ATTACHMENT 1

Letter from Thomas Coutu (NMC)

To

Document Control Desk (NRC)

Dated

January 13, 2003

License Amendment Request 193

Description of Change, Safety Evaluation, Significant Hazards Determination, and Statement of Environmental Considerations

1.0 Introduction

Nuclear Management Company, LLC (NMC), proposes to amend the Facility Operating License and Technical Specifications to increase licensed rated power (RP) level for the Kewaunee Nuclear Power Plant (KNPP). The KNPP is currently licensed to operate at a maximum RP of 1650 megawatts thermal (MWt). Approval is being requested to increase the licensed core RP by 1.4 percent to 1673 MWt. This power increase will be accomplished by using a more accurate main feedwater flow measurement system to calculate the reactor thermal output (RTO) of the unit. Increasing RP by reducing measurement uncertainty is called a measurement uncertainty recapture (MUR) power uprate. The NMC has evaluated the impact of a 1.4 percent uprate to 1673 MWt for the applicable systems, structures, components, and safety analyses at KNPP. The results of this evaluation and the new main feedwater flow measurement system are described in Attachment 2 of this letter, "Summary of Measurement Uncertainty Recapture Power Uprate Evaluation Following Guidance Provided in NRC Regulatory Issue Summary (RIS) 2002-03," (reference 7.4).

2.0 Description of License and Technical Specification Changes

The proposed license amendment will revise the KNPP Facility Operating License and the Technical Specifications (TS) to increase the licensed RP by 1.4 percent from 1650 MWt to 1673 MWt. The proposed changes are described in detail below and are also indicated on the marked up and clean copy Operating License and TS pages in attachments 10 and 11.

2.1 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.

2.2 Revise 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. The basis is two fold. First, the current 28 EFPY limitation is no longer applicable based on the reestablishment of the end of life (EOL) 1/4T and 3/4T reference temperature using the Master Curve-based approach. As stated in the exemption request (reference 7.5) 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. This is described further in attachment 3, section 5.1.

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

2.4 Revise TS 6.9.4, "Core Operating Limits Report (COLR)," as follows:

a) Revise the text of proposed TS 6.9.4.B (reference 7.2) 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," located below, 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 two 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 this amendment request (i.e., using 0.6 percent power measurement uncertainty with a 1.4 percent increase in RP instead of the two percent power measurement uncertainty). To describe this change in applying the power measurement uncertainty, the following text will 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 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:"

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.

3.0 Background

The 1.4 percent power uprate for KNPP is based on eliminating unnecessary analytical margin that is assumed in analyses to account for the measurement uncertainties associated with the heat balance. Kewaunee's current accident and transient analyses include a minimum two percent margin on RP to account for power measurement uncertainty. This power measurement uncertainty was originally required by Title 10 of the Code of Federal Regulations (CFR) Part 50 (10 CFR 50), Appendix K, "ECCS Evaluation Models." The rule required a two percent power margin between the licensed power level and the power level assumed for the emergency core cooling system (ECCS) evaluations. In June 2000, the NRC amended 10 CFR 50, Appendix K to provide licensees the option of maintaining the two percent power margin or applying a reduced margin. For the latter case, the new assumed power level had to account for measurement uncertainties in the power level measurement instrumentation. The revised Appendix K rule had an effective date of July 31, 2000. Uprates taking advantage of this rule change are referred to as MUR power uprates, calorimetric uprates, or mini-uprates.

Uncertainty in the main feedwater flow measurement is one of the most significant contributors to power measurement uncertainty. Based on this fact and on the above Appendix K rule change, the NMC proposes a reduced power measurement uncertainty of 0.6 percent and an increase in RP of 1.4 percent. To accomplish this reduction in uncertainty and increase in power, the KNPP 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 that assumed during the development of the original Appendix K requirements and that of the feedwater flow venturis currently used to calculate RTO. The Crossflow system will measure feedwater mass flow to within ±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 two percent power margin originally required by Appendix K, thereby, allowing an increase in the RP available for electrical generation by 1.4 percent.

In addition to the proposal to increase the RP to 1673 MWt, the NMC also proposes continued use of the topical reports identified in the proposed KNPP Core Operating Limits Report (COLR) (reference 7.2) submittal. The topical reports are located in proposed TS 6.9.4.B. The topical reports describe the NRC approved analytical methodologies used to determine the core operating limits for KNPP. This includes the small and large break loss of coolant accidents. In some of these topical reports, reference is made to the use of a two percent power measurement uncertainty being applied consistent with 10 CFR 50, Appendix K. The NMC requests that these topical reports be approved for use consistent with this license amendment (i.e., 0.6 percent power measurement uncertainty be assumed instead of two percent). The proposed change to TS 6.9.4.B was described in Section 2.0 of this attachment. Additionally, the reduction of the power measurement uncertainty does not constitute a significant change as defined in 10 CFR 50.46(a)(3)(i) regarding ECCS evaluation models.

3.1 Licensing Methodologies for Uprate

The analytical and licensing work supporting the KNPP MUR power uprate is consistent with the methodology established by Westinghouse in WCAP-10263, "A Review Plan for Uprating the Licensed Power of a PWR Power Plant," dated 1983. The methodology in WCAP-10263 establishes the general approach and criteria for uprate projects including the broad categories that must be addressed. These categories include the Nuclear Steam Supply System (NSSS) parameters, systems, components, design transients, accidents, and the interfaces between the NSSS and the Balance of Plant (BOP) systems. This methodology includes the use of well-defined analysis input assumptions and parameter values, the use of currently approved analytical techniques, and the use of currently applicable licensing criteria and standards. This methodology has been successfully used as the basis for power uprate projects on pressurized water reactors, including measurement uncertainty recapture uprates.

The proposed KNPP MUR power uprate is also consistent with topical report CENPD-397-P-A, "Improved Flow Measurement Accuracy Using Crossflow Ultrasonic Flow Measurement Technology." The NRC has approved this topical report for referencing in MUR power uprate submittals. Kewaunee is specifically applying this topical report, and the criteria listed in the NRC SER for the CENPD-397-P-A, for a requested 1.4 percent RP increase.

In addition to the above methodologies, KNPP has taken into account the specific guidance developed by the NRC for the content of MUR power uprate applications. This guidance was published on January 31, 2002, as NRC RIS 2002-03, "Guidance on the Content of Measurement Uncertainty Recapture Power Uprate Applications," (reference 7.4). Attachment 2 of this license amendment application provides an evaluation of the proposed MUR power uprate structured to be consistent with the NRC guidance. The NRC requests for additional information (RAI) for the Point Beach Nuclear Plant MUR power uprate were also reviewed and answers to the applicable RAIs have been incorporated into the text of Attachment 2.

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

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

1) Some of the analyses apply a two percent uncertainty to the licensed power level of 1650 MWt to account solely for the power measurement uncertainty. This results in an assumed core power level of 1683 MWt in the analyses. These analyses have not been reperformed for the 1.4 percent uprate since the sum of the requested core power level (1673 MWt) plus the decreased power measurement uncertainty (0.6 percent) results in an assumed core power level of 1683 MWt. Therefore, these analyses fall within the previously analyzed conditions.

2) Some of the analyses already employ an assumed core power level in excess of the requested 1673 MWt plus the new power measurement uncertainty of 0.6 percent. These analyses were performed at 1683 MWt core power or greater (with or without the two percent measurement uncertainty) during previous plant projects (e.g., steam generator replacement, fuel transition (reference 7.3), alternate source term (reference 7.1)). For these analyses, the available margin envelops the 1.4 percent power increase of the MUR uprate. Consequently, these analyses have not been reperformed and continue to retain sufficient analysis margin. Programs that have submitted analysis changes to the NRC but approval has not been received are also included in this category and include the KNPP submittals to the NRC for alternate source term (reference 7.1) and the fuel transition (reference 7.3).

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

3.3 Conclusion

NMC is requesting a 1.4 percent increase in RP for the KNPP from 1650 MWt to 1673 MWt. This power increase will be accomplished by using a more accurate main feedwater flow measurement system to calculate the RTO. This higher accuracy measurement will be achieved with the use of a Crossflow system. This license amendment request has taken into account industry and NRC accepted methodologies and guidelines for power uprates.

This License Amendment Request (LAR) is made pursuant to 10 CFR 50.90 to modify the Operating License and the TS requirements associated with rated thermal power and the use of the power measurement uncertainty in safety analyses.

4.0 Technical Assessment of the Change in Rated Power

NMC has evaluated the impact of the proposed power uprate on safety analyses, NSSS systems and components, and BOP systems. Attachment 2 summarizes the results of the comprehensive engineering review performed to evaluate the increase in the licensed core rated power. Results of this evaluation are provided in a format consistent with the regulatory guidance provided in NRC RIS 2002-03 (reference 7.4). Additionally, the KNPP MUR power uprate evaluation was consistent with the methodology established in WCAP-10263 described earlier. The results of the NMC's evaluation demonstrate that applicable acceptance criteria will continue to be met following the implementation of the proposed 1.4 percent MUR power uprate.

5.0 No Significant Hazards Determination

In accordance with the requirements of 10 CFR 50.90, Nuclear Management Company (NMC) (licensee) hereby requests an amendment to facility operating license DPR-43, for the Kewaunee Nuclear Power Plant. The NMC proposes to amend the facility operating license and technical specifications to increase licensed rated power (RP) power level for the Kewaunee Nuclear Power Plant (KNPP). The KNPP is currently licensed to operate at a maximum RP of 1650 megawatts thermal (MWt). Approval is being requested to increase the licensed core RP by 1.4 percent to 1673 MWt. This power increase will be accomplished by using a more accurate main feedwater flow measurement system to calculate the reactor thermal output (RTO) of the unit. Increasing RP by reducing power measurement uncertainty is called a measurement uncertainty recapture (MUR) power uprate. The NMC has evaluated the impact of a 1.4 percent uprate to 1673 MWt for the applicable systems, structures, components, and safety analyses at KNPP. The results of this evaluation and the new main feedwater flow measurement system are described in this letter using the NRC RIS 2002-03 guidance.

Nuclear Management Company has evaluated the proposed amendments in accordance with 10 CFR 50.91 against the standards in 10 CFR 50.92 and has determined that the operation of the Kewaunee Nuclear Power Plant in accordance with the proposed amendments presents no significant hazards. Our evaluation against each of the criteria in 10 CFR 50.92 follows.

1. **Operation of the Kewaunee Nuclear Plant in accordance with the proposed amendments does not result in a significant increase in the probability or consequences of any accident previously evaluated.**

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

   The reduction in power measurement uncertainty allows for some of the safety analyses to continue to be used without modification. This is because the safety analyses were performed or evaluated at either 102 percent of 1650 MWt or higher. Analyses at these power levels support a core power level of 1673 MWt with a measurement uncertainty of 0.6 percent. Radiological consequences of USAR Chapter 14 accidents were assessed previously using the alternate source term (AST) methodology (reference 7.1, TAC No. MB4596). These analyses were performed at 102 percent of 1650 MWt and continue to be bounding. The USAR Chapter 14 analyses and accident analyses submitted to the NRC with the fuel transition (reference 7.3, TAC No. MB5718) continue to demonstrate compliance with the relevant accident analyses acceptance criteria. Therefore, there is no significant increase in the consequences of any accident previously evaluated.

The primary loop components (reactor vessel, reactor internals, control rod drive mechanisms, loop piping and supports, reactor coolant pumps, steam generators, and pressurizer) were evaluated at an uprated core power level of 1772 MWt and continue to comply with their applicable structural limits. These analyses also demonstrate the components will continue to perform their intended design functions. Changing the applicability of the heatup and cooldown curves is based on uprated fluence values. This does not have a significant effect on the reactor vessel integrity. Thus, there is no significant increase in the probability of a structural failure of the primary loop components.

All of the NSSS systems will continue to perform their intended design functions during normal and accident conditions. The auxiliary systems and components continue to comply with the applicable structural limits and will continue to perform their intended functions. The NSSS/BOP interface systems were evaluated at 1772 MWt and will continue to perform their intended design functions. Plant electrical equipment was also evaluated and will continue to perform their intended functions. Therefore, there is no significant increase in the probability or consequences of an accident previously evaluated.

2. **Operation of the Kewaunee Nuclear Power Plant in accordance with the proposed amendments does not result in a new or different kind of accident from any accident previously evaluated.**

No new accident scenarios, failure mechanisms, or single failures are introduced as a result of the proposed change. All systems, structures, and components previously required for the mitigation of an event remain capable of fulfilling their intended design function at the uprated power level. The proposed change has no adverse effects on any safety-related systems or component and does not challenge the performance or integrity of any safety-related system. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. **Operation of the Kewaunee Nuclear Power Plant in accordance with the proposed amendments does not result in a significant reduction in a margin of safety.**

   Operation at the 1673 MWt core power does not involve a significant reduction in the margin of safety. The current accident analyses have been previously performed with a two percent power measurement uncertainty or at uprated core powers that exceed the MUR uprated core power. System and component analyses have been completed at a core power in excess of the MUR uprated core power. Analyses of the primary fission product barriers at uprated core powers have concluded that all relevant design basis criteria remain satisfied in regard to integrity and compliance with the regulatory acceptance criteria. As appropriate, all evaluations have been either reviewed and approved by the NRC, are in the process of being approved by the NRC, or are in compliance with applicable regulatory review guidance and standards. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Conclusion

Operation of the KNPP in accordance with the proposed amendment will not result in a significant increase in the probability or consequences of any accident previously analyzed; will not result in a new or different kind of accident from any accident previously analyzed; and does not result in a significant reduction in a margin of safety. Therefore, operation of KNPP in accordance with the proposed amendments does not involve a significant hazards consideration.

6.0 Environmental Considerations

The environmental review, pursuant to 10 CFR 51.22 (b), determined that no environmental assessment or environmental impact statement needs to be prepared for the proposed license amendment. In accordance with RIS 2002-03, the environmental considerations pertaining to this license amendment request are addressed in detail in Attachment 2, Section VII.5, "Environmental Review."

7.0 References

7.1 Letter NRC-02-024 from Mark E. Warner to Document Control Desk, "Revision to the Design Basis Radiological Analysis Accident Source Term," dated March 19, 2002 (TAC No. MB4596).

7.2 Letter NRC-02-064 from Mark E. Warner to Document Control Desk, "License Amendment Request 185 To The Kewaunee Nuclear Power Plant Technical Specifications, 'Core Operating Limits Report Implementation,'" July 26, 2002 (TAC No. MB5717).

7.3 Letter NRC-02-067 from Mark E. Warner to Document Control Desk, "License Amendment Request 187 to the Kewaunee Nuclear Power Plant Technical Specifications Changes for Use of Westinghouse VANTAGE+ Fuel," dated July 26, 2002 (TAC No. MB5718).

7.4 NRC Regulatory Issue Summary 2002-03: "Guidance on the Content of Measurement Uncertainty Recapture Power Uprate Applications," January 31, 2002.

7.5 Letter K-01-027 from NRC to Mark 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.

ATTACHMENT 2

To

Letter from Thomas Coutu (NMC)

To

Document Control Desk (NRC)

Dated

January 13, 2003

License Amendment Request 193

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

I. Feedwater flow measurement techniques and power measurement uncertainty.

1. A detailed description of the plant-specific implementation

The feedwater flow measurement system being installed at the Kewaunee Nuclear Power Plant (KNPP) is an Advanced Measurement and Analysis Group (AMAG) Crossflow Ultrasonic Flow Measurement (UFM) System. The design of the above Crossflow system is addressed in detail in topical report CENP-397-P-A, Revision 1, May 2000 (reference I.1). The Crossflow UFM system consists of four ultrasonic transducers mounted on a metal support frame. The frame attaches externally to the feedwater piping (clamp-on design). Each transducer set is a known distance apart and injects an ultrasonic signal perpendicular to the pipe axis. By measuring the time a unique pattern of eddies takes to pass between the two sets of ultrasonic transducers, the velocity of the fluid is determined. This technique for measuring the fluid velocity is the cross correlation technique and is described in detail in CENP-397-P-A.

In addition to the UFMs, two ultrasonic temperature measurement instruments (UTMs) are attached externally to the feedwater piping for measuring the feedwater temperature. The UTMs will be used in place of the current feedwater resistance temperature detectors (RTDs). When used together, the UFMs and the UTMs make up the Crossflow ultrasonic flow measurement device (UFMD) as described in this submittal text. The UTMs are not described in CENP-397-P-A and can be replaced with the feedwater RTDs if necessary (i.e., when the UTMs are out of service).

A. Identification (by document title, number, and date) of the approved topical report

The approved topical report referenced for the KNPP measurement uncertainty recapture (MUR) power uprate is CENPD-397-P-A, Revision 01, "Improved Flow Measurement Accuracy Using Crossflow Ultrasonic Flow Measurement Technology," Revision 1, May 2000. The non-proprietary version of this report is CENP-397-NP-A, Revision 01, May 2000.

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

Nuclear Regulatory Commission (NRC) approval of topical report CENP-397-P-A is documented in a letter with attached NRC safety evaluation, both dated March 20, 2000, with the subject of, "Acceptance for Referencing of CENPD-397-P, Revision-01-P, "Improved Flow Measurement Accuracy Using Crossflow Ultrasonic Flow Measurement Technology," (TAC No. MA6452)," (reference I.2).

C. Discussion of the plant-specific implementation of the guidelines in the topical report and the staff's SER approving the topical report

The Crossflow UFMD at KNPP consists of a non-intrusive clamp-on ultrasonic flow measurement (UFM) and ultrasonic temperature measuring (UTM) sensors installed on the main feedwater line to each steam generator and an associated Crossflow UFMD electronics control cabinet installed in the plant relay room. In addition to the Crossflow UFMD being installed on the feedwater line to each steam generator, a stand alone UFMD will be installed on the feedwater bypass line. The feedwater bypass line includes a flow straightener, venturi nozzle, and a diffuser. The stand alone installation is only used to calibrate the UFMDs on the individual steam generator lines and is not connected to the Crossflow UFMD electronics cabinet. This installation of a Crossflow UFMD on the feedwater bypass line meets the requirements of CENP-397-P-A (reference I.1, section 1.4.2) for calibrating permanent meters on-line that may not be installed in areas with fully developed flow conditions.

The Crossflow UFMDs on the main feedwater lines to the SGs will be used for continuous calorimetric power determination by data link from the electronics cabinet to the plant process computer system (PPCS). The electronics unit software interfaces with the UFM and the UTM sensors on the feedwater line of each SG to derive individual feedwater flow and temperature signals for each SG. The software compares the individual steam generator feedwater flow and temperature signals derived from the Crossflow UFMD cabinet with the corresponding feedwater venturi flow and feedwater temperature provided from the PPCS through the data link software to develop individual SG feedwater venturi flow and temperature correction factors. Additionally, the RTO computer program will be modified to receive the Crossflow UFMD generated individual venturi flow and temperature correction factors for use in the RTO calculation program. The PPCS program will provide audible and visual alarms to alert plant operators when either the UTMs or Crossflow UFMDs are out of service.

D. Dispositions of the SER criteria that the NRC staff stated should be addressed when implementing the feedwater flow measurement technique

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.

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 Crossflow UFMDs. Plant maintenance and calibration procedures will be revised to incorporate AMAG maintenance and calibration requirements for the Crossflow UFMDs prior to raising the core power to 1673 MWt. The incorporation of, and the continued adherence to, these requirements will assure that the Crossflow UFMDs are properly maintained and calibrated.

Contingency plans for operation of the plant with the Crossflow UFMDs out of service are described in Sections G and H of this attachment.

2. For plants that currently have the Crossflow UFM installed, provide an evaluation of the operational and maintenance history of the installed UFM.

This criterion is not applicable since a Crossflow UFMD 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.**

AMAG provided Crossflow UFMD uncertainties to Westinghouse for performance of the power measurement uncertainty calculations. The bounding feedwater mass flow and temperature uncertainty provided by AMAG was 0.5 percent flow and 1.1°F. AMAG is specifically calculating the feedwater mass flow uncertainties for KNPP. The plant specific uncertainty evaluation will provide verification that the assumed bounding feedwater mass flow and temperature measurement uncertainties used for the power measurement calculation remain bounding. The KNPP will ensure 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.

Westinghouse completed calculations of the power measurement uncertainty for KNPP using the UFM and UTM inputs from AMAG. Westinghouse calculated the power uncertainties using the Westinghouse Revised Thermal Design Procedure (RTDP) instrument uncertainty methodology. The RTDP instrument uncertainty methodology was previously used for the KNPP during the fuel transition safety analysis work. The original RTDP instrument uncertainties were documented in WCAP-15591, revision 0. The WCAP-15591, revision 0, used the venturi measured feedwater flow for the power measurement uncertainty. WCAP-15591, revision 0, was submitted to the NRC on July 26, 2002 (reference I.3). This submittal is currently under NRC review with an expected acceptance date of February 2003. Therefore, the RTDP will be a currently accepted plant setpoint methodology following the approval of the fuel transition license amendment request at KNPP.

The WCAP-15591 has been revised to support the MUR power uprate. WCAP-15591, revision 1, includes three power measurement uncertainty analyses. The first is the original analysis using the feedwater venturis. The second is using the new Crossflow UFMD (UFMs and UTMs). The third analysis is performed using the new Crossflow UFMs with the feedwater RTDs. The report also identifies that the uncertainties calculated are applicable for power levels up to 1772 MWt when the daily heat balance is based on the Crossflow system. The report identifies all power measurement parameters and their individual contribution to the power measurement uncertainty. Proprietary and non-proprietary versions of this report are included with this letter as Attachments 7 and 8 (WCAP-15591, Revision 1, and WCAP-15592, Revision 1, both titled, "Westinghouse Revised Thermal Design Procedure Instrument Uncertainty Methodology - Kewaunee Nuclear Plant (Power Uprate to 1757 MWt-NSSS Power with Feedwater Venturis, or 1780 MWt-NSSS Power with Ultrasonic Flow Measurements, and 54F Replacement Steam Generators)").

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 other 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.**

The Crossflow UFMD was calibrated at the Alden Research Laboratory (ARL) for Reynolds (Rd) numbers ranging from 0.8 million to 7 million. The ARL experimental data was used to establish a curve for the velocity profile correction factor (VPCF) as a function of Rd. This curve is documented in CENP-397-P-A. This curve was then extended to higher Rd numbers, typical of those in nuclear plant feedwater systems. Close agreement was found between the theoretical and experimental VPCF curves.

A stand alone UFMD will be installed on the feedwater bypass line since the KNPP installation on the B feedwater loop did not meet the criteria for fully developed flow. This installation is only used to calibrate the UFMD on the B feedwater loop by subtracting the A feedwater loop UFMD reading from the bypass loop UFMD reading. The bypass UFMD is not connected to the Crossflow system electronics cabinet. This installation of a UFMD on the feedwater bypass line and the installation on the A feedwater loop meets the requirements of CENP-397-P-A (reference I.1, section 1.4.2) for calibrating permanent meters on-line that may not be installed in areas with fully developed flow conditions.

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

Attachment 7 contains WCAP-15591, revision 1, which describes the RTDP methodology for power measurement uncertainty. This WCAP identifies all parameters used for the power measurement uncertainty using the feedwater venturis, the new Crossflow UFMDs, and the Crossflow UFM with the feedwater RTDs. The tables within the specific power measurement uncertainty sections of WCAP-15591 identify the parameters as well as their individual contributions to the uncertainty.

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

i. maintaining calibration

Calibration and maintenance of the Crossflow UFMDs will be performed using site procedures developed from the requirements of the Crossflow technical manuals and procedures. Incorporation of, and adherence to, these Crossflow system requirements will assure that the UFMDs are properly maintained and calibrated. These changes will be made prior to implementing the MUR power uprate.

The instrumentation used in all of the power measurement uncertainty calculations documented in WCAP-15591 are calibrated on specified frequencies through the use of appropriate plant instrument and control procedures. The appropriate plant processes (site work request or corrective action processes) will control any problems encountered during the implementation of the calibration procedures.

ii. controlling software and hardware configuration

Although the UFM's calorimetric input is not nuclear safety-related, the Crossflow software is under configuration control. Verification and validation (V&V) is performed by AMAG using procedures and processes approved by Westinghouse. Westinghouse provides procedures for deficiency reporting and notification. As stated in CENP-397-P-A, the V&V is in accordance with industry standards.

All hardware modifications for the UFMDs or PPCS will be performed in accordance with the site design change process. All software modifications to the UFMDs or the PPCS will be performed in accordance with the software work request process or the design change process, as appropriate.

iii. performing corrective actions

All equipment problems associated with the Crossflow UFMDs, the Crossflow system, or with instrumentation used to develop the power measurement uncertainty will be controlled under the site work request process or the corrective action program, as appropriate. The software error reporting process or the corrective action program will control all software errors.

iv. reporting deficiencies to the manufacturer

Equipment problems for all plant systems, including the new Crossflow UFMD equipment, fall under the site work request process or the corrective action process. Conditions adverse to quality are documented under the corrective action program. Software deficiencies are processed through the software error reporting process or the design change process, as appropriate. Because KNPP relies upon the manufacturer for software support, any deficiencies would require manufacturer notification.

v. receiving and addressing manufacturer deficiency reports

The KNPP Crossflow UFMD is included in AMAG's V&V program and Westinghouse's QA program. Procedures are maintained for user notification of important deficiencies. Any deficiency reports coming to KNPP for the Crossflow UFMD would be screened and addressed through the Operating Experience Assessment (OEA) process. Currently, KNPP is participating in the AMAG user group, which is another forum for receiving manufacturer deficiencies.

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

If the Crossflow UFMD becomes unavailable, plant operation at a core thermal power level of 1673 MWt may continue for a maximum of 24 hours after the last valid Crossflow UFMD correction factor was used in the heat balance (calorimetric) calculation for use in the daily nuclear power range surveillance. The basis of the 24 hour time period is the minimum frequency for the calibration of the nuclear power range channels (Table TS 4.1-1). The calibration frequency is cited as "daily." Completion time for the daily nuclear power range calibration is 24 hours. Calibration of the nuclear power range channels is performed by comparing the nuclear power range channels to the daily heat balance calculation. 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. Plant operations may remain at a reactor thermal output (RTO) of 1673 MWt while continuing to use the last valid Crossflow UFMD correction factor until the next required performance of the surveillance in Table TS 4.1-1.

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

The Crossflow system can calculate correction factors based on two different temperature instruments. The first is using the Crossflow UFMD (UFMs in combination with the UTMs). This yields a power measurement uncertainty of 0.6 percent. If the UFMs are available but the UTMs are out of service, the feedwater RTDs can be used as the temperature input. The power measurement uncertainty for the Crossflow UFMs operable with the RTDs is 0.8 percent. If the Crossflow UFMs are out of service, the power measurement uncertainty will fall back to the feedwater venturis uncertainty of two percent. Therefore, there are two conditions that require explanation for proposed actions: 1) an operable Crossflow system using UFMs and the RTDs instead of the UTMs and 2) an inoperable Crossflow UFMD. The following paragraphs describe the proposed actions for these conditions.

Crossflow UFMD failures will be detected and transmitted to the PPCS and will cause an audible alarm in the control room. Additionally, a control board annunciator will be lit for this alarm. The Crossflow UFMD (UFMs and UTMs) does not perform any safety function and is not used to directly control any plant systems. Therefore, Crossflow system inoperability has no immediate effect on plant operation.

If the UTMs or the Crossflow UFMDs become unavailable, the operators will receive a computer alarm that will generate a control board annunciator alarm. The operators will enter an operations procedure that will contain a step for Crossflow UTM or Crossflow UFMD failure. The procedure will require the UTM or the Crossflow UFMD to be returned to service prior to the next performance of Table TS 4.1-1 (nuclear power range channel surveillance). The basis of the time period is the Table TS 4.1-1 completion time (24 hours). The nuclear power range channel will have been adjusted with the last valid Crossflow UFMD-based heat balance calculation and will be acceptable until the next performance of the surveillance. Plant operations may remain at an RTO of 1673 MWt while continuing to use the last valid Crossflow UFMD correction factor in the heat balance calculation. However, to remain in compliance with the basis for the operation at an RP of 1673 MWt, Crossflow must be returned to service prior to the next performance of the surveillance in Table TS 4.1-1. If the UTMs or the Crossflow system is not returned to service within the above time, power will be reduced and maintained at the appropriate power levels until the UTMs or the Crossflow UFMDs are returned to service. These power levels are described in the table below.

Table I.1, Power Measurement Instrumentation and Corresponding Power Level

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

The basis for reducing power to 1670 MWt rather than the original RP of 1650 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 as opposed to the two percent power measurement uncertainty required by the original Appendix K rule. Applying 0.8 percent power measurement uncertainty allows for a 1.2 percent uprate from the current RP of 1650 MWt to 1670 MWt. This power level with the 0.8 percent power measurement uncertainty results in the same initial power level of 1683 MWt for the accident analyses and is therefore acceptable.

The basis for reducing power to 1650 MWt is the calorimetric uncertainty required of the Appendix K rule. KNPP will continue to operate within the safety analyses and the applicable power measurement uncertainty. When the Crossflow UFMD becomes unavailable, the Appendix K calorimetric uncertainty of two percent will be used and the plant will be operated at or below an RTO of 1650 MWt. Based on a two percent power measurement uncertainty, initial power assumed in certain accident analyses would remain at 1683 MWt.

Once the Crossflow system (using the UFMs with either the RTDs or the UTMs) is returned to service and a heat balance calculation performed with the updated Crossflow correction factor, the nuclear power range channel calibration from Table TS 4.1-1 could then be performed. Once the TS required surveillance is performed using the updated Crossflow correction factor, power can be escalated to the appropriate power level for the power measurement instrumentation available. For an operable Crossflow system using the UTMs for temperature measurement, the allowable power level would be 1673 MWt. For an operable Crossflow system using the RTDs for temperature measurement, the allowable power level would be 1670 MWt.

KNPP will be operated in accordance with the safety analyses and the applicable power measurement uncertainty. When the Crossflow UFMD is available, the Crossflow UFMD-based power measurement uncertainty of 0.6 percent will be used and the plant will be operated at or below a steady state RP of 1673 MWt. When the Crossflow UFMs are available but the UTMs unavailable, the heat balance will be calculated using the feedwater RTDs with a power measurement uncertainty of 0.8 percent. Based on either of the Crossflow power measurement uncertainties, initial power assumed in certain accident analyses would remain at 1683 MWt, which is 102 percent of the current RP.

KNPP operations procedures will be revised to reflect the above responses to the unavailability of the Crossflow system. Additionally, this information will be included in operator training prior to implementation of the MUR uprate license amendment.

2. References for Section I

I.1 CENPD-397-P-A, Revision 01, "Improved Flow Measurement Accuracy Using Crossflow Ultrasonic Flow Measurement Technology," Revision 1, May 2000.

I.2 NRC safety evaluation, "Acceptance for Referencing of CENPD-397-P, Revision-01-P, "Improved Flow Measurement Accuracy Using Crossflow Ultrasonic Flow Measurement Technology, (TAC No. MA6452)," dated March 20, 2000.

I.3 Letter NRC-02-067, "License Amendment Request 187 to the Kewaunee Nuclear Power Plant Technical Specifications Changes for Use of Westinghouse VANTAGE+ Fuel," dated July 26, 2002 (TAC No. MB5718).

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

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

A. Identify the transient or accident that is the subject of the analysis
B. Confirm and explicitly state that:
i. the requested uprate in power level continues to be bounded by the existing analyses of record for the plant
ii. the analyses of record either have been previously approved by the NRC or were conducted using methods or processes that were previously approved by the NRC
C. Confirm that bounding event determinations continue to be valid
D. Provide a reference to the NRC's previous approvals discussed in Item B. above

The analyses referenced in Table II-1 are the existing licensing basis analyses of record for the plant. None of these analyses are changing and all bounding event determinations continue to remain valid for the MUR power uprate. The last column of the matrix states the reference that contains the analysis as well as whether it was approved by an NRC SER, was performed under 10 CFR 50.59, or is part of a KNPP internal document.

Table II-1: Bounding Analyses of Record

USAR Section/Analysis		Analysis of Record Power Assumptions				
		Nominal Core Power	Uncer-tainty	Total Core Power	Bounds MUR	Reference /Method (50.59, etc.)
RIS	A	B, C	B, C	B, C	B, C	D
14.1.10	Loss of Normal FW(2)	1650	± 2%	1683	Yes	II.1/50.59
14.1.11	Anticipated Transient Without Scram (ATWS)	1650	± 2%	1683	Yes	II.1/50.59
14.2.2	Accidental Release – Recycle of Waste Liquid	N/A(1)	N/A(1)	N/A(1)	Yes	II.2/Technical Specification
14.2.4	Steam Generator Tube Rupture Thermal/Hydraulic	1650	± 2%	1683	Yes	II.1/50.59
14.2.5	Rupture of a Steam Pipe: Containment Response	1650	± 2%	1683	Yes	II.1/50.59
14.3.4	Containment Integrity Evaluation	1650	± 2%	1683	Yes	II.1/50.59
14.3.8	Charcoal Filter Ignition Hazard Due to Iodine Absorption	1650	± 2%	1683	Yes	II.3/internal calculation
14.3.9	Generation and Disposition of Hydrogen	1721.4	N/A	1721.4	Yes	II.4 and II.5/internal calculation
8.2.4	Station Blackout	1650	± 2%	1683	Yes	II.6 and II.7/SER
	Spent Fuel Pool Cooling	1650	± 2%	1683	Yes	II.8 and II.9/internal calculation
	Flooding Study	N/A(1)	N/A(1)	N/A(1)	Yes	II.10 and II.11
	Vital Access	1721	N/A	1721	Yes	II.12 and II.13/internal calc and SER
	EQ Radiation Dose – Accident	1721	N/A	1721	Yes	II.11, II.12, and II.13/internal EQ plan

(1) Not dependent on power level.
(2) Although the Loss of Normal Feedwater was submitted with the fuel transition (reference II.14) at a higher core power, a supplement to the fuel transition will change the core power assumptions.

2. Response to Point Beach Nuclear Plant RAI Regarding ATWS

During the August 8, 2002, NRC meeting for KNPP power uprates, it was requested that the KNPP MUR power uprate submittal address the request for additional information given to the Point Beach Nuclear Plant. The following addresses an RAI regarding the Anticipated Transient Without Scram (ATWS).

An ATWS is a postulated anticipated operational occurrence (such as loss of feedwater, loss of condenser vacuum, or loss of off-site power) that is accompanied by a failure of the Reactor Protection System (RPS) to shut down the reactor.

Analyses have shown that the consequences of an ATWS are acceptable as long as the turbine is tripped and auxiliary feedwater is initiated in a timely fashion. Acceptable consequences are defined as reactor coolant system (RCS) pressure remaining below 3200 psig and no fuel failure. The most severe ATWS transients are those that entail a loss of main feedwater.

For ATWS protection, KNPP implemented ATWS Mitigation System Actuation Circuitry (AMSAC), an ATWS mitigation system. The AMSAC is actuated on low steam generator water level and is armed at all power levels. The logic of AMSAC is to trip the turbine and start all three auxiliary feedwater pumps when low steam generator water level signals are present on three of four channels for a specified time period. However, manual initiation of auxiliary feedwater may be required at low power levels (< 15 percent).

In 1998, in response to an engineering evaluation of the AFW system, a plant design change added a Diverse Scram System (DSS). The DSS is initiated on a signal from the existing AMSAC system and de-energizes the rod drive motor generator (MG) set exciter field. Removing the rod drive MG set exciter field will interrupt power to the control rod grippers, allowing the control rods to free fall into the core, ending the ATWS event.

The DSS was installed to ensure the AFW pumps would continue to run throughout a loss of main feedwater ATWS. The DSS in conjunction with the AMSAC system will end the transient before the AFW flow to the steam generators increases to a point where AFW pump NPSH could be lost.

A favorable (consistent with ATWS limitations) core moderator temperature coefficient is ensured for KNPP by Technical Specifications. Technical Specification 3.1.f.4 requires the reactor to have a moderator temperature coefficient (MTC) no less negative then -8.0 pcm/degree F for 95 percent of the cycle time at hot full power (HFP) conditions. This MTC requirement ensures that the ATWS design limits are satisfied for 95 percent of the cycle's time and that the unfavorable exposure time is 5 percent or less of the cycle time. In reference II.15, it was proposed that this technical specification be moved to the core operating limits report (COLR).

The loss of main feedwater ATWS analysis was analyzed with the DSS using the same methodology as the analysis for a loss of main feedwater transient. The analysis was performed to determine the steam generator pressure profile and to ensure the DSS, in conjunction with AMSAC, provides the necessary protection against reactor coolant system (RCS) overpressurization and fuel damage.

The following assumptions are made:

1. The reactor power is initially at a nominal full power value (1650 MWt) plus 2.0 percent.

2. The pressurizer power-operated relief valve is available.

3. All three auxiliary feedwater pumps are operable (two motor driven and one turbine driven). The flow from the three auxiliary feedwater pumps is evenly split between the two steam generators. The auxiliary feedwater flow is assumed at 176 gpm per pump, which is the flow rate assumed for one auxiliary feedwater pump, as used in USAR section 14.1.10, accident analysis for Loss of Normal Feedwater.

4. Reactor Tave is at 578.7 degrees F at the start of the transient. Steam generator initial pressure is 808 psig, which is consistent with the reactor Tave.

5. The turbine trips on the AMSAC signal (steam generator level < 13 percent) with a 25 second time delay.

6. The auxiliary feedwater pumps start on the AMSAC signal (steam generator level < 13 percent) with a 25 second time delay.

7. The reactor trips on the AMSAC signal (steam generator level < 13 percent) with a 37 second time delay. NOTE the additional 12 seconds for the reactor trip is to account for the delay time in the control rod drive MG set exciter field control logic and the decay of the generator field

All of the results of the analysis of the loss of main feedwater ATWS event are within established design basis acceptance criteria. In addition, the analysis demonstrates that the auxiliary feedwater pumps can be relied upon to start and run throughout the transient.

3. References

II.1 10 CFR 50.59 Safety Evaluations for DCR 2858 MOD 1, "Replacement Lower Assemblies and Steam Dome Modifications (for SGR)," Rev. 1, November 11, 2000.

II.2 TS 6.16.b.1, "Radioactive Effluent Controls Program."

II.3 Calculation C10861, "Charcoal Ignition Hazard Due to Iodine," Rev. 1, June 28, 1999.

II.4 Calculation C10950, "Post Accident H2 Generation from Radiolysis of Reactor Coolant," Rev. 0, June 1, 1997.

II.5 Calculation C10948, "Post-Accident Hydrogen Generation in Containment," Rev. 0, June 1, 1997.

II.6 Letter to C. A. Schrock (WPS) from the NRC, "Kewaunee Nuclear Power Plant, Unit No. 1 - Station Blackout Rule (10 CFR 50.63) (TAC # M84521)," November 19, 1992.

II.7 Letter to Ken H. Evers (WPS) from the NRC, "Safety Evaluation of the Kewaunee Nuclear Power Plant Response to the Station Blackout Rule (TAC # 68558)," November 20, 1990.

II.8 Calculation C11168, "SFPCS Heat Removal Capacity," Rev. 0, November 2, 2000.

II.9 Holtec International Report No. HI-992245, "Bulk Temperature Analyses for the Kewaunee Spent Fuel Pools and Transfer Canal," Rev. 0, August 4, 1999.

II.10 Letter to M. L. Marchi (WPS) from the NRC, "Review of Individual Plant Examination for Internal Events – Kewaunee Nuclear Power Plant (TAC # M74424)," January 15, 1997.

II.11 KNPP Environmental Qualification Plan, Rev. 18, December 4, 2002.

II.12 Letter from Steven A. Varga (NRC) to C. W. Giesler (WPS), SE, "NUREG-0737, Item II.B.2.2 – Design Review of Plant Shielding – Access to Vital Areas," 01/12/83.

II.13 Fluor Power Services Report 23-7127-053, "KNPP Design Review of Post-Accident Plant Shielding and Equipment Radiation Qualification," February 13, 1981.

II.14 Letter NRC-02-067 from Mark E. Warner to Document Control Desk, "License Amendment Request 187 to the Kewaunee Nuclear Power Plant Technical Specifications Changes for Use of Westinghouse VANTAGE+ Fuel," dated July 26, 2002 (TAC No. MB5718).

II.15 Letter NRC-02-064 from Mark E. Warner to Document Control Desk, "License Amendment Request 185 To The Kewaunee Nuclear Power Plant Technical Specifications, 'Core Operating Limits Report Implementation,'" dated July 26, 2002 (TAC No. MB5717).

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

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

The analyses in the table below were reperformed at 1683 MWt or higher power level conditions and are currently under review by the NRC. These analyses bound the MUR power uprate. They have been fully described in the corresponding references and will not be redescribed in this amendment request.

Table III.1, Analyses not bounded by analysis of record but currently under review by the NRC.

FSAR Section/Analysis		Analysis of Record Power Assumptions				
		Nominal Core Power	Uncer-tainty	Total Core Power	Bounds MUR	Submittal Reference
RIS	A.	B, C	B, C	B, C	B, C	D
14.1.1	Uncontrolled RCCA Withdrawal from Sub-critical Condition	0	N/A	0	Yes	III.1
14.1.2	Uncontrolled RCCA Withdrawal at Power	1772	RTDP(3)	1772	Yes	III.1
14.1.3	RCCA Misalignment	1772	RTDP(3)	1772	Yes	III.1
14.1.4	Chemical and Volume Control System Malfunction	1772	± 2%	1807	Yes	III.1
14.1.5	Startup of an Inactive Reactor Coolant Loop	N/A	N/A	N/A	Yes	III.1 (1)
14.1.6	Excessive Heat Removal Due to Feedwater System Malfunctions	1772	RTDP(3)	1772	Yes	III.1
14.1.7	Excessive Load Increase Incident	1772	RTDP(3)	1772	Yes	III.1
14.1.8	Loss of Reactor Coolant Flow (Coastdown Events)	1772	RTDP(3)	1772	Yes	III.1
	Locked Rotor: Transient Analysis	1772	± 2%	1807	Yes	III.1
	Locked Rotor: Radiological	1650	± 2%	1683	Yes	III.2
14.1.9	Loss of Electrical Load Overpressure DNB	 1772 1772	 ± 2% RTDP(3)	 1807 1772	 Yes Yes	 III.1 III.1

Table III.1 (continued)

FSAR Section/Analysis		Analysis of Record Power Assumptions				
		Nominal Core Power	Uncer-tainty	Total Core Power	Bounds MUR	Submittal Reference
RIS	A.	B, C	B, C	B, C	B, C	D
14.1.12	Loss of AC Power to the Plant Auxiliaries	1772	± 2%	1807	Yes	III.1
14.2.1	Fuel Handling Accident	1650	± 2%	1683	Yes	III.2
14.2.3	Accidental Release-Waste Gas (Rupture of GDT and VCT)	1650	± 2%	1683	Yes	III.2
14.2.4	Steam Generator Tube Rupture Radiological	1650	± 2%	1683	Yes	III.2
14.2.5	Rupture of a Steam Pipe: Core response	0	N/A	0	Yes	III.1
	Rupture of a Steam Pipe: Radiological	1650	± 2%	1683	Yes	III.2
14.2.6	Rupture of Control Rod Mechanism Housing (RCCA Ejection) – Core response	1772	± 2%	1807	Yes	III.1
	Rupture of Control Rod Mechanism Housing: Radiological	1650	± 2%	1683	Yes	III.2
14.3.1	Loss of Reactor Coolant from Small Ruptured Pipes or From Cracks in Large Pipes which Actuates Emergency Core Cooling System	1772	± 0.6%	1782	Yes	III.1
14.3.2	Major Reactor Coolant System Pipe Ruptures (Loss-of-Coolant Accident)	1772	± 0.6%	1782	Yes	III.1
14.3.3	Core and Internal Integrity Analysis	1772	± 0.6%	1782	Yes	III.1
14.3.5	Off-Site Dose Consequences	1650	± 2%	1683	Yes	III.2
	Natural Circulation Cooldown(2)	1772	± 2%	1807	Yes	III.1

(1) Evaluated in Reference III.1. KNPP Technical Specifications prevent event when reactor power ≥ 2% rated power.

(2) Natural Circulation is addressed in the Loss of AC Power to the Station Auxiliaries accident analysis. There is adequate RCS flow for decay heat removal as well as adequate auxiliary feedwater. Natural circulation cooldown has also been proceduralized (reference III.7).

(3) Revised Thermal Design Procedure (RTDP) Methodology used to incorporate power measurement uncertainty (± 2%)

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

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

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

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

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

   There were no analyses under the NRC approved reload methodology that had to be evaluated for the MUR power uprate.

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

Three analyses had to be evaluated for the MUR power uprate. These analyses are not covered by the reload methodology and are described using the RIS questions of III.3.A through III.3.K. These analyses include annual radiological effluent releases, EQ normal containment radiation doses, and Appendix R safe shutdown analysis. Annual radiological effluents and EQ normal containment equipment doses are not considered accidents or transients. The table below lists the analyses and their power level assumptions. The descriptions immediately follow the table.

Table III.3.1, Analyses Evaluated and Not Covered by the Reload Methodology

Analysis	Analysis Power Assumptions			
	Nominal Core Power	Uncer-tainty	Total Core Power	Bounds MUR
A	B, C	B, C	B, C	B, C
Annual Radiological Effluents	1772	N/A	1772	Yes
EQ Normal Containment Radiation Dose	1772	N/A	1772	Yes
Safe shutdown/Appendix R Cooldown	1772	N/A	1772	Yes

Radiological Effluents

The MUR power uprate has no significant impact on the expected annual radwaste effluent releases or doses. It is, therefore, concluded that following uprate, the liquid and gaseous radwaste effluent treatment system will remain capable of maintaining normal operation offsite doses within the requirements of 10 CFR 50 Appendix I. The evaluation is described in detail below.

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

Normal operation annual radwaste effluent releases were analyzed for a 7.4 percent power uprate. Although not specifically an accident or transient, radwaste effluent releases are regulated. Power uprate will increase the activity level of radioactive isotopes in the primary and secondary coolant. As activity levels in the primary and secondary coolant are increased, the activity levels of radwaste inputs are proportionately increased. Therefore, power uprate could potentially affect radwaste effluent releases and doses. The evaluation is summarized below and further referred to in the evaluation of the radwaste systems (RIS, Section VI) and in the environmental review (RIS, Section VII) both located in this attachment.

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

Analysis inputs included the annual doses from radioactive gaseous effluent reports for the years 1996 through 1999 (references III.13 through III.16). The evaluation accounted for the fact that KNPP did not operate at its licensed power level due to secondary side limitations during the last five years. Additionally, KNPP replaced its steam generators in 2001. Therefore, the uprated system conditions used in this assessment reflect the replacement steam generators. It is also assumed that the layout of the radwaste systems remains the same as before the uprate. The following table list input assumptions for evaluation of effluents for a 7.4 percent power uprate.

Table III.3.2, Input Assumptions for Radiological Effluents

Parameter	Pre-RSG Value(s)	7.4% Uprate Value(s)
Core Power (MWt)	1620 (Cycle 21)(1) 1590 (Cycle 22)(1) 1600 (Cycle 23)(1) 1580 (Cycle 24)(1)	1772 MWt
Steam flow rate (total)	Cycle 21 SG A 3.20 mlbm/hr SG B 3.41 mlbm/hr Tot 6.61 mlbm/hr Cycle 22 SG A 3.23 mlbm/hr SG B 3.42 mlbm/hr Tot 6.65 mlbm/hr Cycle 23 SG A 3.21 mlbm/hr SG B 3.43 mlbm/hr Tot 6.64 mlbm/hr Cycle 24 SG A 3.17 mlbm/hr SG B 3.45 mlbm/hr Tot 6.62 mlbm/hr	7.76E+06 lbm/hr
Weight of water in reactor coolant system	265,600 lb	251,677 lb
Weight of water in steam generators/per generator	95,467 lb	92,493 lb (0% tube plug) 92,006 lb (22% tube plug)
Reactor coolant letdown flow (purification)	2.0E+04 lb/hr	2.0E+04 lb/hr
Reactor coolant letdown flow (yearly average for boron control)	64.6 lb/hr	60.7 lb/hr
Steam generator blowdown flow (total)	5.4E+04 lb/hr	5.4E+04 lb/hr
Flow through the purification system cation demineralizer	~20 min/day of full RCS purification flow	~20 min/day of full RCS purification flow

(1) KNPP did not operate at its licensed power level due to secondary side limitations during cycles 21 through 24. Additionally, KNPP replaced its steam generators in 2001. Therefore, the uprated system conditions used in this assessment reflect the replacement steam generators.

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

This analysis is not for an accident or transient. Therefore, limiting event determinations are not specified. However, the acceptance criteria must be met. The acceptance criteria for this analysis is 10 CFR 20 and 10 CFR 50, Appendix I.

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

The methodology outlined in NUREG-0017 was used to create scaling factors to estimate the change in primary and secondary coolant activity due to a 7.4 percent power uprate. The highest percentage change in activity level was applied to all effluents to bound the impact on offsite releases. The percentage change was then applied to the offsite doses reported in the annual effluent reports for 1996 to 2000 to determine whether the estimated offsite doses following a 7.4 percent uprate would remain below the regulatory limits and the guidelines of 10 CFR 20 and 10 CFR 50, Appendix I.

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

The methodology used was from NUREG-0017, "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Pressurized Water Reactors."

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

Stone & Webster, Incorporated, performed the analyses for a 7.4 percent uprate consistent with the methodologies of NUREG-0017. This methodology has been used successfully in previous power uprate applications.

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

Normal radwaste effluent releases and doses are not accidents or transients, therefore, there is no sequence of events. Discharges of effluents would remain controlled by the processes and procedures currently in place at the KNPP. These processes and procedures will not change as a result of the power uprate.

H. Describe and justify the chosen single-failure assumption

Normal radwaste effluent releases and doses are not considered accidents or transients, therefore, the single failure criterion does not apply.

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

Plots are not applicable.

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

The system of concern with processing the normal radwaste effluents is the radwaste system. The evaluation performed assumed no changes in the design or operation of any of the radwaste system components. The results remained within the acceptance criteria. Therefore, no changes are required to equipment to support this analysis.

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

The acceptance criteria for normal annual radwaste effluents is to show the liquid and gaseous radwaste systems' design remains capable of maintaining the normal operating releases and doses within the requirements of 10 CFR 20 and 10 CFR 50, Appendix I following the 7.4 percent uprate.

a. Expected Reactor Coolant Source Terms

The maximum expected increase in the reactor coolant source (associated with the chemical group with the largest percentage increase) is approximately 17.6 percent for noble gas activity. This increase is primarily a combination of the impact of the core uprate and the reduction in the RCS volume.

b. Liquid Effluents

As discussed above, there is an approximate 17.6 percent increase assumed for the liquid releases as input activities are based on the largest long term RCS activity increase for any chemical grouping and on waste volumes which are essentially independent of power level. Tritium releases in liquid effluents are assumed to increase approximately 17.6 percent.

c. Gaseous Effluents

For all noble gases there will be a maximum 17.6 percent increase in effluent releases due to the core uprate. Gaseous releases of Kr-85 in actuality will increase by approximately the percentage of the power increase. The impact of the uprate on iodine releases is approximated by the power level increase. The other components of the gaseous release are not impacted by power uprate using the NUREG-0017 methodology. Tritium releases in the gaseous effluents increase in proportion to the increased tritium production. For particulates, the NUREG-0017 methodology specifies the release rate per year per unit per building ventilation system. This is not dependent on power level. Thus, there is no change calculated for the uprate. However, a 17.6 percent increase was conservatively addressed.

d. Appendix I Doses

The maximum increase in doses for gaseous and liquid effluents is estimated to be 17.6 percent. A comparison with the 10 CFR 50, Appendix I doses at uprated conditions against the design objectives of 10 CFR 50, Appendix I shows the estimated doses are a small fraction of that allowed under Appendix I.

e. Solid Waste Effluents

Although solid waste is not addressed in 10 CFR 50, Appendix I, it was included here for completeness. Regulatory guidance for a "new" facility estimates the volume and activity of solid waste as being linearly related to the core power level. However, for an existing facility that is undergoing power uprate, the volume of solid waste would not be expected to increase proportionally. This is because the power uprate neither appreciably impacts installed equipment performance nor requires drastic changes in system operation. Only minor, if any, changes in waste generation volume are expected. It is expected that the activity levels for most of the solid waste would increase proportionately to the increase in long half-life coolant activity (i.e., 17.6 percent).

In conclusion, the overall volume increase of waste generation resulting from uprate is expected to be minor. There are no acceptance criteria for solid waste except that disposal of solid radioactive waste must be done with regard to several federal regulations. The KNPP radioactive waste process control program (PCP) (reference III.5) provides the guidance for analyzing, processing, and packing of radioactive wastes in order to produce a final waste form that is acceptable for transportation and burial at a licensed radioactive waste disposal site. This program does not change as a result of the uprate. Solid waste processing and disposal will continue to be controlled by the above program to maintain compliance with regulatory requirements for a 7.4 percent power uprate.

Conclusion for Normal Annual Operating Radwaste Effluents and Doses

A 7.4 percent power uprate has no significant impact on the expected annual radwaste effluent releases or doses. The impact of a 7.4 percent power uprate on the estimated annual radwaste effluent releases/doses is limited to less than 17.6 percent. Based on the estimated fractions of the 10 CFR 50, Appendix I design objectives for the 7.4 percent uprate condition, the liquid and gaseous radwaste system effluent treatment design remains capable of maintaining normal operation offsite releases and doses within the requirements of the KNPP licensing basis. The estimated effluents for the MUR uprate would be substantially less. The activity of solid waste could increase proportionately with the power uprate but the increase in waste generation volume is expected to be minor for a 7.4 percent uprate. Solid waste will continue to be controlled by the process control program (reference III.5) at KNPP to maintain compliance with regulatory requirements. The 7.4 percent evaluation bounds the scaling factors that would be developed for the smaller, 1.4 percent MUR power uprate. Therefore, there is also no significant impact for the 1.4 percent MUR power uprate.

Normal Equipment EQ doses

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

The normal Environmental Qualification (EQ) doses for containment were evaluated for a 7.4 percent uprate. The normal EQ doses for containment are not considered accidents or transients but rather are the radiological environmental conditions that must be considered as part of the qualification of EQ equipment.

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

The 40 year normal operation containment doses reported in the KNPP EQ Plan assumed maximum dose rates from plant survey data. It was assumed that the operation and layout of the plant radioactive system would remain the same following the power uprate. Based on this, the plant survey data could be scaled by the percentage of the power uprate. Additional survey data taken in the containment vault areas determined that an exclusion area needs to be added to the EQ Plan to address the radiological levels in the pressurizer, steam generator, and reactor coolant pump vaults. Therefore, the inputs included the original survey data, the scaling factor, the new exclusion areas for the containment vaults, and the additional survey data.

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

This evaluation is not for an accident or transient. Therefore, limiting event determinations are not specified. Although there are no acceptance criteria, the normal operation radiological environmental parameters for containment must be factored into the EQ program.

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

The methodology used to evaluate the 7.4 percent power uprate relies on scaling factors. The use of scaling factors is a general industry practice that has been accepted by the NRC during other power uprate amendments.

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

The methodology used is a general industry practice that has been accepted by the NRC during other power uprate amendments.

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

There are no limitations or restrictions associated with this methodology since it is simply scaling up the normal radiation doses directly proportional to the percentage of the power uprate. These values will then be used as acceptance criteria in the EQ program.

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

Normal operating radiological environmental parameters are not considered accidents or accident conditions. Therefore, there is no sequence of events.

H. Describe and justify the chosen single-failure assumption

Normal operating radiological environmental parameters are not considered accidents or accident conditions. Therefore, the single-failure assumption does not apply.

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

No plots were developed for this evaluation of normal operation component of the total integrated doses in containment.

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

No changes to equipment are required to implement the evaluation performed for the normal operation component of the total integrated dose used for radiological environmental qualification. However, the EQ Plan will require documentation changes based on the evaluation of the radiological parameters at 1772 MWt. The changes will incorporate the exclusion area for the pressurizer, steam generator, and reactor coolant pump vaults as described above.

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

There are no acceptance criteria for the normal operation component of the total integrated radiation dose used for radiological environmental qualification. Rather, the results become the acceptance criteria for the EQ Plan.

The normal operation radiation component for EQ was evaluated for a 7.4 percent power uprate. This uprate will bound the MUR power uprate of 1.4 percent since the normal dose is expected to increase by the percentage of the uprate.

<u>EQ Zones</u>

For a 7.4 percent uprate, the estimated uprated normal radiation dose rates in all the EQ zones, with the exception of containment, is enveloped by the dose rate assumed in the EQ Plan. Therefore, for those EQ zones other than containment, the current 40 year normal operation radiation doses in the EQ Plan remain valid for a 7.4 percent power uprate.

Currently, the "containment" zone in the KNPP EQ Plan is the entire containment area including the pressurizer, steam generator, and reactor coolant pump vaults. The current established EQ Plan 40 year normal operation radiation dose was reviewed against a current dose rate survey. The comparison indicated that at the current power level, the survey dose rates in the pressurizer, steam generator, and reactor coolant pump vaults are greater than the currently established value in the EQ plan. Therefore, KNPP intends to update the EQ Plan to include an exclusion area within the "containment" zone to address the above listed vaults. The creation of exclusion areas for the vaults would better represent the in-containment normal operation radiological environments. The new 40 year integrated normal radiation dose in the vaults will be based on a dose rate of 50 R/hr that will encompass the current survey data of 35 R/hr and will also accommodate the potential 7.4 percent increase in the normal radiation levels due to core power uprate. The current EQ Plan 40 year normal radiation dose for all other areas of containment remains acceptable.

Conclusion

The KNPP has evaluated normal operation radiation dose in containment for a 7.4 percent uprate. It was found that the "containment" zone radiation levels in the current EQ Plan need to be modified for the 7.4 percent power uprate. This includes the establishment of an exclusion area to address the radiological levels in the pressurizer, steam generator and reactor coolant pump vaults. The 40 year normal operation radiation dose for the new exclusion area will be 50 R/hr. The 50 R/hr will encompass current survey data and the 7.4 percent uprate. This evaluation, and the associated plant document changes, bound the smaller MUR power uprate. The EQ Plan will be updated to include the new containment exclusion areas for the vaults prior to the implementation of the MUR power uprate.

Safe Shutdown/Appendix R Cooldown

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

The Appendix R safe shutdown cooldown was evaluated for the 7.4 percent uprate. In accordance with 10CFR50, Appendix R and the KNPP Appendix R Fire Protection Report, the plant must be capable of achieving cold shutdown in less than 72 hours after reactor shutdown assuming loss-of-offsite power, one train of residual heat removal (RHR), one train of component cooling water (CCW), and a maximum service water (SW) temperature (80°F) (Reference III.6 and III.12).

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

The plant shutdown is a result of a fire coupled with loss of AC power. This scenario implies a natural circulation cooldown since the reactor coolant pumps (RCPs) have tripped. The fire and loss of AC power leaves one train of equipment to support plant cooldown. The cooldown to cold shutdown is performed assuming a SW temperature of 80°F, which is the maximum value used in the safety analysis.

The analysis addresses the impact of the power increase on the current operating conditions and cooldown times.

Important inputs include:

(1) Decay heat taken from ANS 5.1-1979

(2) Reactor power is 1772 MWt

(3) Maximum reactor coolant system (RCS) temperature cooldown rate is 25°F/hour for Westinghouse plants with T_{hot} reactor vessel head during natural circulation cooldown to preclude flashing per generic letter (GL) 81-21 (reference III.11)

(4) Initiation of cooldown is 29 hours, that is the time following reactor trip when a single train of RHR/CCW can accept the sensible and decay heat load.

(5) Service water supply temperature is 80°F which is the maximum allowable service water temperature

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

The Appendix R safe shutdown evaluation is consistent with the KNPP fire program (Reference III.6) and plant emergency procedures (References III.8 and III.9). The analysis was performed because the 7.4 percent increase in power will increase the decay heat produced and the time to reach cold shutdown. The 72 hour limitation is met in the analysis.

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

The analysis was performed using two Westinghouse computer codes:
(1) TSHXB which estimates the performance of heat exchangers and
(2) RHRCOOL used to model plant cooldown via the RHR, CCW, and SW systems. TSHXB was used to determine auxiliary RCS heat loads removed directly by the CCWS (i.e., via the seal water heat exchanger, letdown heat exchanger, sample system heat exchanger, RCP thermal barrier heat exchangers, RHR pump, and waste gas compressor). These loads are input into RHRCOOL and are subtracted from the RCS heat load, which is removed by the RHR heat exchanger. RHRCOOL models heat transfer from the RCS through the RHR heat exchanger to the CCW and from the CCW heat exchanger to the service water.

RHRCOOL throttles the RCS flow through the RHR heat exchanger for two reasons: (1) to maintain the cooldown rate at 25°F/hr during natural circulation cooldown to preclude flashing (References III.7-III.9) and (2) to conservatively limit the discharge CCW temperature to 119°F. KNPP operating procedures limit CCW heat exchanger outlet temperature to 130°F (Reference III.10) at the current licensed power level when the RCS temperature is below 400°F and the RCPs are stopped.

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

The staff normally does not review the specifics of the methodologies used in evaluating system performance. The results of the cooldown analysis which models the cooldown of the RCS by means of the RHR system has been submitted to the NRC for numerous power uprates. System evaluations for RHR and CCW are located in Attachment 3, section 4.1.

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

The methodologies (computer codes) have not specifically received NRC approval. Therefore, there are no limitations or restrictions on this methodology.

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

A fire occurs with simultaneous loss of AC power, which necessitates bringing the plant to cold shutdown within 72 hours. Conditions inside the plant (e.g., the occurrence of the fire) are such that only one train of Appendix R equipment is available to control the shutdown. Shutdown proceeds as described in the Fire Protection Plan.

H. Describe and justify the chosen single-failure assumption

Safe shutdown systems installed to insure positive shutdown capability need not be designed to meet single failure criteria as stated in the Appendix R Design Description (reference III.6).

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

No plots are included for this analysis.

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

Flows through heat exchangers used in removing heat from the RCS were actual flows recorded at the plant. The major effect of the power uprate was the increase in decay heat produced in the core which, in turn, increases the time following reactor trip when a single train of RHR/CCW can begin removing the sensible and decay heat, and the time required to reach cold shutdown.

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

The cooldown analysis concluded that a single train of RHR and CCW at uprated power can reduce the RCS temperature from 350°F to 200°F in 40.2 hours assuming the RHR is placed in service 29 hours after reactor shutdown. Therefore, the total cooldown time is 69.2 hours, which is less than the Appendix R cooldown limit of 72 hours. The RHR, together with the CCW and SW systems, is adequately sized to meet Appendix R regulatory requirements for a 7.4 percent power uprate.

4. References for Section III

III.1 Letter NRC-02-067, from Mark E. Warner to Document Control Desk, "License Amendment Request 187 to the Kewaunee Nuclear Power Plant Technical Specifications Changes for Use of Westinghouse VANTAGE+ Fuel," July 26, 2002 (TAC # MB5718).

III.2 Letter NRC-02-024 from Mark E. Warner to Document Control Desk, "Revision to the Design Basis Radiological Analysis Accident Source Term," March 19, 2002 (TAC # MB4596).

III.3 10 CFR 50.59 Safety Evaluations for DCR 2858, Modification 1, "Replacement Lower Assemblies and Steam Dome Modifications (for SGR)," Rev. 1, November 20, 2000.

III.4 NUREG-0017, "Calculation of Releases of Radioactive materials in Gaseous and Liquid Effluents from Pressurized Water Reactors," Revision 1, April 1985.

III.5 NAD-01.16, "Solid Radioactive Waste Process Control Program (PCP)," Revision E, March 26, 2002.

III.6 KNPP Appendix R Design Description, Safe Shutdown Systems, Rev. 3.

III.7 IPEOP ES-0.2, "Natural Circulation Cooldown," Rev. 0.

III.8 OP E-0-06, "Fire In Alternate Fire Zone," Rev. P.

III.9 OP E-0-07, "Fire In Dedicated Fire Zone," Rev. P.

III.10 OP N-CC-31, "Component Cooling Water System Operation," Rev. V.

III.11 Generic Letter 81-21, "Natural Circulation Cooldown."

III.12 KNPP USAR, Table 14.3.4-19, Rev. 17.

III.13 Letter from M. L, Marchi to Document Control Desk, "Radioactive Effluent Release Report January – December 1996," NRC-97-36, dated April 23, 1997.

III.14 Letter from M. L. Marchi to Document Control Desk, "Radioactive Effluent Release Report January – December, 1997," NRC-98-38, April 23, 1998.

IIII.15 Letter from M. L. Marchi to Document Control Desk, "Radioactive Effluent Release Report January – December, 1998," NRC-99-034, April 29, 1999.

III.16 Letter from M. L. Marchi to Document Control Desk, "Radioactive Effluent Release Report January – December, 1999," NRC-00-036, April 27, 2000.

IV. Mechanical/Structural/Material Component Integrity and Design

The following sections discuss the changes to NSSS design parameters for the MUR uprate and the evaluation of the continued structural integrity of major plant components, reactor vessel integrity, and component inspection and testing programs.

1. A discussion of the effect of the power uprate on the structural integrity of major plant components.

Westinghouse evaluated the structural integrity of the KNPP plant components for power uprate. The evaluations were performed at an increased power level of 1772 MWt. This power level exceeds the power level being requested for this MUR power uprate amendment request. Table IV.B-1, below compares the NSSS design parameters of current operation with the MUR design parameters and the parameters for a 7.4 percent uprate to 1772 MWt. Table IV.A-1 below dispositions the components and will refer the reader to the appropriate section or attachment for review. The table lists the components as found in the RIS, Section IV.1.A.i through IV.1.A.ix. The analysis of record (AOR) or code used is also referred to in Table IV.A-1.

A. This discussion should address the following components:

- **i. reactor vessel, nozzles, and supports**
- **ii. reactor core support structures and vessel internals**
- **iii. control rod drive mechanisms**
- **iv. Nuclear Steam Supply System (NSSS) piping, pipe supports, branch nozzles**
- **v. balance-of-plant (BOP) piping (NSSS interface systems, safety related cooling water systems, and containment systems)**
- **vi. steam generator tubes, secondary side internal support structures, shell, and nozzles**
- **vii. reactor coolant pumps**
- **viii. pressurizer shell, nozzles, and surge line**
- **ix. safety-related valves**

Table IV.A-1, Bounding Evaluations for NSSS Components

Component	Core Power (MWt)	Analysis of Record (Reference)	Evaluation Location (within this submittal letter)
Reactor vessel, nozzles, supports	1772	IV.4	Attachment 3, Section 5.1
Reactor core support structures and vessel internals	1772	IV.12	Attachment 3, Section 5.1
Control Rod Drive Mechanisms	1772	IV.5	Attachment 3, Section 5.4
NSSS piping:			Attachment 3, Section 5.5
Reactor coolant loop piping	1772	IV.1	
Primary equipment nozzles	1772	IV.1	
Primary equipment supports	1772	IV.1	
Pressurizer surge line piping	1772	IV.7	
Reactor coolant loop branch nozzles	1772	IV.1	
NSSS piping supports	1772	IV.8	Attachment 3, Section 5.5
Other NSSS fluid system piping:			Summary directly following this table
Chemical and Volume Control	1772	IV.1	
Residual Heat Removal	1772	IV.1	
Safety Injection	1772	IV.1	
Internal Containment Spray	1772	IV.1	
Component Cooling Water	1772	IV.1	
BOP piping (NSSS interface systems)			Summary directly following this table
Main Steam	1772	IV.1	
Condensate and Feedwater	1772	IV.1	
Auxiliary Feedwater	1772	IV.1	
Steam Generator Blowdown	1772	IV.1	
Unit Model 54F Steam generators:			Attachment 4, Section 5.7
Tubes	1772	IV.13	
Internal support structures	1772		
Shells	1772		
Nozzles	1772		
Reactor Coolant Pumps:			Attachment 3, Section 5.6
Structural	1772	IV.9(1)	
Electrical	1772	IV.10, IV.11	
Pressurizer:			Attachment 4, Section 5.8
Shell	1772	IV.6	
Nozzles	1772	IV.6	
Surge line	1772	IV.6	
Safety-Related Valves	1772	To original specifications see attachment 3	Attachment 3, Section 5.9

(1) Code used only to demonstrate acceptability. KNPP RCPs pre-date the inclusion of pumps into ASME code.

Other NSSS Fluid System Piping

Thermal, pressure, and flow rate change factors were developed during the evaluation of piping systems at KNPP for a 7.4 percent power uprate. If the change factors were less than or equal to a five percent increase, the increase was concluded to be acceptable based on the following rationale. Certain levels of deviation from design basis conditions can be concluded to be permissible if that level of change would not alter the piping system results to an appreciable degree. Relatively small temperature changes can be concluded to be acceptable as the increase in pipe stresses, pipe support loads, nozzle loads and piping displacements are correspondingly small and generally predictable. These increases are somewhat offset by conservatism in analytical methods used to calculate thermal and or fluid transients stresses and loads. Conservatism may include the enveloping of multiple thermal operating conditions, as well as not considering pipe support gaps in thermal analyses. This five percent rationale can also be used for jet impingement and pipe whip. This rationale has been used successfully in several power uprate applications.

If thermal, pressure, and flow rate change factors were greater than five percent, more detailed evaluations were performed to address the specific increase in temperature, pressure, and/or flow rate in order to document design basis compliance.

The piping and supports of the Chemical and Volume Control, Residual Heat Removal, Safety Injection and Internal Containment Spray, and the Component Cooling systems were reviewed to justify acceptability of these systems for a 7.4 percent power uprate (1772 MWt core power). The purpose of the piping review was to evaluate the piping systems for effects resulting from the power uprate conditions to demonstrate design basis compliance in accordance with the USAS B31.1, "Power Piping Code," (reference IV.1). The piping and pipe support evaluations performed concluded that all systems remain acceptable and will continue to satisfy the design basis requirements when considering the temperature, pressure, and flow rate effects resulting from the 7.4 percent power uprate. These evaluations performed at 1772 MWt core power bound the impact of the 1.4 percent MUR power uprate to 1673 MWt.

BOP Piping and Supports Evaluation

As part of the evaluations for an uprate to 1772 MWt core power, balance of plant piping and supports were evaluated using the "change factor" methodology described above. This evaluation included the main steam, feedwater, condensate, auxiliary feedwater, and steam generator blowdown systems. The purpose of the piping review was to evaluate the piping systems for effects resulting from the power uprate conditions to demonstrate design basis compliance in accordance with the USAS B31.1, "Power Piping Code," (reference IV.1). The piping and pipe support evaluations performed for these BOP systems concluded that all systems remain acceptable and will continue to satisfy the design basis requirements when considering the temperature, pressure, and flow rate effects resulting from the 7.4 percent power uprate conditions. These evaluations performed at 1772 MWt core power bound the impact of the 1.4 percent MUR power uprate to 1673 MWt.

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

i. stresses

Addressed in attachments 3 and 4, in the respective component section.

ii. cumulative usage factors

Addressed in attachments 3 and 4, in the respective component section.

iii. flow induced vibration

Addressed in attachments 3 and 4, in the respective component section.

iv. changes in temperature (pre- and post-uprate)
v. changes in pressure (pre- and post-uprate)
vi. changes in flow rates (pre- and post-uprate)

The sections iv, v, and vi regarding changes in parameters are all discussed in the following paragraphs:

The NSSS design parameters developed by Westinghouse are the fundamental NSSS parameters used as primary input to the NSSS system and component analyses, design transient analysis, and USAR Chapter 14 safety analyses. The parameters are established using conservative assumptions in order to provide bounding conditions appropriate for analyses. No uncertainties are incorporated into the values of the parameters.

NSSS component and system analyses were performed for a power level of 1650 MWt either during original design or during the KNPP Replacement Steam Generator (RSG) project. In 2001, the NSSS design parameters were redeveloped to evaluate the transition to Westinghouse 422V+ fuel (the fuel transition that is currently under review by the NRC, reference IV.14). These NSSS design parameters developed for the fuel transition included the uprated power of 1772 MWt. Most of the fuel transition accident analysis were performed at 1772 MWt, as seen in the previous sections II and III of this attachment.

The structural integrity of the components was evaluated using the NSSS parameters at 1772 MWt (a 7.4 percent power uprate). The evaluations for a 7.4 percent power uprate will be used to disposition the components for the 1.4 percent MUR power uprate. The components listed above and evaluated in the attachment used the 7.4 percent uprated power parameters in their evaluations. The 7.4 percent parameters bound those for the 1.4 percent MUR power uprate.

Table IV.B-1 below lists the NSSS design parameters used for the component evaluations at 1650 MWt and for the component evaluations at the full uprated core power of 1772 MWt. Additionally, NSSS design parameters were developed for the 1.4 percent MUR uprate strictly for comparison with the 1650 MWt and 1772 MWt cases to show the values were bounded. No components or accident analyses were specifically evaluated at the 1.4 percent MUR uprated NSSS design parameters.

Table IV.B-1, NSSS Design Parameters at Current, MUR Uprate, and 7.4 Percent Uprated Power Levels

Parameter	Current 1650 MWt	MUR Uprate 1673 MWt	7.4% Uprate 1772 MWt
Core power (MWt)	1650	1673	1772
NSSS power (MWt)	1657.1(1)	1681	1780
RCS Pressure (psia)	2250	2250	2250
Thermal design flow (gpm/loop)	89,000	89,000	89,000
Tavg range (°F)	554.1 – 575.3	556.3 – 573.0	556.3 – 573.0
Vessel Inlet T_{cold} (°F)	521.9 – 543.8	523.7 – 541.0	521.9 – 539.2
Vessel Outlet T_{hot} (°F)	586.3 – 606.8	588.9 – 605.0	590.8 – 606.8
Steam Temperature (°F)	493.8 – 520.3	495.4 –517.1	492.1 – 514.0
Steam Pressure (psia)	644 – 815	653 –792	634 – 771
Steam Flow Rate (lbm/hr)	7.11 E06 – 7.14 E06	7.23 E06 – 7.26 E06	7.73 E06 – 7.76 E06
SG tube plugging (%)	0 – 10	0 – 10	0 – 10
Feedwater Temperature (°F)	427.5	429.3	437.1
FW Flow Rate (assuming no blowdown) (lbm/hr)	7.11 E06 – 7.14 E06	7.23 E06 – 7.26 E06	7.73 E06 – 7.76 E06

(1) Pump heat was 7.1 MWt for RSG. During the fuel transition project, pump heat was rounded up to 8 MWt.

vii. high-energy line break locations

Thermal, pressure, and flow rate change factors were developed during the evaluation of piping systems at KNPP for a 7.4 percent power uprate. The "change factor" methodology was described earlier in section IV.A for NSSS and BOP piping systems. This five percent rationale can also be used for high energy line break locations.

Piping in the following systems are considered part of the high energy line break (HELB) licensing basis at KNPP: the main steam (MS), feedwater (FW), steam generator blowdown (SGBD), and the auxiliary steam to the turbine driven auxiliary feedwater pump. The effect of a 7.4 percent power uprate on HELB locations was evaluated. It was determined that no HELB pipe break locations on the MS, SGBD, and auxiliary steam lines are changed due to changes in the operating conditions associated with the core power uprate, since the uprate pressure and temperature parameters are bounded by the existing pipe stress analysis. For the FW lines, the changes to operating temperatures, pressures, and flow rates were evaluated to be acceptable based on the "change factor" analysis. Therefore, the existing design basis for FW pipe break locations remains acceptable for both the MUR and a 7.4 percent power uprate.

viii. jet impingement and thrust forces

Thermal, pressure, and flow rate change factors were developed during the evaluation of piping systems at KNPP for a 7.4 percent power uprate. The "change factor" methodology was described earlier in section IV.A for NSSS and BOP piping systems. This five percent rationale can also be used for jet impingement and pipe whip.

The changes to the operating temperatures, pressures, and flow rates for the piping systems were determined to be sufficiently small (i.e., less than five percent). Therefore, the existing design basis for pipe whip and jet impingement remains acceptable for both the MUR and a 7.4 percent uprate.

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

i. pressurized thermal shock calculations

Addressed in attachment 3, in section 5.1.2, Reactor Vessel Integrity.

ii. fluence evaluation

Addressed in attachment 3, in section 5.1.2, Reactor Vessel Integrity.

iii. heatup and cooldown pressure-temperature limit curves

Addressed in attachment 3, in section 5.1.2, Reactor Vessel Integrity.

iv. low-temperature overpressure protection

Addressed in attachment 3, in section 5.1.2, Reactor Vessel Integrity.

vi. upper shelf energy

Addressed in attachment 3, in section 5.1.2, Reactor Vessel Integrity.

vii. surveillance capsule withdrawal schedule

Addressed in attachment 3, in section 5.1.2, Reactor Vessel Integrity.

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

The code of record is the design code, including the specific code edition, that is recognized in the KNPP's licensing basis. The codes of record have been identified for each of the components listed in this section in the component reviews completed by Westinghouse and provided as attachments 3 and 4. Additionally, Table IV.A-1 contains a reference to the code of record.

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

Component inspection and testing programs that could be affected by the MUR power uprate were reviewed. The review included the current inservice inspection (ISI) program, the inservice testing (IST) program, the motorized operated valve (MOV) program, the steam generator (SG) inspection program, and the flow accelerated corrosion (FAC) program. Additionally, GL 95-07, which addressed concerns that pressure locking and thermal binding could render redundant safety systems incapable of performing their intended functions, and GL 96-06, which addressed concerns of thermally induced over pressurization of isolated water filled piping sections in containment and two-phase flow and waterhammer in containment air cooling water systems, were both reviewed for uprate impacts.

ISI Program

There are no modifications or replacement of ASME Code Class components that provide safety-related functions for the MUR power uprate. Additionally, there are no new safety-related functions required of existing equipment and no new safety-related equipment will be installed to accommodate the MUR uprate. Therefore, the ISI program at KNPP is not affected by the 1.4 percent MUR power uprate.

IST Program

There are no modifications or replacement of ASME Code Class components that provide safety-related functions for the MUR power uprate. Additionally, there are no new safety-related functions required of existing equipment and no new safety-related equipment will be installed to accommodate the MUR uprate. The program review found that no changes were being made to the safety system flow rate assumptions, system pressures, or the accident analyses assumptions from the current condition or from the fuel transition, the latter of which is currently under review by the NRC (Reference IV.14). Therefore, the current IST program at KNPP is not affected by the MUR power uprate.

Motorized Operated Valves

Generic Letter (GL) 89-10 addressed concerns related to the reliable operation of MOVs. A review of the applicable MOV Program Operating Condition Evaluations (OCEs) was performed to assess the plant operational parameters that will change as a result of the MUR power uprate. The plant operational parameter changes are bounded by the assumptions made in the MOV OCEs. Therefore, the safety-related MOVs at KNPP will continue to be capable of performing their intended functions at MUR uprated conditions.

SG Inspection Program

The steam generator inspection program is addressed in the next section, Section IV.F.

FAC Program

The FAC program at KNPP is affected by the MUR power uprate (i.e., changes in temperature, pressure, and flow). Predictive analysis was performed for a larger, 7.4 percent power uprate using the CHECWORKs computer code, version 1.0 G. Based on the predictive analysis, the wear rates increased in some lines. However, the identified changes are not significant and are not expected to cause wear rates or inspection intervals to change significantly following uprate. In all cases, the additional wear imposed by a 7.4 percent uprate amounts to less than 0.001 inches per year. The system of most concern in regards to FAC is the feedwater system because it contains the most energy. The table below summarizes the change in the final feedwater parameters based on the predictive analysis.

Table IV.E-1, FAC Program Effects on Feedwater System

Parameter	Actual Value (pre up-rate)	Actual Value (post up-rate)
Operating Temperature	431.5°F	436.7°F
Velocity	17.83 ft/sec	20.31 ft/sec
Wear Rate	1.166 mils/yr	1.288 mils/yr

The CHECWORKs model for KNPP will need to be updated following a plant power uprate, but the new power level will have minimal effect on FAC wear rates and inspection intervals. Additionally, wear rate assessments are a part of the FAC program and this program will remain in place following the MUR uprate. Based on the evaluation described above, the changes on the FAC program for a 7.4 percent uprate are insignificant. The evaluation assuming a 7.4 percent power uprate bounds the 1.4 percent MUR power uprate.

Generic Letter 95-07

Generic Letter 95-07 addressed concerns that pressure locking and thermal binding could render redundant safety systems incapable of performing their intended functions. The NRC SER for the GL 95-07 (reference IV.3) dated January 13, 1998, stated that affected valves were either modified or procedures modified to assure that the conditions of concern were adequately addressed. Only two valves had their acceptability for pressure locking based on calculations. The calculation used pressure conditions that would not change with the implementation of the MUR power uprate. Based on this review, there are no changes to the SER conclusions.

Generic Letter 96-06

The Generic Letter 96-06 addressed concerns regarding thermally induced, over pressurization of isolated water-filled piping sections in containment and two-phase flow and waterhammer in containment cooling water systems. The KNPP's current containment integrity analysis is based on a power level of 1650 MWt with two percent uncertainty. This analysis will continue to apply for the MUR power uprate and is bounding. The containment pressure and temperature accident values remain as documented in the analysis of record. Therefore, the MUR power uprate does not impact the GL 96-06 conclusions.

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

NRC Bulletin 88-02, dated February 5, 1988, requested holders of Westinghouse designed nuclear power reactors with steam generators (SGs) having carbon steel support plates to implement actions to minimize the potential for a steam generator tube rupture. In 2001, the Kewaunee SGs were replaced. The replacement steam generators for Kewaunee contain stainless steel support plates with quatrefoil broached tube holes. The information requested by Bulletin 88-02 no longer applies to the Kewaunee SGs because the tube support plate design and material minimizes the potential for denting. The MUR power uprate does not change this. However, flow induced vibration of the SGs is discussed in attachment 4, section 5.7. Additionally, steam generator inspections are performed in accordance with KNPP Nuclear Administrative Directive (NAD) 01.21, "Steam Generator Program," and General Nuclear Procedure (GNP) 01.21.01, "Requirements for Steam Generator Primary Side Activities." These procedures incorporate the requirements of Nuclear Energy Institute (NEI) 97-06, "Steam Generator Program Guidelines," and latest industry guidance. Any degradation due to denting would be identified and evaluated through this program.

The Electrical Power Research Institute (EPRI) Steam Generator Inspection Guidelines define active degradation mechanisms as the combination of at least ten new indications of degradation (greater than or equal to 20 percent through-wall) and previous indications that have an active growth rate that is greater than or equal to 25 percent of the repair limit per cycle in any steam generator. The guidelines also define active degradation mechanisms as new or previously identified indications which have a one cycle growth rate equal to or exceeding the repair limit.

The first inservice inspection of the Kewaunee replacement SGs is scheduled for April 2003. The results of the preservice examination indicated no active damage mechanisms in the SGs prior to being placed in service. The Kewaunee SGs will continue to be assessed for degradation per site directives that meet the EPRI guidelines.

3. References for Section IV:

IV.1 USAS B31.1, Power Piping Code, 1967.

IV.2 ASME Boiler and Pressure Vessel Code Section XI, 1989 Edition.

IV.3 Letter M. L. Marchi from the NRC, "Safety Evaluation of Licensee Response to Generic Letter 95-07, "Pressure Locking And Thermal Binding of Safety-Related power-operated Gate Valves," for the Kewaunee Nuclear Power Plant (TAC No. M93475)," dated January 13, 1998.

IV.4 ASME Boiler and Pressure Vessel Code, Section III, Nuclear Power Plant Components, 1968 Edition with Addenda through the Winter 1968, American Society of Mechanical Engineers, New York.

IV.5 ASME Boiler and Pressure Vessel Code, Section III, "Rules for Construction of Nuclear Vessels", 1965 Edition through Summer 1996 Addenda, The American Society of Mechanical Engineers, New York.

IV.6 "Rules for Construction of Nuclear Power Plant Components," ASME Boiler and Pressure Vessel Code Section III, 1965 Edition, Summer 1966 Addenda, The American Society of Mechanical Engineers, New York, New York, USA.

IV.7 ASME B&PV Code Section III, Subsection NB, 1986 Edition.

IV.8 AISC Specification for the Design, Fabrication & Erection of Structural Steel Buildings, 1969.

IV.9 ASME Boiler and Pressure Vessel Code, Section III, "Rules for Construction of Nuclear Vessels", 1968 Edition, and later Editions and Addenda, The American Society of Mechanical Engineers, New York.

IV.10 Equipment Specification No. E-565614, Revision Q, General Specification for Induction Motor for Shaft Seal Type Pump, Westinghouse Electric Corporation, Atomic Equipment Division, August 9, 1972.

IV.11 Equipment Specification No. E-565626, Revision D, Supplementary Ordering Information for Shaft Seal Type Pump Motor, Westinghouse Electric Corporation, Atomic Equipment Division, January 8, 1971.

IV.12 ASME Boiler and Pressure Vessel Code, Section III, Nuclear Power Plant Components, 1968 Edition with Addenda through the Winter 1968, American Society of Mechanical Engineers, New York.

IV.13 ASME Boiler and Pressure Vessel Code, Rules for Construction of Nuclear Power Plant Components, Section III, 1986 Edition, 1987 Addenda (applies to RSGs), The American Society of Mechanical Engineers, New York.

IV.14 Letter NRC-02-067 from Mark E. Warner to Document Control Desk, "License Amendment Request 187 to the Kewaunee Nuclear Power Plant Technical Specifications Changes for Use of Westinghouse VANTAGE+ Fuel," July 26, 2002 (TAC # MB5718).

V. Electrical Equipment Design

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

A. emergency diesel generators

The loading on the emergency diesel generators 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 BOP loads showed that there were no loads fed by the emergency diesel generators 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 emergency diesel generator 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.

B. station blackout equipment

The only potential impact of the 1.4 percent MUR power uprate on the ability of the plant to withstand and recover from a station blackout (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 uprate. There is no change in the ability of the turbine-driven auxiliary feedwater pump, supplied with steam from the steam generators, to support reactor heat removal due to uprate. The Technical Specification minimum required volume in the condensate storage tanks (CST) is 39,000 gallons. This volume remains acceptable for the MUR power uprate since it is based on 102 percent of the current rated power of 1650 MWt. The TS CST volume and the assumed power level and uncertainty are described in an NRC safety evaluation dated November 20, 1990 (reference V.1) and confirmed in a supplemental safety evaluation and an additional safety evaluation dated October 1, 1991 (reference V.2) and November 19, 1992 (reference V.3), respectively. The two percent uncertainty on the current core power of 1650 MWt bounds the uprate to 1673 MWt (a 1.4 percent uprate with 0.6 percent uncertainty). Therefore, the ability of the KNPP to respond to a SBO will not be altered due to the 1.4 percent MUR power uprate.

C. environmental qualification of electrical equipment

In accordance with 10 CFR 50.49, electrical equipment important to safety must be qualified to survive postulated harsh environments during normal operation and post-accident. This includes conditions of normal operation and design basis events (e.g., LOCA). Changes to normal operating conditions were evaluated for a 1772 MWt core power. The evaluations encompass the 1.4 percent MUR power uprate. The power uprate does not affect the chemical spray or submergence aspects of the KNPP design and, therefore, these parameters are not discussed below.

Normal Operating Conditions

Pressure / Temperature / Humidity

Normal service conditions are those environments that are maintained in each area during normal plant operation. The power uprate results in changes to the RCS, steam generator, main steam, and feedwater parameters. An uprate as high as 7.4 percent corresponding to 1772 MWt has been analyzed for the impact on the normal design temperatures and the environmental conditions in containment, the auxiliary building and the turbine building. The evaluation showed that these areas can be maintained within the normal ranges by the existing HVAC systems. Therefore, the normal aging conditions in these areas will not be impacted and are bounded by the current design.

Radiation

For a 7.4 percent power uprate, areas outside of containment remain bounded by the current EQ Plan. The MUR power uprate will increase the core power to 1673 MWt and is bounded by the evaluation completed for a 7.4 percent power uprate. Therefore, the current EQ Plan is bounding for areas outside of containment.

The effects of the power uprate on the normal containment dose were described in detail in Section III of this attachment. Please refer to that section for a detailed summary.

Accident Conditions

Accident conditions are the most severe environments that may occur in each area following a postulated accident. In general, accidents causing the most severe environments include loss-of-coolant accidents and main steam line breaks inside containment and high energy line breaks outside containment.

Pressure / Temperature / Humidity
The main steam line break results for inside containment bound the conditions for the MUR uprate since the analyses are performed at 1683 MWt (102 percent current power). Mass and energy releases for a steam line break outside containment are also based on a core power of 1650 MWt with two percent uncertainty and remain bounding for the MUR power uprated conditions. Therefore, there are no changes to EQ parameters for a steam line break accident.

The LOCA containment response is currently analyzed at 1683 MWt (102 percent of current rated power). Therefore, the KNPP containment analysis does not change for the MUR uprate and there are no changes in temperature, pressure, and humidity following a LOCA.

Radiation

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.

D. grid stability

American Transmission Company (ATC) assessed thermal loadability, voltage control, and grid stability for the KNPP. The impact study performed by ATC identified no thermal loadability, voltage control, or grid stability issues for the 1.4 percent MUR power uprate for the KNPP.

2. Normal Electrical Power Systems (not part of RIS 2002-03 Guidance)

During the August 8, 2002, NRC meeting for KNPP power uprates, the electrical systems reviewer requested the KNPP submittal address the request for additional information given to the Point Beach Nuclear Plant concerning normal electrical power systems. That information is assembled in Table V-1 below.

Table V-1, Electrical Equipment Information

Equipment	Units	Existing Operating Values	Calculated Existing Operating Values	Existing Design and Procedure Limits	Anticipated Power Uprate (7.4%)	Maximum Equipment Design
Unit Generator	MVA	560	NA	622.389	622.389	622.389
	Power Factor	0.993	NA	0.9	0.957	0.957
	MWe	556	NA	560.15	595.7	595.7
	MVAR	66.1	NA	200	180.3	180.3
Main Transformer	MVA	535	NA	648	627.8	649.599
Main Transformer Isophase Bus	Amps	16145	NA	20000	18900	20000
Main Auxiliary Transformer (MAT)	MVA	26.8	32.1	44.8	32.3	44.8
MAT Iso-phase Bus	Amps	770	923	1600	1360	1600
Reserve Auxiliary Transformer (RAT)	MVA	27.7	35.4	44.8(1)	35.6	40
RAT Primary Cable	Amps	113	143	980	144	980
MAT Y Winding Bus Bar	Amps	2659	2707	3000	2731	3000
MAT X Winding Bus Bar	Amps	1230	1732	2000	1732	2000
RAT B Winding Bus Bar	Amps	2659	2675	3000	2699	3000
RAT A Winding Bus Bar	Amps	1427	1730	2000	1730	2000

(1) A corrective action request has been initiated to investigate why the RAT alarm procedural limit was higher than the RAT maximum equipment design . This corrective action request will be completed prior to the MUR uprate implementation.

Table V-2, Motor Driven Pump Data

Equipment	Existing BHP	Uprate BHP (7.4%)	Design BHP	BHP @ 1.15 Service Factor
Main Feedwater Pump Motor	5000	5150	5000	5750
Condensate Pump Motor	1500	1528	1500	1725
Heater Drain Pump Motor	350	500	500	N/A

Based on the motor driven pump data for the 7.4 percent power uprate above, the main feedwater and condensate pumps will exceed their design brake horse power (BHP). However, the uprated BHP is within the service factor of the motors on both of these pumps. The effect of operating into the motor service factor is that the life of the motor is decreased. The motors will continue to operate properly and within their insulation system design temperatures at the uprated BHP. The data above is for 7.4 percent power uprate. The MUR power uprate would be bounded by this evaluation.

Additionally, a review of the heater drains fluid system information at the current and 7.4 percent uprate values show an increase in pump flow of approximately 13.6 percent. This increase is significantly less than the 30 percent increase in motor horsepower assumed above to require the motor to operate at full load.

3. References Section V.

V.1 Letter to Ken H. Evers from NRC, "Safety Evaluation of the Kewaunee Nuclear Power Plant Response to the Station Blackout Rule (TAC No, 68558)," November 20, 1990.

V.2 Letter to Ken H. Evers from NRC, "Supplemental Safety Evaluation of the Kewaunee Nuclear Power Plant, Response to the Station Blackout Rule (TAC No. 68558)" October 1, 1991.

V.3 Letter to C.A. Schrock from NRC, "Kewaunee Nuclear Power Plant, Unit No. 1 - Station Blackout Rule (10 CFR 50.63) (TAC No. M84521)," November 19, 1992.

VI. System Design

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

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

The NSSS interface systems were evaluated for KNPP at an uprate power of 1772 MWt. The evaluation included the following systems: the main steam system, the steam dump system, the condensate and feedwater system, the auxiliary feedwater system, and the steam generator blowdown system. The evaluations are summarized in attachment 3, Section 4.2, "NSSS/Balance-of-Plant Interface Systems."

B. containment systems

Containment System Structure and Containment Isolation System

No changes to the containment structure or containment isolation systems are being made as part of the MUR power uprate. The systems are periodically tested for containment design integrity. There are no changes in the test programs based on a 1.4 percent power uprate. The containment response for a main steam line break was performed at 1650 MWt with two percent uncertainty. The current loss of coolant accident (LOCA) containment integrity analysis is based on 102 percent of the current licensed power (1650 MWt). Both of these analyses bound operation at the MUR uprated power of 1673 MWt. Therefore, the 1.4 percent MUR uprate does not affect these systems.

Containment Ventilation System

The KNPP containment ventilation system consists of the containment air cooling system and the purge and ventilation system. The function of the containment air cooling system is to remove heat loss from equipment and piping in the containment during normal plant operation. During accident conditions, the containment cooling system is designed to remove sufficient heat from containment to keep the containment pressure from exceeding the design pressure. The function of the purge and ventilation system is to provide fresh, tempered air for comfort during maintenance and refueling operations and to purge contaminated air from containment whenever required for access. The purge and vent system is not used when the reactor is above hot shutdown conditions. Since the purge and vent system is designed for operation during shutdown conditions, it is not affected by the uprate.

The containment ventilation system was evaluated for a 7.4 percent power uprate. This evaluation is bounding since the weather conditions resulting in maximum supply temperatures for the ventilation and air condition systems were used. This evaluation bounds the 1.4 percent MUR power uprate since containment heat load increases would be smaller.

For normal operation at conditions associated with a core power of 1772 MWt (7.4 percent uprate), the heat load to the containment air cooling system is expected to increase approximately two percent. This would correlate to a 1.2°F increase in containment temperature. The increase is based on the increased temperatures of the reactor coolant system, main steam, and feedwater resulting from the power uprate. The highest summer/fall average temperature for uprate is estimated to be 112°F at uprate conditions. This remains below the 120°F limit for containment temperature in the equipment qualification plan. Therefore, at 7.4 percent uprate conditions, the normal operating containment temperature limits are not violated and the two percent increase in heat load would not adversely impact the system operation. The MUR power uprate would be bounded by this evaluation since the ventilation heat load increase would be much smaller.

For accident conditions, the current analysis of record for LOCA containment integrity and the MSLB containment response has been performed at 102 percent of 1650 MWt. These analyses remain bounding for the 1.4 percent MUR power uprate.

Based on the above discussion for normal and accident operations, the containment ventilation system remains capable of performing its functions following the MUR power uprate.

C. safety-related cooling water systems

Residual Heat Removal (RHR) System

The existing RHR System capability has been assessed and found to be adequate for an uprated power condition of 1772 MWt. Both the Appendix R and normal cooldown requirements for the system are satisfied at the uprated power condition. This evaluation bounds the MUR power uprate of 1673 MWt. A summary of the evaluation is provided in attachment 3 of this submittal, section 4.1, "NSSS Fluid Systems."

Safety Injection (SI)/Internal Containment Spray (ICS) System

The existing SI and ICS systems have been evaluated 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 main steam pipe break. The LOCA containment integrity and main steam line break containment response were performed at 102 percent of the current core power of 1650 MWt, as indicated in Section II of this attachment. Both analyses provided acceptable results. Therefore, no changes are required to these systems for the MUR uprated power condition of 1673 MWt. The system evaluation at 1772 MWt bounds the MUR power uprate of 1673 MWt. A summary of the evaluation is provided in attachment 3 of this submittal, section 4.1, "NSSS Fluid Systems."

Component Cooling Water System (CCWS)

The CCWS is adequately sized for normal cooldown heat loads associated with a power uprate of 1772 MWt (7.4 percent uprate). It is also adequately sized to meet the Appendix R cooldown requirements. Small changes in heat loads that are predicted to occur during normal modes of plant operation are well within the system's design capability. Therefore, the CCWS requirements are met for a 1772 MWt uprate. This evaluation bounds the MUR power uprate of 1673 MWt. A summary of the evaluation is provided in attachment 3 of this submittal, section 4.1, "NSSS Fluid Systems."

The current LOCA containment integrity analysis, which defines the maximum emergency core cooling system recirculation heat load, is performed at 102 percent of 1650 MWt. Therefore, the current containment integrity analysis is bounding for the 1.4 percent MUR power uprate.

Service Water (SW) System

The KNPP SW system supplies redundant cooling water to the engineered safeguards equipment required during post-accident conditions and non-redundant cooling water to other systems including the BOP equipment. The SW system was evaluated for a 7.4 percent power uprate. For a 7.4 percent uprate, the required SW flow rates to the engineered safeguards equipment for accident conditions are not impacted and the current analysis is based on conditions that remain bounding. The most significant impact was the turbine building flow rates increase for normal full power operation. The remaining SW heat loads do not require any increase in the SW flow for normal and accident conditions at power levels up to 1772 MWt from that already established for the current power level. No changes or equipment additions are required to the SW system to support a 7.4 percent uprate. The 1.4 percent MUR power uprate is bounded by this evaluation for a 7.4 percent power uprate.

D. spent fuel pool storage and cooling systems

The spent fuel pool (SFP) cooling system is designed to remove decay heat from the spent fuel to maintain the SFP temperatures below a maximum temperature of 150°F. Heat removal is accomplished by drawing water from the surface of the pool, circulating the water through filters, pumping it through a heat exchanger, and returning the cooled water to the pool. The SFP purification components are not affected by the power uprate.

The current spent fuel pool cooling calculation is performed at 1650 MWt with a 2 percent uncertainty added. Therefore, the 1.4 percent MUR power uprate is bounded by the current SFP cooling analysis.

E. radioactive waste systems

The liquid and gaseous radwaste systems' design must be such that the plant is capable of maintaining normal operation offsite releases and doses within the requirements of 10 CFR 20 and 10 CFR 50, Appendix I. The waste systems were evaluated for a 7.4 percent power uprate and it was determined that there was no significant impact on the expected annual radwaste effluent releases or doses. The evaluation of the expected radwaste effluents and doses was summarized in this attachment in section III.3, "Normal Radiological Effluents." Therefore, it can be concluded that following uprate, the liquid and gaseous radwaste effluent treatment system will remain capable of maintaining normal operation offsite doses within the requirements of 10 CFR 50 Appendix I. Additionally, actual performance and operation of the installed equipment and reporting of actual offsite releases and doses continues to be controlled by the requirements of the KNPP Offsite Dose Calculation Manual (ODCM).

F. Engineered safety features (ESF) heating, ventilation, and air conditioning systems

An evaluation of the heating, ventilation, and air conditioning (HVAC) systems was performed at a core power of 1772 MWt. The evaluation concluded the current ventilation systems at the KNPP would be able to maintain operating temperature at or below the maximum normal operating temperatures following the 7.4 percent uprate. Additionally, an uprate to 1772 MWt would not affect the HVAC systems' abilities to perform non-cooling functions (i.e., isolating containment, maintaining negative pressure, heating or providing ventilation). This evaluation bounds the 1.4 percent MUR power uprate.

Engineered safety features (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. Each of these systems is factored into the radiological evaluations performed using the Alternate Source Term (AST) Methodology assuming a core power of 1650 MWt with a calorimetric uncertainty of two percent. Therefore, the radiological analyses are acceptable for a core power of 1673 MWt with 0.6 percent calorimetric uncertainty. All analyses had acceptable results and are currently under review by the NRC (reference VI.1). The table containing the radiological analyses and the initial power levels with uncertainties is contained in table III.1 in Section III of this attachment. Based on these accident analyses, the 1.4 percent MUR power uprate does not affect the ability of the ESF ventilation systems to mitigate the radiological doses of an accident.

The SFP air sweep requirements are not affected by the power uprate and, therefore, this system is not impacted by the uprate. Additionally, the SFP sweep system is not credited in the fuel handling accident.

2. References for Section VI

VI.1 Letter NRC-02-024 from Mark E. Warner to Document Control Desk, "Revision to the Design Basis Radiological Analysis Accident Source Term," dated March 19, 2002 (TAC No. MB4596).

VII. Other

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

   The MUR power uprate is not expected to have any significant effect on the manner in which the operators control the plant (including operator response times) during normal operations or transient conditions. All operator actions that were taken credit for in the safety analysis are still valid following the MUR power uprate since the safety analyses referenced for the MUR uprate are all performed at either 1683 MWt or higher. Additionally, KNPP has performed an evaluation to determine the impact of the MUR power uprate on the KNPP Probabilistic Risk Assessment (PRA) model. This included a review of the Human Reliability Analysis that concluded the MUR power uprate would cause only minor numerical perturbations having negligible impact on the numerical results.

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

   A. **emergency and abnormal operating procedures**

   No changes are required to the KNPP emergency operating procedure (EOP) program as a result of the 1.4 percent MUR power uprate. Abnormal operating procedures 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. These restrictions were discussed earlier in Section I.1.G an I.1.H of this attachment.

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

   The new Crossflow UFMD will interface with the 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.

C. the control room plant reference simulator

The plant simulator is being modified to provide the same information and annunciation that is being changed in the control room. The modifications to the control room simulator are being done in accordance with the appropriate site design change procedures.

D. the operator training program

Overview training regarding the modifications for the power uprate is currently being 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.

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

All modifications associated with the MUR power uprate as discussed above will be completed prior to the MUR power uprate implementation. This includes changes to operating procedures, implementation of the PPCS and control room alarm functions, and operator training, as well as installation of the Crossflow UFMDs.

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

Plant operating procedures will be revised to limit the plant power to less than or equal to the new rated power level of 1673 MWt. Operations procedures will be revised to specify power reductions from the licensed power level. The power reductions will be based on the power measurement uncertainty associated with the available instrumentation being used. The uncertainties and power levels are discussed in detail in section I.1.G and I.1.H of this attachment.

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

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

A review considering the operating license, the current Wisconsin Pollutant Discharge Elimination System (WPDES) permit, and the information contained in the Final Environmental Statement (FES) was performed. Effluents from the plant that could change as a result of the MUR power uprate are thermal discharges to Lake Michigan and radiological effluents. Although increases in discharge amounts associated with the proposed power uprate are possible, they will remain within acceptable limits. Annual radiological discharges will continue to be a small percentage of the allowable limits and the FES estimates. The effluents are described below.

Wisconsin Pollutant Discharge Elimination System Permit, WI-00001571-06-0, and a Wisconsin Department of Natural Resources (WDNR) Order address thermal, chemical and contamination limits, and reporting requirements of KNPP. Chemical and contamination limits are defined on a volumetric basis. There will be no diluting effects based on the power uprate. Therefore, the chemical and contamination limits are not affected. The WDNR order exempted KNPP from thermal mixing zone requirements but imposed alternate effluent limits on the circulating water (CW) system discharge flow rate and the temperature change across the condenser. The WDNR order limits are 450,000 gpm and 30°F, respectively. An evaluation of a 7.4 percent power uprate showed no changes in the current CW flow rate. The evaluation also determined the total temperature rise across the condenser to be 16.7°F. Both remain within the limits of the WDNR order. Therefore, neither the permit nor the WDNR Order requires modification as a result of the 7.4 percent uprate. The 1.4 percent MUR power uprate is expected to have much less of an impact on the thermal discharge and is bounded by the 7.4 percent evaluation.

Normal annual radiological effluents were evaluated for an uprate to 1772 MWt. These effluents were described in Section III.3 of this attachment. Based on the evaluations performed for an uprated power of 1772 MWt, the liquid and gaseous radwaste system design will be capable of maintaining normal operational offsite releases and doses within the requirements of 10 CFR 20 and 10 CFR 50, Appendix I. Additionally, effluent increases are assumed to be proportional to the increase in power. Therefore, effluents from the MUR power uprate (1673 MWt) are bounded by this evaluation. Solid waste volume generation is expected to increase slightly. However, all solid waste is controlled within several state and federal regulatory limits through the KNPP Solid Radioactive Waste Process Control Program (reference VII.1).

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

Normal operation radiation levels were originally evaluated at a core power level of 1721 MWt. Therefore, the original evaluation bounds the power level of the MUR power uprate (i.e., 1673 MWt). Therefore, there will be no changes in radiation zoning in the plant. Additionally, individual worker exposures will be maintained within the acceptable limits of the site ALARA program that controls access to radiation areas.

Environmental Review Conclusions

Thermal effluents may change slightly following the MUR power uprate. However, these changes have been evaluated and the changes remain within the KNPP permit limits. Radiological effluents were evaluated at an increased core power of 1772 MWt. All releases and doses will remain within regulatory limits. Radiation exposure was also reviewed. Original normal dose evaluations were based on a 1721 MWt core power level. Therefore, radiation exposure shielding design does not change. Additionally, the site ALARA program will continue to monitor and control personnel exposure such that the regulatory limits are not exceeded.

Based on the above, the proposed change does not involve a significant change in the types of or significant increase in the amounts of any effluents that may be released offsite or a significant increase in individual or cumulative occupational radiation exposure. Therefore, the proposed change meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22 (b), an environmental assessment of the proposed change is not required.

6. **References**

VII.1 NAD-01.16, "Solid Radioactive Waste Process Control Program (PCP)," Revision E, March 26, 2002.

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

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

   A. a description of the change
   B. identification of analyses affected by and/or supporting the change
   C. justification for the change, including the type of information discussed in Section III, above, for any analyses that support and/or are affected by change

Technical Specification Changes

All technical specification changes were described in detail in attachment 1, section 2.

Protection System Setting Changes

Plant protection system settings for the power range NIs will be rescaled as necessary to support the new rated power. The intermediate range may be rescaled for the new rated power. The setting for ΔT_o will change for the new rated power of 1673 MWt.

Reactor trip interlocks P-7, P-8, and P-10 settings will be rescaled, as necessary, for the new power rating.

Engineered Safety Feature (ESF) Setting Changes

A review of Table TS 3.5-1, "Engineered Safety Features Initiation Instrument Setting Limits," concluded that no ESF TS required changes for the MUR power uprate. There are no plant ESF setting changes necessary for the MUR power uprate.

ATTACHMENT 3

Letter from Thomas Coutu (NMC)

To

Document Control Desk (NRC)

Dated

January 13, 2003

License Amendment Request 193

System and Component Evaluations in Support of Attachment 2

4.1 NSSS Fluid Systems

4.1.1 Introduction

As part of the evaluation for the Kewaunee Nuclear Power Plant (KNPP) 7.4-percent Power Uprate Project the following fluid systems were reviewed to confirm continued compliance with industry codes and standards, regulatory requirements, and applicable performance and design basis requirements:

- Reactor Coolant System (RCS)
- Chemical and Volume Control System (CVCS)
- Residual Heat Removal System (RHR)
- Safety Injection System/Containment Spray System (SIS/CSS)
- Sampling System (SS)
- Component Cooling Water System (CCW)

The fluid systems evaluations described in this section were performed at the system level.

4.1.2 Key Input Assumptions

The review was performed based on the approved range of NSSS design parameters, which were developed to support a NSSS power level of 1,780 MWt (Attachment 2, Table IV.B-1).

The approved range of NSSS design parameters were compared with the non-uprated design parameters previously evaluated for systems and components (Table IV.B-1). The comparison indicated differences that could impact the performance of the above fluid systems. For example, the 7.4-percent Power Uprate would result in a proportional increase in the residual heat load after reactor shutdown that must be removed by the RHRS and CCWS during plant cooldown.

4.1.3 Acceptance Criteria

The evaluations of the above fluid systems relative to compliance with industry codes and standards, regulatory requirements, and applicable performance and design basis requirements

are delineated in subsection 4.1.4. The acceptance criteria are included in subsection 4.1.4, along with the system evaluations, results, and conclusions.

4.1.4 Description of Fluid Systems Evaluations and Results

4.1.4.1 Reactor Coolant System

The changes in NSSS design parameters that impact the Reactor Coolant System (RCS) design bases functions include the increase in core power and the allowable range for average RCS temperature (T_{avg}). The increase in core power affects the total amount of heat transferred to the Main Steam System. This section of the report will discuss the RCS supporting fluid system designs. These system designs include the pressurizer surge line, safety valves inlet and discharge piping, pressurizer relief tank, power-operated relief valve inlet and discharge piping, pressurizer spray sub-system, resistance temperature detector bypass loop piping, and RCS instrumentation setpoints (excluding instrument channels used by the control and protection systems).

4.1.4.1.1 Pressurizer Surge Line, Safety Valves Inlet/Outlet Piping, and Pressurizer Relief Tank

The pressurizer safety valves are required to have adequate capacity to ensure that the RCS pressure does not exceed 110 percent of system design pressure. This is the maximum pressure allowed by the *American Society of Mechanical Engineers Boiler and Pressure Vessel (ASME B&PV) Code* (Reference1) for the worst-case loss-of-heat-sink event, that is, the loss of external electrical load . Based on the range of NSSS design parameters for the Power Uprate, an analysis of the loss-of-external-electrical-load transient was performed. The results of this analysis confirmed that the installed capacity of the pressurizer safety valves (690,000 lbs./hr or 345,000 lbs./hr/valve) is adequate to preclude RCS overpressurization. Based on results of this analysis, it was concluded that the supporting fluid systems design of the surge line, safety valve inlet piping, and safety valve discharge piping is also adequate, since the design of these piping systems is based on safety valve design capacity.

The pressurizer relief tank (PRT) design (including the tank level setpoints) is also based on the total safety valve capacity, and conservatively sized to condense and cool a discharge of pressurizer steam equal to 110 percent of the steam volume above the full-power pressurizer water level setpoint. Since the loss-of-external-electrical-load transient analysis determined that

the actual discharge of steam from the pressurizer into the PRT (540 lbm) is less than the design bases discharge (2965 lbm), the design of the PRT and existing level setpoints remain conservative. The current PRT level setpoints ensure adequate coolant is maintained in the tank to condense and cool the design bases discharge and preclude tank pressure from exceeding 50 psig. Therefore, the tank setpoints are conservative for the Power Uprate.

4.1.4.1.2 Pressurizer Power-Operated Relief Valves Inlet and Outlet Piping

The power-operated relief valves (PORVs) are required to have adequate capacity to prevent pressurizer pressure from reaching the high-pressure reactor trip setpoint for an external load reduction of up to 50 percent of rated electrical load. Based on the range of NSSS design parameters for the Power Uprate, a margin-to-trip analysis was performed. The results of this analysis confirmed that the installed capacity of the PORVs (358,000 lbs./hr, or 179,000 lbs./hr per valve at 2,350 psia) is adequate to preclude a high-pressurizer pressure-reactor trip. Based on these results, it can also be concluded that the supporting fluid systems design of the PORVs inlet and discharge piping is also adequate, since this system piping is designed based on the design capacity of the PORVs.

4.1.4.1.3 Pressurizer Spray Sub-System

The pressurizer spray sub-system (valves and piping) is required to have adequate capacity to maintain the pressurizer pressure below the actuating set pressure of the PORVs (2,350 psia), assuming a 10-percent step-load decrease from full power. Based on the range of NSSS design parameters for the Power Uprate, a margin-to-trip analysis was performed . The analysis concluded that the original design capacity of the spray sub-system (400 gpm or 200 gpm/valve) remains adequate for the 7.4-percent Power Uprate.

The pressurizer spray sub-system (valves and piping) was originally designed to pass the design spray valve flow rate (400 gpm or 200 gpm/valve) with an available pressure drop equal to the pressure drop from the spray flow scoop on each cold leg to the pressurizer surge line connection on the hot leg. The available pressure drop to achieve design spray flow must be based on minimum RCS loop flow, that is, thermal design flow (89,000 gpm) , and the range of RCS design temperatures for T_{cold}. The most significant parameter is thermal design flow and this parameter is not impacted by the Power Uprate. An overall hydraulic evaluation concluded

spray sub-system performance would be equal to, or greater than design for the full range of parameters approved for the Power Uprate.

4.1.4.1.4 Resistance Temperature Detector Bypass Loop Piping

The RCS fast-response temperature detectors that provide temperature signals to the reactor protection system are mounted in manifolds located in small bypass loops around the steam generator and the reactor coolant pump (RCP) of each loop.

The design of the reactor protection system requires that the fluid transport time from the reactor coolant loops (RCLs) to the last resistance temperature detector (RTD) in the RTD manifolds be less, than or equal to 0.5 second. This limits the bypass loop piping size and length, and bypass flows to particular values. The bypass loops are sized to pass sufficient flow rates to meet this fluid transport delay time based on the available pressure drop in the main coolant loops.

With a given bypass loop configuration, the flow through the hot-leg bypass loops is primarily a function of the pressure drop across the steam generators and the flow through the cold-leg bypass loops is primarily controlled by the operating head of the RCPs. The available pressure drop to achieve required cold and hot leg loop bypass flows must be based on RCS flow and the range of RCS design temperatures for T_{cold} and T_{hot}. The most significant parameter is RCS thermal design flow (89,000 gpm) and this parameter is not impacted by the Power Uprate. An overall hydraulic evaluation concluded that loop bypass flows will remain equal to or greater than values required to achieve the required fluid transport time for the full range of parameters approved for the Power Uprate.

4.1.4.1.5 Reactor Coolant System Setpoints (Excluding Channels Used by the Control and Protection Systems)

The pressurizer spray line low-temperature alarm (TIA-422, 423) setpoint is the only RCS setpoint that could be potentially impacted by the Power Uprate NSSS design parameters. The purpose of the alarm is to provide a warning to the operator if the miniflow through the spray lines is decreased. A fixed miniflow is required in the spray lines during normal operation to avoid undesirable temperature transients to the pressurizer spray nozzle and portions of the spray piping. Currently, with a cold-leg temperature lower limit of 525.0°F, the recommended spray line low-temperature alarm set point is 490°F (Reference 2). Since the cold-leg

temperature lower limit (521.9°F) is not significantly impacted by Power Uprate, there is no need to change the alarm setpoint.

4.1.4.2 Chemical and Volume Control System

The changes in NSSS design parameters that could potentially impact the CVCS design bases functions include the increase in core power and the allowable range for RCS full-load design temperatures. The increase in core power and the allowable range for RCS full-load design temperatures may affect the CVCS design bases requirements related to the core re-load boron requirements. Additionally, the allowable range for RCS full-load design temperatures may impact the heat loads that the CVCS heat exchangers must transfer to the CCWS, and in the case of the regenerative heat exchanger, to the charging flow.

System Heat Loads

The CVCS is designed to maintain RCS water inventory, boron concentration, and water chemistry. Other RCS support functions include purification and seal injection flow to the RCPs. During normal operation, the CVCS services the RCS by a letdown-and-charging process. RCS flow is letdown to the CVCS and delivered back to the RCS via charging pumps. This process requires that these feed-and-bleed streams be cooled and re-heated via heat exchangers. Since the Power Uprate alters RCS operating temperatures, the following CVCS heat exchangers were evaluated to assess the impact on the duty of these heat exchangers.

Regenerative Heat Exchanger

The regenerative heat exchanger is designed to recover heat from the letdown flow by reheating the charging flow to minimize RCS heat losses. Heating the charging line fluid before it enters the RCS is also required to minimize thermal transients on the RCS charging line nozzle. The design bases heat load was based on a maximum RCS T_{cold} temperature of 544.5°F. Since the maximum full load T_{cold} (539.2°F) for the Power Uprate is less than 544.5°F, the heat exchanger duty is less severe than the design duty.

Letdown Heat Exchanger

The letdown heat exchanger cools the letdown flow, leaving the regenerative heat exchanger at a temperature that is compatible with the purification demineralizer resins and the RCP seals.

The required temperature of the letdown flow (127.0°F) at the exit of the letdown heat exchanger is automatically controlled by a temperature controller and temperature control valve that regulates the flow of component cooling water that passes through the heat exchanger. The design bases heat load was based on a maximum inlet temperature of 290°F. Since the maximum inlet temperature for the Power Uprate is less than 290°F, the duty on this heat exchanger is less severe than the design duty.

Excess Letdown Heat Exchanger

The excess letdown heat exchanger can be employed either when normal letdown is temporarily out of service to maintain the plant in operation, or it can be used to supplement maximum letdown during the final stages of plant heatup. The design heat load was based on a maximum RCS T_{cold} temperature of 552°F. Since the maximum full-load T_{cold} (539.2°F) for the Power Uprate is less than design, the duty on this heat exchanger is less severe than the design duty.

Seal Water Heat Exchanger

The seal water heat exchanger is designed to cool the fluid from two sources:

- RCP seal water returning to the volume control tank (VCT)
- Reactor coolant discharged from the excess letdown heat exchanger

The first source of heat is not affected by the NSSS design parameters for the Power Uprate. The second heat source is reduced because the maximum RCS temperature (full-load Tcold 539.2°F) entering the excess letdown heat exchanger is less than the original design temperature (552°F). Therefore, the duty on the seal water heat exchanger for the Power Uprate is less severe than the design duty (Reference 3).

In summary, the heat duty on the CVCS heat exchangers is reduced from the design duty based on the NSSS design parameters approved for the Power Uprate.

4.1.4.2.2 Support of RCS (Core) Boron Requirements

The CVCS is designed to support the RCS (core) boron requirements. As part of the Reload Transition Safety Report (RTSR) (Reference 3), the CVCS functional and performance

requirements related to boric acid storage and delivery were evaluated for the Power Uprate. The evaluation concluded the CVCS boron-related capability is adequate for Power Uprate.

4.1.4.3 Residual Heat Removal System

A higher power level results in an increase in the amount of residual heat being generated in the core during normal cooldown, refueling operations and accident conditions. This will result in a higher heat load on the residual heat exchangers during the cooldown and also during the refueling outage. The increased heat loads will be transferred to the CCWS and ultimately to the Service Water System (SWS). Evaluation of the performance of the RHRS in conjunction with the CCWS and SWS with the increased heat loads is addressed in the following subsections.

4.1.4.3.1 Normal Plant Cooldown

The RHRS is designed to reduce the temperature of the RCS to 140°F within 20 hours after reactor shutdown when the service water temperature is 66°F. The residual heat load was based on a core power of 1,650 MWt. The RHRS is also capable of maintaining the RCS temperature at, or below, 140°F when the RCS is open for refueling or maintenance operations. The design basis cooldown was based on earlier plant design studies that concluded that 20 hours was an economically optimum cooldown time relative to the size and cost of the Residual Heat Removal (RHR) and component cooling water (CCW) heat exchangers.

The RHRS is designed to operate 4 hours after reactor shutdown when the RCS pressure and temperature are 400 psig and 350°F, respectively. The maximum heat load removed by the RHRS occurs at this time. This heat load is the sum of the residual heat load produced by the core, the heat load generated by the operation of one RCP, and the sensible heat that must be removed from the RCS. The initial phase of plant cool-down is accomplished by employing the steam generators and condenser steam dump.

Since the residual heat load during plant cooldown would increase by 7.4 percent based on the uprated power, a cooldown analysis was performed to assess the impact of this increased heat load on normal cooldown time. Normal cooldown is accomplished with two RHR heat exchangers and two CCW heat exchangers in service. Several cooldown scenarios were analyzed to assess the impact of Power Uprate on cooldown time. First, the increase in cooldown time was analyzed assuming design CCW flow of 1.25 Mlb/hr to each RHR heat

exchanger, design service water flow of 1.26 Mlbs./hr to each CCW heat exchanger, and the original design SW temperature of 66°F. Additional analysis was performed to determine the impact of Power Uprate on cooldown time with reduced CCW flow (0.747 Mlb/hr) to each RHR heat exchanger, reduced service water flow (1.13 Mlb/hr) to each CCW heat exchanger, and a maximum SW temperature of 80°F. These reduced CCW heat exchanger and RHR heat exchanger flows reflect current operating flows, and the increased SW temperature is currently reflected in the plant safety analysis (Reference 1). The impact of service water temperatures less than the current maximum of 80°F on cooldown time was also accessed with the current reduced flows to the RHR and CCW heat exchangers, since typically service temperature is expected to be significantly less than maximum allowable.

The results and conclusions of this normal plant cooldown analysis are as follows:

- For a core Power Uprate of 7.4-percent (from 1,650 MWt to 1,772 MWt), the Condenser Steam Dump System capacity is more than adequate to establish RHRS cut-in temperature and pressure within the current design basis cooldown time of 4 hours after reactor shutdown.

- Assuming design bases RHR and CCW heat exchanger flows, and the original design SWS temperature of 66°F, a core Power Uprate from 1,650 MWt to 1,772 MWt would increase the time for the RHR to cool the plant down from 350°F to 140°F from 13.9 hours to 17.5 hours, respectively. Therefore total cooldown time, that is, RHRS cooldown time plus 4 hours for steam dump, would increase from 17.9 hours to 21.5 hours, which exceeds the original design bases objective of 20 hours.

- At the current reduced RHR and CCW heat exchanger flows and the current maximum allowable SWS temperature of 80°F, a core Power Uprate from 1,650 MWt to 1772 MWt would increase the RHRS cooldown time from 57.2 hours to 76.2 hours respectively. Therefore, total cooldown time, that is, RHR cooldown time plus 4 hours for steam dump, would increase from 61.2 hours to 80.2 hours. However, at service water temperatures less than 80°F, coupled with current reduced RHR and CCW heat exchanger flows, total cooldown time would be less than 80.2 hours. For example, at uprated power and a service water temperature of 66°F (original design), coupled with current reduced RHR and CCW heat exchanger flows, total cooldown time would be 37.5 hours.

- These longer cooldown times are acceptable, considering the design bases for the 20 hours was simply economic, that is, minimizing the time required for cooldown versus the size and cost of the RHR and CCW components. Any impact that longer plant cooldown times may have on plant economics would be more than compensated for by the economic benefits of the Power Uprate.

- Therefore, the RHR, in conjunction with the CCW and SW, is adequately sized for normal cooldown heat loads associated with the Power Uprate.

4.1.4.3.2 10CFRPart 50, Appendix R, Fire Protection Report

In accordance with the KNPP Appendix R Fire Protection Report, the RHRS must also be capable of achieving RCS cold shutdown (200°F) in less than 72 hours after reactor shutdown, assuming the following:

- Loss-of-offsite power

- Single train of RHR equipment, that is, one RHR pump and one RHR heat exchanger

- Single train of CCWS quipment, that is, one CCW heat exchanger and one CCW pump

- CCW flow (0.747 Mlb/hr) to RHR heat exchanger, and service water flow (1.13 Mlb/hr) to CCW heat exchanger reflect current operating flows

- Maximum SW temperature (80°F) permitted by current safety analysis of record

- Initiation of RHRS operation at 29 hours after reactor shutdown, when a single train of RHR and CCW equipment can match core residual heat

The Appendix R cooldown analysis concluded that a single train of RHR and CCW equipment at uprated power can reduce the RCS temperature from 350°F to 200°F in 40.2 hours, assuming the RHRS is placed in service 29 hours after reactor shutdown. Therefore, total cooldown time is 69.2 hours. This cooldown time is in compliance with the Appendix R cooldown time limit of 72 hours after reactor shutdown. Therefore, the RHR, in conjunction with the CCW and SW, is adequately sized to meet the Appendix R regulatory requirements.

4.1.4.4 Safety Injection System/Containment Spray System

The required volume, duration, and heat rejection capability of the SI and CS flows in the event of a break are determined based on analytical and empirical models that simulate reactor and containment conditions subsequent to the postulated RCS and Main Steam System breaks. As a result of these analyses, the system and component criteria necessary to demonstrate compliance with regulatory requirements at the uprated power level are established. Since the results of these analyses have demonstrated that SIS/CSS provides adequate safety margin, the as-built SI and CS systems are acceptable for the Power Uprate.

4.1.4.5 Sampling System

The change in NSSS design parameters that potentially impact the Sampling System (SS) design bases is the allowable range for average RCS design temperature (T_{avg}). The change in RCS loop operating temperatures may affect the SS design bases requirement related to the maximum heat load that the SS heat exchangers must transfer to the CCW.

4.1.4.5.1 System Heat Loads

The SS provides fluid samples from the RCS (pressurizer and hot leg) for laboratory analysis. The sample flows from the RCS are cooled (pressurizer steam samples condensed and cooled) via heat exchangers. Since the Power Uprate alters RCS loop operating temperatures, the SS heat exchangers were evaluated to assess the impact on the design duty of these heat exchangers.

4.1.4.5.2 Sampling System Heat Exchangers

There are three sample heat exchangers: one for the pressurizer steam sample, one for the pressurizer liquid sample, and one for the RCL hot leg. The design bases heat load, that is, the maximum heat load for sizing these heat exchangers, is based on condensing and cooling pressurizer saturated steam (652.7°F) down to 127°F. Since nominal pressurizer saturated steam temperature (652.7°F) is not impacted by the Power Uprate, the design duty assumed for the SS heat exchangers is not impacted by the Power Uprate.

4.1.4.6 Component Cooling Water System

The CCW is an intermediate system between the RCS and the SW. It ensures that leakage of radioactivity from the components being cooled is contained within the plant. Revised heat rejection rates and /or cooling water flow requirements were assessed due to the Power Uprate.

4.1.4.6.1 Heat Loads

The CCW is designed to:

- Remove residual and sensible heat from the RCS via the RHR during plant cooldown
- Cool the letdown flow to the CVCS during power operation
- Provide cooling to dissipate waste heat from various plant components
- Provide cooling to safeguards loads after an accident

The following primary and auxiliary equipment impose heat loads on the CCWS:

- Residual heat exchangers
- RCPs
- Letdown heat exchanger
- Excess letdown heat exchangers
- Seal water heat exchanger
- Boric acid evaporator
- Evaporator distillate cooler
- SS heat exchangers
- Waste gas compressors
- RHR pumps
- Safety injection pumps
- Containment spray pumps

The total CCW heat load is variable depending on the plant operational mode and the equipment in service.

4.1.4.6.2 Plant Cooldown

The largest heat load on the CCW occurs when the RHR is placed in service during a normal plant cooldown. The increase in this transient heat load due to the higher residual heat load at the uprated power level was evaluated in conjunction with RHR cooldown capability in subsection 4.1.4.3.1. The results of this evaluation concluded that the RHR, in conjunction with the CCW and SW, is adequately sized for normal cooldown heat loads associated with the Power Uprate. Also, single-train cooldown based on 10CFR50, Appendix R fire protection requirements was evaluated (refer to subsection 4.1.4.3.2), and RHRS, in conjunction with the CCWS and SWS, was found to be acceptable in terms of regulatory requirements.

4.1.4.6.3 Plant Heatup/Power Operation/Refueling

The heat loads imposed on the CCW during plant heatup, power operation, and refueling are less limiting with respect to sizing the CCW heat exchangers. As noted previously in the CVCS and the SS evaluations, the majority of the heat loads imposed on the CCWS by these systems will either remain the same, or decrease. Also, during refueling, the heat load imposed on the CCW by the residual heat exchanger(s) will increase due to the proportional increase in residual heat at the uprated power level. The RCP thermal barrier heat loads will decrease due to lower RCS cold-leg operating temperatures.

The changes in heat loads imposed on the CCW during normal modes of plant operation are well within the system design capability.

4.1.4.6.4 Recirculation Phase of Safety Injection

In addition to normal cooldown, the RHR heat exchangers, in conjunction with the CCW heat exchangers, are also used to augment containment cooling during the SI recirculation phase and terminate reactor coolant boiling in the longer term. During the SI recirculation phase, the heat removal capability of the RHR heat exchangers is dependent upon CCWS supply temperature and sump water temperature, and will decrease with time as residual heat generation decreases. The maximum heat load imposed on the CCW is based on maximum sump temperature and maximum CCWS supply temperature during the SI recirculation phase.

The containment integrity analysis defines the maximum recirculation heat load, and this heat load is acceptable from a CCW design standpoint as long as the CCW supply temperature does not exceed the maximum recirculation limit of 130°F. The current containment integrity analysis (which includes a 2% power uncertainty) confirms that the CCW supply temperature will not exceed the maximum recirculation limit.

4.1.5 Conclusions

A summary of the conclusions of the evaluations of the NSSS fluid systems for the KNPP 7.4 percent power uprate is provided below.

4.1.5.1 Reactor Coolant System

The hydraulic design of the pressurizer surge line, safety valve inlet and outlet piping, and the design capacity of the PRT and existing level setpoints are adequate for the Power Uprate.

The hydraulic design of the pressurizer PORV inlet and discharge piping is adequate for the Power Uprate.

The hydraulic design of the pressurizer spray sub-system (piping and valves) is adequate to achieve design spray capacity at Power Uprate NSSS design parameters.

An overall hydraulic evaluation of the RTD bypass loop piping concluded that that loop bypass flows would be equal to, or greater than the values required to achieve the required fluid transport time for the full range of parameters approved for the Power Uprate.

The recommended spray line low-temperature alarm is not impacted based on the range of NSSS design parameters approved for Power Uprate.

4.1.5.2 Chemical and Volume Control System

The heat loads imposed on the CVCS heat exchangers due to the Power Uprate are less than the design basis heat loads.

As part of the RTSR (Reference 3), the CVCS functional and performance requirements related to boric acid storage and delivery were evaluated for Power Uprate. The evaluation concluded the following:

- CVCS boron-related capability is adequate for Power Uprate.

4.1.5.3 Residual Heat Removal System

The RHRS, in conjunction with the CCWS and SWS, is adequately sized for normal cooldown heat loads associated with the Power Uprate. The increase in plant cooldown time is considered acceptable based upon the economic benefits of the Power Uprate.

The Appendix R Fire Protection design basis cooldown requirements can be met at the uprated power level (see subsection 4.1.4.3.2).

4.1.5.4 Safety Injection System/Containment Spray System

The as-built SI and CS are acceptable for the Power Uprate.

4.1.5.5 Sampling System

The heat loads imposed on the SS heat exchangers due to the Power Uprate are less than or equal to the design basis heat loads.

4.1.5.6 Component Cooling Water System

The CCW, in conjunction with the RHR and SW, is adequately sized for normal cooldown heat loads associated with the Power Uprate. The increase in plant cooldown time is considered acceptable based upon the economic benefits of Power Uprate. Also, the CCW in conjunction with the RHR and SW is adequately sized to meet the regulatory requirements of Appendix R with respect to cooldown (see subsection 4.1.4.3.2).

The small changes in heat loads that are predicted to occur during normal modes of plant operation are well within the system design capability.

The containment integrity analysis defines the maximum recirculation heat load, and this heat load is acceptable from a CCWS design standpoint as long as the CCW supply temperature does not exceed the maximum recirculation limit of 130°F. The current containment integrity analysis (including 2% power uncertainty) indicate that the CCWS supply temperature will not exceed the maximum recirculation limit.

4.1.6 References

1. *Updated Safety Analysis Report, Kewaunee Nuclear Power Plant*, Rev. 17, FSAR Update, June 2002.

2. *Kewaunee RSG – Final Licensing Report Submittal*, November 29, 2000.

3. *Kewaunee Nuclear Power Plant, RTSR Program*, Final RTSR Report, July 19, 2002.

4.2 NSSS/Balance-of-Plant Interface Systems

4.2.1 Introduction

As part of the 7.4-percent power uprate, the following balance-of-plant (BOP) fluid systems were reviewed to assess compliance with Westinghouse Nuclear Steam Supply System (NSSS)/BOP interface requirements:

- Main Steam System (MS)
- Condensate and Feedwater System (C&F)
- Auxiliary Feedwater System (AF)
- Steam Generator Blowdown System (SGB)

The review was performed based on the range of NSSS design parameters (shown in Table IV.B-1 of Attachment 2) developed to support a NSSS power level of 1,780 MWt. The various interface systems were reviewed for the purpose of providing interface information that could be used in the more detailed BOP analyses.

4.2.2 Key Input Assumptions

The uprated range of NSSS design parameters were compared with the current design parameters previously evaluated for systems and components for the RSG Program. The comparison indicated significant differences that could impact the performance of the above BOP systems. For example, an increase in NSSS power of 7.4 percent (from 1,657.1 MWt to 1,780 MWt) and the approved lower limit on the average temperature (T_{avg}) (556.3°F) would result in approximately an 8.9-percent increase in steam/feedwater mass flow rates. Additionally, a steam generator tube plugging (SGTP) margin of 10 percent in combination with a T_{avg} in the upper end of the approved T_{avg} operating range (556.3°F to 573.0°F) would result in a reduction of full-load steam pressure from 791 psia to 747 psia.

4.2.3 Acceptance Criteria

The NSSS/BOP system interface requirements are delineated in the *Westinghouse Steam Systems Design Manual* (Reference 1).

4.2.4 Description of Analyses and Results

Evaluations of the above BOP systems relative to compliance with Westinghouse NSSS/BOP interface guidelines were performed to address the approved design parameters for the Power Uprate analyses that include ranges for parameters such as T_{avg} (556.3°F to 573.0°F) and SGTP (0 to 10 percent average). These ranges of NSSS design parameters result in ranges on BOP parameters such as steam generator outlet steam pressure (634 psia to 809 psia). The NSSS/BOP interface evaluations were performed to address the impact of these ranges on NSSS and BOP parameters. The results of the NSSS/BOP interface evaluations are delineated below.

4.2.4.1 Main Steam System

The 7.4-percent Power Uprate, coupled with the potential reduction in full-load steam pressure to the design value of 747 psia, significantly impacts main steam-line pressure drop. At the design steam generator pressure of 747 psia, the full-load steam mass flow rate would increase about 8.7 percent; however, due to the reduced operating pressure and the lower-density steam, the volumetric flow rate would increase by approximately 15.5 percent, and steam-line pressure drop would increase by approximately 25.5 percent. Note that main steam-line pressure drop impacts plant economics, since an increase in pressure drop results in a corresponding increase in plant heat rate over the life of the plant. (An increase in steam-line pressure drop of 1 psi is equivalent to an increase of approximately 2 Btu/KW-hr in plant heat rate.) Initial plant design studies indicated that a pressure drop in the range of 25 to 40 psi at rated load provided an acceptable economic balance between the value of a lower heat rate over the life of the plant, and the capital cost of larger-bore, longer-length pipes (Reference 1).

The NSSS design parameters (Attachment 2, Table IV.B-1) for the current NSSS power (1,657 MWt) resulted in a maximum steam-line pressure drop of about 35.4 psi and a minimum pressure of 608.6 psia at the turbine inlet valves. Based on the NSSS design parameters approved for the Power Uprating to an NSSS power of 1,780 MWt, the lowest design steam generator pressure of 634 psia would result in a steam-line pressure drop of 42.5 psi, and a pressure at the turbine inlet valves of approximately 591.5 psia.

Operating the plant at the highest achievable full-load steam pressure can minimize the impact of main steam-line pressure drop on plant heat rate.

The following summarizes the Westinghouse evaluation of the major steam system components relative to the NSSS design parameters approved for the Power Uprate. The major components of the MSS are the steam generator steam safety valves, the steam generator power-operated atmospheric relief valves (ARVs), the main steam isolation valves (MSIVs) and non-return check valves, the condenser dump valves (CDVs) and atmospheric steam dump valves (ASDVs).

4.2.4.1.1 Steam Generator Safety Valves

The setpoints of the steam generator safety valves are determined based on the design pressure of the steam generators (1,085 psig) and the requirements of the ASME Boiler and Pressure Vessel (B&PV) Code. Since the design pressure of the steam generators has not changed, there is no need to revise the setpoints of the safety valves.

The steam generator safety valves must have sufficient capacity to ensure that main steam pressure does not exceed 110 percent of the steam generator shell-side design pressure (the maximum pressure allowed by the ASME B&PV code) for the worst-case loss-of-heat-sink event ("Loss-of-Load Event"). Based on this requirement, Westinghouse applies the conservative criterion that the valves should be sized to relieve 100 percent of the maximum calculated steam flow at an accumulation pressure not exceeding 110 percent of the MSS design pressure (Reference 1). Additionally, the capacity of any single safety valve is presently limited to 890,000 lbs./hr at 1,100 psia based on the present steam line break analysis of record for a stuck-open steam generator safety valve (Reference 2).

The KNPP has ten safety valves with a total capacity of 7.660×10^6 lbs./hr, which provide about 107.3 percent of the current maximum design full-load steam flow of 7.14×10^6 lbs./hr (Reference2). Based on the proposed range of NSSS design parameters approved for the Power Uprate, the installed safety valves provide about 98.6 percent of the maximum design steam flow of 7.77×10^6 lbs./hr.

The plant safety analysis for the Power Uprate confirms that the installed safety valve capacity of 7.660×10^6 lbs./hr is adequate for overpressure protection.

4.2.4.1.2 Steam Generator Power-Operated Atmospheric Relief Valves (ARVs)

The steam generator ARVs, which are located upstream of the MSIVs and non-return check valves and adjacent to the steam generator safety valves, are automatically controlled by steam-line pressure during plant operations. The steam generator ARVs automatically modulate open and exhaust to atmosphere whenever the steam-line pressure exceeds a predetermined setpoint to minimize safety valve lifting during steam pressure transients. As the steam-line pressure decreases, the steam generator ARVs modulate closed and reseat at a pressure at least 10 psi below the opening pressure. The steam generator ARV set pressure for these operations is between zero-load steam pressure and the setpoint of the lowest-set steam generator safety valve. Since neither of these pressures change for the proposed range of NSSS design parameters, there is no need to change the ARV setpoint.

The steam generator ARVs also provide a means for decay heat removal and plant cool down by discharging steam to the atmosphere when either the condenser, the condenser circulating pumps, or steam dump to the condenser is not available. Under such circumstances, the ARVs, in conjunction with the AFS permit the plant to be cooled down from the pressure setpoint of the lowest-set steam generator safety valve to the point where the Residual Heat Removal System (RHRS) can be placed in service. During cool down, the ARVs are automatically controlled by steam-line pressure with remote manual adjustment of the pressure setpoint from the Control Room.

In the event of a steam generator tube rupture (SGTR) in conjunction with loss of offsite power, the ARVs are used to cool the RCS to a temperature that permits equalization of the primary and secondary pressures at a pressure below the lowest-set steam generator safety valve within 30 minutes. RCS cooldown and depressurization are required to preclude steam generator overfill and to terminate activity release to the atmosphere.

The steam generator ARVs are sized to have a capacity equal to about 10 percent of rated steam flow at no-load pressure (Reference 2). This capacity allows a plant cool down to RHR operating conditions (350°F) in 4 hours (at an average rate of about 50°F/hr) assuming cooldown starts 2 hours after reactor shutdown. Plant operating procedures specify the cooldown rates for the applicable event and plant conditions. This sizing is compatible with normal cooldown capability and minimizes the water supply required by the AFS. This design

basis is limiting with respect to sizing the ARVs and bounds the capacity required for tube rupture.

An evaluation of the installed capacity (745,000 lbs./hr at 1,050 psig) indicates that the original design basis in terms of cool down capability can still be achieved over the full range of NSSS design parameters approved for the Power Uprate.

4.2.4.1.3 Main Steam Isolation Valves and Non-Return Check Valves

The MSIVs in conjunction with non-return check valves are located outside the containment and downstream of the steam generator safety valves and ASDVs. The valves prevent the uncontrolled blowdown of more than one steam generator and minimize the RCS cool down and containment pressure to within acceptable limits following a main steam line break. To accomplish this, the MSIVs must be capable of closure within 5 seconds of receiving a closure signal against steam-line break-flow conditions in the forward direction.

Rapid closure of the MSIVs and non-return check valves following postulated steam line breaks causes a significant differential pressure across the valve seats and a thrust load on the MSS piping and piping supports in the area of the MSIVs and non-return check valves. The worst cases for pressure increase and thrust loads are controlled by the steam line break area (that is, mass flow rate and moisture content), throat area of the steam generator flow restrictors, valve seat bore, and no-load operating pressure. Since these variables and no-load pressure are not impacted by the Power Uprate, the design loads and associated stresses resulting from rapid closure of the MSIVs and non-return check valves will not change. Consequently, the Power Uprate has no significant impact on the interface requirements for the MSIVs or non-return check valves.

4.2.4.2 Steam Dump System

The NSSS RCS and the associated equipment (pumps, valves, heaters, control rods, etc.) were designed to provide satisfactory operation (automatic in the range of 15 to 100 percent power) without reactor trip when subjected to the following load transients:

- Loading at 5 percent of full power per minute with automatic reactor control.
- Unloading at 5 percent of full power per minute with automatic reactor control.

- Instantaneous load transients of plus or minus 10 percent of full power (not exceeding full power) with automatic reactor control.

- Load reductions of 100 percent of full power with automatic reactor control and steam dump.

The Steam Dump System creates an artificial steam load by dumping steam from ahead of the turbine valves to the condenser and the atmosphere. The Westinghouse original sizing criterion recommended that the steam dump system be capable of discharging 85 percent of the rated steam flow at full-load steam pressure to permit the NSSS to withstand an external load reduction of up to 100 percent of plant rated electrical load without a reactor trip (Reference 1). To prevent a trip, all NSSS control systems must be automatic. This includes the RCS, which accommodates 10 percent of the load rejection.

For the power uprate, the large load rejection (LLR) capability was demonstrated to be 50 percent of plant-rated power without reactor trip. The Westinghouse sizing criterion recommends that the Steam Dump System be capable of discharging 40 percent of the rated steam flow at full-load steam pressure to accommodate external load reductions up to 50 percent of plant-rated electrical load. Note that a steam-dump capacity of 40 percent of rated steam flow at full-load steam pressure also prevents steam generator safety valve lifting following reactor trip from full power.

4.2.4.2.1 Condenser/Atmospheric Dump Valves

The KNPP has six condenser and six ASDVs. The valves were sized to provide a total steam flow equal to 85 percent (40 percent to the condenser, and 45 percent to the atmosphere) of the original maximum calculated steam flow (7.5×10^6 lb/hr) at a full-load steam pressure of 750 psia. The valves are grouped into four banks, the first two banks discharging to the condenser, and the last two banks discharging to the atmosphere. At the current minimum full-load steam generator operating pressure, total steam dump is 5.48×10^6 lb/hr, or about 77 percent of rated steam flow (7.11×10^6 lb/hr). This reduced capacity relative to the original sizing criteria was verified to be adequate by a margin-to-trip analysis for load reductions up to 50 percent of rated electrical load. As noted above, the large load rejection (LLR) capability for the power uprate is 50 percent of rated-electrical power without reactor trip. NSSS operation within the approved range of design parameters for Power Uprate at lower steam generator pressures and higher

steam flows will result in a reduced steam dump capability relative to the current minimum capability. An evaluation indicates that the condenser steam dump capacity could be as low as 29.4 percent of rated steam flow (7.73×10^6 lbs./hr), or 2.272×10^6 lbs./hr, at a full-load steam pressure of 634 psia. These operating conditions are based on an assumed SGTP level of 10 percent and a T_{avg} in the lower end of the proposed operating range. If steam dump to the condenser is not adequate, credit can be taken for the actuation of the first bank of the ASDVs to supplement condenser dump, which would provide at least 40 percent steam dump. Therefore, total Steam Dump System capacity is adequate for load reductions up to 50 percent of electrical load.

The control systems margin-to-trip analysis concluded that condenser steam dump is adequate (that is, the ASDVs are not required) to preclude reactor trip for load reductions up to 50 percent of rated-electrical load.

To provide effective control of flow on large step-load reductions or plant trip, the steam dump valves are required to go from full closed to full open in 3 seconds at any pressure between 50 psi less than full-load pressure and steam generator design pressure. The dump valves are also required to modulate to control flow. Positioning response may be slower with a maximum full stroke time of 20 seconds. These requirements are not impacted by the Power Uprate.

4.2.4.3 Condensate and Feedwater System

The Condensate and Feedwater System (C&FS) must automatically maintain steam generator water levels during steady state and transient operations. The range of NSSS design parameters for the Power Uprate will result in a required feedwater volumetric flow increase of up to 9.8 percent during full-power operation. The higher feedwater flow and temperatures will have an impact on system pressure drop, which may increase by as much as 19.5 percent. Also, a comparison of the range of NSSS design parameters for the Power Uprate with the non-uprated design parameters indicates that the steam generator full-power operating steam pressure may be increased by as much as 44 psi (771 psia to 815 psia).

The major components of the C&FS potentially impacted by the Power Uprate are the feedwater isolation valves (FIVs), the feedwater control valves (FCVs), feedwater bypass control valves (FBCVs), and the C&FS pumps.

4.2.4.3.1 Feedwater Isolation Valves

The FIVs are located outside containment and downstream of the FCVs. The valves function in conjunction with the primary isolation signals to the FCVs and backup trip signals to the feedwater pumps. This provides redundant isolation of feedwater flow to the steam generators following a steam line break, or a malfunction in the Steam Generator Level Control System. Isolation of feedwater flow is required to prevent containment overpressurization and excessive RCS cooldowns. To accomplish this function, the FCVs and the backup FIVs must be capable of closure, following the receipt of any feedwater isolation signal (Reference 1).

The closure requirements imposed on the FCVs and the backup FIVs cause dynamic pressure changes that may be of a large magnitude and must be considered in the design of the valves and associated piping. The worst loads occur following a steam line break from no-load conditions with the conservative assumption that both feedwater pumps are in service providing maximum flow following the break. Since these conservative assumptions are not impacted by the Power Uprate, the design loads and associated stresses resulting from closure of these valves will not change.

4.2.4.3.2 Feedwater Control Valves, Feedwater Bypass Control Valves, Condensate and Feedwater System Pumps

The C&FS available head in conjunction with the FCV characteristics must provide sufficient margin for feed control to ensure adequate flow to the steam generators during steady-state and transient operation. A continuous steady feed flow should be maintained at all loads. To ensure stable feedwater control with constant speed feedwater pumps, the pressure drop across the FCVs at rated flow (100-percent power) should be approximately equal to 1.5 times the Feedwater System dynamic losses from the feed pump discharge through the steam generators. In addition, adequate margin should be available in the FCVs at full-load conditions to permit a C&FS delivery of 96 percent of rated flow with a 100-psi pressure increase above the full-load pressure with the FCVs fully open (Reference 1). This margin is required for load rejection. The FBCVs, provided for low-load operation of the Feedwater System, are required to provide enough capacity to enable the plant to obtain 15-percent thermal power.

In light of these NSSS interface requirements, the C&FS piping, pumps, valves, and pressure-retaining components were evaluated to assess their ability to operate at the increased flow

rates, temperatures, and pressures associated with the Power Uprate. To provide effective control of flow, the FCVs and FBCVs are required to stroke open or closed within 20 seconds over the anticipated inlet pressure control range (approximately 0 to 1,600 psig). This requirement is not impacted by the Power Uprate.

4.2.4.4 Auxiliary Feedwater System

The AFS supplies feedwater to the secondary side of the steam generators when the normal Feedwater System is not available, thereby maintaining the heat sink of the steam generators. The system provides feedwater to the steam generators during normal-unit startup, hot-standby, and cool-down operations, and also functions as an Engineered Safeguards System. In the latter function, the AFS is directly relied upon to prevent core damage and system over-pressurization in the event of transients and accidents such as a loss of normal feedwater or a secondary system pipe break.

The minimum flow requirements of the AFS are dictated by safety analyses, and the results of the revised safety analyses for the Power Uprate confirmed that the current AFS performance is acceptable.

4.2.4.4.1 Auxiliary Feedwater Storage Requirements

The AFS pumps are normally aligned to take suction from the condensate storage tanks (CSTs). To fulfill the engineered safety features (ESF) design functions, sufficient feedwater must be available during transient or accident conditions to enable the plant to be placed in a safe shutdown condition.

The limiting transient with respect to CST inventory requirements is the loss-of-all-AC-power transient. Since the Kewaunee plant is required to have a 4-hour coping period, the plant licensing basis requires that sufficient CST useable inventory must be available to remove decay heat to maintain the plant in hot shutdown for 4 hours. In light of this licensing basis, the plant Technical Specifications require a minimum useable inventory of 39,000 gallons during power operation (Reference 3).

Since the minimum CST useable inventory is impacted by the Power Uprate (due to increased decay heat) and the approved range of NSSS operating conditions, a new analysis for the 7.4% uprate was performed as part of the BOP evaluations to determine the required inventory for the

Power Uprate. This new analysis for the loss-of-all-AC-power scenario is based on the following conservative assumptions:

- Reactor trip occurs from 100.6 percent of rated core power (1,772 MWt), from a low-low water level in the steam generators. A 2-second delay is assumed before reactor trip following loss of offsite power.

- Steam is released from the steam generators at the first safety valve setpoint plus setting tolerance and accumulation.

- The CST operating fluid temperature is at the maximum value (that is, 120°F for Power Uprate analyses).

The results of the analysis indicate that a minimum useable inventory should be increased from 39,000 gallons to 41,500 gallons to meet the loss-of-AC-power licensing basis for the range of NSSS operating conditions upon approval of the 7.4% power uprate). The current capacity of 39,000 gallons is acceptable for the 1.4 % MUR power uprate.

4.2.4.5 Steam Generator Blowdown System

The SGBS controls the chemical composition of the steam generator secondary-side water within the specified limits. The SGBS also controls the buildup of solids in the steam generator secondary.

The blowdown flow rates required during plant operation are based on chemistry control and tube-sheet sweep requirements to control the buildup of solids. The blowdown flow rate required to control chemistry and the buildup of solids in the steam generators is tied to allowable condenser in-leakage, total dissolved solids in the plant circulating water system, and allowable primary to secondary leakage. Since these variables are not impacted by the Power Uprate, the blowdown required to control secondary chemistry and steam generator solids will not be impacted by the Power Uprate.

The inlet pressure to the SGBS varies with steam generator operating pressure. Therefore, as steam generator full-load operating pressure decreases, the inlet pressure to the SGBS control valves decreases and the valves must open to maintain the required blowdown flow rate into the system. The present range of NSSS design parameters permits a maximum decrease in steam

pressure from no-load to full-load of 376 psi (that is, from 1,020 psia to 644 psia). Based on the revised range of NSSS design parameters approved for the Power Uprate, the no-load steam pressure (1,020 psia) remains the same, and the minimum full-load steam pressure decreases 10 psi (to 634 psia). This small decrease is not considered to be significant with respect to blowdown flow control. Therefore, the range of design parameters approved for the Power Uprate will not impact blowdown flow control.

4.2.5 Conclusions

A summary of the conclusions of the evaluation of the NSSS/BOP system interfaces for the KNPP 7.4-percent NSSS Power Uprate Program is provided in the following subsections.

4.2.5.1 Main Steam System

The required safety valve capacity is dictated by safety analysis. The results of the safety analyses for the Power Uprate concluded that the installed safety valve capacity is adequate. An evaluation of the capacity of the ASDVs concluded that the original design basis in terms of cool-down capability can still be achieved over the full range of NSSS design parameters approved for the Power Uprate. This cooldown design basis, with respect to sizing the ASDVs, is bounding in regard to the capacity required for tube rupture.

The NSSS/BOP interface systems requirements imposed on the design MSIVs, non-return check valves, and associated pipe loads are not impacted by the Power Uprate.

4.2.5.2 Steam Dump System

The KNPP Steam Dump System was originally sized to accommodate a steam flow equal to about 85 percent of the maximum calculated full-power steam flow (40-percent condenser dump and 45-percent atmospheric dump) to permit external load reductions up to 100 percent of the rated-electrical load. For the Power Uprate, the maximum required load reduction is being relaxed from 100 percent to 50 percent of the plant-rated power. This reduced load reduction capability results in a corresponding reduction in the maximum required steam-dump capacity from 85 percent to 40 percent of the rated-steam flow. In light of this change, the steam dump capacity for the range of NSSS design parameters approved for the Power Uprate 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.

4.2.5.3 Condensate and Feedwater System

To ensure stable feedwater control, with constant speed feedwater pumps, the pressure drop across the FCVs at rated flow (100-percent power) should be approximately equal to 1.5 times the Feedwater System dynamic losses from the feed-pump discharge through the steam generators. In addition, adequate margin should be available in the FCVs at full-load conditions to permit a C&FS delivery of 96 percent of the rated flow, with a 100-psi pressure increase above the full-load pressure with the FCVs fully open. This margin is required for load rejection. The FBCVs, provided for low-load operation of the Feedwater System, are required to provide enough capacity to enable the plant to obtain 15-percent thermal power.

In light of these NSSS interface requirements, the C&FS piping, pumps, valves and pressure-retaining components were evaluated to ensure their ability to operate at the increased flow rates, temperatures, and pressures associated with the Power Uprate. The FCVs and FBCVs stroke time requirement (20 seconds) is not impacted by the Power Uprate.

4.2.5.4 Auxiliary Feedwater System

The minimum flow requirements of the AFS are dictated by safety analysis. The results of the revised safety analyses confirmed that the current AFS performance is acceptable for the Power Uprate. The AFS pumps are normally aligned to take suction from the CSTs. In the event of a loss of all AC power, the plant licensing basis dictates that sufficient CST useable inventory must be available to bring the plant from full-power to hot-standby conditions and maintain the plant at hot standby for 4 hours. In light of this licensing basis, the plant Technical Specifications require a minimum useable inventory of 39,000 gallons during power operation.

4.2.5.5 Steam Generator Blowdown System

The blowdown flow rate required to control the chemistry and the buildup of solids in the steam generators is tied to allowable condenser in-leakage, total dissolved solids in the plant circulating water system, and allowable primary to secondary leakage. Since these variables are not impacted by the Power Uprate, the blowdown required to control secondary chemistry and steam generator solids will not be impacted by the Power Uprate.

The inlet pressure to the SGBS varies with steam generator operating pressure. Since the present range of the approved NSSS design parameters permits a range of operating pressures

that almost bound the Power Uprate design parameters, the range of design parameters approved for the Power Uprate will not impact blowdown flow control.

4.2.6 References

1. WCAP-7451, *Westinghouse Steam Systems Design Manual*, February 1970.

2. *KNPP USAR*, Rev. 17, June 1, 2002.

3. *KNPP Technical Specifications*, Amendment 164, November 7, 2002.

5.1 Reactor Vessel

5.1.1 Structural Evaluation

5.1.1.1 Introduction

Evaluations were performed for the various regions of the Kewaunee reactor vessel to determine the stress and fatigue usage effects of Nuclear Steam Supply System (NSSS) operation at the revised operating conditions of the Kewaunee 7.4-percent power uprate throughout the current plant operating license. The evaluations concluded that the previous structural evaluations performed for the Kewaunee Replacement Steam Generator (RSG) Program continue to bound the effects of operation at the power uprate operating conditions.

5.1.1.2 Input Parameters and Assumptions

The evaluations assess the effects of the design transients on the most limiting locations with regard to ranges of stress intensity and fatigue usage factors. The evaluations consider a worst-case set of design transients from among the high-temperature power uprate conditions, the low-temperature power-uprate conditions, and the original design basis. The reactor vessel normal-operating parameters for the power uprate are bounded by the reactor vessel operating parameters assumed for the RSG analysis. The design transients exhibited no changes that required assessment in addition to the RSG analysis. Furthermore, no new design interface loads were identified as a result of the power uprate.

5.1.1.3 Description of Analysis

Reactor vessel operation from plant startup through implementation of the Power Uprate and any future operation in accordance with the original design basis remain bounded by the stress and fatigue analyses. No additional revised maximum ranges of stress intensity and maximum usage factors were required for the power uprate. In all cases, the stress and fatigue evaluations performed for the previous RSG Program remain conservative so that no new calculational results were identified.

The evaluations for the RSG Program assessed the effects of the revised design transients and operating parameters on the most limiting locations with regard to ranges of primary plus secondary stress intensity and fatigue usage factors in each of the regions as identified in the

reactor vessel stress report and addendum. The evaluations considered a worst case set of operating parameters and design transients from among the high temperature RSG conditions, the low temperature RSG conditions and the original design basis. As a result, all of the RSG parameter cases are fully covered by the evaluations, and reactor vessel operation in accordance with the Kewaunee Replacement Steam Generator Project for the remainder of the current operating license was justified.

In addition, reactor vessel operation from plant startup until implementation of the RSG project and any future operation in accordance with the original design basis is still fully covered by the stress and fatigue analyses in the reactor vessel stress report. Where appropriate, revised maximum ranges of stress intensity and maximum usage factors were calculated for the RSG project. In other cases the original design basis stress analysis remains conservative so that no calculations were necessary, and the maximum ranges of stress intensity and fatigue usage factors reported in the Combustion Engineering, Inc. (CE) stress report and addendum continue to govern.

In addition to the revised operating parameters and design transients for the RSG program, a set of allowable loads was applied at the Kewaunee reactor vessel/reactor internals interfaces. The allowable loads were previously evaluated and justified for application to the Point Beach reactor vessels for their RSG project. Based upon the close similarity between the Point Beach reactor vessels and the Kewaunee reactor vessel, the stresses resulting from the allowable loads were applied to the Kewaunee reactor vessel/reactor internals interfaces at the reactor vessel main closure flanges, outlet nozzles internal projections, and core support pads (lugs). The stresses due to the interface loads were combined by superposition with the thermal and pressure stresses due to normal operation from the Kewaunee reactor vessel stress report.

The evaluation of the reactor vessel for the RSG project shows that it is acceptable for plant operation and these evaluations remain applicable in accordance with the 7.4% power uprate. Therefore, the reactor vessel Power Uprate evaluation addresses reactor operation with the operating temperature ranges and design transients that were calculated for power uprate. Such operation is shown to be acceptable in accordance with the *ASME Boiler and Pressure Vessel Code* (Reference 1) for the remainder of the plant license.

5.1.1.4 Acceptance Criteria

The maximum range of primary-plus-secondary stress intensity resulting from normal and upset condition design transient mechanical and thermal loads should not exceed 3 S_m at operating temperature (Reference 1, Paragraph N-414.4).

The maximum cumulative usage factor resulting from the peak stress intensities due to the normal and upset condition design transient mechanical and thermal loads should not exceed 1.0 in accordance with the procedure outlined in Paragraph N-415 and N-416 of the *ASME Boiler and Pressure Vessel Code* (Reference 1).

5.1.1.5 Results

The reactor vessel Power Uprate evaluation demonstrates that the Power Uprate does not increase the ranges of stress intensity or cumulative fatigue usage factors for any of the various regions of the reactor vessel that were previously evaluated for the Kewaunee RSG Project. The maximum ranges of primary-plus-secondary stress intensity remain as evaluated and justified for the RSG Project. Additionally, the maximum cumulative fatigue usage factors reported for the original design basis are otherwise unaffected by the Power Uprate conditions and remain significantly below the acceptance criterion of 1.00.

Results from the RSG project that remain applicable to the power uprate project evaluation are discussed in the following paragraphs and are shown in Table 5.1-1.

The RSG program affected several of the maximum ranges of primary plus secondary stress intensity reported in the Kewaunee reactor vessel stress report. The evaluations show that for the limiting locations, most of the maximum ranges are unchanged when the revised operating parameters, design transients and design interface loads are incorporated. The exceptions are the outlet nozzles, the closure studs, the CRDM housings, the inlet nozzles and the bottom head instrumentation tubes. The maximum cumulative fatigue usage factors at all of the limiting locations except the CRDM housings, the vent nozzle, the vessel wall transition and the bottom head instrumentation tubes increase somewhat from the values reported. Most of the usage factor increases are minimal, but some of them are relatively large. Most importantly, all of the cumulative fatigue usage factors remain under the 1.0 limit. The greatest increases in usage factor are 0.5592 in the vessel flange, 0.2217 in the safety injection nozzles, 0.193 in the external support brackets and 0.1092 in the closure studs. These large increases resulted from

the application of higher upper head temperatures and reactor vessel/reactor internals interface loads as well as the RSG design transients. The maximum calculated cumulative usage factor in the reactor vessel is 0.900 in the external support brackets. The external support bracket cumulative usage factor increased from 0.707 to this maximum value due of 0.900 due solely to the effect of the revised design transients.

The maximum range of stress intensity at the outlet nozzle safe end increased by 3.0 ksi to 37.4 ksi which is less than the ASME Section III $3S_m$ of 49.2 ksi. The maximum range of stress intensity for the low alloy steel nozzle increased by 2.44 ksi, but the revised value of 48.8 ksi remains less than the $3S_m$ limit of 80.1 ksi. The maximum range of stress intensity at the inlet nozzle safe end increased by 4.57 ksi to 35.38 ksi. This result remains below the $3S_m$ limit of 52.9 ksi. The maximum range of stress intensity for the low alloy steel portion of the inlet nozzle increased by 4.28 ksi to 43.09 ksi and remains less than the limit of 80.1 ksi. The greatest change is in the reported maximum stud service stress which increased by 13.7 ksi to 109.3 ksi, but remains less than the $3S_m$ allowable of 118.8 ksi. This increase in the maximum service stress was largely due to the consideration of a 10 percent contingency that was added to the stud preload.

The updated maximum ranges of primary plus secondary stress intensity and maximum cumulative fatigue usage factors for the Kewaunee reactor vessel accounting for the RSG program are provided in Table 5.1-1.

Allowable reactor vessel/reactor internals interface loads were incorporated using loads and stresses that were previously calculated for the Point Beach. The reported maximum ranges of primary plus secondary stress intensity for the CRDM housings and the bottom head instrumentation tubes are actually reduced. The reduction is due to the exclusion of the Steam Pipe Break transient from consideration in the primary plus secondary stress evaluation. Steam Pipe Break is a Faulted Condition and need not be included in the range of stress intensity evaluation. Exclusion of the Steam Pipe Break keeps the maximum ranges of stress intensity for the CRDM housings and instrumentation tubes below the $3S_m$ limit, and eliminates the need for the simplified elastic-plastic analyses that are in the CE stress report.

The updated maximum ranges of primary plus secondary stress intensity and maximum cumulative fatigue usage factors for the Kewaunee reactor vessel accounting for the RSG

program are the results provided in Table 5.1-1 (These results are unchanged by the 7.4% power uprate).

The RSG evaluation for the reactor vessel is documented in an addendum to the Kewaunee reactor vessel stress report (Reference 2). Based upon the satisfactory results of the evaluations in this addendum report as previously discussed, the Kewaunee reactor vessel is acceptable for plant operation in accordance with the Kewaunee RSG Project.

The effects of the 7.4% power uprate normal operating parameters and design transients are bounded by the corresponding effects considered in the reactor vessel evaluation for the Kewaunee RSG Project. Thus, considering any combination of the design basis, the power uprate, and the RSG NSSS design transients for the specified numbers of occurrences, the Kewaunee reactor vessel stress and fatigue analyses and evaluations justify operation with a range of vessel outlet temperature from 586.3°F up to 606.8°F and a range of vessel inlet temperature (T_{cold}) from 521.9°F up to 543.8°F. Therefore, the reactor vessel evaluation for the RSG project, in conjunction with the reactor vessel stress report, bounds the effects of reactor operation in accordance with the 7.4% uprate Such operation is acceptable in accordance with the 1968 Edition of Section III of the ASME Boiler and Pressure Vessel Code with Addenda through the Winter 1968 (Reference 1) for the remainder of the plant license.

5.1.1.6 Conclusions

The reactor vessel structural evaluation concludes that all acceptance criteria continue to be met for the 7.4% power uprate.

5.1.1.7 References

1. *ASME Boiler and Pressure Vessel Code*, Section III, Nuclear Power Plant Components, 1968 Edition with Addenda through the Winter 1968, American Society of Mechanical Engineers, New York.

2. WCAP-15345, "Addendum to the Analytical Report for the Kewaunee Nuclear Power Plant Reactor Vessel (Replacement Steam Generator Evaluation)," dated December 1999.

Table 5.1-1 Stress Intensities and FatigueUsage Factors for the Kewaunee Reactor Vessel as Evaluated at 1780 MWt		
Location	**Maximum Range of Primary plus Secondary Stress Intensity ($P_L + P_b + Q$)**	**Maximum Cumulative Fatigue Usage Factor (U_c)**
Outlet Nozzles	Nozzle: 48.8 ksi < 3 S_m = 80.1 ksi Safe End: 37.4 ksi < 3 S_m = 49.2 ksi	Nozzle: 0.126 < 1.0
Inlet Nozzles	Nozzle: 43.1 ksi < 3 S_m = 80.1 ksi Safe End: 35.4 ksi < 3 S_m = 52.9 ksi	Nozzle: 0.033 < 1.0
Main Closure Flange Region		
Closure Head Flange	70.0 ksi < 3 S_m = 80.1 ksi	0.012 < 1.0
Vessel Flange	67.4 ksi < 3 S_m = 80.1 ksi	0.576 < 1.0
Closure Studs	109.3 ksi < 3 S_m = 118.8 ksi	0.593 < 1.0
Vessel Shell		
Vessel Wall Transition	32.2 ksi < 3 S_m = 80.1 ksi	0.004 < 1.0
Bottom Head to Shell Juncture	28.6 ksi < 3 S_m = 80.1 ksi	0.004 < 1.0
Core Support Guides	57.5 ksi < 3 S_m = 69.9 ksi	0.204 < 1.0
CRDM Housings	45.7 ksi < 3 S_m = 69.9 ksi	0.293 < 1.0
Bottom Head Instrumentation Tubes	57.8 ksi < 3 S_m = 69.9 ksi	0.384 < 1.0
Safety Injection Nozzles	46.8 ksi < 3 S_m = 80.1 ksi	0.295 < 1.0
External Support Brackets	36.5 ksi < 3 S_m = 80.1 ksi	0.900 < 1.0

5.1.2 Reactor Vessel Integrity-Neutron Irradiation

5.1.2.1 Introduction

Reactor vessel integrity is affected by any changes in plant parameters that affect neutron fluence levels or temperature/pressure transients. Note that the temperature/pressure transients have been reviewed and do not change as a result of the Kewaunee Power Uprate Program. The balance of Section 5.1.2 addresses the potential impact of the neutron fluence increase, resulting from the Kewaunee Power Uprate Program, on reactor vessel integrity. The reactor vessel integrity evaluations for the Power Uprate Program included the following evaluations:

- Review of the reactor vessel surveillance capsule removal schedule to determine if changes are required as a result of changes in the vessel fluence due to the Power Uprate Program.

- Review of the existing pressure-temperature (P-T) limit curves to determine if a new applicability date needs to be calculated due to the effects of the uprated fluence projections.

- Review of the existing RT_{PTS} values to determine if the effects of the uprated fluence projections result in an increase in RT_{PTS} for the beltline materials in the Kewaunee reactor vessels at the end of license (EOL) (33 effective-full-power year [EFPY]) and end of life extension (EOLE) (51 EFPY).

- Review the upper shelf energy (USE) values at EOL for all reactor vessel beltline materials in the Kewaunee reactor vessels to assess the impact of the uprated fluence projections.

- Review the LTOP limitations to determine if a new applicability date needs to be calculated due to the effects of the uprated fluence projections.

5.1.2.2 Input Parameters and Assumptions

Uprated Fluence Projections

The calculated fluence projections on the vessel were evaluated for the uprated power level for input into the reactor vessel integrity evaluations. Typically, fluence values are used to evaluate end-of-life (EOL) transition temperature shift (EOL ΔRT_{NDT}) for development of surveillance capsule withdrawal schedules, determining EOL upper shelf energy (USE) values, adjusted reference temperature (ART) values for determining applicability of heatup and cooldown curves and LTOP limitations, ERG limits, and RT_{PTS} values. The calculated fluence projections used in the Power Uprate Program evaluation comply with Regulatory Guide 1.190 (Reference 1). The fluence projections for the KNPP Power Uprate Program evaluation are generated using the methodologies of Regulatory Guide 1.190 (Reference 1).

Inlet Temperature

The reactor vessel inlet temperature at uprated power is evaluated to verify its compliance with Regulatory Guide 1.99, Revision 2 (Reference 10).

Low Temperature Overpressure Protection System

The LTOPS (Low Temperature Overpressure Protection System) is intended to provided protection against violations of the reactor vessel Appendix G pressure vs. temperature limits during hot or cold shutdown operation at relatively low RCS temperatures (generally below about 350 Deg. F). This protection is provided via a spring loaded relief valve with a pressure setpoint to avoid violation of these Appendix G limits for the design basis transients.

The existing LTOPS setpoint is evaluated based on power uprate for one or more of the following reasons:

- A change in the NSSS resulting in a change in the RCS volume
- A change in the steam generators resulting in a change in the heat transfer coefficient during shutdown conditions
- A change in the design basis mass input or heat input transients

- A change in the spring loaded relief valve operating characteristics (spring constant, flow capability, etc.)

- A change in the reactor vessel Appendix G pressure vs. temperature limits used as an upper pressure limit that the LTOPS must protect against.

5.1.2.3 Description of Analysis/Evaluations

The reactor vessel integrity evaluation for the Kewaunee uprating included the following five objectives. Results for each of these five objectives are discussed in more detail under Sections 5.1.2.4 and 5.1.2.5.

1. Review of the reactor vessel surveillance capsule removal schedule to determine if changes are required as a result of changes in the vessel fluence due to the Power Uprate Program.

2. Review of the existing pressure-temperature (P-T) limit curves to determine if a new applicability date needs to be calculated due to the effects of the uprated fluence projections.

3. Review of the existing RT_{PTS} values to determine if the effects of the uprated fluence projections result in an increase in RT_{PTS} for the beltline materials in the Kewaunee reactor vessels at the end of license (EOL) (33 effective-full-power year [EFPY]) and EOLE (51 EFPY).

4. Review the upper shelf energy (USE) values at EOL for all reactor vessel beltline materials in the Kewaunee reactor vessels to assess the impact of the uprated fluence projections.

5. Review reactor vessel inlet temperature for Kewaunee to verify that it maintains an acceptable level after the uprated condition takes affect.

6. Review the LTOP limitations to determine if a new applicability date needs to be calculated due to the effects of the uprated fluence projections.

<u>Surveillance Capsule Withdrawal Schedules</u>

A surveillance capsule withdrawal schedule is developed to periodically remove surveillance capsules from the reactor vessel. This procedure is followed to effectively monitor the condition

of the reactor vessel materials under actual operating conditions. The current surveillance capsule scheduled defined in the KNPP USAR is consistent with the recommended number of surveillance capsules and withdrawal schedule cited in ASTM E185-82. It is noted that the KNPP surveillance capsule program is based upon ASTM E185-70 (Reference 12) and WCAP-8107 (Reference 13), "Wisconsin Public Service Corp. Kewaunee Nuclear Power Plant Reactor Vessel Radiation Surveillance Program." Testing is performed in accordance with the methods specified in ASTM E185-82 and additional provisions identified in Reference 8.

The surveillance capsule withdrawal schedule is in terms of effective full-power years (EFPY) of plant operation, with a design life of 33 EFPY. Other factors that must be considered in establishing the surveillance capsule withdrawal schedule are maximum fluence values at the vessel inside surface and 1/4-thickness (1/4T) location.

The first surveillance capsule is typically scheduled to be withdrawn early in vessel life to verify initial predictions of surveillance material response to the actual radiation environment. It is generally removed when the predicted shift exceeds expected scatter by sufficient margin to be measurable. Normally, the capsule with the highest lead factor is withdrawn first. Early withdrawal also permits verification of the adequacy and conservatism of reactor vessel pressure-temperature operational limits.

The withdrawal schedule for the maximum number of surveillance capsules to be withdrawn is adjusted by the lead factor so that:

- exposure of the *second* surveillance capsule withdrawn occurs when the accumulated neutron fluence of the capsule corresponds to a value midway between that of the first and third capsules,

- exposure of the *third* surveillance capsule withdrawn does not exceed the peak EOL 1/4T fluence,

- exposure of the *fourth* surveillance capsule withdrawn does not exceed the peak EOL reactor vessel fluence, and

- exposure of the *fifth* surveillance capsule withdrawn does not exceed twice the peak EOL reactor vessel fluence.

Per ASTM E185-82, the four steps used for development of a surveillance capsule withdrawal schedule are as follows:

1. Estimate peak vessel inside surface fluence at EOL and the corresponding transition temperature shift (ΔRT_{NDT}). This identifies the number of capsules required.

2. Obtain the lead factor for each surveillance capsule relative to peak beltline fluence.

3. Calculate the EFPY for the capsule to reach peak vessel EOL fluence at the inside surface and 1/4T locations. These results are used to establish the withdrawal schedule for all but the first surveillance capsule.

4. Schedule surveillance capsule withdrawals at the nearest vessel refueling date.

A current surveillance capsule withdrawal schedule for the Kewaunee reactor vessel is documented in the KNPP USAR and WCAP-14279 Rev. 1 (Reference 2). This schedule has been evaluated for the Power Uprate Project due to increased neutron fluence. In addition, the supplemental requirements imposed under the Master Curve Process (Reference 8) were also reviewed and are not impacted by the Kewaunee Power Uprate Program.

Heatup and Cooldown Pressure-Temperature Limit Curves

A review of the applicability dates of the heatup and cooldown curves, for Kewaunee, was performed. The curves are currently contained in WCAP-14278, Revision 1 (Reference 4). This review was carried out by comparing the fluence projections used in the current calculations of adjusted reference temperature for all the beltline materials in the Kewaunee reactor vessels to the fluence based on the uprated condition.

Pressurized Thermal Shock (PTS)

Pressurized Thermal Shock (PTS), a limiting condition on reactor vessel integrity, is postulated to occur during a severe system transient such as a loss-of-coolant-accident (LOCA) or a steam line break. Such transients may challenge the integrity of a reactor vessel under the following conditions:

- severe overcooling of the inside surface of the vessel wall followed by high repressurization,

- significant degradation of vessel material toughness caused by radiation embrittlement, and

- the presence of a critical-size defect in the vessel wall.

The PTS concern arises if one of these transients acts on the beltline region of a reactor vessel where a reduced fracture resistance exists because of neutron irradiation. Such an event may produce the propagation of flaws postulated to exist near the inner wall surface, thereby potentially affecting the integrity of the vessel.

In 1985, the Nuclear Regulatory Commission (NRC) issued a formal ruling on PTS. It established screening criterion on pressurized water reactor (PWR) vessel embrittlement as measured by nil-ductility reference temperature, termed RT_{PTS}. RT_{PTS} screening criteria values were set (using conservative fracture mechanics analysis techniques) for beltline axial welds, plates, and beltline circumferential weld seams for end-of-life plant operation. All PWR vessels in the United States have been required to evaluate vessel embrittlement in accordance with these criteria through EOL.

The Nuclear Regulatory Commission amended its regulations for light water nuclear power plants to change the procedure for calculating radiation embrittlement. The revised PTS Rule was published in the Federal Register, December 19, 1995 with an effective date of January 18, 1996 (Reference 6). This amendment makes the procedure for calculating RT_{PTS} values consistent with the methods given in Regulatory Guide 1.99, Revision 2.

The Rule establishes the following requirements for all domestic, operating PWRs:

- For each pressurized water nuclear power reactor for which an operating license has been issued, the licensee shall have projected values of RT_{PTS}, accepted by the NRC, for each reactor vessel beltline material for the EOL fluence of the material.

- The assessment of RT_{PTS} must use the calculation procedures given in the PTS Rule and must specify the bases for the projected value of RT_{PTS} for each beltline material. The report must specify the copper and nickel contents and the fluence values used in the calculation for each beltline material.

- This assessment must be updated whenever there is a significant change in projected values of RT_{PTS} or upon the request for a change in the expiration date for operation of the facility. Changes to RT_{PTS} values are significant if either the previous value or the current value, or both values, exceed the screening criteria prior to expiration of the operating license, including any renewal term (if applicable), for the plant.

- The RT_{PTS} screening criteria values for the beltline region are:

  270°F for plates, forgings and axial weld materials, and

  300°F for circumferential weld materials.

RT_{PTS} must be calculated for each vessel beltline material using a fluence value, f, which is the EOL fluence for the material. Equation 1 must be used to calculate values of RT_{NDT} for each weld, plate or forging in the reactor vessel beltline.

Equation 1: $$RT_{NDT} = RT_{NDT(U)} + M + \Delta RT_{NDT}$$

Where,

$RT_{NDT(U)}$ = Reference Temperature for a reactor vessel material in pre-service or non-irradiated condition

M = Margin added to account for uncertainties in values of $RT_{NDT(U)}$, copper and nickel content, fluence, and calculation procedures. M is evaluated from Equation 2.

Equation 2: $$M = \sqrt{\sigma_U^2 + \sigma_\Delta^2}$$

σ_U is the standard deviation for $RT_{NDT(U)}$.

σ_U = 0°F when $RT_{NDT(U)}$ is a measured value.

σ_U = 17°F when $RT_{NDT(U)}$ is a generic value.

σ_Δ is the standard deviation for RT_{NDT}. σ_Δ is not to exceed one half of ΔRT_{NDT}

For plates and forgings:

σ_Δ = 17°F when surveillance capsule data is not used.

σ_Δ = 8.5°F when surveillance capsule data is used.

For welds:

σ_Δ = 28°F when surveillance capsule data is not used.

σ_Δ = 14°F when surveillance capsule data is used.

ΔRT_{NDT} is the mean value of the transition temperature shift, or change in RT_{NDT}, due to being irradiated and must be calculated using Equation 3.

Equation 3: $\Delta RT_{NDT} = (CF) * f^{(0.28 - 0.10 \log f)}$

CF (°F) is the chemistry factor, which is a function of copper and nickel content. CF is determined from Tables 1 and 2 of the PTS Rule (10 CFR 50.61). Surveillance data deemed credible must be used to determine a material-specific value of CF. A material-specific value of CF is determined in Equation 5.

f is the calculated neutron fluence, in units of 10^{19} n/cm^2 (E > 1.0 MeV), at the clad-base-metal interface on the inside surface of the vessel at the location where the material in question receives the highest fluence. The 33 EFPY EOL fluence are used in calculating RT_{PTS}.

Equation 4 must be used for determining RT_{PTS} using Equation 3 with EOL fluence values for determining RT_{PTS}

Equation 4: $RT_{PTS} = RT_{NDT(U)} + M + \Delta RT_{PTS}$

To verify that RT_{NDT} for each vessel beltline material is a bounding value for the specific reactor vessel, licensees shall consider plant-specific information that could affect the level of embrittlement. This information includes but is not limited to reactor vessel operating temperature and any related surveillance program results. Results from the plant-specific surveillance program must be integrated into the RT_{NDT} estimate if the plant-specific surveillance data has been deemed credible. According to Regulatory Guide 1.99, Revision 2, in order to use surveillance data there has to be "...two or more credible data sets...from the reactor in

question." A material-specific value of CF for surveillance materials is determined from Equation 5.

$$\text{Equation 5:} \quad CF = \frac{\sum [A_i * f_i^{(0.28-0.10 \log f_i)}]}{\sum [f_i^{(0.56-0.20 \log f_i)}]}$$

In Equation 5, "A_i" is the measured value of ΔRT_{NDT} and "f_i" is the fluence for each surveillance data point. If there is clear evidence that the copper and nickel content of the surveillance weld differs from the vessel weld, i.e., differs from the average for the weld wire heat number associated with the vessel weld and the surveillance weld, the measured values of RT_{NDT} must be adjusted for differences in copper and nickel content by multiplying them by the ratio of the chemistry factor for the vessel material to that for the surveillance weld.

It should be noted here that the above methodology was used for evaluation of the KNPP Power Uprate Program for all the beltline materials except the intermediate to lower shell circumferential weld. In this case, the Master Curve technology was used as described in WCAP-15075 (Reference 11) and with modifications as described in the NRC's Safety Evaluation (Reference 8) of the request for exemption by Kewaunee. Equation 6 was used to determine RT_{PTS} for the intermediate to lower shell circumferential weld using Master Curve technology.

$$\text{Equation 6:} \quad RT_{PTS} = RT_{TO} + M + Bias$$

Where, RT_{TO} = $T_O + 33°F$

M = Margin = 62.5°F

Bias = 8.5°F

(All the above terms were prescribed or adjusted by the NRC in Reference 8.)

New T_o values and adjusted reference temperatures have been calculated for the intermediate to lower shell circumferential weld using the methodology described in Reference 8 corresponding to EOL and EOLE fluence for the KNPP Uprate.

Emergency Response Guideline (ERG) Limits

Emergency Response Guideline (ERG) pressure-temperature limits were developed to establish guidance for operator action in the event of an emergency, such as a PTS event. Generic categories of limits were developed for the guidelines based on the limiting inside surface RT_{NDT} at EOL. These generic categories were conservatively generated for the Westinghouse Owners Group (WOG) to be applicable to all Westinghouse plants.

The highest EOL RT_{NDT} for which the generic category ERG pressure-temperature limits were developed is 250°F for a longitudinal flaw and 300°F for a circumferential flaw. Thus, if the limiting vessel material has an EOL RT_{NDT} that exceeds 250°F for a longitudinal flaw or 300°F for a circumferential flaw, plant-specific ERG pressure-temperature limits are required.

A comparison of the current RT_{PTS} calculation (which is the EOL RT_{NDT} value (33 EFPY)) has been made to the uprated RT_{PTS} values for Kewaunee to determine if the applicable ERG category would change. The ERG Categories are presented in Table 5.2-4.

Upper Shelf Energy (USE)

The integrity of the reactor vessel may be affected by changes in system temperatures and pressures resulting from the power uprate. To address this consideration, an evaluation was performed to assess the impact of the power uprate on the USE values for all reactor vessel beltline materials in the Kewaunee reactor vessel. The USE assessment used the results of the neutron fluence evaluation for the power uprate and Figure 2 of Regulatory Guide 1.99, Revision 2 (Reference 10), to determine if a further decrease in USE at EOL would occur due to the effects of the uprate on fluence projections.

Inlet Temperature

Paragraph 1.3.2 of Regulatory Guide 1.99, Revision 2, which is the basis of10 CFR 50.61 and used in all the analyses described herein, stipulates that these equations are valid only in the temperature range of 525°F to 590°F. Therefore, the reactor vessel inlet temperature must be maintained within this range to retain validity of all existing analyses.

Low Temperature Overpressure Protection System

The LTOP limitations are evaluated to determine if any change is required due to power uprate.

5.1.2.4 Acceptance Criteria

Surveillance Capsule Withdrawal Schedule

The proposed surveillance capsule withdrawal schedules developed for Kewaunee following the uprating shall meet the requirements of ASTM-185-82. A satisfactory number of surveillance capsules shall remain in the reactor vessel so that further analysis, such as for life extension, can be completed as necessary.

Heatup and Cooldown Pressure-Temperature Limit Curves

Fluence projections for Kewaunee increase from that used in the current heatup and cooldown curves. Revised applicability dates for the new heatup and cooldown curves are generated.

Pressurized Thermal Shock (PTS)

The uprated RT_{PTS} values for all beltline materials shall not exceed the screening criteria of the PTS Rule. Specifically, the RT_{PTS} values of the base metal (plates or forgings) shall not exceed 270°F, while the girth weld metal RT_{PTS} values shall not exceed 300°F through the end-of-license (33 EFPY) and EOLE (51 EFPY).

Emergency Response Guideline (ERG) Limits

The ERG limits shall be developed to establish guidelines for operator action in the event of an emergency, such as a PTS event. The ERG Categories are presented in Table 5.2-4. The category that Kewaunee must follow will be presented in Section 5.1.2.5.

Upper Shelf Energy (USE)

At power uprate conditions, the EOL USE values for all reactor beltline materials shall meet the requirements of 10 CFR 50, Appendix G (Reference 7).

Inlet Temperature

The inlet temperature must be maintained in the range of 525°F to 590°F for the current analyses described herein to remain valid.

Low Temperature Overpressure Protection System

The LTOPS setpoint protect the reactor vessel so there are no violations of the reactor vessel Appendix G pressure-temperature limits.

5.1.2.5 Results

An evaluation of the impact of uprating on reactor vessel integrity was performed for Kewaunee. The neutron fluence projections for Kewaunee after the power uprating have increased from previous analyses.

Surveillance Capsule Withdrawal Schedule

The revised fluence projections considering the Power Uprate Program have exceeded the fluence projections used in the development of the current withdrawal schedule for Kewaunee (Reference 2). A calculation of ΔRT_{NDT} at 33 EFPY was performed to determine if the increased fluences alter the number of capsules to be withdrawn for Kewaunee. This calculation determined that the maximum ΔRT_{NDT} using the uprated fluences for Kewaunee at EOL is greater than 200°F. Per Reference 3, these ΔRT_{NDT} values would require five capsules to be withdrawn from Kewaunee. The number of capsules has not changed from the current withdrawal schedule. However, due to changes in capsule fluences, the schedule has been updated as shown in Table 5.2-1. It should also be noted that the Kewaunee withdrawal schedule is acceptable since it meets the requirements of ASTM E185-82 (Reference 3).

Applicability of Heatup and Cooldown Pressure-Temperature Limit Curves

Kewaunee is currently operating to P-T limit curves documented in Reference 4. A review based on uprated fluences was conducted on the current heatup and cooldown curve applicability date for Kewaunee. 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, which is projected to be reached during cycle 31 on approximately April 23rd 2011.

Pressurized Thermal Shock

The pressurized thermal shock (PTS) calculations were performed for the Kewaunee intermediate and lower shell forgings using the latest procedures required by the Nuclear Regulatory Commission (NRC) in Reference 6 and using master curve technologies for the circumferential weld. The calculated neutron fluence values for the Power Uprate Program

condition at Kewaunee have increased over the current fluences. Therefore, to evaluate the effects of the Power Uprate Program, the PTS values for the beltline region materials from Kewaunee were re-evaluated using the uprated fluences. Based on this evaluation, all RT_{PTS} values will remain below the NRC screening criteria values using the projected Power Uprate Program fluence values through EOL (33 EFPY) and EOLE (51 EFPY) for Kewaunee, as shown in Tables 5.2-2 and 5.2-3.

Emergency Response Guideline Limits

The current peak-inside surface RT_{NDT} values at EOL, that were calculated in Tables 5.2-2 and 5.2-3, are above 250°F for Kewaunee. The limiting material for Kewaunee was the circumferential weld 1P3571. The RT_{NDT} values prior to the Power Uprate Program, which are documented in Reference 5, put Kewaunee in emergency response guideline (ERG) Category IIIb per Table 5.2-4. Even though the revised fluence projections after the Power Uprate Program have increased over the fluence projections used in development of the current peak inside surface RT_{NDT} values at EOL (33 EFPY), Kewaunee will remain in ERG Category IIIb.

Upper Shelf Energy

All beltline materials are expected to have a USE greater than 50 ft-lb through the EOL (33 EFPY), as required by 10 CFR 50 (Reference 7). The EOL USE was predicted using the EOL 1/4T fluence projection.

The revised fluence projections associated with the Power Uprate Program have increased the fluence projections used in developing the current predicted EOL USE values (Reference 2). However, it has only affected the 1/4T fluence by 1.074 percent. This 1.07-percent increase has a slight measurable effect on the percent decrease in USE. Therefore, the current predicted USE values for Kewaunee have been updated as shown in Table 5.2-5. Note that all USE values for Kewaunee will maintain a level above the 50 ft-lb screening criterion at end of license (33 EFPY).

Inlet Temperature

The inlet temperature for Cases 1 and 2 are below 525°F, while cases 3 and 4 are within the range of 525°F to 590°F. Kewaunee will operate at a reactor inlet temperature above 525°F.

Low Temperature Overpressure Protection System

The KNPP uprating is not changing any of the parameters described earlier that could potentially affect the LTOPS setpoint except indirectly the pressure vs. temperature limits. The uprating will not be changing the actual Appendix G pressure vs. temperature limits for which the LTOPS must provide protection. However, the maximum time in plant lifetime for which these pressure vs. temperature limits are applicable will be changed from 33.0 EFPY (Effective Full Power years) to 31.1 EFPY. Therefore, since the actual Appendix G pressure vs. temperature limits are not changing, the existing LTOPS setpoints will remain applicable for the uprating. They will remain applicable until the actual pressure vs. temperature limits (not the maximum EFPY for which they are applicable) require revision.

5.1.2.6 Conclusions

The fluence projections associated with the Power Uprate Program, while considering actual power distributions incorporated to date, will exceed the current fluence projections used in the evaluations of record (References 2, 4, 5 and 6) (withdrawal schedules, ERG category, PTS, LTOP limitations and USE). The effect of the higher fluence values is minimal for PTS and has not changed the ERG limits. As for the withdrawal schedule and predicted EOL USE, the effect of the higher fluences is shown in Tables 5.2-1 and 5.2-5, respectively. With respect to the P-T curves, the current curves are documented in Reference 4. These P-T curves used fluences that were developed without the current Power Uprate Program. The new applicability date for the Kewaunee P-T curves is now 31.1 EFPY, which is projected to be reached during cycle 31 on approximately April 23rd 2011. The LTOP limitations are tied directly to the P-T curves and are therefore valid through 31.1 EFPY. Lastly, the inlet temperature will be maintained above 525°F for normal operation.

It is concluded that the Kewaunee Power Uprate Program will not have significant effect on the reactor vessel integrity.

5.1.2.7 References

1. Regulatory Guide 1.190 "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence."

2. E. Terek, G. N. Wrights and J. F. Williams, "Analysis of Capsule S From the Wisconsin Public Service Corporation Kewaunee Nuclear Plant Reactor Vessel Radiation Surveillance Program," WCAP-14279, March 1995.

3. ASTM E185-82, Annual Book of ASTM Standards, Section 12, Volume 12.02, "Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels."

4. T. Laubham and C. Kim, "Kewaunee Heatup and Cooldown Limit Curves for Normal Operation," WCAP-14278 Rev. 1, September 1998.

5. E. Terek, R. Lott and C. Kim, "Evaluation of Pressurized Thermal Shock for the Kewaunee Reactor Vessel," WCAP-14280 Rev. 1, September 1998.

6. 10 CFR 50.61, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events," Federal Register, Volume 60, No. 243, December 19, 1995.

7. 10 CFR 50, Appendix G & H, "Reactor Vessel Material Surveillance Program Requirements," Federal Register, Volume 60, No. 243, December 19, 1995.

8. NRC Docket No. 50-305 "Kewaunee Nuclear Power Plant – Request for Exemption from the Requirements of 10 CFR Part 50," Appendices G and H, and 10 CFR 50.61 for the Kewaunee Nuclear Power Plant (KNPP).

9. ASTM E900, "Standard Guide for Predicting Neutron Radiation Damage to Reactor Vessel Materials, E 706 (IIF)," Reapproved 1994.

10. Regulatory Guide 1.99, Revision 2, May 1988, "Radiation Embrittlement of Reactor Vessel Materials."

11. WCAP-15075, "Master Curve Strategies for RPV Assessment," R.G. Lott, M.T. Kirk, C.C. Kim, September 1998.

12. ASTM E185-70 "Recommended Practice for Surveillance Tests on Nuclear Reactor Vessels."

13. WCAP-8107 "Wisconsin Public Service Corp. Kewaunee Nuclear Power Plant Reactor Vessel Radiation Surveillance Program," S. E. Yanichko, et. al, April 1973.

Table 5.2-1 Recommended Surveillance Capsule Withdrawal Schedule				
Capsule	**Capsule Location**	**Lead Factor**	**Withdrawal EFPY(1)**	**Fluence (n/cm^2)**
V	77°	3.03	1.3	5.86×10^{18}
R	257°	3.03	4.6	1.76×10^{19}
P	247°	2.00	11.1	2.61×10^{19}
S	57°	2.08	16.2	3.67×10^{19}
T	237°	2.18	EOL	See note 2
N	67°	2.13	Standby	- - -

Notes:

1. EFPYs from plant startup.
2. Capsule T should be removed before it receives a fluence of 7.12×10^{19} n/cm^2(E>1.0 MeV) (i.e., twice the peak vessel EOL surface fluence of 3.56×10^{19} n/cm^2 (E>1.0 MeV)). This capsule may be held without testing following withdrawal. Capsule T will reach a fluence of approximately 5.63×10^{19} n/cm^2 (E> 1.0 MeV) at 22.23 EFPY, which was reached during cycle 25 on approximately February 26th, 2002. This is equal to the reactor vessel peak surface fluence of 5 63 x 10^{19} n/cm^2(E>1.0 MeV) at 51 EFPY (Approximately December 31st 2033 during fuel cycle 46,60 calendar year life).

Table 5.2-2

RT_{PTS} Calculations for Kewaunee Intermediate and Lower Shell Materials at 33 EFPY and 51 EFPY with Uprated Fluences Using Charpy-Based Data

Material	Fluence (n/cm^2, E>1.0 MeV)	FF	CF (°F)	ΔRT_{PTS} (3) (°F)	Margin (°F)	$RT_{NDT(U)}$ (1) (°F)	RT_{PTS} (2) (°F)
33 EFPY							
Intermediate Shell Forging 122X208VA1	3.56	1.33	37	49.2	34	60	143
→ Using S/C Data	3.56	1.33	23.8	31.6	34	60	126
Lower Shell Forging 123X167VA1	3.56	1.33	37	49.2	34	20	103
→ Using S/C Data	3.56	1.33	21.2	28.2	34	20	82
51 EFPY							
Intermediate Shell Forging 122X208VA1	5.63	1.425	37	52.7	34	60	147
→ Using S/C Data	5.63	1.425	23.8	33.9	34	60	128
Lower Shell Forging 123X167VA1	5.63	1.425	37	52.7	34	20	107
→ Using S/C Data	5.63	1.425	21.2	30.2	34	20	84

Notes:

1. Initial RT_{NDT} values are measured values.
2. $RT_{PTS} = RT_{NDT(U)} + \Delta RT_{PTS}$ + Margin (°F)
3. ΔRT_{PTS} = CF * FF

Table 5.2-3 RT_{PTS} Calculations for Kewaunee Circumferential Weld 1P3571 at 33 EFPY and 51 EFPY with Uprated Fluences Using Master Curve Technology							
Material	Fluence (n/cm^2, E>1.0 MeV)	FF	T_o (°F)	$RT_o = T_0 + 33°F$ [(2)]	Margin [(1)] (°F)	Bias [(1)] (°F)	RT_{PTS} (°F)
33 EFPY							
Circumferential Weld 1P3571	3.56	1.330	167.5	200.5	62.5	8.5	271.5
51 EFPY							
Circumferential Weld 1P3571	5.63	1.425	190.5	223.5	62.5	8.5	294.5

Notes:

1. Per NRC in Reference 8, margin term is 62.5°F and bias term is 8.5°F.
2. Per NRC in Reference 8, $RT_o = T_o + 33°F$.

Table 5.2-4 ERG Pressure-Temperature Limits[6]	
Applicable RT_{NDT} (ART) Value [(a)]	ERG Pressure-Temperature Limit Category
$RT_{NDT} < 200°F$	Category I
$200°F < RT_{NDT} < 250°F$	Category II
$250°F < RT_{NDT} < 300°F$	Category IIIb

Notes:

(a) Longitudinally oriented flaws are applicable only up to 250°F, the circumferentially oriented flaws are applicable up to 300°F

Table 5.2-5 Predicted EOL (33 EFPY) USE Calculations for all the Beltline Region Materials					
Material	**Weight % of Cu**	**1/4T EOL Fluence (10^{19} n/cm^2)**	**Unirradiated USE (ft-lb)**	**Projected USE Decrease[1] (%)**	**Projected EOL USE (ft-lb)**
Intermediate Shell Forging 122X208VA1	0.06	2.41	92	7.5	85
Lower Shell Forging 123X167VA1	0.06	2.41	97	3.5	94
Circumferential Weld 1P3571	0.287	2.41	126	46	68

Note:

1. Values are deduced from Figure 2 of Regulatory Guide 1.99, Revision 2, "Predicted Decrease in Upper Shelf Energy as a Function of Copper and Fluence."

5.2 Reactor Pressure Vessel System for Kewaunee

5.2.1 Introduction

The reactor pressure vessel (RPV) system consists of the reactor vessel, reactor internals, fuel, and control rod drive mechanisms (CRDMs). The reactor internals support and orient the reactor core fuel assemblies and control rod assemblies, absorb control rod assembly dynamic loads, and transmit these and other loads to the reactor vessel. The reactor vessel internal components also direct coolant flow through the fuel assemblies, provide adequate cooling flow to the various internal structures, and support in-core instrumentation. The reactor vessel internals are designed to withstand forces due to structural deadweight, preload of fuel assemblies, control rod assembly dynamic loads, vibratory loads, earthquake accelerations, and loss-of-coolant accident (LOCA) loads.

Evaluation of the uprated conditions (7.4 percent uprate) requires that the reactor vessel/internals/fuel system interface be assessed to ensure compatibility and that there are no adverse effects on the structural integrity of the reactor vessel/internals/fuel system. In addition, thermal-hydraulic analyses are required to determine plant-specific core bypass flows, pressure drops, and upper head temperatures for input to LOCA and non-LOCA safety analyses and Nuclear Steam Supply System (NSSS) performance evaluations.

Generally, the areas of concern most affected by changes in system operating conditions are:

- Reactor internals system thermal-hydraulic performance
- Rod control cluster assembly (RCCA) scram performance
- Reactor internals system structural response and integrity

5.2.2 Input Parameters and Assumptions

5.2.1 The operating parameters (pressure, temperature, flow, power level) are provided in Table IV.B-1 (Attachment 2). All analyses and evaluations were performed at 1772 MWt.

5.2.2 A full core of Westinghouse 422 V+ fuel is assumed.

5.2.3 Description of Analyses/Evaluations

Descriptions of the various analyses and evaluations are given in the individual sections and subsections of 5.2.5 through 5.2.7

5.2.4 Acceptance Criteria

Acceptance criteria are typically listed in each individual section. The most important acceptance criteria are grouped together below.

The core bypass flow must be within the 7.0 percent upper limit

Hydraulic forces on the reactor internals must be limited such that the internal will remain seated and stable.

The cumulative fatigue usage factor for the most critically stressed components should be less than 1.0

5.2.5 Thermal-Hydraulic System Evaluations

5.2.5.1 System Pressure Losses

The principal Reactor Coolant System (RCS) flow route through the RPV system at the Kewaunee unit begins at the two reactor inlet nozzles. At this point, flow turns downward through the reactor vessel core barrel annulus. After passing through this downcomer region, flow enters the lower reactor vessel dome region. This region is occupied by the internals energy absorber structure, lower support columns, bottom-mounted instrumentation columns, and supporting tie plates. From this region, flow passes upward through the lower core plate and into the core region. After passing up through the core, the coolant flows into the upper plenum, turns, and exits the reactor vessel through the two reactor outlet nozzles. The support columns and rod control cluster assembly (RCCA) guide columns occupy the upper plenum region.

A key area in evaluating core performance is to determine the hydraulic behavior of coolant flow within the reactor internals system, that is, vessel pressure drops, core bypass flows, RPV fluid temperatures and hydraulic lift forces. The power uprate analyses determined the distribution of pressure and flow within the reactor vessel, internals, and the reactor core for the uprated conditions.

5.2.5.2 Bypass Flow Analysis

Bypass flow is the total amount of reactor coolant flow bypassing the core region. Bypass flow is not considered effective in the core heat transfer process. Since variations in the size of the bypass flow paths occur during manufacturing, such as gaps at the outlet nozzles and core barrel, or changes due to different fuel assembly designs or changes in RCS conditions, plant-specific as-built dimensions are used to demonstrate that bypass flow limits are not violated. Analyses are performed to determine core bypass flow values. This ensures that either the design bypass flow limit for the plant will not be exceeded or a revised design core bypass flow is required.

The present design core bypass flow limit is 7.0 percent (with thimble plugs removed) of the total reactor vessel flow. The purpose of this evaluation is to ensure that the design value of 7.0 percent can be maintained at the uprated RCS conditions. The principal core bypass flow paths are the:

- Baffle-barrel region
- Vessel head cooling spray nozzles
- Core barrel – reactor vessel outlet nozzle gap
- Fuel assembly – baffle plate cavity gap
- Fuel assembly thimble tubes

Fuel assembly hydraulic characteristics and system parameters, such as reactor coolant inlet temperature, pressure, and flows were used in conjunction with the THRIVE computer code to determine the impact of uprated conditions on total core bypass flow. THRIVE solves the mass and energy balances for the reactor internals fluid system and has been used in the original design basis of KNPP.

A summary of the bypass flow is as follows:

- The total best-estimate core bypass flow values (including uncertainties) at 7.4% uprate were determined to be 5.1 percent, which is below the limit of 7.0 percent.
- At 7.4% uprate, the component of bypass flow which travels from the internals inlet plenum to the upper head region keeps the head temperature at 595 degrees F on a

best-estimate basis, assuming a core inlet temperature of 539.2 degrees F. This is considered an acceptable temperature with respect to CRDM j-groove weld service life.

5.2.5.3 Hydraulic Lift Forces

The reactor internals hold-down spring is essentially a large diameter belleville-type spring of rectangular cross section. The purpose of this spring is to maintain a net clamping force between the reactor vessel head and upper internals flanges, and the reactor vessel shell flange and core barrel flange of the internals.

An evaluation was performed for the 7.4% uprated conditions to determine the effects of hydraulic lift forces on the various reactor internal components. The results show that the uplift forces are less than previously analyzed and are therefore acceptable.

5.2.5.4 RCCA Scram Performance Evaluation

The RCCAs represent the interface between the fuel assemblies and other internal components. An evaluation was performed to determine the potential impact due to power uprating on RCCA scram characteristics that are used in the safety analyses. This evaluation is based on the newly incorporated Westinghouse 422 V+ fuel assemblies.

Calculations demonstrated that, for even the most severe case, the current limit for drop-time-to-dashpot entry of 1.8 seconds is bounding and remains conservatively applicable for accident analyses.

5.2.5.5 Momentum Flux and Fuel Rod Stability

Baffle jetting is a hydraulically induced instability or vibration of fuel rods caused by a high-velocity jet of water. This jet is created by high-pressure water being forced through gaps between the baffle plates surrounding the core. To guard against fuel rod failures from flow-induced vibration (FIV), the crossflow from baffle joint gaps must be limited to a specific momentum flux, V^2h, that is, the product of the gap width, h, and the square of the baffle joint jet velocity, V.

To assess the impact of the uprated RCS conditions on baffle jetting margins of safety at the Kewaunee unit, the ratio of the margins of safety between the present plant configuration and the uprated configuration was determined. The results show that, based on mechanical design flow, the margins of safety for momentum flux at uprated conditions do not change significantly from those at the present conditions.

5.2.6 Mechanical System Evaluations

The RCS mechanical response, subjected to auxiliary line breaks of a LOCA transient, is performed in three steps. First, the RCS is analyzed for the effects of loads induced by normal operation, which includes thermal, pressure, and deadweight effects. From this analysis, the mechanical forces acting on the RPV, which would result from release of equilibrium forces at the break locations, are obtained. In the second step, the loop mechanical loads and reactor internals hydraulic forces are simultaneously applied, and the RPV displacements due to the LOCA are calculated. Finally, the structural integrity of the reactor coolant loop (RCL) and component supports relied upon to deal with the LOCA are evaluated by applying the calculated reactor vessel displacements to a mathematical model of the RCL. Thus, the effects of vessel displacements upon the loop and reactor vessel/internals are evaluated.

5.2.6.1 Loss-of-Coolant Accident Loads

LOCA loads applied to the Kewaunee RPV system consist of reactor internal hydraulic loads (vertical and horizontal), and pressure loads acting on the baffle plates. All loads are calculated individually and combined in a time-history manner.

The severity of a postulated break in a reactor vessel is related to two factors: the distance from the reactor vessel to the break location, and the break opening area. The nature of the reactor vessel decompression following a LOCA, as controlled by the internals structural configuration previously discussed, results in larger reactor internal hydraulic forces for pipe breaks in the cold leg than in the hot leg (for breaks of similar area and distance from the RPV). Pipe breaks farther away from the reactor vessel are less severe because the pressure wave attenuates as it propagates toward the reactor vessel.

Since the Kewaunee reactor takes credit for leak-before-break (LBB) applied to the primary loop, LOCA analyses of the RPV system for postulated ruptures of primary loop piping are not required. The next limiting breaks to be considered are branch-line breaks, such as in the

accumulator line, pressurizer surge line, and residual heat removal (RHR) line. With consideration of LBB on the primary lines, such auxiliary line breaks are not as severe as the main line breaks (for example, RPV inlet nozzle or RCP outlet nozzle break).

The LOCA forces generated for Kewaunee for these branch line breaks are much lower than those originally considered for the reactor pressure vessel inlet nozzle (RPVIN) and reactor coolant pump outlet nozzle (RCPON) breaks. Therefore, critical stresses in baffle and former plates and associated bolting are bounded by existing analyses.

5.2.6.2 Flow Induced Vibrations

FIVs of pressurized water reactor internals have been studied at Westinghouse for a number of years. The objective of these studies was to ensure the structural integrity and reliability of reactor internal components.

Results from scale model and in-plant tests indicate that the primary cause of lower internals excitation is flow turbulence generated by expansion and turning of flow at the transition from the inlet nozzle to the barrel-vessel annulus, and wall turbulence generated in the downcomer.

The design parameters, which could potentially influence FIV response of the reactor internals, include inlet nozzle flow velocities, vessel/core inlet temperatures, and vessel outlet temperatures. Generally, the inlet nozzle velocity for FIV response during hot functional testing is calculated using mechanical design flows, which are approximately 15 percent higher than thermal design flows.

The other parameter, which would influence the FIV response, is core inlet temperature. For the most limiting case at uprated conditions, the vessel/core inlet temperature is 521.9°F. For the uprated conditions , it was determined that FIV loads on the guide tubes, upper support columns, and the lower internals are within 5 percent of previous FIV analyses. Similar evaluations show that there is sufficient margin to accommodate this increase in FIV loads. Consequently, the structural integrity of the Kewaunee reactor internals remains acceptable with regard to FIVs.

5.2.7 Structural Evaluation of Reactor Internal Components

In addition to supporting the core, a secondary function of the reactor vessel internals assembly is to direct coolant flows within the vessel. While directing primary flow through the core, the internals assembly also establishes secondary flow paths for cooling the upper regions of the reactor vessel and the internals structural components. Some of the parameters influencing the mechanical design of the internals lower assembly are the pressure and temperature differentials across its component parts and the flow rate required to remove heat generated within the structural components due to radiation (for example, gamma heating). The configuration of the internals provides adequate cooling capability. Also, the thermal gradients resulting from gamma heating and core coolant temperature changes are maintained below acceptable limits within and between the various structural components.

Structural evaluations demonstrate that the structural integrity of reactor internal components is not adversely affected either directly by the uprated RCS conditions and transients, or by secondary effects on reactor thermal-hydraulic or structural performance. Heat generated in reactor internal components, along with the various fluid temperature changes, results in thermal gradients within and between components. These thermal gradients result in thermal stresses and thermal growth, which must be considered in the design and analysis of the various components.

Since the Kewaunee reactor internals were designed prior to introduction of Subsection NG of the ASME Boiler and Pressure Code Section III, a plant-specific stress report on the reactor internals was not required. However, the design of the Kewaunee reactor internals was evaluated according to Westinghouse internal criteria, which is similar to the ASME Code (Reference 1). Moreover, the structural integrity of the Kewaunee reactor internals design has been ensured by analyses performed on both generic and plant-specific bases. These analyses were used as the basis for evaluating critical Kewaunee reactor internal components for uprating and revised design transients.

5.2.7.1 Lower Core Plate

Structural evaluations were performed to demonstrate that the structural integrity of the lower core plate is not adversely affected either by the uprated RCS conditions or by secondary

effects on reactor thermal-hydraulic or structural performance. For this lower core plate evaluation the criteria described in Section III, Subsection NG of the ASME Code were utilized.

Primarily because of the higher gamma heating rates associated with the uprated conditions, the lower core plate is the most critically stressed component in the entire reactor internals assembly. The conclusion of these evaluations is that structural integrity of the lower core plate is maintained. The uprated RCS conditions produced acceptable margins of safety and fatigue utilization factors for all ligaments under all loading conditions. The limiting (highest) cumulative usage factor for the Kewaunee lower core plate at 7.4% uprate is 0.482, whereas 1.0 is the ASME Code limit (Reference 1).

5.2.7.2 Upper Core Plate Evaluations

The upper core plate positions the upper ends of the fuel assemblies and the lower ends of the control rod guide tubes, thus serving as the transitioning member for the control rods in entry and retraction from the fuel assemblies. It also controls coolant flow exiting the fuel assemblies and serves as a boundary between the core and exit plenum. The upper core plate is restrained from vertical movement by the upper support columns, which are attached to the upper support plate assembly. Four equally spaced core plate alignment pins restrain lateral movement.

The stresses in the upper core plate are mainly due to hydraulic and thermal loads. The total thermal stresses are due to thermal bending moments through the thickness and surface peak stresses. Evaluations were performed to determine the impact of uprating on the structural integrity of the upper core plate. As a result of this evaluation, it is concluded that the upper core plate is structurally adequate for the uprated RCS conditions at the Kewaunee unit. The limiting (highest) cumulative usage factor for the Kewaunee upper core plate at 7.4% uprate is 0.161, whereas 1.0 is the ASME Code limit (Reference 1).

5.2.7.3 Baffle-Barrel Region Components

The Kewaunee lower internals assembly consists of a core barrel into which baffle plates are installed, supported by interconnecting former plates. A lower core support structure is provided at the bottom of the core barrel and a neutron panel surrounds the core barrel. The components

comprising the lower internals assembly are precision-machined. The baffle and former plates are bolted into the core barrel. The reactor vessel internals configuration for Kewaunee utilizes downflow in the barrel-baffle region.

5.2.7.3.1 Core Barrel Evaluation

The thermal stresses in the core active region of the core barrel shell are primarily due to temperature gradients through the thickness of the core barrel shell. Evaluations were performed to determine the thermal bending and skin stresses in the core barrel for the uprated RCS conditions. These evaluations indicated that the actual number of fatigue cycles, based on all normal/upset conditions, was well below the allowable value. From these conservative results, it has been concluded that the core barrel is structurally adequate for the Kewaunee uprated RCS conditions.

5.2.7.3.2 Baffle-Barrel Bolt Evaluation

The bolts are evaluated for loads resulting from hydraulic pressure, seismic loads, preload, and thermal conditions. The temperature difference between baffle and barrel produces the dominant loads on the baffle-former bolts. Hydraulic pressure and seismic loads produce the primary stresses, whereas bolt preloading and thermal conditions produce the secondary stresses. The uprated RCS conditions do not affect deadweight or preload forces.

Since these bolts are qualified by test, the evaluation of the revised loads consisted of comparing the existing operating loads to those developed with the uprated RCS conditions. The results indicate that the thermal baffle-former and barrel-former bolt loads, with the currently analyzed condition, bound those developed with the uprated RCS conditions. Therefore, it is concluded that the baffle-former and barrel-former bolts are structurally adequate for the uprated RCS conditions. Specific degraded bolt combinations are documented.

5.2.7.4 Additional Components

A series of assessments were performed on reactor internal components that were not significantly impacted by the power uprating (and the resulting internal heat generation rates), but are affected by the uprated RCS conditions due to primary loop design transients. These components are:

- Lower core support plate
- Lower support columns
- Core barrel outlet nozzle
- Core barrel flange
- Lower radial restraints (clevis inserts)
- Upper core plate alignment pin
- Upper support columns
- Upper support plate
- Guide tubes and support pins

The results of these assessments demonstrate that the above listed components are structurally adequate for the uprated RCS conditions.

5.2.8 Conclusions

Analyses have been performed to assess the effect of changes due to 7.4% power uprate power uprate. The results of these analyses are as follows:

- The total core bypass flow (with uncertainties) was determined to be 5.1 percent. Therefore, the design core bypass flow value of 7.0 percent of the total vessel flow is maintained for the uprating.

- Hydraulic forces at the uprated conditions were evaluated for effects on the reactor internals. It was determined that the Kewaunee reactor internals assemblies will remain seated and stable at the uprated conditions.

- An RCCA performance evaluation was completed and indicated that the current 1.8-second RCCA drop-time-to-dashpot entry limit (from gripper release of the drive rod) is satisfied at power uprate conditions.

- Baffle plate momentum flux margins of safety due to power uprate conditions are relatively unchanged from present conditions for mechanical design flow, and remain acceptable.

- Evaluations were completed and indicated that the uprated RCS conditions will not adversely impact the response of reactor internals systems and components due to seismic/LOCA excitations and FIVs.

- Evaluations of the critical reactor internal components were performed, which indicated that the structural integrity of the reactor internals is maintained at the uprated RCS conditions. Limiting cumulative usage factors are per the table below

ITEM	CUF AT 7.4% UPRATE
LOWER CORE PLATE	0.482
UPPER CORE PLATE	0.161

- Upper head temperature of 595 degrees F is considered acceptable

- The maximum core exit temperature of 611.3 degrees F is considered acceptable with respect to PWSCC of guide tube split pins since this phenomena is primarily a function of temperature. Many similar reactors operate with exit temperatures of 620 – 630 degrees F.

5.2.9 References

1. ASME Code Section III, Appendices, 1989 Edition (used as a guideline, there is no code of record for the Kewaunee Reactor Internals).

5.4 Control Rod Drive Mechanisms

5.4.1 Introduction

This section addresses the ASME Code of record structural considerations for the pressure boundary components of the Westinghouse full-length L-106A control rod drive mechanisms (CRDMs). The CRDMs are evaluated for the design parameters shown in Table IV.B-1 of Attachment 2 and the Nuclear Steam Supply System (NSSS) design transients which assumed a 7.4 percent increase in core power.

5.4.2 Input Parameters and Assumptions

The Model L-106A CRDMs were originally designed and analyzed to meet the ASME Code, 1965 Edition through the Summer 1966 Addenda or later (Reference 1). A later evaluation of the Model L-106A CRDMs was conducted for the Steam Generator Replacement (RSG) Project for the Kewaunee plant. The RSG Project evaluation supplements the original evaluation and is contained in Reference 2.

The current Kewaunee Power Uprating Program modifies the design parameters and the NSSS design transients that were considered in these previous evaluations. The seismic loading has not been changed for the Kewaunee Power Uprating Program.

The Kewaunee CRDMs are of the hot head type, defined by the vessel outlet reactor coolant temperature on the design parameters (Table IV.B-1, Attachment 2), and must be analyzed for the NSSS design transients defined for the hot leg.

The differences associated with the uprating requirements are discussed in subsection 5.4.3.

5.4.3 Description of Analysis

5.4.3.1 Operating Pressure and Temperature

There are no changes from the current reactor coolant pressure of 2,250 psia for any of the uprating cases. The hot-leg temperature (T_{hot}) defined by the vessel outlet temperature on the design parameters for the power uprate is a maximum of 606.8°F, which is the same as the temperature defined for the RSG Project. Since none of the temperatures exceeds the

previously considered temperature, and the pressure does not change, the uprated design parameters are bounded by the original and RSG analyses.

From Table 5.4-1, the present (RSG) conditions provide an RCS T_{hot} range of 586.3°F to 606.8°F, compared to 590.8°F to 606.8°F for the Power Uprating Program. Therefore, the present RSG operation range bounds the range of T_{hot} for the 7.4% power uprate.

5.4.3.2 Transient Discussion

The only hot-leg transients that have been modified to become more severe for the uprating are the large-step decrease and the reactor trip from full power. For the large-step decrease, the change in temperature for the low-temperature operating condition becomes -78.5°F. For the RSG evaluation, the controlling temperature change for this transient was -77°F for the high-temperature operating condition. For the reactor trip transient, the temperature change for the low temperature operating case becomes -68.8°F. For the RSG evaluation, as well the low-temperature operating condition, the controlling temperature change for this transient was -64.8°F. These changes are addressed in Section 5.4.5.

There are no changes in the pressure transients associated with these system transients.

5.4.4 Acceptance Criteria

The acceptance criteria for the ASME Code structural analysis of the CRDM pressure boundary are that the analyzed stresses do not exceed the stress allowables of the ASME Code and that the cumulative usage factors from the Code fatigue analysis remain less than 1.0.

When the RSG evaluation was performed, the stresses and the cumulative usage factors were increased where changes to the design transients indicated that an increase was necessary. However, where changes to the design transients would have allowed a decrease in stresses or cumulative usage factors, no decrease was calculated, and no credit was taken for such a decrease. Therefore, the CRDM components are still acceptable for the new NSSS design transients if the new NSSS design transients are bounded by either the original NSSS design transients, or the NSSS design transients used in the RSG evaluation. If the NSSS design transients are shown to be bounded by those considered for either the RSG Project or the original design, then the stresses and cumulative usage factors calculated for the CRDMs for the RSG Project remain bounding and applicable for the Power Uprating Program. Since the

CRDMs were acceptable for the RSG Project, the CRDMs remain acceptable for the Kewaunee Power Uprating Program.

5.4.5 Results

The only difference from the previous (RSG) evaluation and the 7.4% power uprate., as discussed in subsection 5.4.3.2, is the modification of the large-step decrease and the reactor-trip transients.

The operating temperature and pressure discussion presented above showed that the operating pressures and temperatures were bounded by those considered for the RSG Project.

In the original evaluation, the large-step decrease had a temperature change of -84°F, and the reactor trip had a temperature change of -78°F (-200°F for the reactor trip – rod drop). These values are more severe than the -78.5°F and -68.8°F temperature changes defined for these transients for the power uprate. As discussed in Section 5.4.4, the stresses and cumulative usage factors calculated for the RSG Project bound both the RSG Project and the original analysis conditions. Since the current power uprating design parameters and the NSSS design transients are all bounded by the original evaluation inputs or the RSG evaluation inputs, the 7.4% power uprate operating parameters are acceptable for the CRDMs.

A summary of the results of the analysis performed for the RSG program is presented in Tables 5.4-2 and 5.4-3. The highest recalculated stresses, as compared to the associated allowables, are presented in Table 5.4-2 for the upper, middle, and lower joints of the CRDM pressure boundary. The cumulative usage factors that were recalculated for the RSG program are given in Table 5.4-3. It is noted that the highest cumulative usage factor, 0.996 at the upper joint canopy, was calculated in a conservative manner where the applied transients were grouped for analysis and the allowable number of cycles considered for each group was based on the most severe transient in the group. These results are applicable to KNPP power uprate operation.

5.4.6 Conclusions

The design parameters and NSSS design transients for the 7.4% power uprate have been shown to be bounded by the parameters and transients considered for either the RSG Project or the original design analysis. Therefore, the conclusions of the RSG Project evaluation are still valid and applicable to the power uprate. The CRDMs are acceptable from a structural

standpoint. The CRDM pressure boundary parts still satisfy the ASME Code of record. Therefore, the evaluation results for the Power Uprating Program are consistent with and continue to comply with the current licensing basis/acceptance requirements for Kewaunee.

5.4.7 References

1. *ASME Boiler and Pressure Vessel Code*, Section III, "Rules for Construction of Nuclear Vessels", 1965 Edition through Summer 1996 Addenda, The American Society of Mechanical Engineers, New York.

2. *Kewaunee RSG – Final Licensing Report Submittal*, November 29, 2000.

Table 5.4-1 Temperatures Used to Bracket All Operating Conditions for Kewaunee 7.4 Percent Power Uprate				
	Present		Uprating	
Parameter	High T_{avg}	Low T_{avg}	High T_{avg}	Low T_{avg}
T_{hot}	606.8°F	586.3°F	606.8°F	590.8°F

Table 5.4-2 Highest Stresses, Compared to Allowables, for CRDM Joints, as Recalculated for the RSG Project and Remaining Applicable for Kewaunee 7.4 Percent Power Uprate			
	Stress (psi)		
CRDM Joint and Component	Value Calculated for the RSG Project	Allowable Value	Source of the Values
Upper Joint Threaded Area	19,962	20,620	Reference 2, page 5.3-11, Table 5.2-3
Middle Joint Canopy	39,223	45,900	Reference 2, page 5.3-11, Table 5.2-3
Lower Joint Canopy	38,269	45,900	Reference 2, page 5.3-11, Table 5.2-3

Table 5.4-3 Cumulative Usage Factors for CRDM Joints, as Recalculated for the RSG Project and Remaining Applicable for Kewaunee 7.4 Percent Power Uprate			
	Cumulative Usage Factor		
CRDM Joint and Component	Value Calculated for the RSG Project	Allowable Value	Source
Upper Joint Canopy	0.996	1.00	Reference 2, page 5.3-5
Upper Joint Canopy Weld	0.619	1.00	Reference 2, page 5.3-6
Upper Joint Threaded Area	0.091	1.00	Reference 2, page 5.3-6
Middle Joint Canopy Weld	0.088	1.00	Reference 2, page 5.3-7
Lower Joint Canopy Weld	0.083	1.00	Reference 2, page 5.3-7

5.5 RCL Piping and Supports

5.5.1 Reactor Coolant Loop Piping

5.5.1.1 Introduction

The parameters associated with the 7.4 percent Power Uprate Project, which also support the KNPP Reload Transition Safety Report (RTSR) Uprate Program, were reviewed for impact on the existing design basis replacement steam generator (RSG) reactor coolant loop (RCL) analysis for the following components:

- RCL piping stresses and displacements
- Primary equipment nozzle loads
- Primary equipment support loads
- Pressurizer surge line piping stresses and displacements including the effects of thermal stratification
- Impact on RCL branch nozzle loads
- Impact on Class 1 auxiliary piping systems

5.5.1.2 Inputs and Assumptions

The following significant assumptions are considered for the evaluation of the RCL for the 7.4 percent Power Uprate Program:

- There is no change to the input response spectra due to the uprate, the response spectra utilized in the current design basis is still applicable
- The are no changes to the primary equipment support stiffnesses due to the uprate, stiffnesses utilized in the current design basis are still applicable.
- The breaks at the loop branch nozzle connections are assumed to envelope all breaks in the auxiliary branch lines. KNPP has been licensed for Leak-Before-Break on the main loop piping and the pressurizer surge line. However, Hydraulic Forcing Functions (HFF) utilized in the loss-of-coolant accident (LOCA) evaluations for the 7.4 percent power uprate are

conservatively based on breaks at the 10-inch surge line nozzle on the hot leg, and at the 12-inch accumulator line nozzle on the cold leg.

- The analysis methodologies and criteria utilized in the original design basis analysis and the RSG program continue to be applicable for the uprate.

The following three basic sets of input parameters were considered in the evaluation:

- Nuclear Steam Supply System (NSSS) design parameters shown in Table IV.B-1 of Attachment 2.
- NSSS design transients
- Loss-of-coolant accident (LOCA) hydraulic forcing functions loads and associated reactor pressure vessel (RPV) motions

The acceptance criteria for the RCL analysis are as specified in USAS B31.1 Power Piping Code, 1967 Edition (Reference 3).

The acceptance criteria for the pressurizer surge line thermal stratification analysis are as specified in ASME B&PV Code Section III, Subsection NB, 1986 Edition, (Reference 4).

The acceptance criteria for the Primary Equipment supports are as specified in AISC Specification for the Design, Fabrication & Erection of Structural Steel Buildings, 1969 (Reference 5).

The parameters associated with the 7.4% power uprate were reviewed for impact on the existing RCL piping and subsequent impact to the RCL branch nozzles and the Class 1 auxiliary lines attached to the RCL. The conclusions of this review are summarized below.

NSSS Design Parameters

The NSSS design parameters as identified in Table IV.B-1 were compared with the design basis thermal analysis for the RCL. The changes in these parameters for the 7.4% uprate, as compared to the corresponding design parameters for the RSG program are very small (See Table 5.5.1-1). The RCL is evaluated for two temperature cases as identified in the table. These two cases bound the 100% power operating window as defined by the uprated parameters.

Therefore, the impact on the RCL analysis is determined to be insignificant. It is also noted that, because maximum and minimum temperatures for the uprate are bounded by those of the RSG, there is no adverse impact to the primary equipment support thermal gaps and shims. The shimming of the primary equipment supports performed for the RSG Program is still applicable. Additionally, the temperatures in the current design basis analysis of the pressurizer surge line envelopes the temperatures for the uprate. Therefore, the design basis results summarized in Reference 2 for the pressurizer surge line remain applicable.

NSSS Design Transients

The impact on design transients due to the changes in full-power operating temperatures for the 7.4% power uprate was evaluated. For all primary side components seeing the hot leg temperature (T_{hot}, based on the high T_{avg} condition) and cold leg temperature (T_{cold}, based on the low T_{avg} condition), the limiting cases remain unchanged from the Steam Generator Replacement (SGR) Program transients. Additionally, per Reference 1, the design criteria for the RCL piping is USAS B31.1 Power Piping Code, 1967 Edition; thus no fatigue analysis is required for the RCL.

For the pressurizer surge line, the impact of the design transients with respect to the thermal stratification and fatigue analysis is controlled by the ΔT between the pressurizer temperature and the hot leg temperature. The SGR program had the larger temperature window for design purposes and the limiting full power T_{hot} and T_{cold} values for the uprating are bounded by the ones used in the SGR Program. Therefore, the 7.4-percent uprate has no adverse impact on either the thermal stratification or the fatigue analysis for the pressurizer surge line and the results in Reference 2 remain valid.

LOCA Hydraulic Forcing Functions Loads And Associated RPV Motions

The impact on the loss-of-coolant accident (LOCA) hydraulic forcing functions due to the 7.4% power uprate was evaluated. Leak-Before-Break is applicable for the RCL main loop piping and the pressurizer surge line (refer to Section 5.5.2). However, it is noted that the RCL is evaluated for LOCA using Hydraulic Forcing Functions (HFF) generated for the uprate program conservatively based on breaks at the 10-inch surge line nozzle on the hot leg, and at the 12-inch accumulator line nozzle on the cold leg. Additionally, RPV motions corresponding to the surge line break and accumulator line break are also included.

Additionally, the RCL was also evaluated for secondary side breaks at the main steam line and feedwater line terminal end nozzle locations at the steam generator. Based on the design parameters identified in Table IV.B-1, the secondary side breaks at the main steam line and feedwater line terminal end nozzle locations remain bounded by the evaluation performed for the SGR Program due to the reduction in Steam Generator operating pressure.

5.5.1.3 Analysis Methods

As previously defined in Reference 1, the system analysis of the RCL piping was performed using program WESTDYN for all applicable deadweight, thermal expansion, seismic, and loss-of-coolant-accident (LOCA) loads. The analysis of the RCL primary equipment supports was performed using program STADD-III.

5.5.1.4 Reactor Coolant Loop Piping Analysis and Results

The evaluation of the RCL includes performing deadweight, thermal expansion, seismic, and loss-of-coolant-accident (LOCA) analyses.

The deadweight analysis is not impacted by the uprate. Therefore the evaluation as performed for the RSG program is still applicable.

The thermal analysis considered the range of operating temperatures for 100% power. The RCL Primary equipment supports were shimmed to accommodate the range of operating temperatures as specified by the RSG program. Because these temperature ranges bound corresponding uprated temperature ranges, the shimming performed for the RSG program is not impacted by the uprate program.

The seismic analysis methods and input response spectra are not impacted by the 7.4% uprate parameters. Seismic analysis for the RSG program was performed for the operating basis earthquake (OBE) and a factor of 2.0 applied to obtain safe shutdown earthquake (SSE) conditions. The seismic analyses considered multiple cases based on various primary equipment support activity, and accounted for the range of operating temperatures.

The LOCA analysis for the RCL was performed considering time history hydraulic forces distributed throughout RCL system. The analysis was performed for the breaks at the auxiliary nozzles for the 8-in RHR line on the hot leg, and the 12-inch accumulator line on the cold leg.

As previously noted, KNPP has been licensed for Leak-Before-Break on the main loop piping and the pressurizer surge line. However, the Hydraulic Forcing Functions (HFF) utilized in the LOCA evaluations for the 8-inch RHR line are conservatively based on a 10-inch surge line break for the 7.4 percent Power Uprate Program. The LOCA analyses considered multiple cases based on various primary equipment support activity, and accounted for the range of operating temperatures for the uprated parameters.

Based on these evaluations for the uprated NSSS design parameters, NSSS design transients, LOCA hydraulic forcing functions and associated RPV motions; it is concluded that 7.4 percent uprate does not adversely affect the current design basis RSG Reactor Coolant Loop (RCL) analyses.

The maximum RCL piping stresses for the RCL piping and the corresponding Code allowable stress values are presented in Table 5.5.1-2. The stresses are combined in accordance with the methods and criteria described in Reference 1.

The primary equipment nozzle loads are compared to the allowables as defined in the equipment design specifications and the loads previously evaluated for the RSG program. The nozzle loads are acceptable. The uprate program has no adverse impact analysis results.

The applicable reactor coolant loop piping loads resulting from the range of operating temperatures, as defined by the uprated NSSS parameters and bound by the RSG parameters were provided for evaluation and confirmation of Leak-Before-Break (See Section 5.5.2).

The impact on RCL piping displacements at the RCL Branch nozzles and corresponding Class 1 auxiliary piping systems, is insignificant due to the uprated parameters and revised LOCA hydraulics. Therefore, the subsequent impact on the RCL Branch nozzles and the corresponding Class 1 piping systems due to the uprate, including the Primary Sampling System (PSS), Chemical and Volume Control System (CVCS), Residual Heat Removal System (RHR) and Safety Injection System (SIS), is considered negligible. Updated RCL piping and equipment displacements as previously provided to KNPP for the RSG Program remain valid.

Additionally, the current design basis pressurizer surge line analysis including the effects of thermal stratification (Reference 2) is applicable to the 7.4 percent power uprate.

5.5.1.5 Primary Equipment Supports Analysis and Results

The primary equipment supports have been evaluated for the revised loads associated with the 7.4% Power Uprate Program. The equipment supports are evaluated to meet the stress criteria defined in Reference 1. The stress limits correspond to those provided in Reference 6. Support stresses are evaluated for normal, upset, emergency, and faulted conditions.

The equipment support loads associated with the uprate were compared with the loads previously evaluated and qualified for the RSG program. The stress ratios were adjusted, where necessary, to reflect these new loads due to the uprate.

Based on the evaluations, it is concluded that the primary equipment support stresses remain within the acceptance criteria. Stress ratios are less than 1.0 and summarized in Table 5.5.1-3.

In addition, it has been determined that building interface loads do not require further evaluation since these loads are enveloped by previously analyzed loads. Therefore, the loads previously used for the RSG program, remain valid.

Furthermore, since the high and low RCL temperatures associated with the 7.4% power uprate are bounded by the high and low temperatures associated with the replacement steam generator, the equipment support thermal gap and shim evaluation performed for the RSG program remains valid.

5.5.1.6 Conclusions

The parameters associated with the 7.4% power uprate have been evaluated for the following components:

- RCL piping stresses and displacements
- Primary equipment nozzle loads
- Primary equipment support loads
- Pressurizer surge line piping stresses and displacements including the effects of thermal stratification
- Impact on RCL branch nozzle loads
- Impact on Class 1 auxiliary piping systems

The evaluation indicates that the parameters associated with the 7.4% power uprate have no adverse effect on the analysis of the RCL piping system, including impacts to the primary equipment nozzles and primary equipment supports. RCL piping stresses meet the required stress criteria as summarized in Table 5.5.1-2. RCL piping loads for leak-before-break (LBB) evaluation for the 7.4% power uprate were evaluated and found to be acceptable (refer to section 5.5.2). RCL primary equipment support loads met the required stress criteria summarized in Table 5.5.1-3.

RCL piping displacements at branch nozzles previously provided for the RSG program remain valid for the uprate. Therefore, the 7.4 percent power uprate has no subsequent impact to either the RCL branch nozzle loads or the Class 1 auxiliary piping systems that are attached to the RCL.

Additionally, the current design basis analysis, for the pressurizer surge line, (Reference 2), including the effects of thermal stratification, are still applicable and remain valid for the 7.4% uprate.

5.5.1.7 References

1. "Structural Analysis of Reactor Coolant Loop/Support System For NSP (Prairie Island) and WPS (Kewaunee) Nuclear Power Plants," WCAP-7840, Report No. SD 103, February 1972.

2. "Structural Evaluation of the Kewaunee Pressurizer Surge Line, Considering the Effects of Thermal Stratification," WCAP-12841, March 1991.

3. USAS B31.1 Power Piping Code, 1967 Edition.

4. ASME B&PV Code, Section III, Subsection NB, 1986 Edition.

5. AISC Specification for the Design, Fabrication & Erection of Structural Steel Buildings, 1969.

6. "Specification for the Design, Fabrication, & Erection of Structural Steel Buildings," American Institute of Steel Construction, 2/12/69.

TABLE 5.5.1-1 Comparison of Design Parameters – 7.4 percent Uprate vs. RSG			
	Hot Leg	Crossover Leg	Cold Leg
Thermal Case	Degrees F	Degrees F	Degrees F
7.4% Uprate – Case 1 & 2	590.8	521.6	521.9
7.4% Uprate – Case 3 & 4	606.8	538.9	539.2
RSG Case 1 & 2	586.3	521.6	521.9
RSG Case 3 & 4	606.8	543.6	543.8
RCL Thermal Analysis			
Thermal Case A (1) (High T_{avg})	612.0	548.0	548.0
Thermal Case B (2) (Low T_{avg})	590.8	521.6	521.6

(1) Case A Original design basis thermal, envelopes High T_{avg} Thermal case

(2) Case B: represents low T_{avg} Thermal case

TABLE 5.5.1-2 Reactor Coolant Loop Stress Analysis Summary – 7.4 percent Uprate Program						
	Hot Leg		Crossover Leg		Cold Leg	
Stress Condition	Maximum ksi	Allowable ksi	Maximum ksi	Allowable ksi	Maximum ksi	Allowable ksi
Equation 9 design stress (ksi) (DW, P)	6.650	14.950	6.350	14.950	6.220	14.950
Allowable Stress Limit	(1.0 S)		(1.0 S)		(1.0 S)	
Eq 9 upset stress (ksi) (DW, P, OBE)	7.070	17.940	6.494	17.940	6.762	17.940
Allowable Stress Limit	(1.2 S)		(1.2 S)		(1.2 S)	
Eq 9 faulted stress (ksi) (DW, P, SSE, LOCA)	18.131	26.910	17.845	26.910	20.129	26.910
Allowable Stress Limit (1)	((1.5 x 1.2S))		((1.5 x 1.2S))		((1.5 x 1.2S))	
Thermal (ksi)	12.600	25.612	2.410	25.612	3.576	25.612
	(1.25 x Sc + 0.25 x Sh)		(1.25 x Sc + 0.25 x Sh)		(1.25 x Sc + 0.25 x Sh)	

Notes:

DW = Deadweight

P = Pressure

(1) Faulted stresses conservatively evaluated against Emergency limits.

Table 5.5.1-3

Summary of RSG and RCP Support Member Stress Ratios – 7.4 percent Uprate Program

Support	Support Item	Normal	Upset	Emergency	Faulted
SGLS Vertical	SGLS columns	0.48	0.53	0.44	0.50
	SGLS column foot bolt modification	0.79	0.85	0.92	0.83
	SGLS column adapter modification	0.73	0.76	0.53	0.76
SGLS Lateral	SGLS bumpers	N/A	0.02	0.03	0.30
	SGLS bumper guide mod.	N/A	0.07	0.09	0.86
	SGLS cross-compartment beams	N/A	0.02	0.03	0.32
SGUS Lateral	SGUS snubber	N/A	0.12	0.22	0.71
	SGUS bumpers	N/A	0.02	0.05	0.18
SGUS Ring Girder	SGUS ring girder	N/A	0.08	0.11	0.51
	SGUS ring girder splice plate mod.	N/A	0.11	0.21	0.93
RCP Vertical	RCP columns	0.42	0.47	0.38	0.90
RCP Lateral	RCP tie rods	N/A	0.14	0.23	0.65
	RCP tie rods bracket modification	N/A	0.30	0.42	0.80

NOTES : SGLS = Steam Generator Lower Support

SGUS = Steam Generator Upper Support

All stress ratios are less than 1.0 and are therefore acceptable.

5.5.2 Application of Leak-Before-Break (LBB) Methodology

The current structural design basis of Kewaunee Nuclear Power Plant (KNPP) includes the application of leak-before-break (LBB) methodology to eliminate consideration of the dynamic effects resulting from pipe breaks in the Reactor Coolant System (RCS) primary loop piping and the pressurizer surge line piping. This section describes the analyses and evaluations performed to demonstrate that the elimination of these breaks continues to be justified at the operating conditions associated with the 7.4-percent Power Uprate Program.

5.5.2.1 Reactor Coolant System Primary Loop Piping

5.5.2.1.1 Introduction

Westinghouse performed analyses for the LBB of KNPP primary loop piping in 1987. The results of the analyses were documented in WCAP-11411 (Reference 1) and WCAP-11619 (Reference 2) and approved by the NRC (Reference 3). Westinghouse also performed analyses to support Steam Generator Replacement (SGR) and also performed evaluation for the SGR shim support gap. The results of the SGR analysis were documented in WCAP-15311 (Reference 4).

To demonstrate the elimination of Reactor Coolant System (RCS) primary loop pipe breaks, the following objectives were achieved:

- Margin exists between the "critical" crack size and a postulated crack that yields a detectable leak rate.
- There is sufficient margin between the leakage through a postulated crack and the leak detection capability.
- Margin exists on the applied load.
- Fatigue crack growth is negligible.

To support the 7.4% power uprate at KNPP, the current LBB analyses was updated to address the power uprating conditions.

5.5.2.1.2 Input Parameters and assumptions

The loadings, operating pressure, and temperature parameters for the uprating were used in the re-evaluating LBB.

The parameters, which are important in the evaluation, are the piping forces, moments, normal operating temperature, and normal operating pressure.

5.5.2.1.3 Description of Analyses and Evaluations

The recommendations and criteria proposed in Reference 7 are used in this evaluation. The primary loop piping dead weight, normal thermal expansion, and Safe Shutdown Earthquake (SSE) and pressure loads due to the 7.4% power uprate were used. The normal operating temperature and pressure for the 7.4-percent uprate were used in the evaluation. The evaluation showed that all the LBB recommended margins were satisfied. The margins from Reference 7 are described below.

5.5.2.1.4 Acceptance Criteria and Results

The LBB acceptance criteria is based on the SRP 3.6.3 (Reference 7). The recommended margins are as follows:

- Margin of 10 on leak rate.
- Margin of 2.0 on flaw size
- Margin on loads of 1.0 (Using faulted load combinations by absolute summation method).

The evaluation results showed the following at all the critical locations:

Leak Rate- A Margin of 10 exists between the calculated leak rate from the leakage flaw and the leak detection capability of 1 gpm.

Flaw size- A margin of 2.0 or more exits between the critical flaw and the flaw having a leak rate of 10 gpm (the leakage flaw).

Loads- A margin of 1.0 on loads exists.

The evaluation results show that the LBB conclusions provided in References 1 through 2 for KNPP remain unchanged for the uprate conditions.

5.5.2.1.5 Conclusions

The LBB acceptance criteria are satisfied for the KNPP primary loop piping at the 7.4% power uprate. All the recommended margins are satisfied and the conclusions shown in References 1 and 2 remain valid. It is therefore concluded that the dynamic effects of RCS primary loop pipe breaks need not be considered in the structural design basis at the uprated conditions.

5.5.2.2 KNPP Pressurizer Surge Line Piping

The KNPP pressurizer surge line analysis for the application of LBB was documented in WCAP-12875 (Reference 6) and approved by the NRC (Reference 7). Section 5.5.1.4 has indicated that the current design basis pressurizer surge line analysis results including the effects of thermal stratification are applicable for the uprate. Therefore the conclusions of the previous LBB analysis shown in Reference 6 for KNPP for the pressurizer surge line also remain valid for the 7.4% power uprate.

It is therefore concluded that the dynamic effects of the pressurizer surge line pipe break need not be considered in the structural design basis of KNPP at the 7.4% power uprate conditions.

5.5.2.3 LBB analyses for the auxiliary lines

LBB analyses performed for the auxiliary lines are applicable for the 7.4-percent Power Uprate conditions.

5.5.2.4 References

1 "Technical Bases for Eliminating Large Primary Loop Pipe Rupture as the Structural Design Basis for Kewaunee," WCAP-11411, Rev. 1, April 1987.

2 "Additional Technical Bases for eliminating Large Primary Loop Pipe Rupture as the Structural Design Basis for Kewaunee," WCAP-11619, October 1987.

3 NRC Docket No. 50-305," Application of Leak-Before-Break Technology as a Basis for Kewaunee Nuclear Power Plant Steam Generator Snubber Reduction," Dated February 16, 1988.

4 WCAP-15311," Technical Justification for eliminating Large Primary Loop Pipe Rupture as the Structural Design Basis for the Kewaunee Nuclear Power Plant after SG Replacement," June 2000.

5 Not Used.

6 Technical Justification for Eliminating Pressurizer Surge Line Rupture as the Structural Design Basis for Kewaunee Nuclear Plant," WCAP-12875, June 1991.

7 NRC Docket No. 50-305," Kewaunee Nuclear Power Plant: Leak-Before-Break Evaluation of Pressurizer Surge Line (TAC No. M72140)," Dated January 3, 1992.

5.6 Reactor Coolant Pumps

The reactor coolant pumps (RCPs) at Kewaunee were evaluated for the 7.4% power uprate in two separate areas: the structural adequacy of the pumps (subsection 5.6.1), and the acceptability of the RCP motors (subsection 5.6.2).

5.6.1 Reactor Coolant Pumps (Structural)

5.6.1.1 Introduction

This section addresses the ASME Code structural considerations for the pressure boundary components of the Westinghouse Model 93A RCPs. The Kewaunee RCP equipment specification requires that the design analysis, materials, welding, inspection, and testing of the pumps be per the requirements of the ASME Code, Section III, 1968 Edition or later (Reference 1). The RCPs predate the inclusion of pumps in the ASME Code, Section III, and are not Code stamped. As the Code is used for guidance, Code addenda and Editions later than the Code in effect at the time of the order are used for the analyses. The Code editions used in the generic stress reports applicable to the Kewaunee pumps range from the 1968 Edition with Winter 1970 Addenda, to the 1971 Edition with Winter 1972 Addenda.

The RCPs were evaluated for the uprate design parameters (Attachment 2, Table IV.B-1) and the Nuclear Steam Supply System (NSSS) design transients assuming a 7.4% increase in core power.

Continued adequacy of the structural supports for the reactor coolant pumps is addressed in Section 5.5.1.

A structural evaluation of the pump non-pressure components was performed. The analyses are independent of the design parameters and the NSSS design transients. The analyses remain valid for the uprate conditions.

5.6.1.2 Input Parameters and Assumptions

The Model 93A RCPs were originally designed and analyzed to meet the requirements of the equipment specification and the ASME Code (Reference 1). A later evaluation of the RCPs was

performed for the Steam Generator Replacement (RSG) Project. The RSG Project evaluation supplements the original evaluation and is contained in Reference 2.

The current power uprate modifies the design parameters (Attachment 2, Table IV.B-1) and the NSSS design transients that were considered in these previous evaluations. Seismic loadings, nozzle loadings on main and auxiliary nozzles, and auxiliary nozzle transients are either unchanged or remain bounded by the original or the RSG Project basis.

The RCPs are installed in the Reactor Coolant System (RCS) cold leg, between the steam generator outlet and the reactor vessel inlet. The temperatures and pressures used as inputs to the RCP Code structural analysis are those defined for the reactor vessel inlet in Attachment 2, Table IV.B-1. The reactor coolant pumps must be evaluated for the NSSS design transients, as defined for the RCS cold leg by the equipment specification, and updated by the RSG Project and this Power Uprating Program.

The differences associated with the uprating requirements are discussed in subsection 5.6.1.3.

5.6.1.3 Description of Analysis

5.6.1.3.1 Operating Temperature and Pressure

The current reactor coolant pressure of 2,250 psia is used for all the uprate cases. The RCS cold-leg temperature (T_{cold}), defined by the vessel inlet (RCP outlet) temperature is a maximum of 539.2°F. The maximum uprating RCS T_{cold} is less than the corresponding RSG Project T_{cold} temperature of 543.8°F, or the lowest temperature considered in the original analyses, 552°F. Since none of the temperatures exceeds the previously considered temperatures, and the pressure does not change, the uprate design parameters are bounded by those used as inputs to the original and RSG analyses.

Table 5.6-1 summarizes the cold-leg uprate temperatures. From Table 5.6-1, the present (RSG) conditions provide an RCS T_{cold} range of 521.9°F to 543.8°F, compared to an RCS T_{cold} range of 521.9°F to 539.2°F for the power uprate. Therefore, the present operation range bounds the RCS T_{cold} range for the Power Uprating Program.

5.6.1.3.2 Transient Discussion

Several of the cold-leg transients defined for the uprate have small increases in the temperature change associated with the transient, when compared to the NSSS cold leg design transients defined for the RSG Project. These transients are summarized in Table 5.6-2. The unit-loading/unloading, loss-of-load, loss-of-power, loss-of-flow, and reactor-trip-from-full-power transients all show increases in the amount of temperature change. In no case does the increased temperature change differ from the temperature change defined for the RSG Project by more than 1.3°F. The largest temperature change associated with any of these transients is 47.1°F for the loss-of-power transient. These changes are shown to be acceptable for the power uprate in Section 5.6.1.5.

There are no changes in the pressure transients associated with these system transients.

5.6.1.4 Acceptance Criteria

The acceptance criteria for the ASME Code structural analysis of the rector coolant pump pressure boundary are that the analyzed stresses do not exceed the stress allowables of the ASME Code and that the cumulative usage factors from the Code fatigue analysis remain less than 1.0.

When the RSG evaluation was performed, no increases were required in the stresses that had been determined in the original generic stress analyses (Reference 2). This is because the stresses associated with the temperature and pressure changes in the original analyses bounded the stresses associated with temperature and pressure changes defined for the RSG Project. This occurs because, in the original analyses, the design transients were grouped and only the most severe transient in each group was analyzed. The total number of transient cycles for all transients in the group was considered for each group, along with the worst-case stresses. This was done to reduce the number of analyses required

The cumulative usage factors were increased where changes to the design transients indicated that an increase was necessary. The fatigue usage factors were increased for the RSG Project due to the addition of two transients that were not considered in the original analyses. However, where changes to the design transients would have allowed a decrease in stresses or cumulative usage factors, no decrease was calculated, and no credit was taken for such a decrease. Therefore, the RCP components are still acceptable for the new NSSS design

transients if the new NSSS design transients are bounded by either the original NSSS design transients, or the NSSS design transients used in the RSG evaluation. If the NSSS design transients are shown to be bounded by those considered for either the RSG Project or the original design, then

- The cumulative usage factors calculated for the RCPs for the RSG Project remain bounding and applicable for the power uprate, and
- The stresses calculated in the original design analyses remain bounding and applicable for the power uprate.

Since the RCPs were acceptable for the RSG Project, the RCPs remain acceptable for the 7.4% power uprate.

5.6.1.5 Results

The differences between the previous (RSG) evaluation and the power uprate, as discussed in subsection 5.6.1.3.2, are some modifications to the temperature changes associated with a few of the NSSS design transients.

The operating temperature and pressure discussion presented above in subsection 5.6.1.3.1 showed that the operating pressures and temperatures are bounded by those considered for the RSG Project.

The temperature transients with slight increases in the temperature differences for the power uprate, shown in Table 5.6-2, are all transients that were considered in the same grouping in the original design reports. The applicable transients used in the original design reports to represent this transient group considered a temperature change of at least 52.5°F. This is larger than the maximum temperature change of 47.1°F defined for any of the uprate transients that showed an increase in temperature change from the RSG Project transients. Since the uprate design parameters and the NSSS design transients are all bounded by the original evaluation inputs or by the RSG evaluation inputs, the operating parameters are acceptable for the RCPs.

As a summary of the analysis performed for the RSG Project, the cumulative usage factors that were recalculated or verified for the RSG Project are presented in Table 5.6-3. It is noted that the fatigue waiver, allowed by Article 4, paragraph N-415.1, of the 1968 Code (Reference 1) or

by NB-3222.4 (d) of the 1971 Code, is cited for many of the components of the RCPs. This indicates that, by the Code rules, a detailed fatigue calculation leading to a value of the cumulative usage factor is not required for those components.

5.6.1.6 Conclusions

The uprate design parameters and NSSS design transients are bounded by the parameters and transients considered for either the RSG Project or the original design analyses. Therefore, the conclusions of the RSG Project evaluation are still valid and applicable to the 7.4% power uprate. The RCPs are acceptable from a structural standpoint. The RCP pressure boundary parts still comply with the ASME Code originally specified or later editions. Therefore, the evaluation results of the Power Uprating Program for the RCPs are consistent with and continue to comply with the current licensing basis/acceptance requirements for Kewaunee.

5.6.1.7 References

1. *ASME Boiler and Pressure Vessel Code*, Section III, "Rules for Construction of Nuclear Vessels", 1968 Edition, and later Editions and Addenda, The American Society of Mechanical Engineers, New York.

2. *Kewaunee RSG – Final Licensing Report*, November 29, 2000.

Table 5.6-1 Design Conditions Used to Bracket All Operating Conditions for Kewaunee 7.4% Power Uprate Program				
Parameter	Present		7.4% Uprate Program	
	High T_{avg}	Low T_{avg}	High T_{avg}	Low T_{avg}
T_{cold} (vessel inlet)	543.8°F	521.9°F	539.2°F	521.9°F

Table 5.6-2 Cold Leg Thermal Transient Summary for RCP Evaluation for Kewaunee 7.4% Power Uprate Program				
	7.4% Uprate Program		RSG Project[1]	
	Max ΔT (°F) High T_{avg}	Max ΔT (°F) Low T_{avg}	Thermal Transient (°F)	Max ΔT (°F)
Normal Condition				
Unit Loading/Unloading	8.1	**25.4**	546.5-522-546.5	24.5
Upset Condition				
Loss of Load	37.3	**44.4**	521.9-565-560 543.8-581.1-560.5 (See note 1)	43.1 37.3 (See note 1)
Loss of Power	22.2	**47.1**	521.9-568-558	46.1
Loss of Flow (no RCP)	18.8	**30.4**	522-517-547	30.0
Loss of Flow (w/ RCP)	14.6	25.7		
Reactor Trip from Full Power	**13**	5	543.8-555-543.3	11.7

Notes:

1. These transients include both the high- and low-operating temperatures within the RCS T_{avg} window, with the exception of the loss-of-load transient. For the loss-of-load transient, the first transient listed is for the high-temperature case, and the second transient listed is for the low-temperature case.
2. See the discussion in Section 5.6.1.5 to establish the acceptability of these transients.

Table 5.6-3 RCP Fatigue Evaluation for Kewaunee RSG Project and Kewaunee 7.4% Power Uprate Program		
RCP Component	Cumulative Usage Factor from Original Analyses	Cumulative Usage Factor RSG Project and 7.4% Power Uprate Program
Casing	Fatigue Waiver	Fatigue Waiver
Main Flange	Fatigue Waiver	Fatigue Waiver
Thermowell	Negligible	0.076
Main Flange Bolts	0.67	0.746
Thermal Barrier Flange	0.0002	0.076
Suction/Discharge Nozzles	Fatigue Waiver	Fatigue Waiver
Casing Feet	Fatigue Waiver	Fatigue Waiver

5.6.2 Reactor Coolant Pump Motors

5.6.2.1 Introduction

This section addresses the performance of the reactor coolant pump (RCP) motors. The RCP motors are evaluated for uprate design parameters (Attachment 2, Table IV.B-1) and best estimate flows which assumed a 7.4 percent increase in core power.

5.6.2.2 Input Parameters and Assumptions

The input parameters considered in the evaluation of the RCP motors are the steam generator outlet temperatures (Section 2) and the best estimate flows defined for the Kewaunee Nuclear Power Plant (KNPP) 7.4% power uprate. These parameters are considered for the three Model 93A reactor coolant pumps at KNPP (two installed pumps and one spare).

5.6.2.3 Description of Analysis

The steam generator outlet temperatures and best estimate flows are considered in a hydraulic analysis using the operating characteristics of the Kewaunee reactor coolant pumps. This hydraulic analysis calculates the power requirements for the impeller that operates at the highest cold power. For the 7.4% Power Uprate, the power requirements from this analysis for hot loop operation and for cold loop operation were compared to the power requirements considered for the RSG evaluation contained in Reference 1. Based on a negligible increase in the power requirements between the power uprating and the RSG Project, the RCP motors are considered acceptable for the Power Uprating based on the RSG Project evaluation.

The RSG Project report (Reference 1) presented evaluations of the RCP motor loading in four areas. These areas were

- continuous operation at hot-loop temperatures and flows,
- continuous operation at cold-loop temperatures and flows,
- starting across the line with a minimum 80% starting voltage, and
- loads on the thrust bearings.

5.6.2.4 Acceptance Criteria

For the power uprate, the acceptance of the RCP motor loading is based on the change from the loading previously evaluated in the RSG Project being negligible. The acceptance criteria used for evaluating the motor loading for the RSG Project were taken from the equipment specifications for the motor.

Per the Equipment Specification, the motor is required to drive the pump continuously under hot-loop conditions without exceeding a stator winding temperature rise of 75°C. This corresponds to the National Electric Manufacturers' Association (NEMA) Class B temperature rise limit in a 50°C ambient temperature.

Per the Equipment Specification, the motor is required to drive the pump for up to 50 hours (continuous) and 3000 hours maximum over the 40-year design life under cold-loop conditions without exceeding a stator winding temperature rise of 100°C. This corresponds to the National Electric Manufacturers' Association (NEMA) guaranteed limit for a Class F winding in a 50°C ambient temperature.

Per the Equipment Specification, the motor is required to start across the line with a minimum 80% starting voltage against the reverse flow of the other pump running at full speed under cold-loop conditions. The limiting component for this type of loading is the rotor cage winding, which has design limits of a 300°C temperature rise on the bars and 50°C temperature rise on the rings.

The thrust bearing loading used for the design of the motor is given in the equipment specification. Performance of the thrust bearings in an RCP motor can be adversely affected by excessive or inadequate loading. The thrust bearing loading for the revised conditions is compared to the design thrust bearing loading to determine continued acceptability.

5.6.2.5 Results

The worst case loads for the RCP motors were calculated for the power uprate operating conditions, considering the revised design parameters (Attachment 2, Table IV.B-1) and best estimate flows. The new worst-case hot-loop load under the revised operating conditions is 5,942 HP. The new worst-case cold-loop load under the revised operating conditions is 7,656 HP. These loadings are not significantly different from the motor loadings of 5940 HP for hot-

loop operation and 7653 HP for cold-loop operation that were previously evaluated for the RSG Project in Reference 1. Thus, the revised motor loadings were judged to be acceptable on the basis of the previous evaluation.

The evaluations of the RCP motors that were the basis of the RSG Project conclusions were as follows:

Continuous Operation at Hot-Loop Conditions

The worst-case hot-loop operating load for the RSG Project was 5940 HP. This is below the nameplate rating of the motor, which is 6000 HP. Since the loading is within the nameplate rating of the motor, it is acceptable without further calculations. It is noted that the power uprate hot-loop loading of 5942 HP is also less than the nameplate rating of 6000 HP.

Continuous Operation at Cold-Loop Conditions

The worst-case cold-loop operating load of 7653 for the RSG Project exceeded the nameplate cold-loop rating of the motor, 7500 HP, by 2%. Testing on duplicate motors has shown a stator temperature rise no greater than 77.7°C at the cold-loop nameplate rating of 7500 HP. Analysis indicated that the cold-loop temperature rise of the stator at the RSG Project loading of 7653 will be approximately 79°C, which is well below the NEMA limit given in Section 5.6.2.4. The cold-loop loading for the power uprate is 7656 HP, an increase of less than 0.04% from the RSG Project loading. The stator temperature will thus also remain well below the NEMA limit for the 7.4% uprate.

Starting

The starting temperature rise of the rotor cage winding for the RSG conditions was calculated using a conservative all-heat-stored analysis. The results of that analysis indicated temperature rises of 226.2°C and 37.21°C, respectively, for the rotor bars and the resistance rings. These did not exceed the design limits given in Section 5.4.2.4. The temperature rise with the insignificant changes in motor loading associated with the power uprate will likewise not exceed the design limits.

Thrust Bearing Loading

The thrust bearing loadings for the RSG Project conditions indicated a reduction in thrust bearing load of 1923 pounds for hot-loop operation and a reduction in thrust bearing load of 4578 pounds for cold-loop operation. In comparison to the normal operating thrust bearing load of 104,400 pounds given in the equipment specification, these changes were not considered significant and the thrust bearings were considered acceptable for the RSG Project loads. The insignificant changes in motor loading associated with the uprate are also considered acceptable for the thrust bearings.

5.6.2.6 Conclusions

The RCP motors were previously evaluated in four areas for the RSG Project conditions under loadings of 5,940 HP for worst-case hot-loop operation, and 7,653 HP for worst-case cold-loop operation. Since the new RCP motor loads show no significant change from the bounding loads considered in the RSG analysis, no further evaluation is required. The RCP motors are still considered acceptable for the 7.4% power uprate.

5.6.2.7 References

1. *Kewaunee RSG – Final Licensing Report Submittal*, November 29, 2000.

5.9 NSSS Auxiliary Equipment

5.9.1 Introduction

This section discusses evaluations performed for the KNPP auxiliary tanks, heat exchangers, pumps, and valves impacted by the thermal transients and maximum operating temperatures, pressures, and flow rates associated with the 7.4% power uprate. The systems affected by the uprate include are the Reactor Coolant System (RCS), Chemical and Volume Control System (CVCS), Safety Injection System (SIS), Residual Heat Removal (RHR) System, and the Component Cooling Water (CCW) System. The evaluation consists of a structural fatigue review and flow capacity review of the component pressure boundaries. The review does not include a structural evaluation or a performance/controllability evaluation of the sub-components for any of the components discussed in this report (e.g., valve actuators, controllers, electronics, or pump motors) unless specifically noted.

5.9.2 Input Parameters and Assumptions

The auxiliary system heat exchangers and tanks evaluated for uprate are listed in Tables 5.9-1 and 5.9-2. The auxiliary system pumps are listed in Table 5.9-3. All valves originally supplied by Westinghouse were also evaluated but are not tabulated in this submittal because of the large number. Component design information from equipment specifications, drawings, and data sheets includes pressure and temperature design conditions as well as design transients applicable to each individual identified component. The uprate design parameters are listed in Attachment 2, Table IV-B-1. This information was applied where applicable for evaluation of the auxiliary equipment maximum operating temperatures and pressures. Evaluations were performed to determine the impact of the NSSS transients on the auxiliary tanks, heat exchangers, pumps, and valves at the uprated conditions.

5.9.3 Description of Analyses and Evaluations

The design parameters were reviewed for the auxiliary tanks, heat exchanger, pumps, and valves. The specific criteria included design temperature, pressure, thermal transients, and flow rates. These parameters were compared to those used in the power uprate to determine if the design parameters still enveloped those for the uprating. The uprated design requirements were compared, as applicable to the equipment, with the original design requirements to determine the acceptability of the components for the uprating conditions.

5.9.3.1 Auxiliary System Tanks

The SIS accumulators are the only tanks affected by the transients. The evaluation of operating temperatures and pressures for these vessels remain within the design basis. Therefore, the safety injection accumulators remain bounded by the original design transients. As a result, none of the auxiliary tanks are impacted by the uprating conditions.

Note that the pressurizer relief tank sparger is not included in this evaluation because it is not impacted by the power uprate.

5.9.3.2 Auxiliary System Heat Exchangers

The NSSS auxiliary heat exchangers evaluated for the uprating conditions are listed in Table 5.9-1. The equipment specifications for these heat exchangers identify the applicable design transients, and the data sheets identify the design temperature and pressures.

Based on the uprate design parameters, there is no impact on the auxiliary systems heat exchangers listed in Table 5.9-1. The operating temperature and pressure ranges for these vessels remain bounded by the original design parameters. Evaluations indicate that the original design transients for the auxiliary equipment bound the transients associated with the uprate. The heat exchangers identified as having transients in the original design specifications are the regenerative, letdown, excess letdown, and residual heat removal (RHR) heat exchangers. All of these temperatures remain bounded by the original design conditions.

5.9.3.3 Auxiliary System Pumps

The NSSS auxiliary pumps evaluated for the uprated conditions are listed in Table 5.9-3. There is no impact on the auxiliary system pumps. The operating temperature and pressure ranges for these pumps remain bounded by the original design parameters. Evaluations also indicate that the original design transients for the auxiliary equipment bound the transients associated with the uprating.

5.9.3.4 Auxiliary System Valves

The NSSS auxiliary system valves were evaluated for the uprated conditions.

Evaluations indicate that there is no impact upon the auxiliary systems valves. The operating temperature and pressure ranges for the valves remain bounded by the original design parameters. The original design transients for the auxiliary equipment remain bounded for the transients associated with the power uprating.

5.9.4 Acceptance Criteria

In order to demonstrate the qualification of equipment, the maximum system operating temperatures, pressures, and flow rates for the 7.4% power uprate must be bounded by or equal to the original system design conditions as well as those applicable to the RSG. Then, no further effort would be required to qualify the auxiliary tanks, heat exchangers, pumps, and valves for this aspect of the uprate. Any values in excess of the design values will be addressed in this report.

In addition the original design transients and those applicable to the RSG must bound the revised auxiliary tanks, heat exchangers, pumps, and valve transients, with fatigue usage factors being less then 1.0, then no further effort would be required to qualify the equipment for this aspect of the uprate. If the original equipment design does not bound the revised transients, then each affected piece of equipment would need to be re-qualified for the new transient conditions on a case-by-case basis.

5.9.5 Results

The evaluations for the auxiliary system equipment indicate that maximum operating temperatures and pressures for the evaluated equipment are bounded by the existing design basis. Since the auxiliary system tanks, heat exchangers, pumps, and valves were designed and manufactured consistent with the system design and applicable codes and standards, this equipment is acceptable for the maximum system operating temperatures and pressures associated with the uprate. Therefore, the auxiliary tanks, heat exchangers, pumps, and valves remain acceptable for the thermal transients resulting from the power uprate. This is also applicable to equipment that was replaced in accordance with the original Westinghouse technical and quality assurance requirements.

5.9.6 Conclusions

The KNPP auxiliary tanks, heat exchangers, pumps, and valves have been evaluated for the 7.4% power uprate. These components are acceptable for continued use at the uprated conditions because no change has been identified to the auxiliary system operating conditions.

These results are consistent with, and continue to comply with, the current KNPP licensing basis/acceptance requirements. The original NSSS design parameters and auxiliary design transients remain bounding for the conditions associated with the 7.4% power uprate.

References

None.

Table 5.9-1

Kewaunee (WPS) Auxiliary Heat Exchangers

Component Supplier/System
Regenerative Hx Joseph Oat./CVCS
Residual Hx Joseph Oat./RHR
Seal Water Hx Atlas Ind./CVCS
Excess Letdown Hx Sentry Equip. Corp./CVCS
Letdown Hx Atlas Ind./CVCS
CC Water Hx Eng. & Fab./CCW

Table 5.9-2

Kewaunee (WPS) Tanks

Component/System
Component Cooling Surge Tank /CVCS
Boric Acid Tank/CVCS
Boric Acid Batching Tank/CVCS
Concentrates Holding Tank/CVCS
Chemical Mixing Tank/CVCS
Holdup Tank/CVCS
Monitor Tank/CVCS
Resin Fill Tank/CVCS
VCT/CVCS
Boric Acid Filter/CVCS
Concentrates Filter/CVCS
Seal Water Injection Filter/CVCS
Seal Water Return Filter/CVCS
PRT/RCS
SIS Accumulator/SIS

Table 5.9-3

Kewaunee (WPS) Auxiliary Pumps

Description/System
Boric Acid Transfer Pump/CVCS
Component Cooling Water Pump/CCW
Concentrate Holdup Tank Pump/CVCS
Gas Stripper Feed Pump/CVCS
Holdup Tank Recirc. Pump/CVCS
Monitor Tank Pump
Positive Displacement Charging Pump/CVCS
Residual Heat Removal Pump/RHR
Safety Injection Pump/SIS