acid to shut down the core.

Because the most limiting case of this accident occurs at no load conditions, and because the SI system terminates the accident independent of power level, 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 1673 MWt.

3.2.2.2.13 Rupture of Control Rod Mechanism Housing (RCCA Ejection - Core Response)

A control rod drive mechanism (CRDM) pressure housing rupture may result in the ejection of an RCCA and drive shaft to their fully withdrawn position. The consequences of this failure include a rapid positive reactivity insertion together with an adverse core power distribution, which could lead to localized fuel rod damage.

The licensee submitted their analyses for this accident to the NRC by letter dated July 26, 2002. By letter dated April 4, 2003, 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 at 1772 MWt with a 2 percent uncertainty (1807 MWt) for the Hot Full Power case. The results indicate that the KNPP licensing basis acceptance criteria continue to be met, i.e., the total rods in DNB continue to be less than 10 percent, the peak RCS pressure remains below the faulted condition stress limits, and the maximum average fuel pellet enthalpy at the hot spot is less than 200 calories per gram (cal/gm).

Since the licensee performed the analyses based upon a core power level of both 0 MWt and 1807 MWt using an approved methodology, the NRC staff finds that it bounds the requested core power level of 1673 MWt. Therefore, the NRC staff finds the analyses acceptable for the 1.4 percent MUR power uprate.

3.2.2.2.14 Loss of Reactor Coolant from Small Ruptured Pipes or From Cracks in Large Pipes which Actuates ECCS

Ruptures with small cross sections cause 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. The reactor will trip when the pressurizer low-pressure trip setpoint is reached, and SI will occur when an appropriate SI initiation setpoint is reached.

The licensee submitted their analyses for this accident to the NRC by letter dated July 26, 2002. By letter dated April 4, 2003, the NRC staff found that the licensee performed the analyses

assuming a reactor core power level of 1772 MWt with an 0.6 percent uncertainty (1782 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 1782 MWt using an approved methodology, the NRC staff finds that they bound the requested core power level of 1673 MWt. Therefore, the NRC staff finds the analyses acceptable for the 1.4 percent MUR power uprate.

3.2.2.2.15 Major Reactor Coolant System Pipe Ruptures LOCA

For KNPP, a large break loss-of-coolant-accident (LBLOCA) includes a rupture of the RCS piping from 1.0 square feet (ft$^{2)}$ up to a double-ended rupture of the largest pipe. Should a major break occur, the RCS rapidly depressurizes until the pressure nearly equals the containment pressure. SI initiates upon receipt of either a high containment pressure or a low pressurizer pressure setpoint. After the end of the blowdown, the ECCS will reflood the reactor.

The licensee submitted their analyses for this transient to the NRC by letter dated July 26, 2002. By letter dated April 4, 2003, the NRC staff found that the licensee performed the analyses assuming a reactor core power level of 1772 MWt with an 0.6 percent uncertainty (1782 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 1782 MWt using an approved methodology, the NRC staff finds that it bounds the requested core power level of 1673 MWt. Therefore, the NRC staff finds the analyses acceptable for the 1.4 percent MUR power uprate.

3.2.2.2.16 Core and Internal Integrity Analysis

The core and internal integrity analysis examines the effects of an excitation produced by a simultaneous complete severance of a reactor coolant pipe and a seismic excitation of the core internals. Because of these postulated horizontal and vertical movements, the accident subjects the core to significant internal stresses.

The licensee submitted this analysis to the NRC by letter dated July 26, 2002. By letter dated April 4, 2003, the NRC staff found that the licensee performed the analysis assuming a reactor core power level of 1772 MWt with an 0.6 percent uncertainty (1782 MWt). The results indicate that the acceptance criteria of 10 CFR 50.46 continue to be met, i.e., the core remains in a coolable geometry and the long-term core coolability is maintained.

Since the licensee performed this analysis based upon a core power level of 1782 MWt using an approved methodology, the NRC staff finds that it bounds the requested core power level of

1673 MWt. Therefore, the NRC staff finds the analysis acceptable for the 1.4 percent MUR power uprate.

3.2.2.2.17 Natural Circulation Cooldown

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

The licensee submitted their analysis for natural circulation cooldown as part of their loss of ac power to the plant auxiliaries transient by letter dated July 26, 2002. By letter dated April 4, 2003, the NRC staff found that the licensee performed the analysis assuming a reactor core power level of 1772 MWt with a 2.0 percent uncertainty (1807 MWt). The results indicate that KNPP has adequate RCS flow and auxiliary feedwater for decay heat removal.

Since the licensee performed the analysis based upon a core power level of 1807 MWt using an approved methodology, the NRC staff finds that it bounds the requested core power level of 1673 MWt. Therefore, the NRC staff finds the analysis acceptable for the 1.4 percent MUR power uprate.

3.2.2.3 Residual Heat Removal System (RHR)

Operation at a higher power level increases the amount of decay heat being generated in the core, which results in a higher heat load to the residual heat exchangers during cooldown and refueling. The licensee evaluated the RHR system for the normal cooldown requirements for a core power level of 1772 MWt and found that this power level causes an increase to the RHR cooldown times. For example, the licensee determined that when both RHR pumps work, to cooldown from 350 ^{o}F to 140 ^{o}F, the total cooldown time would increase from 17.9 hours to 21.5 hours. However, even with an increased core decay heat load, KNPP would still be capable of meeting its TS requirements for 36 hour cooldown to cold shutdown conditions. Since the RHR system remains capable of meeting its core cooldown requirements at 1772 MWt, the NRC staff finds the RHR system acceptable for the uprated power level of 1673 MWt with a 0.6 percent uncertainty.

3.2.2.4 Safety Injection

The SI system provides water inventory and cooling to the RCS in the event of a 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 1772 MWt and determined that the system performance remains acceptable. Since the system remains adequate up to a power level of 1772 MWt, the NRC staff finds it acceptable for the uprated power level of 1673 MWt with a 0.6 percent uncertainty.

3.2.2.5 Changes to Protection System Settings and Emergency System Settings

Upon examination of the protection and emergency system settings for the power uprate, NMC determined that changes to engineered safety feature (ESF) settings are not necessary for the MUR power uprate. However, the licensee does need to calibrate and scale some nuclear instrumentation. For the overpower delta T and overtemperature delta T setpoints, the licensee

proposed changing the full power Δt_o inputs to the predicted value based on the best estimate evaluations for the power uprate to 1673 MWt. Also, NMC proposed performing gain adjustments to the power range nuclear instruments based upon a secondary heat balance for the new 100 percent power level of 1673 MWt. This adjustment would ensure that the power range reactor trips, rod stops, and permissives (P-7, P-8, and P-10) function at their appropriate values. Because these setpoints relate to thermal power, the NRC staff agrees with the licensee's decision to recalibrate and rescale the above parameters for the MUR power uprate.

On the other hand, for the intermediate range and source range nuclear instrumentation, NMC proposed keeping its current setpoints. A power uprate has no effect on the source range setpoints; however, TS Table 3.3.2-1 requires that the intermediate range high flux trip actuates at $\leq$ 40 percent of rated thermal power. Currently, the intermediate range rod stop setpoint and reactor trip setpoints are set at 34 percent and 39 percent power, respectively. Adjusting these setpoints for the uprated power would cause a slight increase to their value and would cause them to activate later in an accident. Therefore, maintaining their current settings would be conservative. Since it is conservative not to change the intermediate range setpoints, the NRC staff finds the licensee's proposal acceptable.

3.2.2.6 Nuclear Steam Supply System (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 1650 MWt, 1673 MWt, and 1772 MWt. The key parameters included core power, NSSS power, reactor coolant system pressure, thermal design flow, Tavg range, steam pressure, steam temperature, and steam flow rate. The differences between the parameters at 1650 MWt and 1673 MWt included an increased core power level, increased minimum value for Tavg, lower maximum steam pressure, lower maximum steam temperature, and a higher steam flow rate. The NRC staff evaluated these changes to the plant conditions and found the changes to adequately represent the plant behavior at the specified power levels; therefore, the NRC staff finds the NSSS design parameters acceptable.

3.2.2.7 Reactor Vessel Integrity-Neutron Irradiation

Power uprates affect reactor vessel integrity due to increasing beltline material embrittlement from increasing neutron fluence. To ensure vessel integrity following a power uprate, the licensee reviewed the existing P-T limit curves, the pressurized thermal shock reference temperature (RT_{PTS}), and the Low Temperature Overpressurization (LTOP) limits.

NMC evaluated the fluence projections using calculations that adhere to the guidance of Regulatory Guide (RG) 1.190 (RG 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," March 2001). The fluence evaluation resulted in higher values at 33 EFPYs than the previous estimates. Therefore, the current P-T curves should either be recalculated or adjusted downward to the exposure corresponding to the fluence previously calculated for 33 EFPYs. The licensee decided to adjust the life projection, which equated to 31.1 EFPYs. Accordingly, the LTOP setpoints would also be valid for 31.1 EFPYs. This change is acceptable because the fluence evaluation methodology follows the provisions of RG 1.190, which satisfies the requirements of GDC 30 and GDC 31.

The NRC staff reviewed the submittal to determine the applicability of the P-T limit curves, the LTOP limits, and the RT_{PTS}. The NRC staff review determined that the fluence calculations adhere to the guidance of RG 1.190; therefore, the calculated values are acceptable. Based on this conclusion, the NRC staff also finds that the P-T limits and the LTOP limits are valid until 31.1 EFPYs. In addition, the RT_{PTS} limits are valid for 33 EFPYs (the end of the current license).

3.2.2.8 RCCA Scram Performance Evaluation

NMC evaluated the Westinghouse 422 VANTAGE Plus (V+) fuel design for impacts on RCCA drop times. The licensee performed a drop time analysis under worst case conditions, assuming a power uprate up to 1772 MWt core power. The licensee determined the maximum RCCA drop time with seismic allowance to be 1.59 seconds in the letter from M. E. Warner, Nuclear Management Company to USNRC, “Licence Amendment Request 187 to the Kewaunee Nuclear Power Plant Technical Specifications, Conforming Technical Specification Changes for Use of Westinghouse VANTAGE + Fuel,” Docket No. 50-305, License No. DPR-43, Letter No. NRC-02–067, dated July 26, 2002, and found that it satisfies the KNPP TS limit of 1.80 seconds. Since the current TS limits bound drop times for a power uprate to 1772 MWt, it bounds the drop times for the proposed power uprate to 1673 MWt. Therefore, the NRC staff finds the drop times acceptable.

3.2.2.9 Momentum Flux and Fuel Rod Stability

High velocity jets created by high-pressure water being forced through the gaps between the baffle plates and the core can cause hydraulically induced vibration of the fuel rods. This phenomenon is called baffle jetting. The licensee evaluated the impact of the uprated reactor coolant system on the margin of safety for baffle jetting and determined that the margins of safety for momentum flux at the uprated conditions do not change significantly from those at the present conditions.

The NRC staff reviewed the licensee’s analyses related to the effects of the power uprate on the hydraulic design of the core. In the letter from John Lamb, USNRC to Thomas Coutu, Site Vice President, Kewaunee Plant, “Kewaunee Nuclear Power Plant - Issuance of Amendment (TAC No. 5718),” April 4, 2003, the NRC staff concluded that NMC adequately accounted for the effects of the power uprate to 1673 MWt on the hydraulic design and demonstrated that the design is not susceptible to thermal-hydraulic instability. Therefore, the design will continue to meet the requirements of GDC 10 following implementation of the power uprate. Because the design will continue to meet GDC 10 requirements, the NRC staff finds the power uprate to 1673 MWt acceptable.

3.2.2.10 LOCA Loads

To evaluate the internal forces generated by a LOCA, NMC uses a leak-before-break (LBB) methodology and a branch-line break location (i.e., pressurizer surge line, accumulator line, or residual heat removal line). In the letter from M. E. Warner, Nuclear Management Company to USNRC, “Licence Amendment Request 187 to the Kewaunee Nuclear Power Plant Technical Specifications, Conforming Technical Specification Changes for Use of Westinghouse VANTAGE + Fuel,” Docket No. 50-305, License No. DPR-43, Letter No. NRC-02–067, dated July 26, 2002, using the square-root-of-sum-of-squares method identified in Appendix A to SRP

4.2, the licensee demonstrated that the combined loadings were less than the allowable limits, thus ensuring maintenance of a coolable core geometry. Therefore, in the letter from John Lamb, USNRC to Thomas Coutu, Site Vice President, Kewaunee Plant, “Kewaunee Nuclear Power Plant - Issuance of Amendment (TAC No. 5718),” April 4, 2003, the NRC staff found these loads acceptable. Additionally, for the power uprate, NMC states that the critical stresses in the core are bounded by existing analyses. Because the current analyses are bounding, the NRC staff finds them acceptable for the power uprate to 1673 MWt.

3.2.2.11 Long-Term Core Cooling

10 CFR 50.46(b)(5), “Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors - Long-Term Cooling,” establishes the long-term cooling requirements following a LOCA. 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 their operating procedures to ensure adequate hot leg switch-over times.

However, KNPP is an upper plenum injection (UPI) plant, where the low head ECCS pumps (RHR pumps) deliver flow to the core deluge nozzles directly into the upper plenum. Because of this design, the hot-leg switchover procedure that is applied at some Westinghouse plants does not apply for KNPP. As the RCS pressure is below the RHR shutoff head, the RHR pumps will continuously inject ECCS fluid directly above the core. This injection, in turn, will cause flushing and mixing of the upper plenum and upper core. Therefore, boron precipitation during the long-term cooling phase of a LBLOCA should not occur. Since KNPP is a UPI plant, the NRC staff concludes that boron precipitation is not an issue during the design-basis LBLOCA. Therefore, the NRC staff finds long-term core cooling acceptable for the power uprate to 1673 MWt.

3.2.2.12 WCAP-15591, “Calorimetric RCS Flow Measurement Uncertainty”

The NRC staff audited the RCS aspects of Section 3.1.a of WCAP-15591, “Calorimetric RCS Flow Measurement Uncertainty” from the January 13, 2003, submittal, to assess the determination of RCS flow rate via a plant secondary side calorimetric heat balance.

The licensee uses the equivalent of the following equation for determination of RCS flow rate (Note: Dimensional information and the number of loops have been removed for simplicity.):

$$W = \{Q_{SG} - Q_P + Q_L\} / (h_H - h_C) \qquad (1)$$

where:

W = flow rate
Q_{SG} = calorimetrically-determined steam generator thermal output
Q_P = RCP heat addition rate
Q_L = RCS net heat loss rate
h_H = hot-leg enthalpy (determined at T_H and P)
h_C = cold-leg enthalpy (determined at T_C and P)
T_C = cold-leg temperature (nominal value = 542.1 °F)
T_H = hot-leg temperature (nominal value = 608.5 °F)
P = pressurizer pressure (nominal value = 2250 psia).

The nominal values correspond to a 1757 MWt NSSS power. Note that Q_{SG} and Q_P are associated with heat outside of the reactor vessel, whereas Q_L includes all heat loss. Yet, h_H - h_C is the enthalpy change between the locations of measurement of T_C and T_H, which includes sections of the hot and cold-leg pipes and the reactor vessel. This raises a question of consistency in treating the terms.

The Q_L term contributors are stated to consist of the following:

charging flow (+)
letdown flow (-)
seal injection flow (+)
RCP thermal barrier cooler heat removal (-)
pressurizer spray flow (-)
pressurizer surge line flow (+)
component insulation heat losses (-)
component support heat losses (-)
CRDM heat losses (-).

A single calculated sum for 100 percent reactor thermal power operation is used for these losses or heat inputs.

To assess the calculation basis, the NRC staff considered a general control volume and the sum of the forms of energy entering and leaving the control volume boundary:

heat + [mass flow rate]$_{in}${internal energy + flow energy + kinetic energy + potential energy}$_{in}$ =
work + [mass flow rate]$_{out}${internal energy + flow energy + kinetic energy + potential energy}$_{out}$

The control volume is constant (the boundaries are rigid), and heat is the only form of energy passing through the control volume surface. This equation may be written as follows:

$$Q + W_{in} \{ u + [P\,V + v^2 / (2\,g) + Z\,] / f\, \}_{in} = \omega + W_{out} \{ u + [P\,V + v^2 / (2\,g) + Z\,] / f\, \}_{out} \qquad (2)$$

where:

Q =	heat addition rate
u =	internal energy per unit weight
P =	pressure
V =	volume per unit weight
v =	velocity
g =	gravitation constant
Z =	elevation
f =	conversion factor
ω =	work performed by the fluid

Since, by definition, enthalpy is:

$$h = u + P\,V / f \qquad (3)$$

Equation 2 may be written as:

$$Q + W_{in} \{ h + [v^2 / (2 g) + Z] / f \}_{in} = \omega + W_{out} \{ h + [v^2 / (2 g) + Z] / f \}_{out} \quad (4)$$

Now, the NRC staff selected the control volume to enclose half of the reactor vessel and the corresponding loop pipes between the reactor vessel and the locations of T_h and T_c. The NRC staff assumed there was no fluid addition or removal other than via the hot and cold-legs at the control volume boundary, so that $W_{in} = W_{out}$. As there is no work done by the system within this control volume, $\omega = 0$. As the hot and cold-leg pipe elevations are identical, $Z_{in} = Z_{out}$. The NRC staff also assumed that $v_{in} = v_{out}$; an assumption confirmed to be correct by considering the density and flow areas. Equation 4, therefore, reduces to:

$$Q + W\ h_c = W\ h_h \quad (5)$$

But Q is the heat generated in the core, $Q_{core,}$ minus that portion of Q_L associated with the reactor vessel and the pipes between the locations of T_h and T_c, $Q_{loss\Delta T}$. Thus,

$$W = (Q_{core} - Q_{loss\Delta T}) / (h_h - h_c) \quad (6)$$

A straightforward RCS heat balance shows that:

$$Q_{SG} = Q_{core} + Q_P - Q_L \quad (7)$$

Equations 6 and 7 combine to yield:

$$W = \{Q_{SG} - Q_P + Q_L - Q_{loss\Delta T}\} / (h_H - h_C) \quad (8)$$

The $Q_{loss\Delta T}$ term does not appear in the licensee's Equation 1. Thus, the licensee over-predicts the flow rate when it uses Equation 1. If the NRC staff limits consideration to loss through insulation and structure, assumes the total heat loss is about a quarter of the RCP heat input, takes the RCP heat input as 5 MWt/RCP, and assumes half of the heat loss occurs between T_H and T_C, the NRC staff estimates the over-prediction to be about 150 gpm or 0.08 percent.

During a telephone conference call on May 22, 2003, for which the meeting summary is contained in ADAMS Accession No. ML031470688, the NRC staff learned that the licensee made several additional assumptions when calculating W. These assumptions and the NRC staff's assessment are as follows:

Assumption	Assessment
h_H and h_C are determined at pressurizer pressure as opposed to determination at the T_H and T_C locations.	Kewaunee's nominal value $\Delta h = 87.961$. The staff estimates a correct calculation provides $\Delta h = 87.974$. Kewaunee over-predicts RCS flow rate by 26 gpm, which is a very small effect.

<table>
<tr><th>Assumption</th><th>Assessment</th></tr>
<tr><td>RTD manifolds are installed at Kewaunee, but flow through the manifolds is neglected.</td><td>Manifold flow bypasses the elbow tap flow measurement locations and cold leg manifold flow bypasses the reactor vessel. Thus, the elbow taps are calibrated with a local flow that differs from the reactor vessel flow. However, the RCS is then operated under conditions identical to the elbow tap calibration conditions. Consequently, the elbow taps provide the correct reactor vessel flow rate.</td></tr>
<tr><td>Letdown and makeup are included as total values without consideration of which loops are actually affected.</td><td rowspan="2">The NRC staff does not know the locations of the RCS connections with respect to the temperature measurement locations, and the NRC staff does not know the flow rates when Kewaunee makes its calorimetric determination. Note also that the NRC staff analysis was developed with an assumption that there was no mass flow into or out of the control volume except for flow in the hot and cold legs. The analyses may need to be developed with consideration of these effects. The NRC staff's judgement is that the effect of the items omitted from the analyses will be of little consequence for purposes of a 1.4 percent power increase.</td></tr>
<tr><td>Pressurizer spray flow and pressurizer surge line flow are included without consideration of which loops are actually affected.</td></tr>
</table>

On the basis of the audit, the NRC staff concludes that the licensee's flow determination appears to contain non-conservative biases of about 0.1 percent that were not considered in the licensee's determination of flow rate bias. The NRC staff also did not evaluate the licensee's analysis of the effect of letdown, makeup, RCP cooling, RCP seal injection, and the pressurizer on predicted RCS flow rate. The NRC identified potential areas of concern will be addressed during the NRC staff's review of the licensee's 6-percent stretch power uprate submitted by the licensee on May 22, 2003.

WCAP-15591, page 50 states that the RCS flow uncertainty used in the safety analyses is ±4.3 percent random with an allowance for 0.1 percent bias for the calorimetric measurement based on the venturi in the feedwater. The allowed bias is approximately the same as the bias that the NRC staff identified. The stated calculated uncertainty is substantially less than the uncertainty used in the safety analysis and the difference is more than sufficient to cover any residual concerns with the identified bias. The NRC staff further judges that the ±4.3 percent uncertainty is sufficient to address any letdown, makeup, RCP cooling, RCP seal injection, and pressurizer considerations. Consequently, the NRC staff concludes that the licensee's determination of RCS flow rate, as audited above, is acceptable for purposes of the requested 1.4 percent MUR power uprate.

3.2.3 Summary

The NRC staff has reviewed the licensee's safety analyses of 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 NSSS, and (5) LOCA and non-LOCA transient analyses. The NRC staff concludes that the results of licensee's analyses related to these areas continue to meet the applicable acceptance criteria following implementation of the proposed MUR power uprate. Where additional evaluations/analyses were necessary, the NRC staff has reviewed these evaluations and analyses and finds that the licensee has satisfactorily addressed the areas discussed above, the supporting safety analyses were performed using NRC-approved methods, the input parameters of the analyses adequately represent the plant conditions at the proposed uprated power level, and the analytical results meet the applicable acceptance criteria. Based on the above, the NRC staff finds the proposed MUR 1.4 percent power uprate acceptable with respect reactor systems performance.

3.3 Electrical Systems

3.3.1 Regulatory Evaluation

The NRC staff review in the area of electrical engineering covers the impact of the proposed MUR power uprate on (1) grid stability, including performance of the main generator, main transformer, isophase bus, and unit auxiliary transformer/reserve auxiliary transformer, (2) emergency diesel generator loading, (3) station blackout, and (4) environmental qualification of electrical equipment (NRC RIS 2002-03, Attachment 1, Section V). This review was conducted to verify that the results of licensee analyses related to these areas continue to meet the requirements of the AEC criterion as issued in the "Safety Evaluation of the Kewaunee Nuclear Power Plant" on July 24, 1972, supplemented December 18, 1972, and May 10, 1973; 10 CFR 50.63; and 10 CFR 50.49 following implementation of the proposed MUR power uprate.

3.3.2 Technical Evaluation

3.3.2.1 Technical Evaluation for Environmental Qualification for Electrical Equipment

The term environmental qualification (EQ) 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. Although the amendment is for power uprate of 1.4 percent, the licensee analyzed the impact of 7.4 percent power uprate on the normal design temperatures and the environmental conditions in the containment, auxiliary building and the turbine building. The evaluation showed that these areas can be maintained within normal ranges by the existing heating, ventilation, and air conditioning (HVAC) systems. Therefore, the normal pressure, temperature and humidity conditions in these areas will not be impacted and are bounded by the current design. The licensee also analyzed the impact of 7.4 percent power uprate on the environmental conditions in the areas outside of containment and these conditions remain bounded by the current EQ plan. This 7.4 percent power uprate encompasses the 1.4 percent MUR power uprate. The effects of the power uprate on the normal containment dose were also evaluated and the KNPP EQ Plan will be updated to include the new containment exclusion areas for the pressurizer, steam generator, and reactor coolant pump vaults.

Accidents causing the most severe environments include the main steamline breaks (MSLBs) inside containment, high-energy line breaks (HELB) outside containment, and LOCAs. The

licensee performed the analyses at 1683 MWt (102 percent current power) for the MSLB inside as well as outside containment and the results bound the conditions for the 1.4 percent MUR uprate. Therefore, there are no changes to EQ parameters for a steam line break accident. The licensee analyzed the LOCA containment response at 1683 MWt (102 percent of current rated power) and the results bound the conditions for the 1.4 percent MUR uprate. Therefore, the containment analysis does not change for the MUR uprate and there are no changes in temperature, pressure, and humidity following a LOCA. The MUR power uprate does not affect the post-accident radiation environments, as they were developed using a source term that assumed a core power of 1721 MWt (4.3 percent). This encompasses the 1.4 percent power uprate.

The NRC staff has reviewed the licensee's submittal regarding the effects of the proposed power uprate on EQ of the electrical equipment and concludes that the electrical equipment continues to meet the relevant requirements of 10 CFR 50.49. Therefore, the NRC staff finds the proposed 1.4 percent MUR power uprate acceptable with respect to EQ of electrical equipment.

3.3.2.2 Technical Evaluation for the Offsite Power System

The offsite power system includes two or more physically independent circuits capable of operating independently of the onsite standby power sources. The NRC staff's review covers the information, analyses and documents for the offsite power system and the stability studies for the electrical transmission grid. The focus of the review relates to the basic requirement that the loss of the nuclear unit, the largest operating unit on the grid or the loss of the most critical transmission line will not result in the loss of offsite power to the plant.

As described by the licensee's amendment request dated January 13, 2003, the main generator is rated at 622.389 megavars-ampere (MVA) {(560.15 MWe at 0.9 power factor [pf])}. The main generator operates at 560 MVA (556 MWe at 0.993 pf). The main generator provides power through the isolated phase bus at 22 kV to both the main transformer and the unit auxiliary transformer. The generator voltage is stepped up through the main transformer to a 345 kV transmission system. The preferred ac power source provides offsite ac power to the auxiliary power distribution system for the startup, operation, or shutdown of the station. The preferred ac power also provides a source of offsite ac power to all emergency loads necessary for the safe shutdown of the reactor. The electrical distribution system has been previously evaluated to conform to the AEC criterion as issued in the "Safety Evaluation of the Kewaunee Nuclear Power Plant" on July 24, 1972, supplemented December 18, 1972, and May 10, 1973.

3.3.2.2.1 Technical Evaluation for Grid Stability for the Offsite Power System

By letter dated March 14, 2003, the licensee provided additional information in support of the requested change to the Kewaunee Operating License. American Transmission Company (ATC) performed the grid stability study for the licensee on December 12, 2002, for 38 MWe increase. The study evaluated a 38 MWe uprate implemented in two phases: a 10 MWe addition in 2003, and the remaining 28 MWe addition in 2004. The study done at 10 MWe encompasses the power uprate of 1.4 percent. ATC did not identify any grid stability issues or facility upgrades based on the MUR power uprate of 1.4 percent.

The NRC staff reviewed the licensee's submittal and concluded that there is no impact of the power uprate on the grid stability. Therefore, the plant continues to meet the requirements of the AEC criterion as issued in the "Safety Evaluation of the Kewaunee Nuclear Power Plant" on July 24, 1972, supplemented December 18, 1972, and May 10, 1973, for grid stability with the 1.4 percent MUR power uprate.

3.3.2.2.2 Technical Evaluation for the Main Generator for the Offsite Power System

The main generator is rated at 622.389 MVA (560.15 MWe at 0.9 pf). With the anticipated power uprate of 7.4 percent, the main generator will operate at 622.4 MVA (595.7 MWe at 0.957 pf). The anticipated power uprate of 7.4 percent encompasses the 1.4 percent MUR power uprate. The power uprate of 1.4 percent does not affect the generator auxiliaries since the generator will continue to operate below its design rating. The main generator performance is bounded by existing design and is not impacted by the 1.4 percent MUR power uprate.

The NRC staff reviewed the licensee's submittal and concluded that KNPP will continue to operate the main generator within its design rating at the anticipated 1.4 percent MUR power uprate and, therefore, the design is acceptable.

3.3.2.2.3 Technical Evaluation for the Main Power Transformer for the Offsite Power System

The main power transformer is rated at 649.6 MVA. With the anticipated power uprate of 7.4 percent, the main power transformer will operate at 627.8 MVA. The anticipated power uprate of 7.4 percent is below the design rating of 649.6 MVA. The anticipated power uprate of 7.4 percent encompasses the 1.4 percent MUR power uprate.

The NRC staff reviewed the licensee's submittal and concluded that the anticipated power uprate of 1.4 percent is below the maximum main transformers design rating of 649.6 MVA and, therefore, operating the main power transformer at the 1.4 percent MUR uprated power condition is acceptable.

3.3.2.2.4 Technical Evaluation for the Isophase Bus for the Offsite Power System

The isophase bus is rated at 20 kiloamps (kA) for the main section and 1600 amps (A) for the branch section. With the power uprate of 7.4 percent, the current in the main section will be 18.9 kA and the current in the branch section will be 892 A. This is below the design rating of 20 kA for the main section and 1600 A for the branch section. The anticipated power uprate of 7.4 percent encompasses the 1.4 percent MUR power uprate.

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 isophase bus and, therefore, operating the isophase bus at the 1.4 percent MUR uprated power condition is acceptable.

3.3.2.2.5 Technical Evaluation for the Reserve Auxiliary Transformer for the Offsite Power System

The reserve auxiliary transformer (RAT) is rated at 40 MVA. The load on the RAT will be 35.6 MVA with the anticipated power uprate of 7.4 percent power. The increase in load due to power

uprate is bounded by its design rating of 40 MVA. The anticipated power uprate of 7.4 percent encompasses the 1.4 percent MUR power uprate.

The NRC staff reviewed the licensee's submittal and concluded that the startup transformer loading resulting from the 1.4 percent power uprate is below its maximum design rating and, therefore, operating the RAT at the 1.4 percent MUR uprated power condition is acceptable.

3.3.2.2.6 Technical Evaluation for the Main Auxiliary Transformer for the Offsite Power System

The main auxiliary transformer (MAT) is rated at 44.8 MVA. The load on the MAT will be 32.3 MVA with the anticipated power uprate of 7.4 percent power. The increase in the load due to power uprate is bounded by its design rating of 44.8 MVA. The anticipated power uprate of 7.4 percent encompasses the 1.4 percent MUR power uprate.

The NRC staff reviewed the licensee's submittal and concluded that the unit auxiliary transformer loading resulting from the 1.4 percent power uprate is below its maximum design rating and, therefore, operating the MAT at the 1.4 percent MUR uprated power condition is acceptable.

3.3.2.2 Technical Evaluation for the AC Onsite Power Systems

The 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 staff's review covers the descriptive information, analyses, and referenced documents for the ac onsite power system.

The emergency diesel generators (EDGs) supply the source of ac power following a loss of offsite power or under offsite power degraded voltage conditions. The EDGs automatically supply ac power to the Class 1E buses in order to provide motive and control power to equipment required for a safe shutdown of the plant. The loading on the EDGs was evaluated for full 7.4 percent power uprate for maximum loading for a design-basis accident (DBA) (i.e., LOCA/loss of offsite power). The evaluation was to identify any load changes, the impact of the load changes for existing analysis, and confirm the diesel generator would remain capable of performing its safety-related functions.

Review of the NSSS loads and the balance of plant (BOP) loads showed that there were no loads fed by the EDGs under DBA conditions that would increase for uprated conditions. Therefore, there was no impact on the existing analysis and it remains bounding at uprated conditions. Additionally, there were no load additions or modifications to the EDG loading. Therefore, the existing protection schemes are acceptable and it is concluded that the EDGs are adequate for the 1.4 percent MUR power uprate.

3.3.2.3 Technical Evaluation for the Direct Current (DC) Onsite Power Systems

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.

The licensee reviewed the dc loading requirements and did not identify any additional reactor power-dependent loads. Operation at the 1.4 percent MUR uprated power level does not increase any loads or revise control logic. Therefore, the existing dc onsite power systems are acceptable and it is concluded that the dc onsite power systems are adequate for the 1.4 percent MUR power uprate.

3.3.2.4 Technical Evaluation for Station Blackout (SBO)

A SBO refers to the complete loss of ac electric power to the essential and nonessential switchgear buses in a nuclear power plant. A 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 of the proposed power uprate on the plant's ability to cope with and recovery from an SBO event are based on 10 CFR 50.63.

The only potential impact of 1.4 percent MUR power uprate on the ability of the plant to withstand and recover from an SBO is the increased decay heat that must be removed from the RCS. The methodology and assumptions associated with the SBO analysis with regard to equipment operability are unchanged with the uprate. There is no change in the ability of the turbine-driven auxiliary feedwater pump to support reactor heat removal due to uprate. The TS minimum required volume in the condensate storage tank is 39,000 gallons. The volume remains acceptable for the MUR power uprate since it is based on 102 percent of the current rated power of 1650 MWt. The 2 percent uncertainty on the current core power of 1650 MWt bounds the uprate of 1673 MWt (a 1.4 percent uprate with 0.6 percent uncertainty). Therefore, the ability of the plant to respond to a SBO will not be altered due to 1.4 percent MUR power uprate.

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 NRC staff concludes that the licensee has adequately evaluated the effects of the proposed 1.4 percent MUR 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 1.4 percent MUR power uprate. Therefore, the NRC staff finds the proposed 1.4 percent MUR power uprate acceptable with respect to a SBO.

3.3.3 Summary

The NRC staff has reviewed the licensee's safety analyses of the impact of the proposed 1.4 percent MUR power uprate on (1) grid stability, including performance of the main generator, main transformer, isophase bus, and unit auxiliary transformer/reserve auxiliary transformer, (2) EDGs, (3) SBO, and (4) environmental qualification of electrical equipment. The NRC staff concludes that the results of licensee's analyses related to these areas continue to meet the requirements of the AEC criterion as issued in the "Safety Evaluation of the Kewaunee Nuclear Power Plant" on July 24, 1972, supplemented December 18, 1972, and May 10, 1973, 10 CFR 50.63, and 10 CFR 50.49 following implementation of the proposed 1.4 percent MUR power uprate. Therefore, the NRC staff finds the proposed 1.4 percent MUR power uprate acceptable with respect to electrical engineering.

3.4 Civil and Engineering Mechanics

3.4.1 Regulatory Evaluation

The NRC staff review in the area of mechanical and civil engineering covers the structural and pressure boundary integrity of NSSS and BOP systems and components (NRC RIS 2002-03, Attachment 1, Section IV, Items 1.A, 1.B, and 1.D). The NRC staff's review focuses on the impact of the proposed MUR power uprate on NSSS piping, components, and supports; BOP piping, components, and supports; reactor vessel and supports; CRDM; SG and supports; RCPs and supports; pressurizer and supports; reactor pressure vessel internals and core supports; and safety-related valves. Technical areas covered by this review include stresses, cumulative usage factors, flow induced vibration, HELB locations, jet impingement and thrust forces, and safety-related valve programs. The review is conducted to confirm that (1) the results of the analyses continue to meet code allowable limits of the American Society of Mechanical Engineers (ASME) code of record for the plant, (2) the safety-related valves will continue to perform acceptably, and (3) the safety-related valve programs will continue to be adequate. Guidance for the NRC staff's review of the topics within the mechanical and civil engineering area are contained in Chapters 3 and 5 of NUREG-0800.

3.4.2 Technical Evaluation

The NRC staff reviewed the KNPP 1.4 percent MUR power uprate amendment, as it relates to the effects of the power uprate on the structural and pressure boundary integrity of the NSSS and BOP systems. Affected components in these systems included piping, in-line equipment and pipe supports, the reactor pressure vessel (RPV), core support structures, reactor vessel internals, SGs, CRDMs, RCPs, and pressurizer.

3.4.2.1 Reactor Vessel

The proposed power uprate will increase the core power by approximately 1.4 percent above the currently licensed level of 1650 MWt. The licensee reported that the power increase will result in changing the design parameters given in Table IV.B-1, Attachment 2 of the January 13, 2003, submittal. Table IV.B-1 provides a comparison of the current design parameters, the revised design parameters at the proposed uprated power level of 1673 MWt, and the design parameters at the core power level of 1772 MWt that were used as the bounding power uprate analysis for this amendment request.

The licensee evaluated the reactor vessel for the effects of the revised design conditions provided in Table IV.B-1 of the January 13, 2003, submittal, with respect to the core power level of 1772 MWt. The evaluation was performed for the limiting vessel locations with regard to stresses and cumulative fatigue usage factors (CUFs) in each of the regions, as identified in the reactor vessel stress reports for the core power uprated conditions. The regions of the reactor vessel affected by the power uprate include outlet and inlet nozzles, the RPV (main closure head flange, studs, and vessel flange), CRDM housing, safety injection nozzles, external supports brackets, bottom head to shell juncture, core support guides, and the instrumentation tubes. In its amendment request, the licensee indicated that the evaluation of the reactor vessel was performed in accordance with the ASME Boiler and Pressure Vessel (B&PV) Code, Section III, 1968 Edition with Addenda through the Winter 1968, which is the code of record. Table 5.1-1 of the January 13, 2003, submittal, provides the calculated maximum stresses and

CUFs for the reactor vessel critical locations. The results indicate that the maximum primary plus secondary stresses are within the code allowable limits, and the CUFs remain below the allowable ASME Code limit of 1.0. Therefore, the NRC staff agrees with the licensee's conclusion that the current design of the reactor vessel continues to be in compliance with licensing basis codes for the proposed 1.4 percent MUR power uprate condition.

3.4.2.2 Reactor Core Support Structures and Vessel Internals

The licensee evaluated the reactor vessel core support and internal structures. The limiting reactor internal components evaluated include the lower core plate, lower support columns, core barrel, baffle plates, baffle/barrel region bolts, guide tubes and support pins, upper core plate, and upper support columns. The licensee indicated that the reactor internal components were not licensed to the ASME B&PV Code; however, the design of the reactor internals was evaluated in accordance with requirements of Subsection NG of the 1989 Edition of the ASME Section III Code.

The licensee evaluated these critical reactor internal components considering the revised design conditions provided in Table IV.B-1 of the January 13, 2003, submittal for KNPP for a core power of 1772 MWt, which is bounding for the requested power level of 1673 MWt. The licensee indicated that the calculated stress for the limiting reactor internals are acceptable within the Code allowable limits. The calculated CUFs as provided in the amendment request are less than the ASME code allowable limit of 1.0. In addition, the licensee evaluated the flow induced vibration, which was found to remain within the allowable limits for the power uprate condition. Based on the above evaluations, the NRC staff agrees with the licensee's conclusion that the reactor internal components at KNPP will be structurally adequate for the proposed 1.4 percent MUR power uprate.

3.4.2.3 Control Rod Drive Mechanisms

The pressure boundary portion of the CRDMs are those exposed to the vessel/core inlet fluid. Both of the KNPP units have the L-106A CRDMs, full-length mechanisms manufactured by Westinghouse. The licensee evaluated the adequacy of the CRDMs by reviewing the original E-Specification and the generic evaluation for L-106A CRDMs to compare the design-basis input parameters against the revised design conditions in Table IV-B-1 of Appendix 2 to the January 13, 2003, submittal for the power uprate. The licensee also indicated in the January 13, 2003, amendment request that the key input parameters such as the hot-leg maximum temperature, the maximum pressure fluctuation and the maximum temperature fluctuation for the uprated power condition are bounded by the design-basis analysis. The power uprate evaluation was performed using Section III of ASME B&PV Code, 1965 Edition with addenda through Summer 1966, which is the Code of record. Tables 5.4-2 and 5.4-3 of Attachment 3 to the January 13, 2003, submittal, provides the calculated stresses and CUFs for the critical CRDM locations at the proposed power uprate conditions, which are less than the ASME Code allowable limits.

On the basis of its review, the NRC staff concurs with the licensee's conclusion that the current design of CRDMs continues to be in compliance with licensing basis codes and standards for the proposed 1.4 percent MUR power uprate.

3.4.2.4 Steam Generators

The licensee reviewed the existing structural and fatigue analyses of the SGs at KNPP and compared the power uprate conditions with the design parameters of the analysis of record for the Model 54F SGs at KNPP. The comparison of key parameters are shown in Table 5.7-1 of the January 13, 2003, submittal, for the current rated power and the proposed power uprate conditions.

As a result of its review, the licensee indicated that all components, except the feedwater nozzle, thermal sleeve, and the J-nozzle-to-feed ring weld fatigue analyses, experience the primary or secondary side temperature and pressure gradients when operating at the power uprate condition and are bounded by the KNPP existing design-basis analyses. The licensee performed evaluations of these affected components for the power uprate condition. The calculated maximum ranges of stress intensities are provided in Tables 5.7-2, 5.7-3 and 5.7-4 of the January 13, 2003, submittal. The fatigue calculations were revised to reflect the power uprate condition. As a result of its evaluation, the licensee indicated that the stress intensity ranges and fatigue usage factors provided are in compliance with the requirements of the ASME Code, Section III, 1986 Edition through the Winter 1987 Addenda, which is the Code of record at KNPP, and are, therefore, acceptable. The NRC staff concurs with the licensee's conclusion.

In addition, the licensee evaluated the flow induced vibration of the U-bend tubes for Model 54F SGs at KNPP. The licensee indicated that the calculated fluid-elastic stability ratio is less than the allowable limit of 1.0, and that the maximum fluid induced displacement values due to turbulence and the vortex shedding are insignificant. As a result, the licensee concluded that the flow induced vibration of SG tubes will remain within the allowable limits for the power uprate. The NRC staff concurs with the licensee's conclusion.

On the basis of its review, the NRC staff concludes that the licensee has demonstrated the maximum stresses and CUFs for the limiting SG components to be within the Code allowable limits and, therefore, acceptable for the proposed 1.4 percent MUR power uprate.

3.4.2.5 Reactor Coolant Pumps

The licensee reviewed the existing design basis analyses of the KNPP RCPs to determine the impact of the revised design conditions in Table IV.B-1. The licensee indicated that the Kewaunee RCPs predate the inclusion of pumps in the ASME Code, Section III, and are not Code stamped. Code editions used for the power uprate evaluation for the Kewaunee pumps range from the 1968 Edition with Winter 1970 Addenda, to the 1971 Edition with 1972 addenda.

After the core power uprate, the RCS pressure remains unchanged. The licensee indicated that the design parameter of the RCP temperature (reactor pressure vessel inlet) as provided in Table 5.6-1 of the January 13, 2003, submittal, for the power uprate condition is less than the present design-basis. Also, there are no significant changes to the design thermal transients. CUFs for RCP limiting components shown in Table 5.6-3 of the January 13, 2003, submittal are below the allowable limit of 1.0. Stresses for the RCP vertical and lateral supports as shown in Table 5.5.1-3 of the January 13, 2003, amendment request are less than the allowable. As a result of the evaluation, the licensee concluded that the current KNPP Model 93A RCPs remain

in compliance with the applicable ASME Code requirements for structural integrity at the proposed power uprate.

On the basis of its review, the NRC staff concurs with the licensee's conclusion that the RCPs, when operating at the proposed 1.4 percent MUR power uprate conditions, will remain in compliance with the requirements of the Codes and standards under which the KNPP were originally licensed and the NRC staff finds this acceptable.

3.4.2.6 Pressurizer

The licensee evaluated the limiting design locations of the pressurizer components. The components in the lower end of the pressurizer (such as the surge nozzle, lower head well and penetration, and support skirt) are affected by the pressure and the hot leg-temperature. The components in the upper end of the pressurizer (such as the spray nozzle, instrument nozzle, safety and relief nozzle, and upper head and shell) are affected by the pressure and the cold-leg temperature for operation at the uprated conditions. The evaluation was performed using the ASME Code, Section III, 1965 Edition, through Summer 1966 addenda, which is the Code of record for KNPP pressurizer.

The key parameters in the current KNPP pressurizer stress report were compared against the revised design conditions in Table IV.B-1 for the January 13, 2003, proposed power uprate amendment. The limiting operating conditions of the pressurizer occur when the RCS pressure is high and the RCS hot-leg (T_{hot}) and cold-leg (T_{cold}) temperatures are low. Because the proposed power uprate does not change the maximum RCS pressure and the pressurizer temperature (T_{sat}), the existing design-basis analyses with the lowest T_{hot}, that maximizes the thermal stresses in components at the lower end of the pressurizer, remain bounding for the proposed power uprate. However, there is a slight increase in thermal stress due to lower T_{cold} at the power uprate condition. The evaluation was performed to demonstrate the adequacy of the components in the upper end of the pressurizer. The calculated CUFs for limiting pressurizer locations at the uprated condition were found to be below the code allowable limit of unity as shown in Table 5.8-2, Appendix 4 of the January 13, 2003, submittal. As a result of the above evaluation, the licensee concluded that the existing pressurizer components will remain adequate for plant operation at the proposed 1.4 percent power increase while the RCS pressure remains unchanged. The NRC staff agrees with the licensee's conclusion and the NRC staff finds the proposed 1.4 percent MUR power uprate acceptable related to the pressurizer.

3.4.2.7 Nuclear Steam Supplying System Piping and Pipe Supports

The proposed power uprate of KNPP involves the increase of temperature difference across the RCS. The licensee evaluated the NSSS piping and supports by reviewing the design-basis analysis against the uprated power design system parameters, transients and the LOCA dynamic loads. The evaluation was performed for the reactor coolant loop (RCL) piping, primary equipment nozzles, primary equipment supports, and the pressurizer surge line piping. USAS B31.1 Power Piping Code, 1967 Edition was used for the power uprate evaluation of KNPP RCS piping, except the surge line which was evaluated in accordance with requirements of the ASME B&PV Code Section III, 1986 Edition, which is the Code of record. The calculated stresses and CUFs are provided in Table 5.5.1-2 of the January 13, 2003, amendment request for the primary loop piping for the power uprate. The maximum calculated stresses and CUFs

are less than the code allowable limits. In its response to the NRC staff's RAI, the licensee provided a summary of the stresses and CUF for the surge line and they are below the code allowable stress limits and the fatigue usage factor limit of 1.0.

The licensee also indicated that the design transients used in the evaluation of the RCS piping systems and equipment nozzles are unchanged for the KNPP power uprate. The proposed power uprate does not change the maximum RCS pressure. The design-basis LOCA forces due to postulated primary loop guillotine breaks have been eliminated using the loop LBB methodology for KNPP. With the use of LBB technology, LOCA forces for the power uprate condition were derived based on postulation of breaks in three branch lines at the surge line nozzle on the hot-leg, the accumulator line nozzle at the cold-leg, and the RHR line nozzle on the hot-leg. As such, the design-basis LOCA hydraulic forcing functions are bounding for the LOCA loads at the uprated power condition. Furthermore, the deadweight and seismic loads are not affected by the power uprate. The licensee concluded that the existing stresses, fatigue usage factors and loads remain bounding for the power uprate for the NSSS components including the reactor cooling loop piping, the primary equipment nozzles, the primary equipment supports, pipe supports and the auxiliary equipment (i.e., heat exchangers, pumps, valves and tanks). Therefore, these components will continue to be in compliance with the Code of record at KNPP.

On the basis of its review of the licensee's submittal, the NRC staff concurs with the licensee's conclusion that the existing NSSS piping and supports, the primary equipment nozzles, the primary equipment supports, and the auxiliary lines connecting to the primary loop piping will remain in compliance with the requirements of the design bases criteria, as defined in the KNPP final safety analysis report and are, therefore, acceptable for the proposed 1.4 percent MUR power uprate.

3.4.2.8 Balance of Plant (BOP) Systems and Motor-Operated-Valves (MOVs)

The licensee evaluated the adequacy of the BOP systems based on comparing the existing design-basis parameters with those for the core power uprate conditions. The BOP piping systems that were evaluated for the power uprate include main steam, feedwater, SG blowdown, steam extraction, and auxiliary feedwater systems. The licensee evaluated these affected systems at the uprated power level using the change factors which were calculated based on ratios of the temperature, pressure, or flow rate at the power uprate condition, to the corresponding value at the current rated power condition. In its April 30, 2003, response to the NRC staff's RAI, the licensee provided the change factors for main steam, condensate, and feedwater piping systems. As a result of its evaluation, the licensee concluded that the existing design-basis analyses for the BOP piping, pipe supports, and components for operation at the proposed 1.4 percent power uprate condition at KNPP, will be in compliance with the Code of record USAS B31.1, Power Piping Code, 1967 Edition. The NRC staff agrees with the licensee's conclusion, because the calculated stresses provided by the licensee for the affected limiting main steam and feedwater systems are below the allowable stress limits, and are therefore, acceptable for the proposed 1.4 percent MUR power uprate.

The licensee also reviewed the programs, components, structures, and non-NSSS system issues as they relate to the power uprate. In Attachment 2 of the January 13, 2003, the licensee indicated that the NMC MOV program used the plant operational parameters that are bounding for the proposed power uprate. The maximum operating design system pressure

does not change as a result of the 1.4 percent power uprate. Therefore, the licensee concluded that the safety related MOVs at KNPP will continue to be capable of performing their intended functions at the uprated power condition.

The licensee reviewed the evaluation of generic letter (GL) 95-07 associated with the pressure locking and thermal binding for safety related gate valves. The licensee found that the existing analysis used pressure conditions that would not be affected by the 1.4 percent power uprate. The licensee reviewed the evaluation of the NMC GL 96-06 program regarding the over-pressurization of isolated piping segments. The licensee concluded that the existing evaluation for GL 96-06 was based on the containment integrity analysis performed at 102 percent of the current rated power and is therefore bounding, for the proposed power uprate of 101.4 percent rated power level. On the basis of the above review, the NRC staff concurs with the licensee's conclusions that the power uprate will have no adverse effects on the safety-related valves and that conclusions of the NMC GL 95-07, and GL 96-06, as well as GL 89-10 programs, remain valid, and are therefore, acceptable for the proposed 1.4 percent MUR power uprate.

As a result of the above evaluation, the NRC staff concludes that the BOP piping, pipe supports, equipment nozzles and valves, remain acceptable and continue to satisfy the design-basis requirements for the proposed power uprate and are therefore, acceptable for the proposed 1.4 percent MUR power uprate.

3.4.3 Summary

The NRC staff has reviewed the licensee's evaluation of the impact of the proposed MUR power uprate on NSSS and BOP systems and components with regard to stresses, CUFs, flow induced vibration, HELB locations, jet impingement and thrust forces, and safety-related valve programs and concludes that the these areas will continue to be acceptable following implementation of the proposed MUR power uprate. Therefore, the NRC staff finds the proposed 1.4 percent MUR power uprate acceptable with respect to the areas of civil and mechanical engineering.

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, 10 CFR 50.67, and/or 10 CFR Part 50, Appendix A, GDC-19, as applicable, following implementation of the proposed MUR power uprate.

3.5.2 Technical Evaluation

The NRC staff reviewed the impact of the proposed 1.4 percent MUR power uprate on DBA radiological analyses.

In March 2002, the licensee requested revisions to the radiological consequence analyses for DBAs in the Kewaunee Updated Final Safety Analysis Report. The proposed revisions

implemented the AST as described in RG 1.183, "Alternative Radiological Source Terms for Evaluating Design-Basis Accidents at Nuclear Power Reactors" and pursuant to 10 CFR 50.67, "Accident Source Terms." In its re-evaluation of the DBA radiological consequence analyses in the proposed revision, the licensee used the reactor core and fission product activities which were based on a reactor thermal power level of 1683 MWt (2 percent above the licensed reactor power level of 1650 MWt).

The licensee demonstrated in the proposed revision that after implementing the AST, the Kewaunee engineered safety feature system will continue to provide assurance that the total radiological consequences of DBAs will meet the dose criteria specified in 10 CFR 50.67 and GDC 19. The NRC staff verified the licensee's determination with its independent radiological consequence dose calculations. On March 17, 2003, the NRC issued License Amendment No. 166 for Facility Operating License No. DPR-43 for the Kewaunee Nuclear Power Plant. This amendment accepts the revision to the radiological consequence analyses which accounts for a reactor thermal power level of 1683 MWt.

3.5.3 Summary

The current licensed Kewaunee DBA radiological consequence analyses are based on a reactor power level (1683 MWt) that bounds the requested uprate power (1673 MWt). Therefore, the NRC staff concludes that Kewaunee will continue to meet the applicable dose acceptance criteria following implementation of the proposed 1.4 percent MUR power uprate. The NRC staff finds the proposed MUR power uprate acceptable with respect to dose consequence analyses.

3.6 Materials and Chemical Engineering

3.6.1 Regulatory Evaluation

The NRC staff review in the area of materials and chemical engineering covers reactor vessel integrity, SG tube integrity, and erosion corrosion programs (NRC RIS 2002-03, Attachment 1, Section IV, Items 1.C through 1.F). The NRC staff's review in this area focuses on the impact of proposed MUR power uprate on pressurized thermal shock calculations, fluence evaluations, heatup and cooldown P-T limit curves, low-temperature overpressure protection, upper-shelf energy, surveillance capsule withdrawal schedules, licensee programs for addressing SG tube degradation mechanisms, and erosion/corrosion. This review is conducted to verify that the results of licensee analyses related to these areas continue to meet the requirements of 10 CFR 50.60, 10 CFR 50.61, and 10 CFR 50.55a; and 10 CFR Part 50, Appendices G and H, following implementation of the proposed MUR power uprate. 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.

3.6.2 Technical Evaluation

3.6.2.1 Reactor Pressure Vessel

Regarding the KNPP RPV surveillance program and capsule withdrawal schedule, the licensee concluded in Section 5.1.2.5 of Attachment 3, page 5.1.2-13 of the January 13, 2003, letter:

> A calculation of [reference temperature nil ductility] RT_{NDT} at 33 [effective full power years] EFPY was performed to determine if the increased fluences alter the number of capsules to be withdrawn from Kewaunee. This calculation determined that the maximum [change of reference temperature nil ductility] ΔRT_{NDT} using the uprated fluence for Kewaunee at [end-of-life] EOL is greater than 200 ^{0} F. These ΔRT_{NDT} values would require five capsules to be withdrawn from Kewaunee. However, due to changes in capsule fluence, Capsule T should be removed before it receives a fluence of $7.12x10^{19}$ n/cm^2 [neutron per centimeter squared] (E>1.0 MeV [million electron volts]) (i.e., twice the peak vessel EOL fluence of $3.56x10^{19}$ n/cm^2 (E>1.0 MeV). This capsule may be held without testing following withdrawal.

Consistent with the requirements of Appendix H to 10 CFR Part 50, the licensee utilizes the guidance in the 1982 edition of American Society for Testing and Materials (ASTM) E185 (E185-82) to define the number of surveillance capsules in the KNPP surveillance program and their withdrawal requirements. For RPVs that demonstrate shifts in material transition temperature (ΔRT_{NDT}) in excess of 200 ^{0} F, E185-82 requires that the surveillance program have 5 capsules in it with the last pulled at a fluence between one and two times the end of license fluence at the RPV inside diameter. ASTM E185-82 permits the last surveillance capsule to be held without testing following withdrawal. The licensee's RPV surveillance capsule withdrawal schedule will continue to be consistent with the provisions of Appendix H to 10 CFR Part 50, and ASTM E185-82.

By letter dated February 21, 2001, the NRC staff granted exemptions from the requirements of 10 CFR Part 50, Appendices G and H, and 10 CFR 50.61 for KNPP. The NRC staff stated that using the NRC staff's Master Curve-based approach would justify the KNPP cooldown P-T limit curves from 28 EFPY to 33 EFPY of operation if requested by the licensee. The licensee used the NRC staff's Master Curve-based approach in its January 13, 2003, submittal.

Regarding the topic of the RPV P-T limits, the licensee concluded in Section 5.1.2.5 of Attachment 3, page 5.1.2-13 of the January 13, 2003, letter that:

> This review indicates that the revised adjusted reference temperature (ART) after the power uprate program will be more restrictive than that used in developing the current ART values for Kewaunee at 33 EFPY. Therefore, a change in applicability date is required. The 33 EFPY P-T curves for Kewaunee will be applicable to 31.1 EFPY after the uprating.

The KNPP TSs contain 33 EFPY P-T limit curves. Based on the uprated fluence, the current vessel end of life (EOL) (33 EFPY) fluence will be reached at 31.1 EFPY. Hence, the ART value upon which the existing P-T limit curves are based will also be reached at 31.1 EFPY instead of 33 EFPY, and the current P-T limits will be applicable up to 31.1 EFPY. Therefore, the licensee's proposal to limit the existing heatup and cooldown curves to a period of applicability through 31.1 EFPY of operation is consistent with the requirements of Appendix G to 10 CFR Part 50.

The licensee provided information regarding changes to operating temperature, flow rates, and neutron fluences which result from the power uprate. The licensee's evaluations of the critical components indicated that the structural integrity of the reactor internals will be maintained at the uprated RCS conditions.

Based on the information provided by the licensee regarding insignificant changes to operating temperature, flow rates, and neutron fluences which result from the power uprate, the NRC staff agrees that the integrity of the RPV internals will be maintained such that the licensee's ability to meet the regulatory requirements in 10 CFR 50.46 regarding ECCS performance, and maintaining a coolable core geometry will not be adversely impacted.

Finally, regarding the PTS and upper shelf energy (USE) analyses for the KNPP RPV (Section 5.1.2.5 of Attachment 3, page 5.1.2-13 of the January 13, 2003, letter), the licensee provided RT_{PTS} and USE values for the beltline materials of the KNPP, vessel and concluded:

> **Pressurized Thermal Shock**
>
> The calculated neutron fluence values for the Power Uprate Program condition at Kewaunee have increased over the current fluence. Based on licensee's evaluation all [reference temperature pressurized thermal shock] RT_{PTS} values will remain below the NRC screening criteria values using projected Power Uprate Program fluence through EOL (33 EFPY).
>
> **Upper Shelf Energy**
>
> The revised fluence projections associated with the Power Uprate Program have increased the fluence projections used in developing the current predicted EOL USE values. All USE values for Kewaunee will maintain a level above the 50 ft-lb. level at end of license (33 EFPY).

The NRC staff has evaluated the information provided by the licensee as well as information contained in the NRC staff's Reactor Vessel Integrity Database. Based on the revised fluence values noted, the NRC staff independently confirmed that the KNPP RPV materials would continue to meet the PTS screening criteria requirements of 10 CFR 50.61 and the USE requirements of Appendix G to 10 CFR Part 50 though EOL.

3.6.2.2 Flow accelerated corrosion (FAC) Program

FAC is a corrosion mechanism causing wall thinning of high energy pipes in the power conversion system which may lead to their failure. Since failure of these pipes may result in undesirable challenges to the plant's safety systems, the licensee has a program for predicting, inspecting, and repairing or replacing the components whose wall thinning exceeds the values required for their safe operation. In the submittal, the licensee stated predictive analysis was performed for a larger 7.4 percent power uprate using the CHECWORKs computer code developed by the Electric Power Research Institute. Although wear rates increased in some lines, the identified changes were not significant and were not projected to cause wear rates or inspection intervals to change significantly. The licensee stated the CHECWORKs model for KNPP will be updated following the plant power uprate, and that the results of the upgraded code would be factored into the ongoing FAC program surveillance/pipe repair plans. The NRC staff considers this licensee's action adequate for ensuring integrity of the high energy pipes; therefore, the NRC staff finds that the FAC program acceptable for the 1.4 percent MUR power uprate.

3.6.2.3 Structural Integrity and Primary-to-Secondary Pressure Differential Evaluation

The licensee performed a power uprate evaluation for the structural integrity of the SGs based on an existing analysis from a previous KNPP SG replacement project. KNPP SGs were replaced in 2001. The majority of the structural analyses performed in support of the KNPP SG replacement remained applicable for the uprated conditions. Therefore, only those components impacted by the revised feedwater temperature and flow rates associated with plant uprating required further evaluation. These included the feedwater nozzle, thermal sleeve, and J-nozzle-to-feedring weld. Stress analysis results from the limiting locations in the feedwater nozzle, thermal sleeve, and J-nozzle-to-feed ring weld were shown to remain within the ASME Code allowable limits.

Maximum primary-to-secondary side pressure differentials under normal conditions and upset transient conditions were analyzed and compared to the design pressure requirements in ASME B&PV Code, Section III. This analysis was performed using power uprate operating parameters and SG tube plugging levels of 0 percent and 10 percent. A 10 percent tube plugging level represents the maximum allowable tube plugging for a single KNPP SG. Analysis showed that the ASME Code requirements were met.

The NRC staff finds the licensee's evaluation to be acceptable and, therefore, the NRC staff concluded that the proposed 1.4 percent MUR power uprate for Kewaunee will not have significant impact on SG structural integrity.

3.6.2.4 Tube Vibration, Wear, and Repair Hardwear

An analysis was performed to evaluate the potential for increased tube wear resulting from the operation of the SGs in an uprated power condition. The licensee also evaluated the acceptability of various SG tube plug designs for the uprated power operating conditions. Results from the current design-basis vibration and wear analysis were modified to account for anticipated changes in secondary side thermal- hydraulic operating conditions due to the uprated power conditions. The licensee determined that the maximum projected increase in the wear that could occur for the SG tubes increased from approximately 3 mils to approximately 4 mils at the uprated condition. Any increase in wear would progress over many cycles, and would be observed during routine eddy current inspections.

The NRC staff finds the licensee's evaluation to be acceptable since the maximum projected increase in wear is small. Therefore, any additional wear that could challenge tube integrity would occur over many cycles and would be detected during routine tube inspection.

A structural evaluation was performed for mechanical and welded plug designs. Mechanical plug design included the short and long ribbed original equipment manufacturer plug design. Although there are no shop welded tube plugs in the Kewaunee SGs, this design was also evaluated to ensure it is acceptable for the power uprate conditions based on the 1989 ASME Code. Evaluation included the applicable transient stresses associated with plant uprating, plus cumulative fatigue design criteria per the ASME Code, Section III. Results from the analysis concluded that both mechanical plug designs satisfy all applicable stress and retention criteria for the power uprate condition. The welded plug calculations for uprated power operation were also shown to meet the allowable ASME Code values for stress and fatigue usage.

Some tube repair circumstances (e.g. removal of a tube plug by drilling/reaming prior to sleeve installation) may result in removal of a portion of the tube and weld metal. Therefore, an analysis was performed to evaluate the acceptability of a tube with 40 percent undercut, (i.e., removal of 40 percent tube/weld), operating at a 7.4 percent uprated power condition. Analysis of the 40 percent tube undercut demonstrated all stresses and fatigue usage values were within acceptable limits of the ASME Code.

The NRC staff finds the licensee's evaluation to be acceptable since analysis showed a tube in this condition meets ASME Code limits and does not exceed the technical specification tube repair limits.

3.6.2.5 Secondary Side Foreign Object Evaluation

The licensee indicated that the Kewaunee SGs are at an early stage in their service life and there have been no loose parts in the SGs at this time. Since the licensee recognized there is a potential for loose parts in the future, a generic loose parts evaluation was performed that addressed undefined loose parts in the SG under uprated power conditions.

The NRC considers the licensee's action acceptable since there have been no loose parts in the steam generator to date. The licensee also performs routine inspections that would detect the presence of loose parts in the future.

3.6.2.6 Regulatory Guide 1.121 Analysis

NRC RG 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes," describes an acceptable method for establishing the limiting safe condition of degradation in the tubes beyond which tubes found defective by the established in-service inspection shall be removed from service. The allowable tube repair limit, in accordance with RG 1.121, is obtained by incorporating into the resulting structural limit a growth allowance for continued operation and an allowance for eddy current measurement uncertainty.

The structural limits for the Kewaunee SGs are defined in a Westinghouse topical report, WCAP-15325, assuming a uniform thinning mode of degradation in both the axial and circumferential directions. The assumption of uniform thinning is generally regarded to result in a conservative structural limit for all flaw types occurring in the field. A revised analysis was performed to document applicable tube structural limits for the uprated conditions. Although the primary-to-secondary pressure gradients are increased for the uprated conditions, analysis showed the changes were not significant enough to result in an appreciable change to the structural limits. Therefore, the licensee concluded the existing plugging limit contained in the TSs is adequate. The NRC staff finds the licensee's evaluation to be acceptable because it follows the guidance in RG 1.121.

3.6.2.7 Tube Degradation

The potential effects of the 1.4 percent power uprate on SG tube degradation (e.g., axial and/or circumferential stress corrosion cracking, intergranular attack, etc.) were evaluated. The licensee concluded that the 1.4 percent power uprate is not expected to have a significant impact on tube degradation. Degradation resistance is based on the use of the Alloy 690 thermally treated (TT) SG tubing in the Kewaunee SGs. The Alloy 690 TT is expected to be an

improvement over Alloy 600 TT SG tubing, which has been shown to be much more resistant to degradation than the original Kewaunee Alloy 600 mill annealed SG tubing. Based on similar design SGs operating experience, accumulated EFPY of operation, and operating temperature, the licensee's analysis projects very low percentages of tubes plugged at the end of the current license under uprated power operation. Therefore, none of the potential degradation mechanisms are significantly affected by the power uprate conditions.

Based on the above rationale, the NRC staff finds the licensee's evaluation to be acceptable for the 1.4 percent MUR power uprate.

3.6.2.8 Reactor Coolant Loop Primary Piping

The licensee evaluated the RCL primary piping in Section 5.5.1 of its January 13, 2003, submittal. The evaluation was accomplished by reviewing the existing design-basis analysis against the uprated power conditions. The licensee's evaluation indicated that the parameters associated with the power uprate have no adverse effects on the analysis of the RCL piping, including impacts to the primary equipment nozzles. RCL piping was previously evaluated for the Replacement Steam Generator (RSG) program. The licensee noted that the maximum and minimum temperatures for the uprate are bounded by those used in the evaluation of the RSG project and, therefore, the results of the analysis performed for the RSG bound the proposed 1.4 percent MUR power uprate. Table 5.5.1-1 of the licensee's January 13, 2003, submittal, includes a comparison of design parameters used for the RSG project.

The NRC staff reviewed the information provided by the licensee. On the basis of its review of the licensee's submittal, the NRC staff concurs with the licensee's conclusion that the proposed 1.4 percent MUR power uprate will have no adverse effects on the RCL piping. Further, the results of the analysis performed for the RSG bound the proposed 1.4 percent MUR power uprate and thus, the temperatures used in the current design-basis envelopes the temperatures for the proposed 1.4 percent MUR power uprate. The NRC staff concludes that the proposed 1.4 percent MUR power uprate is acceptable for the KNPP.

3.6.3 Summary

The NRC staff has reviewed the licensee's evaluation of the impact of the proposed MUR power uprate on reactor vessel integrity, SG tube integrity, and FAC programs. The technical areas reviewed by the NRC staff are those discussed in Section 3.6.1 of this SE. 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 applicable requirements following implementation of the proposed MUR power uprate. Therefore, the NRC staff finds the proposed 1.4 percent MUR power uprate acceptable with respect to materials and chemical engineering.

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 January 13, 2003, application. 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 analysis 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 KNPP 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 (EOPs) that need to be changed as a result of the 1.4 percent MUR power uprate. Abnormal Operating Procedures (AOPs) will be modified to contain, or refer to an additional procedure containing the administrative restrictions for the plant operating power level based on the availability of the Crossflow UFMD (See Attachment 12, "List of Regulatory Commitments," of the January 13, 2003, application). Based on the above, the NRC staff finds that procedures will be changed or updated as necessary 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 new Crossflow UFMD will interface with the plant process computer system (PPCS). The PPCS will be used to monitor and display parameters associated with the Crossflow UFMD inputs. The PPCS will also provide input to visual and audible alarms on the control panel in the control room to alert the operator of problems or out of normal conditions associated with the Crossflow UFMD. Modifications associated with the MUR power uprate will be completed prior to implementation; this includes the installation of the Crossflow UFMDs and implementation of the PPCS and control room alarm functions and the licensee will provide appropriate training to the necessary plant staff for changes associated with the installation of the Crossflow UFMD and the implementation of the new rated power (See Attachment 12, "List of Regulatory Commitments," of the January 13, 2003, application). This will be finalized prior to implementing the proposed MUR power uprate. The NRC staff finds this acceptable.

3.7.2.4 Control Room Plant Reference Simulator

The KNPP Simulator Certification was submitted in a letter from C. A. Schrock, Wisconsin Public Service Corporation, to NRC Document Control Desk, dated January 27, 1992. The KNPP simulator will be modified to provide the same information and annunciation that will be provided in the control room. The modifications to the control room simulator will be done in accordance with the licensee's site design change procedures. Modifications associated with the MUR power uprate will be completed prior to implementation (See Attachment 12, "List of Regulatory Commitments," of the January 13, 2003, application). This will be finalized prior to implementing the proposed MUR power uprate. The NRC staff finds this acceptable.

3.7.2.5 Operator Training Program

Overview training regarding the modifications for the MUR power uprate will be provided to the operators. Specific training will be performed associated with the plant procedure changes as determined by the KNPP operations department in accordance with the appropriate plant processes. The licensee will provide appropriate training to the necessary plant staff for changes associated with the installation of the Crossflow UFMD and the implementation of the new rated power (See Attachment 12, "List of Regulatory Commitments," of the January 13, 2003, application). This will be finalized prior to implementing the proposed MUR power uprate. 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 the 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 Plant Systems

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) ESF HVAC 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

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 MSLB was performed by the licensee at its current power of 1650 MWt with two percent uncertainty or 1683 MWt. The licensee's current LOCA containment integrity analysis is based on 102 percent of the current licensed power of 1650 MWt or 1683 MWt. Both of these analyses bound operation at the MUR uprated power of 1673 MWt. The NRC staff finds this acceptable.

The licensee evaluated the containment ventilation system for a 7.4 percent power uprate or 1772 MWt. For normal operation with a core power of 1772 MWt, the licensee analyzed that the heat load to the containment air cooling system would increase approximately 2 percent. This would correlate to a 1.2 ^{o}F increase in containment temperature, which is based on the increased temperatures of the RCS, main steam, and feedwater resulting from the power uprate. The highest summer/fall temperature is estimated by the licensee to be 112 ^{o}F at uprate power conditions; this remains below the 120 ^{o}F for containment temperature contained in the licensee's equipment qualification plan. 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 1650 MWt or 1683 MWt. This bounds the 1.4 percent power uprate of 1673 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 cooldown analysis concluded that a single train of RHR and component cooling water (CCW) at uprated power can reduce the RCS temperature from 350 degrees ^{o}F to 200 ^{o}F in 40.2 hours assuming the RHR system is placed in-service 29 hours after reactor shutdown. The total cooldown time is 69. 2 hours (40.2 hours + 29 hours). The total cooldown time of 69.2 hours is less than the Appendix R cooldown limit of 72 hours. The NRC staff finds the safe shutdown fire analyses and required systems acceptable for the 1.4 percent MUR power uprate.

By letter dated June 30, 2003, the licensee informed the NRC staff that it had identified a discrepancy between the analysis assumptions for the 10 CFR Part 50 Appendix R safe shutdown analysis and the procedurally directed plant lineup. The licensee indicated that it will revise plant procedures to make them consistent with the analysis described in the application for the power uprate. The revisions would require two new local manual operator actions to achieve the RCS cooldown from 350 °F to 200 °F. The first action is to locally manually isolate the component cooling water heat exchanger in the train assumed inoperable. The second action is to locally manually throttle the component cooling water loads to achieve the flow rate assumed and to obtain the required cooldown. The analysis shows that completion of these steps would not be required until more than a day following the event, when the fire has been extinguished, and when additional emergency response personnel are available. The staff has reviewed the new information presented in the licensee's June 30, 2003, letter. Based on its review, the NRC staff has determined that (1) sufficient time is available for operators to complete the new local manual actions consistent with the analysis assumptions, and (2) the

licensee's commitments to complete revisions to plant procedures and provide the necessary training to plant personnel prior to implementing the power uprate are appropriate. As described in Section 5.0 of this SE, the NRC staff has conditioned the implementation of the proposed MUR power uprate on completion of the regulatory commitments made by the licensee including the commitment related to completion of procedure changes and training.

3.8.2.3 Spent Fuel Pool Cooling Analyses and Systems

The licensee's current spent fuel pool cooling analysis is performed at 1650 MWt with a 2 percent uncertainty added or 1683 MWt. This bounds the 1.4 percent power uprate of 1673 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, contained in the KNPP Environmental Qualification Plan, Revision 18, dated December 4, 2002, and the letter to M.L. Marchi (Wisconsin Public Service Corporation) from the NRC, "Review of Individual Plant Examination for Internal Events - Kewaunee Nuclear Power Plant (TAC No. M74424," dated January 15, 1997, 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 at 7.4 percent power uprate for the NSSS interface systems. The licensee evaluated the following BOP fluid systems for NSSS/BOP interface: main steam (MS), steam dump system, auxiliary feedwater (AFW) system, steam generator blowdown system (SGBD), condensate (CD) and feedwater (FW) system.

The licensee performed an analysis of the main steam system at the 7.4 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 steam dump valves (ASDVs) concluded that the original design-basis in terms of cooldown capability can still be achieved over the full range of NSSS design parameters. The cooldown design basis, with respect to sizing ASDVs, is bounding in regard to the capacity required for a SG tube rupture. The licensee determined that the design of the main steam isolation valves, MS non-return check 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 steam dump system at the 7.4 percent power uprate conditions. The licensee's analysis of the steam dump system capacity for the range of NSSS design parameters for the 7.4 percent power uprate conditions exceeds the minimum recommended capacity of 40 percent of rated-steam flow for load reductions up to 50 percent of electrical load and remains acceptable for uprated conditions. The NRC staff finds the steam dump system acceptable for the 1.4 percent MUR power uprate.

The licensee performed an analysis of the AFW system at the 7.4 percent power uprate conditions. The licensee's analysis confirmed that the current AFW system performance is acceptable for the power uprate conditions. The licensee's analysis confirmed that the existing condensate storage tank TS required minimum useable inventory is still 39,000 gallons during

power operation. The NRC staff finds the AFW system acceptable for the 1.4 percent MUR power uprate.

The licensee performed an analysis of the SGBD system at the 7.4 percent power uprate conditions. The licensee performed an evaluation of the SGBD rate required to control the chemistry and buildup of solids in the SGs. The SGBD flow rate is tied to allowable condenser inleakge, total dissolved solids in the KNPP circulating water system, and allowable primary-to-secondary leakage. The licensee determined that these variables are not impacted by the power uprate; therefore, the SGBD required to control secondary chemistry and SG solids will not be 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 7.4 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 flow control valves and flow control bypass valves stroke time requirement of 20 seconds is not impacted by the power uprate. The NRC staff finds the CD and FW systems acceptable for the 1.4 percent MUR power uprate.

3.8.2.6 Radioactive Waste Systems

The licensee evaluated the liquid and gaseous radioactive waste systems for a 7.4 percent power uprate, and the licensee determined that there was no significant impact on the expected annual radwaste effluent releases or doses. The licensee stated that the liquid and gaseous radioactive waste effluent treatment system will remain capable of maintaining normal operation offsite doses within the requirements of 10 CFR Part 50, Appendix I; therefore, the NRC staff finds the liquid and gaseous radioactive waste systems acceptable for the 1.4 percent MUR power uprate.

3.8.2.7 ESF Heating, Ventilation, and Air Conditioning Systems

ESF ventilation systems at KNPP include the control room post accident recirculation system, the auxiliary building special ventilation system (Zone SV), and the shield building ventilation (SBV) system. The licensee performed an evaluation of the HVAC systems at the core power of 1772 MWt . The licensee's evaluation concluded that the current ventilation systems at the KNPP would be able to maintain operating temperature at or below the maximum normal operating temperatures at the 7.4 percent power uprate conditions. The NRC performed an evaluation of the control room post accident recirculation system, the auxiliary building special ventilation system (Zone SV), and the SBV system as part of Amendment No. 166 for AST. KNPP AST Amendment No. 166 was approved by the NRC by letter dated March 17, 2003 (ADAMS Accession No. ML030210062). 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 7.4 percent power uprate conditions or 1772 MWt: RHR system, safety injection (SI) system, internal containment spray (ICS) system, CCW system, and service water (SW) system.

The licensee performed a cooldown analysis of the RHR system to assess the impact of the increased heat load on normal cooldown time. The licensee analyzed the cooldown time assuming various design conditions and determined that the RHR system is adequately sized for normal cooldown heat loads associated with the power uprate. The NRC staff finds the RHR cooling water system acceptable for the 1.4 percent MUR power uprate.

The existing SI and ICS systems have been evaluated by the licensee for an uprated power of 1772 MWt. Required volume, duration, and heat rejection capability of SI and ICS flows are based on analytical and empirical models that simulate reactor and containment conditions following a postulated RCS or MS pipe break by the licensee. The licensee stated that the analysis provided acceptable results. The NRC staff finds the SI and ICS cooling water systems 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.

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 the 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-43

The licensee proposes to revise paragraph 2.C.(1) of the operating license, DPR-43, to authorize operation at reactor core power levels not in excess of 1673 MWt (100-percent power).

Based on the evaluation provided in Section 3.0 above, the NRC staff finds the proposed change acceptable.

4.2 Change to TS vi, TS 3.1-6, Figure TS 3.1-1, Figure TS 3.1-2, TS B3.1-6 and TS B3.1-7

The licensee proposes to revise the note on the following pages regarding the KNPP P-T Limitation Curves: TS vi, TS 3.1-6, Figure TS 3.1-1, Figure TS 3.1-2, TS B3.1-6 and TS B3.1-7. The note will be revised to read, "[1]The curves are limited to 31.1 EFPY due to changes in vessel fluence associated with operation at uprated power."

The value for EFPY will change from the current value of 28 to 31.1 EFPY. The basis is two fold. First, the current 28 EFPY limitation is no longer applicable based on the reestablishment of the EOL 1/4T and 3/4T reference temperature using the Master Curve-based approach. As stated in the exemption letter from the NRC to M. Reddemann, “Kewaunee Nuclear Power Plant - Request for Exemption from the Requirements of 10 CFR Part 50, Appendix G and H, and 10 CFR 50.61 (TAC No. MA8585),” dated February 21, 2001, it is justified that the current KNPP P-T limitation curves are applicable through 33 EFPY. The second change lowers the EFPY to 31.1. This is based on changes in vessel fluence associated with operation at an uprated core power condition of 1772 MWt.

Based on the evaluation provided in Section 3.0 above, the NRC staff finds the proposed changes acceptable. The NRC staff has no objection to the licensee's proposed changes to the TS Bases.

4.3 <u>Change to TS 1.0m</u>

The licensee proposes to revise the definition of “RATED POWER” in TS 1.0m to reflect the increase from 1650 MWt to 1673 MWt.

Based on the evaluation provided in Section 3.0 above, the NRC staff finds the proposed change acceptable.

4.4 <u>Change to TS 6.9.4</u>

The licensee proposes to revise TS 6.9.4, “Core Operating Limits Report (COLR),” as follows:

4.4.a Revise the text of proposed TS 6.9.4.B of the letter from M. E. Warner to NRC Document Control Desk, “License Amendment Request 185 to the Kewaunee Nuclear Power Plant Technical Specifications (Core Operating Limits Report Implementation),” dated July 26, 2002, to explain the use of the Crossflow system power measurement uncertainty in other topical reports listed in the COLR. As stated in Section 3.0, “Background,” in the licensee’s January 13, 2003, submittal, KNPP proposes continued use of the topical reports identified in proposed TS 6.9.4.B. These reports describe NRC-approved methods that support the KNPP safety analyses. In some of these topical reports, reference is made to the use of the 2 percent power uncertainty that is consistent with the original Appendix K rule. KNPP proposes these topical reports be approved for use consistent with the new Appendix K rule and the Janauary 13, 2003, amendment request (i.e., using 0.6 percent power measurement uncertainty with a 1.4 percent increase in reactor power instead of the 2 percent power measurement uncertainty). To describe this change in applying the power measurement uncertainty, the licensee proposed the following text to be inserted just prior to the listing of topical reports in proposed TS 6.9.4.B:

> “The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC. When an initial assumed power level of 102 percent of the original rated power is specified in a previously approved method, 100.6 percent of uprated

rated power may be used only when the main feedwater flow measurement (used as the input for reactor thermal output) is provided by the Crossflow ultrasonic flow measurement system (Crossflow system) as described in report (15) listed below. When main feedwater flow measurements from the Crossflow system are unavailable, a power measurement uncertainty consistent with the instrumentation used shall be applied.

"Future revisions of approved analytical methods listed in this Technical Specification that currently reference the original 10 CFR Part 50, Appendix K uncertainty of 102 percent of the original rated power should include the condition given above allowing use of 100.6 percent of uprated rated power in the safety analysis methodology when the Crossflow system is used for main feedwater flow measurement.

"The approved analytical methods are described in the following documents:"

4.4.b Add reference (15) to proposed TS 6.9.4.B for topical report, CENPD-397-P-A, "Improved Flow Measurement Accuracy Using Crossflow Ultrasonic Flow Measurement Technology," May 2000.

Based on the evaluation provided in Section 3.0 above, the NRC staff finds the proposed changes acceptable.

4.5 Change to TS Table of Contents

The licensee proposed to revise the Table of Contents page TS iv to reflect the administrative change for the page number for Section 6.9.b from 6.9-5 to 6.9-6.

The NRC staff reviewed the proposed changes and finds the proposed changes acceptable.

5.0 REGULATORY COMMITMENTS

To support the proposed KNPP 1.4 percent MUR power uprate, the licensee made the following commitments (as stated) to be completed prior to the MUR power uprate implementation:

KNPP will complete revisions to affected documents (i.e., procedures) and provide appropriate training to the necessary plant staff for changes associated with the installation of the Crossflow UFMD and the implementation of the new rated power.

The KNPP will ensure the plant-specific analysis has been completed and that plant specific uncertainties are equal to or less than those provided to Westinghouse for the calculation of the power measurement uncertainty.

KNPP will complete revisions to affected operations procedures and provide appropriate training to operations for the implementation of the new rated power and the administrative restrictions for inoperable Crossflow UFMDs.

The KNPP EQ Plan will be updated to include the new containment exclusion areas for the pressurizer, steam generator, and reactor coolant pump vaults.

A corrective action request has been initiated to investigate the RAT procedural limit. This will be completed prior to the MUR power uprate implementation.

Modifications associated with the MUR power uprate will be completed prior to implementation. This includes the installation of the Crossflow UFMDs and implementation of the PPCS and control room alarm functions.

Rescaling and setting changes of the protection system will be completed as necessary.

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.

6.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Wisconsin 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 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 5679). 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
AMSAC	ATWS mitigation system actuation circuitry
AOP	Abnormal Operating Procedures
ART	adjusted reference temperature
ASDV	atmospheric steam dump valve
ASME	American Society of Mechanical Engineers
AST	alternate source term
ASTM	American Society for Testing and Materials
ATC	American Transmission Company
ATWS	anticipated transient without scram
B&PV	Boiler and Pressure Vessel
BOP	balance-of-plant
CCW	component cooling water
CD	condensate
CENP	Combustion Engineering Nuclear Power, LLC
CFR	*Code of Federal Regulations*
COLR	core operating limits report
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
EDG	emergency diesel generator
EFPY	effective full power years
EOL	end of life

EOP	emergency operating procedure
EQ	environmental qualification
ESF	engineered safety feature
FAC	flow-accelerated corrosion
FW	feedwater
GDC	general design criteria
GL	generic letter
HELB	high energy line break
HVAC	heating ventilation and air conditioning
ICS	internal containment spray
KNPP	Kewaunee Nuclear Power Plant
LBB	leak-before-break
LEFM	leading edge flowmeter
LBLOCA	large break LOCA
LOCA	loss-of-coolant accident
LTOP	low temperature overpressurization
MAT	main auxiliary transformer
MOV	motor-operated valve
MS	main steam
MSLB	main steam line break
MSSV	main steam safety vaves
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
P-T	pressure-temperature
PPCS	plant process computer system
PPCS SCRN	PPCS screen
PTS	pressurized thermal shock
RAI	request for additional information

RAT	reserve auxiliary transformer
RCCA	rod cluster control assembly
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
RSG	replacement steam generator
RTD	resistance temperature detector
RTO	reactor thermal output
RTP	rated thermal power
SBO	station blackout
SBV	shield building ventilation
SCP	signal conditioning/processing unit
SCU	signal conditioning unit
SE	safety evaluation
SG	steam generator
SGBD	steam generator blowdown
SGTR	steam generator tube rupture
SI	safety injection
SRP	Standard Review Plan
SW	service water
TS	Technical Specification
TT	thermally treated
UFM	Ultrasonic Flow Measurement
UFMD	Ultrasonic Flow Measuring Device
UFSAR	Updated Final Safety Analysis Report
UPI	upper plenum injection
USE	upper shelf energy
UTM	Ultrasonic Temperature Measurement