June 30, 2008

Mr. Stewart B. Minahan
Vice President-Nuclear and CNO
Nebraska Public Power District
72676 648A Avenue
Brownville, NE 68321

SUBJECT: COOPER NUCLEAR STATION - ISSUANCE OF AMENDMENT RE: MEASUREMENT UNCERTAINTY RECAPTURE POWER UPRATE (TAC NO. MD7385)

Dear Mr. Minahan:

The U.S. Nuclear Regulatory Commission (the Commission) has issued the enclosed Amendment No. 231 to Facility Operating License No. DPR-46 for the Cooper Nuclear Station. The amendment consists of changes to the license and Technical Specifications in response to your application dated November 19, 2007, as supplemented by letters dated March 6, March 12, April 4, and May 9, 2008.

The amendment revises the license and Technical Specifications to reflect an increase in the rated thermal power from 2381 to 2419 megawatts thermal (1.62 percent increase). The increase is based upon increased feedwater flow measurement accuracy, achieved by using high-accuracy Caldon CheckPlus™ Leading Edge Flow Meter ultrasonic flow measurement instrumentation.

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

Sincerely,

/RA/

Carl F. Lyon, Project Manager
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-298

Enclosures: 1. Amendment No. 231 to DPR-46
2. Safety Evaluation

cc w/encls: See next page

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ADAMS Accession No(s).: Pkg ML081540278, Amdt/License ML081540280, TS Pgs ML081540285 (*)memo dated (**)email dated (***)previously concurred

OFFICE	LPL4/PM	LPL4/LA	EICB/BC	EEEB/BC	EMCB/BC	CVIB/BC	CSGB/BC	CPTB/BC
NAME	FLyon	GLappert(***)	WKemper(*)	GWilson(*)	KManoly(*)	MMitchell(*)	AHiser(*)	JMcHale(*)
DATE	6/12/08	6/5/08	5/29/08	4/28/08	3/20/08	3/26/08	3/24/08	3/4/08
OFFICE	AADB/BC	AFPB/BC	CVIB/BC	SRXB/BC	SCVB/BC	ITSB/BC	IOLB/BC	SNPB/BC
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DATE	3/17/08	3/13/08	3/26/08	3/26, 4/16/08	6/11/08	6/10/08	6/11/08	6/12/08
OFFICE	SBPB/BC	CPNB/BC	OGC, NLO	LPL4/BC	DORL/DD	DORL/D		
NAME	DHarrison(**)	TChan(**)	DRoth	THiltz	RNelson	JGiitter, TJM for		
DATE	6/10/08	6/6/08	6/26/08	6/30/08		6/30/08		

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NEBRASKA PUBLIC POWER DISTRICT

DOCKET NO. 50-298

COOPER NUCLEAR STATION

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 231
License No. DPR-46

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

   A. The application for amendment by Nebraska Public Power District (the licensee), dated November 19, 2007, as supplemented by letters dated March 6, March 12, April 4, and May 9, 2008, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and Paragraph 2.C.(2) of Facility Operating License No. DPR-46 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A as revised through Amendment No. 231 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. The license amendment is effective as of its date of issuance and shall be implemented within 90 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA by Timothy J. McGinty for/

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

Attachment: Changes to the Facility
Operating License No. DPR-46
and Technical Specifications

Date of Issuance: June 30, 2008

ATTACHMENT TO LICENSE AMENDMENT NO. 231

FACILITY OPERATING LICENSE NO. DPR-46

DOCKET NO. 50-298

Replace the following pages of the Facility Operating License No. DPR-46 and Appendix A Technical Specifications with the enclosed revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Facility Operating License

REMOVE	INSERT
Page 3	Page 3

Technical Specification

REMOVE	INSERT
1.1-4	1.1-4
3.1-9	3.1-9
3.1-18	3.1-18
3.3-2	3.3-2
3.3-5	3.3-5
3.3-6	3.3-6
3.3-8	3.3-8
3.3-17	3.3-17
3.3-18	3.3-18
3.3-19	3.3-19
3.3-51	3.3-51
3.4-23	3.4-23
3.4-24	3.4-24
3.4-25	3.4-25

(5) Pursuant to the Act and 10 CFR Parts 30, 40, and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by operation of the facility.

C This license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR Chapter I: Part 20, Section 30.34 of Part 30, Section 40.41 of Part 40, Sections 50.54 and 50.59 of Part 50, and Section 70.32 of Part 70; is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

The licensee is authorized to operate the facility at steady state reactor core power levels not in excess of 2419 megawatts (thermal).

(2) Technical Specifications

The Technical Specifications contained in Appendix A as revised through Amendment No. 231, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

(3) Physical Protection

The licensee shall fully implement and maintain in effect all provisions of the Commission-approved physical security, training and qualification and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822) and to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The combined set of plans, which contain Safeguards Information protected under 10 CFR 73.21, are entitled: "Cooper Nuclear Station Safeguards Plan," submitted by letter dated May 17, 2006.

(4) Fire Protection

The licensee shall implement and maintain in effect all provisions of the approved fire protection program as described in the Cooper Nuclear Station (CNS) Updated Safety Analysis Report and as approved in the Safety Evaluations dated November 29, 1977; May 23, 1979; November 21, 1980; April 29, 1983; April 16, 1984; June 1, 1984; January 3, 1985; August 21, 1985; April 10, 1986; September 9, 1986; November 7, 1988; February 3, 1989; August 15, 1995; and July 31, 1998, subject to the following provision:

The licensee may make changes to the approved fire protection program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 231 TO

FACILITY OPERATING LICENSE NO. DPR-46

NEBRASKA PUBLIC POWER DISTRICT

COOPER NUCLEAR STATION

DOCKET NO. 50-298

1.0 INTRODUCTION

By application dated November 19, 2007 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML073300570), as supplemented by letters dated March 6, March 12, April 4, and May 9, 2008 (ADAMS Accession Nos. ML080720065, ML080780497, ML080990523, and ML081340352, respectively), Nebraska Public Power District (NPPD, the licensee), requested changes to the license and Technical Specifications (TSs) for Cooper Nuclear Station (CNS).

The proposed changes revise the license and TSs to reflect an increase in the rated thermal power from 2381 to 2419 megawatts thermal (MWt) (1.62 percent increase). The increase is based upon increased feedwater flow measurement accuracy, achieved by using high-accuracy Caldon CheckPlus™ Leading Edge Flow Meter (LEFM) ultrasonic flow measurement (UFM) instrumentation. This type of application is commonly referred to as a measurement uncertainty recapture (MUR) power uprate. The licensee developed the application using the guidance of U.S. Nuclear Regulatory Commission (NRC) Regulatory Issue Summary (RIS) 2002-03, "Guidance on the Content of Measurement Uncertainty Recapture Power Uprate Applications."

The supplements dated March 6, March 12, April 4, and May 9, 2008, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on January 29, 2008 (73 FR 5224).

Specifically, the licensee proposes the following changes:

- Paragraph 2.C.(1) in Facility Operating License DPR-46 (page 3) is revised to authorize operation at a steady state reactor core thermal power level not in excess of 2419 MWt.

- The definition of RATED THERMAL POWER (RTP) in TS 1.1, page 1.1-4, is revised to reflect the increase from 2381 MWt to 2419 MWt.

- Reference to “10% RTP” has been scaled down to “9.85% RTP” in the following TS:
    - 3.1.3 CONDITION D (page 3.1-9),
    - 3.1.6 APPLICABILITY (page 3.1-18),
    - Surveillance Requirements (SRs) 3.3.2.1.2 and 3.3.2.1.3 (page 3.3-17),
    - SR 3.3.2.1.6 (page 3.3-18),
    - Footnote (f) of Table 3.3.2.1-1 (page 3.3-19).

- Reference to “30% RTP” has been scaled down to “29.5% RTP” in the following TS:
    - 3.3.1.1 Reactor Protection (RPS) Instrumentation REQUIRED ACTION E.1 (page 3.3-2), and related TS SR 3.3.1.1.14 (page 3.3-5);
    - Table 3.3.1.1-1, FUNCTION 8 and 9, APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS (page 3.3-8).

- TS Table 3.3.1.1-1, Average Power Range Monitors ALLOWABLE VALUE of FUNCTION 2.b, Neutron Flux-High (Flow Biased), page 3.3-6, referenced by Limiting Condition for Operation (LCO) 3.4.1c (Recirculation Loops Operating), is revised from “≤ 0.66 W + 71.5% RTP [(b)]” to “≤ 0.75 W + 62.0% RTP [(b)].” Footnote (b) is revised from “0.66 W + 71.5% - 0.66” to “0.75W + 62.0% - 0.75.”

- ALLOWABLE VALUE on page 3.3-51 for TS Table 3.3.6.1-1, FUNCTION 1.c., Main Steam Line Flow – High, is revised from “≤ 144% rated steam flow” to “≤ 142.7% rated steam flow.”

- Validity of the TS Pressure/Temperature Figures 3.4.9-1, 3.4.9-2, and 3.4.9-3, pages 3.4-23, 3.4-24, and 3.4-25, respectively, will be revised from “30 EFPY” to “28 EFPY.”

The proposed changes reflect the impact on RTP of installing higher accuracy feedwater flow instrumentation, and incorporate adjustments required by the associated setpoint and plant analyses. The licensee also proposed corresponding changes to the licensee-controlled TS Bases, Technical Requirements Manual (TRM), and TRM Bases.

2.0 BACKGROUND

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

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

However, the final rule by itself did not allow increases in licensed power levels. Because the licensed power level for a plant is a TS limit, proposals to raise the licensed power level must be reviewed and approved under the license amendment process. CNS was originally licensed to operate at a maximum power level of 2381 MWt, which includes a 2 percent margin in the ECCS evaluation model to allow for uncertainties in core thermal power measurement as was previously required by 10 CFR 50, Appendix K.

CNS will install a Caldon LEFM CheckPlus™ System for feedwater flow measurement. This will be in addition to the venturi-based feedwater flow measurement system CNS currently uses to obtain the daily calorimetric heat balance measurements. Use of the LEFM CheckPlus™ System will reduce the calorimetric core power measurement uncertainty to ≤ ± 0.31 percent. Based on this, CNS is proposing to reduce power measurement uncertainty, while meeting the requirements of 10 CFR 50, Appendix K, to permit an increase of 1.62 percent in licensed power level.

Uncertainty in feedwater flow measurement is the most significant contributor to core power measurement uncertainty. The licensee states that use of the LEFM CheckPlus™ System provides a more accurate measurement of feedwater flow that supplements accuracy of the venturi-based instrumentation originally installed at CNS. Caldon Engineering Report ER-80P, "Improving Thermal Power Accuracy and Plant Safety While Increasing Operating Power Level Using the LEFM Check™ System," documents the theory, design, and operating features of the system and its ability to achieve increased accuracy of flow measurement. In a Safety Evaluation (SE) dated March 8, 1999, (ADAMS Accession No. 9903190065), ER-80P was approved by the NRC staff for use in justification of measurement uncertainty recapture (MUR) power uprates up to 1 percent. ER-80P was supplemented by Caldon Engineering Report ER-157P, "Basis for a Power Uprate With the LEFM Check™ or LEFM CheckPlus™ System." On December 20, 2001, the NRC issued an SE (ADAMS Accession No. ML013540256) approving ER-157P for use in justifying MUR power uprates up to 1.7 percent. The NRC reviewed and approved ER-80P and ER-157P again on July 5, 2006 (ADAMS Accession No. ML061700222), as part of its generic assessment of the hydraulic aspects of UFM application to increase licensed thermal power.

3.0 EVALUATION

3.1 Human Factors

The NRC staff reviewed the licensee's human factors evaluation to determine if it conforms to the staff's guidance in Section VII of RIS 2002-03. RIS 2002-03 provides guidance to the licensee in evaluating the need for changes to the areas of operator manual actions, procedures, human-system interfaces, and operator training related to the MUR power uprate.

The NRC staff human factors evaluation was conducted to confirm that operator performance would not be adversely affected as a result of system and procedure changes made to implement the proposed MUR power uprate.

The NRC staff developed a standard set of questions for human factors reviews in RIS 2002-03, Attachment 1, Section VII, Items 1 through 4). The following sections evaluate the licensee's response to these questions in its application.

3.1.1 Operator Actions

The licensee stated in its application that operator response to transient, accident, and special events is not affected for uprate conditions and that operator actions for maintaining safe shutdown, core cooling, and containment cooling do not change for the uprate. The licensee's response satisfies Section VII.1 of RIS 2002-03, Attachment 1.

3.1.2 Emergency and Abnormal Operating Procedures

The licensee committed in its application that procedure changes governing normal operation, emergency operation, and off-normal operation, as well as equipment changes that may be affected by power uprate, will be made. The licensee stated that procedures governing normal operation, emergency operation, and off-normal operation, as well as equipment changes that may be affected by the power uprate, will be identified in the design change process and revised prior to implementation of uprated power. Appropriate personnel will receive training on the Caldon LEFM CheckPlus™ System, as well as on the affected procedures. The training consists of briefings, required reading, classroom sessions, and a simulator demonstration, as needed, and will be conducted prior to operation at the uprated power. The licensee's response satisfies Sections VII.2.A and VII.3 of RIS 2002-03, Attachment 1.

3.1.3 Control Room Controls, Displays, and Alarms

The licensee stated in its application that the LEFM CheckPlus™ System features automatic self-checking. A continuously operating on-line test is provided to verify that the digital circuits are operating correctly and within the specified accuracy envelope. The on-line monitoring and diagnostics tests include the acoustic processing unit transmitters, timing circuits, signal quality, path sound velocity, hydraulic profile as represented by path velocities, and active computation as reported by watchdog timers. The system provides display and storage of verification test results. Failure messages are generated and monitored in the control room, if system failure events are detected.

If the LEFM CheckPlus™ System or a portion of the system becomes inoperable, control room operators are promptly alerted by control room computer indications. Feedwater flow input to the core thermal power calculation would then be provided by the existing flow nozzles, or a combination of flow nozzle(s) and LEFM flow data. Calculations have been performed to support the uncertainty of LEFM and flow nozzle inputs to the core thermal power calculation. The TRM will be revised prior to implementation of the uprated power to include LEFM CheckPlus™ System out-of-service administrative controls. The licensee's response satisfies Section VII.2.B of RIS 2002-03, Attachment 1.

3.1.4 Control Room Plant Reference Simulator

The licensee committed in its application that simulator changes and validation for the power uprate will be performed in accordance with ANSI/ANS 3.5-1985, "Nuclear Power Plant Simulators for Use in Operator Training and Examination," prior to implementation of the requested license amendment. The licensee's response satisfies Section VII.2.C of RIS 2002-03, Attachment 1.

3.1.5 Operator Training Program

The licensee stated in its application that no additional training, apart from normal training for plant changes, is required to operate the plant in the uprate condition. For uprate conditions, operator response to transient, accident, and special events is not affected. Operator actions for maintaining safe shutdown, core cooling, and containment cooling do not change for the uprate. Minor changes to the power/flow map and flow-referenced setpoint will be communicated through normal operator training. Simulator changes and validation for the uprate will be performed in accordance with established plant certification testing procedures. The licensee committed that appropriate personnel will receive training on Caldon LEFM CheckPlus™ System, and on affected procedures prior to operation at uprated power. The licensee's response satisfies Sections VII.2.D and VII.4 of RIS 2002-03, Attachment 1.

3.1.6 Conclusion

As described above, the NRC staff has reviewed the licensee's planned actions related to the human factors area, and concludes that NPPD adequately considered the impact of the proposed MUR power uprate on changes to operator actions, procedures, plant hardware, and associated training programs to ensure that operators' performance is not adversely affected by the proposed MUR power uprate.

3.2 Dose Consequences Analysis

3.2.1 Regulatory Evaluation

The NRC staff reviewed the impact of the proposed changes on previously analyzed design-basis-accident (DBA) radiological consequences and the acceptability of the revised analysis results. The regulatory requirements for which the NRC staff based the acceptance are the accident dose guidelines in 10 CFR 100.11, as supplemented by regulatory guidance accident-specific criteria in Section 15 of NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," and 10 CFR Part 50 Appendix A, General Design Criterion 19 (GDC-19), "Control Room," as supplemented by Section 6.4 of NUREG-0800. The NRC staff also considered relevant information in the CNS updated safety analysis report (USAR) and technical specifications. The 1967 Proposed GDC as described in the USAR, Appendix F, are the licensing basis for CNS; however, the NRC staff concluded in its 1973 Safety Evaluation Report for CNS that the intent of the 1971 Final Rule for 10 CFR Part 50, Appendix A, had also been met.

RIS 2002-03 recommends that, to improve efficiency of the NRC staff's review, licensees requesting an MUR uprate should identify existing DBA analyses of record which bound plant operation at the proposed uprated power level. For any DBA for which the existing analyses of

record do not bound the proposed uprated power level, the licensee should provide a detailed discussion of the reanalysis. The NRC staff review covers the impact of the proposed MUR power uprate on the results of dose consequence analyses as noted in RIS 2002-03, Attachment 1, Sections II and III.

3.2.2 Technical Evaluation

The NRC staff reviewed the regulatory and technical analyses, as related to the radiological consequences of DBAs, performed by CNS in support of its proposed license amendment. Information regarding these analyses was provided in Enclosure 1, Section 9.0 of the submittal.

The NRC staff reviewed the impact of the proposed 1.62 percent power uprate on DBA radiological analyses, as documented in Chapter 14 of the CNS USAR. In the Licensee's application, the licensee stated that each of the current DBA dose analyses of record for CNS were unaffected by the requested power uprate because they were performed assuming 102 percent of 2381 MWt. Analyses performed at this power are applicable to the requested uprated power of 2419 MWt with a 0.38-percent power measurement uncertainty. Using the current CNS USAR documentation in addition to information in the licensee's application dated November 19, 2007, the NRC staff verified that the existing CNS USAR Chapter 14 radiological analyses source term and release assumptions bound the conditions for the proposed 1.62 percent power uprate to 2419 MWt, considering the higher accuracy of the feedwater measurement instrumentation. The analyses of record show that, for the proposed power uprate, the radiological consequences of postulated DBAs continue to meet the dose limits given in 10 CFR 100.11 and the plant-specific design criteria as described in the USAR, and continues to meet the intent of 10 CFR 50, Appendix A, GDC-19, as well as applicable dose acceptance criteria given in NUREG-0800, Section 15, for the radiological consequences of DBAs.

3.2.3 Conclusion

The NRC staff reviewed the licensee's assessment of the impact of the proposed 1.62 percent MUR power uprate on dose consequences analyses for the CNS. As discussed above, the NRC staff determined that the results of the licensee's analyses of the radiological consequences of DBAs continue to meet the applicable acceptance criteria following implementation of the proposed MUR power uprate. Therefore, the NRC staff finds the proposed MUR power uprate acceptable with respect to the dose consequences of DBAs.

3.3 Fire Protection

3.3.1 Regulatory Evaluation

The purpose of the fire protection program is to provide assurance, through a defense-in-depth design, that a fire will not prevent the performance of necessary plant safe-shutdown functions nor will it significantly increase the risk of radioactive releases to the environment. The NRC staff review focused on the effects of the increased decay heat on the plant's safe-shutdown analysis to ensure that structures, systems, and components (SSCs) required for the safe-shutdown of the plant are protected from the effects of the fire and will continue to be able to achieve and maintain safe-shutdown following a fire. The NRC's acceptance criteria for the fire protection program are based on 10 CFR 50.48, "Fire protection," insofar as it requires the development of a fire protection program to ensure, among other things, the capability to safely

shutdown the plant. The 1967 Proposed General Design Criterion (GDC) as described in the USAR, Appendix F, are the licensing basis for CNS; however, the NRC staff concluded in its 1973 Safety Evaluation Report for CNS that the intent of the 1971 Final Rule for 10 CFR Part 50, Appendix A, had also been met. The staff therefore reviewed the proposed changes to verify that the intent of GDC 3 of Appendix A to 10 CFR Part 50, insofar as it requires that [a] SSCs important to safety be designed and located to minimize the probability and effect of fires, [b] noncombustible and heat resistant materials be used, and [c] fire detection and suppression systems be provided and designed to minimize the adverse effects of fires on SSCs important to safety; and of GDC 5 of Appendix A to 10 CFR Part 50, insofar as it requires that SSCs important to safety not be shared among nuclear power units unless it can be shown that sharing will not significantly impair their ability to perform their safety functions, were still met.

RIS 2002-03, Attachment 1, Sections II and III, recommends that, to improve the efficiency of the NRC staff's review, licensees requesting an MUR power uprate should identify current accident and transient analyses of record which bound plant operation at the proposed uprated power level. For any design-basis-accident for which the existing analyses of record do not bound the proposed uprated power level, the licensee should provide a detailed discussion of the reanalysis.

3.3.2 Technical Evaluation

The licensee developed its application consistent with the guidelines in RIS 2002-03. In its application, the licensee re-evaluated the applicable SSCs and safety analyses at the proposed MUR core power level of 2419 MWt against the previously analyzed core power level of 2381 MWt. The NRC staff's review of the licensee's November 19, 2007, application and General Electric Company (GE)-Hitachi Safety Analysis Report for CNS Thermal Power Optimization, NEDC-33385P, Revision 0, November 2007, identified areas in which additional information was necessary to complete the review of the proposed MUR. The licensee responded to the NRC staff request for additional information (RAI) as discussed below.

In the RAI, the NRC staff noted that the GE-Hitachi Safety Analysis Report for CNS Thermal Power Optimization [TPO], NEDC-33385P, Revision 0, November 2007, Section 6.7, "Fire Protection," states that operation of the plant at the TPO level does not affect fire detection and suppression systems. The NRC staff requested that the licensee address the impact of TPO uprate conditions on other fire protection program elements, at a minimum, the following: (1) administrative controls, (2) fire barriers, (3) fire protection responsibilities of plant personnel, and (4) procedures and resources necessary for systems required to achieve and maintain safe-shutdown.

By letter dated March 6, 2008, the licensee provided the following response.

> A review was conducted of the Fire Protection Program as related to administrative controls, fire barriers, fire protection responsibilities of plant personnel and resources necessary for systems required to achieve and maintain safe-shutdown. The review looked at the impact of TPO uprate and how it would impact these areas. The TPO uprate will have no impact on fire protection administrative controls, fire barriers, fire protection responsibilities of plant personnel, or resources necessary for systems required to achieve and maintain safe-shutdown.

The licensee's response satisfactorily addresses the NRC staff concerns, and the RAI issue is considered resolved based on the following. The licensee conducted a review of the CNS fire protection program for the proposed MUR power uprate condition with the current operating power level. The licensee's review indicated that the proposed MUR power uprate would not impact the fire protection program elements, i.e., fire protection administrative controls, fire barriers, fire protection responsibilities of plant personnel, and resources necessary for systems required to achieve and maintain post-fire safe-shutdown capability.

In the RAI, the NRC staff noted that the GE-Hitachi Safety Analysis Report for CNS Thermal Power Optimization, NEDC-33385P, Revision 0, November 2007, Section 6.7, "Fire Protection," states that the operator actions required to mitigate the consequences of a fire are not affected. The NRC staff requested the licensee to verify that additional heat in the plant environment from the MUR power uprate will not interfere with required operator manual actions being performed at their designated time.

By letter dated March 6, 2008, the licensee provided the following response.

> The operator manual actions that are being used for compliance with 10 CFR 50, Appendix R were reviewed. No operator manual actions have been identified in areas where environmental conditions, such as heat, would challenge the operator. Since this uprate is being performed at a constant pressure and temperature, the normal temperature environments are not affected by the MUR. Therefore, the MUR power uprate will have no impact on operator manual actions.

The licensee's response satisfactorily addresses the NRC staff concerns, and the RAI issue is considered resolved based on the following. The licensee indicated that the proposed MUR power uprate does not impact the previous operator manual actions in the fire safe-shutdown analysis. The licensee stated that (1) no operator manual actions have been identified in areas where environmental conditions, such as heat, would challenge the operator, and (2) the proposed MUR power uprate does not impact operator manual actions.

In the RAI, the NRC staff noted that the results of the Appendix R evaluation for MUR power uprate are provided in Section 6.7, "Fire Protection," of the GE-Hitachi Safety Analysis Report for CNS Thermal Power Optimization, NEDC-33385P, Revision 0, November 2007. However, this section does not discuss the time necessary for the repair of systems required to achieve and maintain cold shutdown nor the increase in decay heat generation following plant trips. The NRC staff requested the licensee to verify that the plant can meet the 72-hour requirements in both 10 CFR Part 50, Appendix R, Sections III.G.1.b and III.L, with increased decay heat at MUR power uprate conditions.

By letter dated March 6, 2008, the licensee provided the following response.

> A review was conducted of all activities that are credited to obtain and maintain cold shutdown. The CNS Appendix R analysis demonstrates that the station can reach cold shutdown with significant margin to the 72 hour requirements in 10 CFR 50, Appendix R, Section III.G.1.b and III.L. No "time-critical" repair would be required to reach or maintain cold

shutdown. The MUR power uprate and the additional decay heat removal would not impact the ability to reach and maintain cold shutdown within 72 hours.

The licensee's response satisfactorily addresses the NRC staff concerns, and the RAI issue is considered resolved based on the following. For the MUR power uprate condition, the licensee reviewed its systems to obtain and maintain plant in cold shutdown condition. The results demonstrate that CNS can be placed in cold shutdown following a fire within the required 72 hours. Further, the licensee indicated that (1) no time-critical repair would be required to reach or maintain cold shutdown, and (2) additional decay heat removal would not impact the ability to reach and maintain cold shutdown within 72 hours.

3.3.3 Conclusion

The NRC staff reviewed the licensee's fire-related safe-shutdown assessment and concludes that the licensee adequately accounted for the effects of the 1.62 percent increase in decay heat on the ability of the required systems to achieve and maintain safe- shutdown conditions. The NRC staff finds this aspect of the capability of the associated SSCs to perform their design-basis functions at an increased core power level of 2419 MWt acceptable with respect to fire protection.

3.4 Chemical Engineering

3.4.1 Flow Accelerated Corrosion (FAC)

3.4.1.1 Regulatory Evaluation

FAC is a corrosion mechanism occurring in carbon steel components exposed to flowing single-phase or two-phase water flow. Components made from stainless steel are immune to FAC, and FAC is significantly reduced in components containing small amounts of chromium or molybdenum. The rates of material loss due to FAC depend on velocity of flow, fluid temperature, steam quality, oxygen content, and pH. During plant operation, control of these parameters is limited and the optimum conditions for minimizing FAC effects, in most cases, cannot be achieved. Loss of material by FAC will, therefore, occur.

The NRC staff reviewed the effects of the proposed MUR power uprate on FAC and the adequacy of the licensee's FAC program to predict the rate of material loss so that repair or replacement of damaged components could be made before they reach critical thickness. The licensee's FAC program consists of predicting loss of material using the CHECWORKS computer code, and visual inspection and volumetric examination of the affected components. The NRC's acceptance criteria are based on the structural evaluation of the minimum acceptable wall thickness for the components undergoing degradation by FAC.

3.4.1.2 Technical Evaluation

Operation at the proposed MUR power uprate conditions will result in changes to parameters affecting FAC in some systems at CNS. The licensee stated that evaluation of and inspection for FAC in affected systems is performed under the existing plant FAC program. Assurance of the integrity of high-energy piping systems is maintained through continued monitoring. For

those systems that will experience changes in process variables (i.e. moisture content, temperature, oxygen, and flow velocity) as a result of the proposed MUR power uprate, the licensee indicated that changes to piping inspection frequency will be implemented to ensure adequate margin in component wall thickness exists. Adjustments to inspection frequency will be based in part on adjustments to the predicted material loss rates that are used to project the need for component repair or replacement.

The CNS FAC program uses the CHECWORKS predictive code in conjunction with actual wall thickness measurements to model wear rates of piping and components. In order to determine the past performance and accuracy of the CNS FAC program and predictive model, the NRC staff requested a sample list of components, included in the FAC program, for which actual thickness measurements as well as the thickness predicted by CHECWORKS were available. By letter dated March 6, 2008, the licensee provided a list of 40 components including the nominal thickness, the thickness predicted by the code, and the actual measured thickness of each component. For all cases provided by the licensee the actual measured thickness of each component was greater than the thickness predicted by the code. This indicates that the predictive code is conservative in its estimation of component wear rates, thus resulting in predicted thicknesses that are less that the actual component thickness. This will trigger inspections of the components prior to the point at which the margins in component wall thickness are challenged.

The NRC staff also requested a list of the systems that were predicted to experience the greatest increase in wear rate as a result of the proposed MUR power uprate, as well as the magnitude of the predicted increase, and magnitude of change in the process variable responsible for the increase in wear rate. By letter dated March 12, 2008, the licensee provided a response to the NRC staff question. Using the CHECWORKS predictive model, the licensee input parameters for the pre-uprate and post-uprate conditions. Using the output of the model, the licensee calculated the average wear rate increase for the most FAC susceptible components. Three components are expected to experience a wear rate increase of greater than 2 mils per year. The components are the extraction steam piping to the third stage feed water (FW) Heater (2.8 mils/yr increase), the drain piping from the fourth stage FW Heater (2.1 mils/yr increase), and the drain piping from the fifth stage FW Heater (4.7 mils/yr increase). The predicted increase in average wear rate of the extraction steam piping is a result of the increased wall velocity and reduced steam quality resulting from the proposed MUR power uprate. The predicted increase in average wear rate for the drain piping is due to increased velocity for the fourth stage drains and due to approaching a more critical temperature for wear on the fifth stage drains. None of the other piping systems are expected to have an increase of greater than 1 mil per year.

3.4.1.3 Conclusion

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

3.4.2 Reactor Water Cleanup System

3.4.2.1 Regulatory Evaluation

The reactor water cleanup system (RWCU) provides a means for maintaining reactor water quality by filtration and ion exchange and a path for removal of reactor coolant when necessary. Portions of the RWCU comprise the reactor coolant pressure boundary (RCPB). The NRC staff review of the RWCU included component design parameters for flow, temperature, pressure, heat removal capability, and impurity removal capability; as well as the instrumentation and process controls for proper system operation and isolation. The review consisted of evaluating the adequacy of the plant TSs in these areas under the proposed MUR power uprate conditions. The 1967 Proposed GDC as described in the USAR, Appendix F, are the licensing basis for CNS; however, the NRC staff concluded in its 1973 Safety Evaluation Report for CNS that the intent of the 1971 Final Rule for 10 CFR Part 50, Appendix A, had also been met. The NRC reviewed the RWCU to verify that the intent of the following GDC continued to be met: (1) 10 CFR Part 50, Appendix A, General Design Criteria (GDC)-14, insofar as it requires that the RCPB be designed, fabricated, erected, and tested so as to have an extremely low probability of rapidly propagating fracture; (2) GDC-60, insofar as it requires that the plant design include means to control the release of radioactive effluents; and (3) GDC-61, insofar as it requires that systems that contain radioactivity be designed with appropriate confinement. Specific review criteria are contained in NUREG-0800, Section 5.4.8.

3.4.2.2 Technical Evaluation

Since the RWCU continuously withdraws a portion of the reactor water, the NRC staff evaluated potential changes to the system resulting from the power uprate. The RWCU is a normally operating system with no safety functions other than containment isolation. The licensee’s evaluation of the RWCU system concluded that changes to the system resulting from the MUR power uprate would be negligible and insignificant to the system performance. There is no significant effect on operating temperature and pressure in the high-pressure portion of the system. The flow through the RWCU is not significantly affected by reactor power and recirculation flow conditions. Operation at uprated power will cause an insignificant change in the quantity of fission and corrosion products, and other soluble and insoluble impurities in the reactor water. Based on the NRC staff review of the CNS analysis and based on operating experience with other MUR power uprates as well as larger power uprates increases (i.e., Stretch Power Uprates and Extended Power Uprates), the NRC staff agrees the changes imparted by the proposed MUR power uprate are insignificant with respect to system performance at CNS.

3.4.2.3 Conclusion

The NRC staff reviewed the licensee's evaluation of the effects of the proposed MUR power uprate on the RWCU and concludes that the licensee adequately addressed changes in impurity levels and pressure and their effects on the RWCU. The NRC staff further concludes that the licensee demonstrated that the RWCU will continue to be acceptable following implementation of the proposed MUR power uprate and will continue to meet the intent of GDC-14, GDC-60, and GDC-61. Therefore, the NRC staff finds the proposed MUR power uprate acceptable with respect to the RWCU.

3.5 Mechanical and Civil Engineering

3.5.1 Regulatory Evaluation

The NRC staff review in the area of mechanical engineering covers the structural and pressure boundary integrity of nuclear steam supply system (NSSS) and balance-of-plant (BOP) systems and components. The review focuses on the impact of the proposed MUR power uprate on (1) NSSS piping, components, and supports; (2) BOP piping, components, and supports; (3) the reactor pressure vessel (RPV) and its supports; (4) reactor internals and core supports; and (5) safety related valves. Technical areas covered by this review include stresses, cumulative usage factors (CUFs), flow-induced vibration, high-energy line break locations, jet impingement and thrust forces, and safety-related valve programs.

The above affected piping systems, components and their supports, including core support structures, are designed in accordance with the rules of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME B&PVC), Section III, and the United States of America Standards (USAS) B31.1 Code for Pressure Piping. The NRC staff evaluation considered draft General Design Criteria (GDC) 1, 2, 9, 33, 34, and 51 which are located in Appendix F of the CNS Updated Safety Analysis Report (USAR). The NRC staff review focused on verifying that the licensee has provided reasonable assurance of the structural and functional integrity of piping systems, components, component internals, and their supports under normal and vibratory loadings, including those due to fluid flow, postulated accidents, and natural phenomena such as earthquakes.

The acceptance criteria is based on continued conformance with the requirements of the following regulations: (1) 10 CFR 50.55a, and draft GDC 1 as they relate to structures and components being designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety function to be performed; (2) draft GDC 2 as it relates to structures and components important to safety being designed to withstand the effects of earthquakes combined with the effects of normal or accident conditions; (3) draft GDC 9 and draft GDC 34 as they relate to the reactor coolant pressure boundary (RCPB) being designed and constructed to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture; and (4) draft GDC 33 as it relates to the RCPB being capable to accommodate, without rupture and with only a limited allowance for energy absorption through plastic deformation, the loads imposed on the boundary by a sudden release of energy to the coolant; and (5) draft GDC 51 as it relates to the design of the RCPB outside containment being designed such that its rupture does not jeopardize public health and safety.

The review also includes the licensee's safety-related valves analysis. The NRC's acceptance criteria for reviewing the safety-related valves analysis are based on 10 CFR 50.55a, "Codes and Standards." Additional information is provided by the plant-specific evaluations for Generic Letter (GL) 95-07, "Pressure Locking and Thermal binding of Safety-Related Power-Operated Gate Valves," GL 96-06, "Assurance of Equipment Operability and Containment Integrity During Design-Basis-Accident Conditions," regarding the over-pressurization of isolated piping segments, GL 89-10, "Safety-Related Motor-Operated Valve Testing and Surveillance," and GL 96-05, "Periodic Verification of Design-Basis Capability of Safety-Related Motor-Operated Valves."

3.5.2 Technical Evaluation

The NRC staff review focused on the effects of power uprate on the structural and pressure boundary integrity of RCS piping, components, and their supports, the reactor vessel and internal components, BOP piping systems, and safety related valves.

The proposed 1.62 percent power uprate will increase the RTP level from 2,381 MWt to 2,419 MWt. The power uprate will be achieved by an increase in reactor power along the current rod and core flow control lines. An increase in steam flow and FW flow will accompany this increase in reactor power.

Table 1-2 of Enclosure 1 (proprietary) and Enclosure 3 (non-proprietary) in the licensee's November 19, 2007, application shows the pertinent temperatures, pressures, and flow rates for the current and proposed (uprated) conditions. At full power, the dome temperature remains at a constant 547.0 degrees Fahrenheit (^{o}F) from current to uprated conditions. At full power, the dome pressure remains at a constant 1020.0 pounds per square inch absolute (psia) from current to uprated conditions. The steam flow increases from 9.556 million pounds per hour (Mlbm/hr) to 9.707 Mlbm/hr. The FW flow rate increases from 9.521 Mlbm/hr to 9.672 Mlbm/hr. The FW temperature remains at a constant 367.1^{o}F from current to uprated conditions. There is no change to the maximum allowable core recirculation drive flow. The proposed uprate does not change heatup or cooldown rates or the number of cycles assumed in the design analyses. In addition, the limiting analyses for design transients are still bounding.

The design parameters for the RCS are located in the CNS USAR. Table IV-2-1 of the CNS USAR indicates that the RPV and its associated components were designed to operate at 575°F and 1250 psig. Table IV-3-1 of the CNS USAR shows the Reactor Recirculation System (RRS) design parameters including the suction piping (562°F and 1148 psig) and the discharge piping (575°F and 1274 psig). The RCS components, including the reactor vessel and all associated internals, were designed to operate at a core power level of 2428 MWt (102 percent of the current licensed thermal power [CLTP]).

3.5.2.1 Reactor Pressure Vessel

The original code of record for the RPV, nozzles, and associated supports is the ASME B&PVC, Section III, 1965 Edition with addenda through Winter, 1966. The licensee noted in Section 3.2.2 of Enclosure 1 of the LAR application that reactor vessel components' designs that have been modified were analyzed against the governing code for those particular components and not the original code of record. Modified components included: recirculation outlet nozzle (N1), recirculation inlet nozzle (N2), FW nozzle (N4), core spray nozzle (N5), jet pump instrumentation nozzle (N8), control rod drive (CRD) hydraulic system return nozzle (N9), core differential pressure (ΔP) and liquid control nozzle (N10), CRD penetration, and an intermediate and source range monitor (IRM/SRM) dry tube, power range detector. The proposed power uprate does not change the operating reactor pressure and temperature from the current operating condition. In addition, the current design basis transient analyses remain valid for the proposed power uprate. The licensee concluded that the current design-basis stress and calculated stresses and fatigue usage factors (CUFs) analyses for the reactor pressure vessel components will continue to meet the code limits and are, therefore, acceptable for the proposed power uprate. The NRC staff concurs with the licensee's assessment that the RPV, nozzles, and supports are acceptable for operation at the uprated power level.

3.5.2.2 Reactor Internals

The licensee evaluated the reactor internal components, including core support structures and non-core support structures, considering the changes in the design input parameters and loads due to the proposed 1.62 percent power uprate. The loads applicable to the internal components include reactor internal pressure difference (RIPD), RPV design pressure, deadweight, acoustic and flow induced loads, and thermal effects. The staff noted that the reactor internal components are not certified to the ASME Code. However, the licensee indicated that the ASME Code requirements were used as guidelines during the stress evaluations of the reactor internals. The core support structures evaluated for the proposed power uprate include: shroud, shroud support, core plate, tope guide, control rod drive housing, control rod guide tube, orificed fuel support, and the fuel channel. The non-core support structure components evaluated include the steam dryer, FW sparger, jet pump assembly, core spray line and sparger, access hole cover, shroud head and steam separator assembly, in-core housing and guide tube, jet pump instrument penetration seal, and the core ΔP/standby liquid control system. The steam separator and steam dryer performance are specifically addressed by the licensee noting that the generic evaluation detailed in Section 5.5.1.6 of GE Topical Report (TR) NEDC-32938P, “Generic Guidelines and Evaluations for General Electric Boiling Water Reactor Thermal Power Optimization,” is applicable to CNS and no further evaluation is necessary.

The proposed power uprate does not change the operating reactor pressure and temperature from the current operating condition. In addition, the current design basis transient analyses remain valid for the proposed power uprate. The MUR power uprate conditions are bounded by the design conditions. In addition, the operating transients continue to bound the uprate conditions and no additional transients have been proposed. The existing loads, stresses, and fatigue CUF values for the reactor vessel internals remain valid for the proposed power uprate. The NRC staff concurs with the licensee’s assessment that the reactor internals are acceptable for operation at the uprated power level.

3.5.2.3 Reactor Coolant Pressure Boundary Piping and Supports

The code of record for the RCPB piping is USAS 31.1, 1967 Edition, Code for Power Piping. The methods used by the licensee for the piping and pipe support evaluations are described in GE TR NEDC-32938P, Appendix K, “Methods and Assumptions for Piping Evaluation of TPO Uprate.” The NRC staff notes that the methods described in Appendix K of the GE TR have been used in piping evaluations for BWR power uprates of up to 20 percent. The licensee evaluated the effects of the proposed 1.62 percent power uprate condition on the RCPB piping, components and their supports with regard to changes in flow rate, temperature and pressure. The piping systems evaluated by the licensee included the RRS, main steam (MS) and attached piping systems (inside containment), reactor core isolation cooling (RCIC) piping, MS drain lines, RPV head vent line, FW piping (inside containment), RPV bottom head drain line, and residual heat removal (RHR), low pressure core spray, high pressure core spray, reactor water cleanup (RWCU), and standby liquid control (SLC) piping systems. The evaluation was summarized in a table in Section 3.5.1 of Enclosure 1 of the licensee's application.

Based on the tabulated results of the evaluation paired with the constant vessel dome pressure at the uprated conditions, it is concluded that a negligible effect on a majority of the RCPB piping

exists at the proposed power level. However, the licensee indicated that the analysis shows a slight effect on portions of the MS and FW flow lines in addition to piping connected with these affected portions due to the slight increase in MS and FW flows. The licensee determined the extent of the effects on these affected portions caused by the uprate utilizing analytical methodologies found in Section 5.5.2 and Appendix K of the aforementioned GE TR NEDC-32938P. The results of this evaluation showed that the affected portions of the MS and FW systems continue to be bounded by the design conditions. In addition, the operating transients continue to bound the uprate conditions and no additional transients have been proposed. The existing loads, stresses, and fatigue CUF values for the RCPB piping and supports remain valid for the proposed power uprate. The NRC staff concurs with the licensee's assessment that the RCPB piping and supports are acceptable for operation at the uprated power level.

3.5.2.4 BOP Piping (NSSS Interface Systems, Safety Related Cooling Water Systems, and Containment Systems) and Safety Related Valves

The code of record for the balance-of-plant (BOP) piping is the USAS 31.1, 1967 Edition, Code for Power Piping. The licensee evaluated the BOP piping and supports by comparing the original design basis conditions with those for the proposed power uprate. The analyzed portions of the BOP piping systems included the sections of the MS and FW systems located outside containment. The evaluation of BOP piping was performed in a similar manner to the evaluation of RCPB piping and supports for operation at the proposed power uprate (i.e. using analytical methodologies found in Section 5.5.2 and Appendix K of GE TR NEDC-32938P).

The MUR power uprate conditions are bounded by the design conditions. In addition, the operating transients continue to bound the uprate conditions and no additional transients have been proposed. The existing loads, stresses, and fatigue CUF values for the BOP piping and supports remain valid for the proposed power uprate. The NRC staff concurs with the licensee's assessment that the BOP piping and supports are acceptable for operation at the uprated power level.

The licensee stated that the revised conditions were reviewed for impact on the existing design-basis analyses for the safety related valves. The review showed that the flow increases due to the MUR uprate are bounded by those used in the existing analyses. Safety analyses confirmed that the installed capacities and lift set points of the Safety Relief Valves (SRVs) continue to be valid for the MUR conditions. None of the safety related valves required a change to their design or operation as a result of the MUR. The existing loads, stresses, and fatigue CUF values remain valid.

The licensee's evaluation also showed that the temperature changes due to MUR uprate are insignificant and bounded by those used in the existing analyses, and no changes in reactor coolant system design or operating pressure are made as part of MUR uprate. In Enclosure 1, Section 3.1, "Nuclear System Pressure Relief/Over Protection," the licensee confirms that there is no increase in normal operating pressure for the MUR uprate, and therefore, no changes in the safety/relief valve setpoints. In Section 3.8, "Main Steam Isolation Valves (MSIVs)," the licensee confirms that the requirements for the MSIVs remain unchanged for the MUR uprate conditions; and all safety and operational aspects of the MSIVs are within previous evaluations. In Section 4.1.2, "Generic Letter 95-07 Program," the licensee states that commitments relating to GL-95-07 have been reviewed and no changes are identified as a result of MUR uprate, and valves in the RHR, RCIC, HPCI, and CS were included in the evaluation. In Section 4.1.3,

"Generic Letter 96-06," the licensee states that the containment design temperature and pressure are not exceeded under post-accident conditions for the MUR uprate, and therefore, the CNS responses to GL-96-06 remains valid under MUR uprate conditions. In Section 6.5, "Standby Liquid Control System (SLCS), the licensee states that the SLCS relief valve margin is adequate for the MUR uprate and the uprate does not affect shutdown or injection capability of the SLCS.

The licensee did not identify any changes to the plant-specific provisions of GL 89-10, related to motor operated valves, GL 95-07, related to pressure locking and thermal binding of safety-related gate valves, or GL 96-06, related to over-pressurization of isolated piping segments. The NRC staff does not anticipate any changes to the analysis of over-pressurization of isolated piping segments because the analyses of record for containment temperature and pressure was performed at 102 percent of CLTP and remains bounding for the uprate conditions. Therefore, the NRC staff does not expect any changes to the plant-specific provisions of GL 89-10, GL 95-07, or GL 96-06.

The licensee concluded that the CNS safety related valves will remain acceptable for operation at the uprated conditions. Due to the insignificant changes in temperature and operating pressure, none of the safety-related valves required a change to their design or operation as a result of the MUR uprate. Based on the above, the NRC staff agrees with the licensee's conclusion that the proposed 1.62 percent power uprate will not have adverse effects on safety-related valves.

3.5.2.5 Flow Induced Vibration

The licensee assessed flow induced vibration (FIV) for the proposed power uprate for limiting reactor internal components including the shroud head and separator, steam dryers, core spray (CS) line, low pressure coolant injection (LPCI) coupling, control rod guide tube, in-core guide tubes, in-core instrumentation fuel channel, jet pumps, jet pump sensing lines, and FW sparger. The licensee utilized vibration data obtained from startup testing at FitzPatrick Nuclear Power Plant (FNPP) to estimate the effect of the MUR power uprate on reactor internals at CNS. The licensee indicated that there is a slight increase in flow induced vibration for the shroud, shroud head and separator, and FW sparger because of an approximately 2 percent increase in steam and FW flow due to the power uprate. Other internal components are not affected since the maximum core flow and the maximum recirculation drive flow remain unchanged following the proposed 1.62 percent power uprate. As a result of its evaluation, the licensee concluded that vibration of safety related internal components due to flow induced vibration loads will remain within the GE acceptable stress limits of 10 ksi. The licensee committed in its application that, prior to exceeding the current licensed power rate (2381 MWt), it will ensure compliance with the methodology contained in Regulatory Guide (RG) 1.20 for vibration assessment.

The NRC staff finds that the increases in FIV in the shroud head and separator, and FW sparger, due to the proposed power uprate, remain within acceptable limits. Therefore, the NRC staff finds that the reactor internals will remain adequate and acceptable for the proposed 1.62 percent power uprate, considering the GE stress limit of 10 ksi at 10^{11} service cycles is much more conservative than the ASME allowable stress limit of 13.6 ksi. The NRC staff concurs with the licensee's conclusion that the reactor internals design remains acceptable for the FIV at the proposed power uprate condition.

3.5.2.6 High Energy Line Break Locations (HELB)

The licensee stated that an engineering evaluation was performed to determine the impact of power uprate on HELB systems. The HELB evaluations were performed at 2,429 MWt (102 percent of CLTP) to bound the expected range of operation resulting from the MUR uprate. There are no new line breaks postulated for current HELB systems due to the fact that the piping configuration will not change as a result of the power uprate. The licensee stated that the impact of the MUR uprate on HELB systems remains bounded by the values in existing analyses. Also, there are no new systems that qualify as HELB systems as a result of the uprate. The NRC staff agrees with the licensee's conclusion regarding HELB.

3.5.3 Conclusion

The NRC staff reviewed the licensee's assessment of the impact of the proposed MUR power uprate at CNS on NSSS and BOP systems and components with regard to stresses, CUFs, and safety related valve programs. On the basis of the review described above, the NRC staff finds that the proposed MUR power uprate will not have an adverse impact on the structural integrity of the piping systems, components, their supports, reactor internals, core support structures, BOP piping, or safety-related valves.

3.6 Reactor Systems

References for Section 3.6 are listed in Section 3.6.3.6.

3.6.1 Introduction

The licensee's application (Ref. 1) summarizes the results of safety analyses that justify increasing the licensed thermal power at CNS to 2419 MWt. The requested license power level is 1.62 percent above the current licensed thermal power (CLTP) level of 2381 MWt. The licensee's safety analyses were performed at 2421 MWt, i.e., 1.7 percent above CLTP; however, the actual power increase will be governed by the results of the core thermal power uncertainty calculation, which was done to allow an increase to 2419 MWt, 1.62 percent above CLTP.

The Caldon LEFM CheckPlus™ System is similar to the LEFM Check™ System, except that it has 16 transducers on eight acoustic measurement paths grouped into two orthogonal planes with four measurement paths in each plane. The LEFM CheckPlus™ System essentially combines two LEFM Check™ Systems. In order to ensure independence, each measurement plane employs its own timing clock in the LEFM CheckPlus™ System. The LEFM CheckPlus™ System is expected to provide feedwater flow measurement that is more accurate than that provided by a LEFM Check™ System. It can support a power uprate of up to 1.7 percent. The LEFM CheckPlus™ System is described in Caldon Report ER- 157P (Ref. 3). The NRC staff approved ER-157P in its December 20, 2001, safety evaluation (Ref. 4).

In its application, the licensee included licensing report NEDC-33385P, "Safety Analysis Report for Cooper Nuclear Station Thermal Power Optimization" (Ref. 5), to support the proposed power uprate. The application evaluated the impact of the increased operating power on the facility's safety analyses and on the capabilities and performance of the NSSS and its components. The power-dependent safety analyses, which are based on 102 percent of the current reactor thermal power, will remain applicable and bounding at the uprated condition;

however, equipment and system qualifications and analyses performed at nominal power must be reevaluated. The power uprate will be achieved by increasing the FW flow to produce higher steam flow from the reactor vessel and by adjusting the turbine control valve position to reduce the main steam line flow resistance.

The licensee's power uprate application follows the generic format and content of the BWR power uprate licensing topical report NEDC-32938P, "Generic Guidelines and Evaluations for General Electric Boiling Water Reactor Thermal Power Optimization (TPO)," Thermal Power Optimization Licensing Topical Report (TLTR), July 2000 (Ref. 6). The TLTR was referred to in several sections of the CNS plant-specific TPO report even though the TLTR covers power uprates only up to 1.5 percent. As discussed in the safety evaluation, the licensee confirmed that the generic evaluation at the 1.5 percent uprate was valid for the CNS 1.62 percent uprate. CNS used one of three approaches for each of its power-dependent analyses of the TPO: (a) the existing evaluation was conducted at 102 percent or greater of CLTP and therefore, it bounded uprate conditions; (b) a new plant-specific evaluation was conducted; or (c) the licensee confirmed that the generic evaluation presented in the TLTR is applicable to CNS at the uprated power level.

The CNS fuel reload analysis is based on the NRC-approved GE methodology described in NEDE-24011-P-A-US, "General Electric Standard Application for Reactor Fuel, Supplement for United States (GESTAR II)," Section 2.2.3.1 (using the revision specified in the CNS core operating limits report). The NRC-approved codes and methodologies used for the licensing safety analyses are also referred to in Section 5 of the CNS TSs. The limiting anticipated operational occurrence (AOO) and accident analyses are reanalyzed for every reload and the safety analyses are documented in Chapter 15 of the CNS updated safety analysis report (USAR). Limiting AOOs and accidents are events that could potentially affect the core operating and safety limits that ensure the safe operation of the plant.

3.6.2 Regulatory Evaluation

In its application, the licensee stated that the measured feedwater (FW) flow is used in the calorimetric calculation of core thermal power and that a more accurate feedwater flow meter will be installed. Previously, a core thermal power uncertainty of 2 percent was assumed with the flow nozzles which are currently used to measure the feedwater flow. The Caldon LEFM CheckPlus™ System will result in a core thermal power uncertainty of ≤ 0.3 percent. The licensee stated that the LEFM CheckPlus™ System uncertainty range (≤0.3 percent) is based on a 95 percent confidence limit. The licensee therefore proposed to operate at a power level closer to the power assumed in the accident analyses, while maintaining a 0.3 percent power measurement uncertainty.

In large part, the basis for acceptability of this proposed amendment is that the MUR power level conditions are bound by the current analyses of record. The NRC staff finds that the licensee's proposed changes were developed consistent with the guidelines in RIS 2002-03 and comply with Appendix K of 10 CFR Part 50.

3.6.3 Technical Evaluation

3.6.3.1 Reactor Core and Fuel Performance

The core thermal-hydraulic design and fuel performance characteristics are evaluated for each fuel cycle in accordance with the NRC-approved GE design criteria, analytical models, and methods described in GESTAR II.

The following sections address the effect of the power uprate on fuel design performance, thermal limits, the power/flow map, and reactor stability.

3.6.3.1.1 Fuel Design and Operation

Fuel bundles are designed to ensure that (1) the fuel bundles are not damaged during normal steady-state operation and AOOs, (2) any damage to the fuel bundles will not be so severe as to prevent control rod insertion when required, (3) the number of fuel rod failures during accidents is not underestimated during accidents, and (4) the coolability of the core is always maintained. For each fuel vendor, the NRC-approved fuel design acceptance criteria and analysis methodology assure that the fuel bundles comply with the objectives of Sections 4.2 and 4.3 of NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants" (the SRP), and the applicable general design criteria (GDC) of 10 CFR Part 50, Appendix A. The 1967 Proposed GDC as described in the USAR, Appendix F, are the licensing basis for CNS; however, the NRC staff concluded in its 1973 Safety Evaluation Report for CNS that the intent of the 1971 Final Rule for 10 CFR Part 50, Appendix A, had also been met. The fuel vendors perform thermal-mechanical, thermal-hydraulic, neutronic, and material analyses to ensure that the fuel system design can meet the fuel design limits during steady-state, AOO, and accident conditions.

The licensee stated that the uprated core for CNS will consist of 548 GE-14 fuel bundles with a batch size of 140 fresh GE-14 bundles, the MUR core will not be a mixed core, GE-14 fuel has been used at CNS since 2000, and that no new fuel types will be introduced in conjunction with the proposed power uprate. The fuel design criteria are based on the NRC-approved methodology described in GESTAR II. The licensee also stated that a new mechanical fuel design is not needed to achieve the 1.62 percent power uprate, though new fuel designs may be used in the future to obtain additional operating flexibility or to maintain the fuel cycle length. The NRC staff concurs with the licensee. The staff finds that the current GE-14 fuel meets the NRC-approved acceptance criteria, and notes that any new fuel designs that do not comply with the NRC-approved fuel design criteria given in GESTAR II would be subject to NRC review and approval.

The slightly higher operating power and the increased steam void content will affect the core and fuel performance. The licensee may also change the power distribution in the reload design to allow more operating flexibility or to maintain the fuel cycle length. This would also affect the core and fuel performance. However, the steady-state and transient design linear heat generation rate limits for each fuel bundle ensure that the fuel plastic strain design limit or the fuel centerline melt limit will not be exceeded. The thermal-hydraulic design and the operating limits will also ensure that the probability of boiling-transition fuel failures will not increase at the uprated conditions.

When a new fuel type is introduced, numerous evaluations are performed as part of the reload process. These evaluations not only confirm that the approved burn-up limits are not exceeded, but address all other impacts this new fuel type may have on operation at the TPO power level, including impacts on stability, thermal-hydraulic compatibility, radiological analyses, and hydrogen generation. The licensee stated that it will follow acceptable methods and processes described in NRC-approved fuel vendor topical reports to perform these analyses and evaluations.

3.6.3.1.2 Thermal Limits Assessment

GDC 10 of 10 CFR Part 50, Appendix A, requires that the reactor core and the associated control and instrumentation systems be designed with appropriate margin to ensure that the specified acceptable fuel design limits are not exceeded during normal operation, including AOOs. Operating limits are established to assure that regulatory and/or safety limits are not exceeded for a range of postulated events (transients and accidents). The safety limit minimum critical power ratio (SLMCPR) protects 99.9 percent of the fuel rods from boiling transition during steady-state operation. The operating limit minimum critical power ratio (OLMCPR) assures that the SLMCPR will not be exceeded as result of an AOO. The operating linear heat generation rate (LHGR) is the core operating limit that assures the fuel thermal-mechanical performance limit (i.e., the 1 percent fuel plastic strain design limit or the no-fuel-centerline-melt criterion) will not be exceeded as a result of an AOO.

The SLMCPR is calculated for every reload at the rated thermal power using NRC-approved methodologies. The SLMCPR is dependent upon the nominal average power level and the uncertainty in its measurement. Consistent with approved practice, a revised SLMCPR is calculated for the first TPO fuel cycle and confirmed for each subsequent cycle.

The OLMCPR is determined on a cycle-specific basis from the results of the reload transient analysis and this approach will not change. AOOs are analyzed at various points in the allowable operating domain, depending on the type of transient. The change in the MCPR is combined with the SLMCPR to establish the OLMCPR, which ensures that 99.9 percent of the rods will not reach boiling transition in the event of an anticipated transient. The licensee will calculate the OLMCPR at the uprated condition for CNS.

The steady-state and transient LHGR limits are established for every fuel design to protect against fuel centerline melt throughout the operating cycle. The licensee will determine the LHGR limits for the uprated cycle in the reload analysis for future cycles, and these limits will be maintained during operation.

The maximum planar linear heat generation rate (MAPLHGR) operating limit is based on the most limiting loss-of-coolant accident (LOCA) and ensures compliance with the ECCS acceptance criteria in 10 CFR 50.46. For every new fuel type, the licensee performs LOCA analyses to confirm compliance with the LOCA acceptance criteria, and for every reload the licensee confirms that the MAPLHGR operating limit for each reload fuel bundle design remains applicable.

Thus, the licensee will calculate the OLMCPR, the SLMCPR, the LHGR, and the MAPLGHR for the uprated conditions as part of the reload analysis using NRC-approved methodologies. The

licensee will propose appropriate changes to the limits in the TSs and/or the core operating limit report (COLR). Therefore, the licensee will continue to meet the intent of GDC 10.

3.6.3.1.3 Reactivity Characteristics

The core analysis done for each fuel reload ensures that the minimum shutdown margin requirements will be met for each core design.

3.6.3.1.4 Stability

CNS is an Option I-D plant. Option I-D is a solution combining prevention and detect-and-suppress elements (Ref. 7). The prevention portion of the solution is an administratively controlled exclusion region. The exclusion region calculation is a confirmation that regional mode instabilities are not probable. The flow-biased Average Power Range Monitor (APRM) scram provides automatic detection and suppression of core wide instabilities. The scram ensures the fuel cladding integrity safety limit is met for thermal hydraulic oscillations. TPO may affect the exclusion region slightly. However, the exclusion region is dependent upon the core loading, then reviewed and adjusted as required, for each reload core. The confirmation that regional mode reactor instability is not probable is also reevaluated when the exclusion region is recalculated. These features will be analyzed for the first core reload analysis that incorporates the new rated power level.

TPO may also affect the SLMCPR protection confirmation slightly. Changes to the nominal flow-biased APRM trip setpoint or the rated rod line require that the hot channel oscillation magnitude portion of the detect-and-suppress calculation be recalculated. The calculation is not dependent upon the core and fuel design. However, the SLMCPR protection calculation is dependent upon the core and fuel design and is performed for each reload. These features will be analyzed for the first reload analysis that incorporates the new rated power level. Therefore, a separate evaluation for Option I-D plants is not required for TPO.

3.6.3.1.5 Reactivity Control

The generic discussion in TLTR Sections 5.6.3 and Appendix J.2.3.3 applies to the CNS plant. The licensee determined that the Control Rod Drive (CRD) and CRD hydraulic systems and supporting equipment are not affected by the TPO uprate and no further evaluation of CRD performance is necessary.

The NRC staff agrees that the proposed power uprate will not have a significant impact on the operation of the CRD system for the following reasons:

1. The operating dome pressure will not change, and the scram timing at steady-state power conditions will not be affected.

2. There must be a minimum pressure differential of 250 psid between the hydraulic control unit (HCU) and the vessel bottom head for normal CRD insertions and withdrawals. Since the operating dome pressure will not increase, the power uprate will have little impact on the CRD pump capacity.

Therefore, the NRC staff agrees that the CRD system will continue to perform all its safety-

related functions at the proposed uprated conditions.

3.6.3.2 Reactor Coolant System and Connected Systems

3.6.3.2.1 Nuclear System Pressure Relief / Overpressure Protection

The safety/relief valves (SRVs) provide overpressure protection for the NSSS during abnormal operational transients. The steam flow associated with the 1.62 percent power uprate can be regulated adequately by adjusting the turbine control valve position; therefore, the operating dome pressure will not increase, and the SRV setpoints and the number of valve actuation groups will not be changed.

Evaluations and analyses for CNS was performed at 102 percent of CLTP to demonstrate that the reactor vessel conformed to ASME B&PV Code and plant Technical Specification requirements. There is no increase in nominal operating pressure for the CNS TPO uprate. There are no changes in the SRV setpoints or valve out-of-service options. There is no change in the methodology or the limiting overpressure event. Therefore, the generic evaluation contained in the TLTR is applicable.

The analysis for each fuel reload will confirm the capability of the system to meet the ASME design criteria. Since the SRVs will actuate at the current setpoints and the current ASME overpressure protection analysis is based on operation at 102 percent power, the NRC staff concurs with the licensee's assessment that the SRVs will have sufficient capacity to handle the increased steam flow associated with the proposed uprate.

3.6.3.2.2 Reactor Recirculation System

The reactor recirculation system (RRS) evaluation described in TLTR Section 5.6.2 applies to CNS. The power uprate will be accomplished by operating along extensions of the rod and core flow lines on the power/flow map. The TPO uprate has a minor effect on the RRS and its components. The TPO uprate does not require an increase in the maximum core flow.

The NRC staff concurs that the changes associated with the 1.62 percent power uprate will have an insignificant impact on the function of the recirculation system. The cycle-specific reload analyses will consider the full range of the power and flow operating region.

3.6.3.2.3 Reactor Core Isolation Cooling System

The generic discussion in TLTR Section 5.6.7 is applicable to CNS. The reactor core isolation cooling (RCIC) system provides core cooling when the reactor pressure vessel (RPV) is isolated from the main condenser and the RPV pressure is greater than the maximum allowable for starting a low-pressure core cooling system. The RCIC system is designed to provide rated flow over a range of reactor pressures.

In the generic 5 percent uprate topical report (Ref. 8), GE performed a generic loss-of-feedwater analysis for all BWR types, core sizes, and RCIC flows to demonstrate that the RCIC system can perform its design basis function for the limiting plant designs.

Since the proposed 1.62 percent power uprate does not increase the steady-state operating pressure or the SRV actuation setpoints, the NRC staff concludes that the RCIC performance would not be affected.

3.6.3.2.4 Residual Heat Removal System

The generic discussion in TLTR Sections 5.6.4 and J.2.3.13 is applicable to CNS. The residual heat removal system (RHR) is designed to restore and maintain the coolant inventory in the reactor vessel and to provide primary-system decay heat removal after reactor shutdown for both normal and post-accident conditions. The RHR system is designed to operate in the low-pressure coolant injection (LPCI) mode, the shutdown cooling mode, the suppression pool and the containment spray cooling mode, and fuel pool cooling assist mode.

The slightly higher decay heat has a small effect on the operation of the RHR system in the shutdown cooling mode. The ability of the RHR system to perform required safety functions was demonstrated with analyses based on 102 percent of CLTP. Therefore, all safety aspects of the RHR system are within previous evaluations. The NRC staff, therefore, concludes that the requirements for the RHR system remain unchanged for TPO uprate conditions.

3.6.3.3 Emergency Core Cooling System (ECCS)

The ECCS is designed to provide protection in the event of a loss-of-coolant accident (LOCA) due to a rupture of the primary-system piping. Although design basis accidents (DBAs) are not expected to occur during the lifetime of a plant, plants are designed and analyzed to ensure that the radiological dose from a DBA will not exceed the 10 CFR Part 100 limits. For a LOCA, 10 CFR 50.46 specifies design acceptance criteria based on (1) the peak cladding temperature, (2) local cladding oxidation, (3) total hydrogen generation, (4) coolable core geometry, and (5) long-term cooling. The LOCA analysis considers a spectrum of break sizes and locations, including a rapid circumferential rupture of the largest recirculation system pipe. Assuming a single failure of the ECCS, the LOCA analyses identify the break sizes that most severely challenge the ECCS systems and the primary containment. The maximum average planar linear heat generation rate (MAPLHGR) operating limit is based on the most limiting LOCA analysis, and the licensees perform LOCA analyses for each new fuel type to demonstrate that the 10 CFR 50.46 acceptance criteria can be met.

The ECCS for CNS includes the high-pressure coolant injection (HPCI) system, the low-pressure coolant injection (LPCI) mode of the RHR system, the low-pressure core spray (CS) system, and the automatic depressurization system (ADS).

3.6.3.3.1 HPCI System

The HPCI system is a turbine driven system designed to pump water into the reactor vessel over a wide range of operating pressures. For the TPO uprate, there is no change to the nominal reactor operating pressure or the SRV setpoints. The primary purpose of the HPCI is to maintain reactor vessel coolant inventory in the event of a small break LOCA that does not immediately depressurize the RPV. The generic evaluation of the HPCI system provided in TLTR Section 5.6.7 is applicable to CNS. The ability of the HPCI system to perform required safety functions is demonstrated with previous analyses based on 102 percent of CLTP. Therefore, the NRC staff concludes that all safety aspects of the HPCI system are within

previous evaluations and the requirements are unchanged for the TPO uprate conditions.

3.6.3.3.2 CS System

The Core Spray (CS) system sprays water into the reactor vessel after it is depressurized. The primary purpose of the CS system is to provide reactor vessel coolant makeup for a large break LOCA and for any small break LOCA after the RPV has depressurized. It also provides spray cooling for long-term core cooling in the event of a LOCA. The generic evaluation of the CS system provided in TLTR Section 5.6.10 is applicable to CNS. The ability of the CS system to perform required safety functions is demonstrated with previous analyses based on 102 percent of CLTP. Therefore, all safety aspects of the CS system are within previous evaluations and the requirements are unchanged for the TPO uprate conditions.

3.6.3.3.3 LPCI Mode of the RHR System

The LPCI mode of the RHR system is automatically initiated in the event of a LOCA. The primary purpose of the LPCI mode is to provide reactor vessel coolant makeup during a large break LOCA or small break LOCA after the RPV has depressurized. The generic evaluation of the LPCI mode provided in TLTR Section 5.6.4 is applicable to CNS. The ability of the RHR system to perform required safety functions required by the LPCI mode is demonstrated with previous analyses based on 102 percent of CLTP. Therefore, all safety aspects of the RHR system LPCI mode are within previous evaluations and the requirements are unchanged for the TPO uprate conditions.

3.6.3.3.4 ADS

The ADS uses the safety/relief valves (SRVs) to reduce reactor pressure after a small-break LOCA with HPCI failure, allowing low-pressure coolant injection (LPCI) and core spray (CS) to provide cooling flow to the vessel. The plant design requires SRVs to have a minimum flow capacity. After a delay, the ADS actuates either on low water level plus high drywell pressure or on low water level alone. The licensee states that the ADS's ability to perform these functions is not affected by the power uprate. Since the small-break LOCA analyses assume that the ADS actuates at a bounding vessel pressure and power, the NRC staff concurs with the licensee's assessment that the proposed power uprate does not affect the capability of the ADS to perform its function.

The generic evaluation of the ADS provided in TLTR Section 5.6.8 is applicable to CNS. The ability of the ADS system to perform required safety functions has been demonstrated with previous analyses based on 102 percent of CLTP. Therefore, all safety aspects of the ADS are within previous evaluations and the requirements are unchanged for the TPO uprate conditions.

3.6.3.3.5 ECCS Performance Evaluation

The ECCS is designed to provide protection against postulated LOCAs caused by ruptures in the primary system piping. The current 10 CFR 50.46, or LOCA, analyses for CNS were performed at 102 percent of CLTP, which is consistent with 10 CFR Part 50, Appendix K.

Experience with power uprates up to 20 percent has shown that there is substantial margin to the 10 CFR 50.46 criteria, including peak cladding temperature (PCT), local cladding oxidation,

core wide metal water reaction, coolable geometry and long-term core cooling.

There are no changes in the plant configuration that would affect the PCT and there is no change in the CNS licensing basis PCT. The analysis presented in Reference 9 demonstrates that for GE-14 fuel, the limiting break and single failure combination is the maximum recirculation line break with battery failure for both nominal and Appendix K assumptions. Based on the limiting large break, and applying the SAFER/GESTR-LOCA methodology, the CNS ECCS-LOCA analysis was performed for the limiting LOCA event for GE-14 fuel. The LOCA results were summarized in the licensee's application (Ref. 5). The results meet all requirements of 10 CFR 50.46.

The CNS LOCA analyses resulted in a licensing basis PCT of 2040 degrees F for GE-14 fuel, maximum cladding oxidation less than 1.0 percent, and maximum hydrogen generation less than 0.1 percent. The results comply with the 10 CFR 50.46 requirements of PCT of less than 2200 degrees F, maximum cladding oxidation less than 17 percent, and maximum hydrogen generation less than 1 percent. The NRC staff accepts the licensee's ECCS performance evaluation, because the analytical models and codes are based on the NRC-approved methodology described in GESTAR II, and the ECCS-LOCA analyses are based on bounding power and flow conditions.

Therefore, the pre-TPO SAFER/GESTR LOCA analysis for GE-supplied fuel bounds a 1.7 percent TPO uprate for CNS. Furthermore, the LOCA analyses of record demonstrate that the HPCI system, the LPCI mode of RHR, the CS system, and the ADS have the capabilities to provide core cooling during a LOCA. The capabilities do not change for operation at the uprated conditions. The ECCS will, therefore, continue to meet the ECCS-LOCA analysis assumptions and design criteria at the uprated conditions.

Since the LOCA analysis is based on an NRC-approved methodology and codes, and the assumed power is bounding, the NRC staff concurs with the licensee's assessment that the ECCS will perform as designed and analyzed at the uprated conditions.

3.6.3.4 Standby Liquid Control (SLC) System

The SLC system evaluation is applicable to and consistent with the evaluation in TLTR Section 5.6.5. The SLC system provides the alternate means of attaining and maintaining cold shutdown conditions, assuming no control rod movement, as required by GDC 26.

The shutdown capability of the SLC system and the boron solution necessary are evaluated each reload cycle. Since the SRV setpoints are not changed for the proposed power uprate, the uprate will have no effect on the rated injection flow. The licensee determined that the capability of the SLC system to provide its backup shutdown function is unchanged and it will continue to meet the requirements of 10 CFR 50.62. Because the uprate will not change the operating parameters of the SLC system, the NRC staff concurs that the SLC will perform acceptably during TPO operation.

3.6.3.5 Reactor Safety Performance Features

3.6.3.5.1 Reactor Transients

AOOs are abnormal transients which are expected to occur one or more times in the life of a plant and are initiated by a malfunction, a single failure of equipment, or a personnel error. The applicable acceptance criteria for the AOOs are based on 10 CFR Part 50, Appendix A, GDC 10, 15, and 20. The 1967 Proposed GDC as described in the USAR, Appendix F, are the licensing basis for CNS; however, the NRC staff concluded in its 1973 Safety Evaluation Report for CNS that the intent of the 1971 Final Rule for 10 CFR Part 50, Appendix A, had also been met. GDC 10 requires that the reactor core and associated control and instrumentation systems be designed with sufficient margin to ensure that the specified acceptable fuel design limits are not exceeded during normal operation and during AOOs. GDC 15 requires that sufficient margin be included to ensure that the design conditions of the reactor coolant pressure boundary are not exceeded during normal operating conditions and AOOs. GDC 20 specifies that a protection system be provided that automatically initiates appropriate systems to ensure that the specified fuel design limits are not exceeded during any normal operating condition and AOOs.

NUREG-0800 provides further guidelines: (1) pressure in the reactor coolant and main steam system should be maintained below 110 percent of the design values according to the ASME Code, Section III, Article NB-7000, “Overpressure Protection”; (2) fuel cladding integrity should be maintained by ensuring that the reactor core is designed to operate with appropriate margin to specified limits during normal operating conditions and AOOs; (3) an incident of moderate frequency should not generate a more serious plant condition unless other faults occur independently; and (4) an incident of moderate frequency, in combination with any single active-component failure or single operator error, should not result in the loss of function of any fission product barrier other than the fuel cladding. A limited number of fuel cladding perforations are acceptable.

Chapter 15 of the CNS updated safety analysis report contains the design basis analyses that evaluate the effects of an AOO resulting from changes in the system parameters such as (1) a decrease in core coolant temperature, (2) an increase in reactor pressure, (3) a decrease in reactor coolant flow rate, (4) reactivity and power distribution anomalies, (5) an increase in reactor coolant inventory, and (6) a decrease in reactor coolant inventory. The facility’s responses to the most limiting transients are analyzed each reload cycle and corresponding changes in the MCPR are added to the safety limit MCPR to establish the operating limit MCPR. A potentially limiting event is an event or an accident that has the potential to affect the core operating and safety limits.

The licensee states that reload analyses are performed at the uprated conditions using an NRC-approved methodology. The licensee has determined that the thermal limits to ensure the fuel cladding integrity will be maintained for operation at the uprated conditions during AOOs and accidents. Since the reload analysis to determine acceptable thermal limits was and will be performed using NRC-approved methodology, the NRC staff finds it acceptable.

3.6.3.5.2 Anticipated Transient Without Scram (ATWS)

ATWS is an AOO with failure of the reactor protection system to initiate a reactor scram to terminate the event. The requirements for ATWS are specified in 10 CFR Part 62. The

regulation requires BWR facilities to have the following mitigating features for an ATWS event:

1. a standby liquid control (SLC) system with the capability of injecting a borated water solution with reactivity control equivalent to the control obtained by injecting 86 gpm of a 13 weight percent sodium pentaborate decahydrate solution at the natural boron-10 isotope abundance into a 251-inch inside-diameter reactor vessel

2. an alternate rod injection (ARI) system that is designed to perform its function in a reliable manner and that is independent all the way from sensor output to the final actuation device

3. equipment to trip the reactor coolant recirculation pumps automatically under conditions indicative of an ATWS

The licensee stated, and the NRC staff concurs, that CNS complies with the ATWS mitigation requirements defined in 10 CFR 50.62. CNS has a SLC capable of boron injection equivalent to 86 gpm, and has installed an Alternate Rod Insertion (ARI) system and automatic Recirculation Pump Trip (RPT) logic.

BWR facilities are also analyzed against certain ATWS acceptance criteria to demonstrate their ability to withstand an ATWS event. These criteria include maintaining fuel integrity (the core and fuel must maintain a coolable geometry), primary system integrity (the peak reactor vessel pressure remains below 1500 psig), and containment integrity (the containment temperature and pressure must not exceed the design limit).

TLTR Section 5.3.5 and Appendix L present a generic evaluation of the sensitivity of BWRs to an ATWS event after a TPO uprate. The topical report provides an ATWS acceptance criterion margin to determine whether a plant-specific evaluation is needed at the TPO power level. In a supplement to the TLTR, GENE revised the ATWS peak pressure margin criterion based on additional sensitivity studies and evaluations. The TLTR states that if the 2°F suppression pool temperature margin criterion is not met, a plant-specific ATWS containment analysis is required. The CNS ATWS analysis, performed at 100 percent of CLTP, demonstrated a margin of 193 psi to the peak vessel bottom head pressure limit and a margin of 16°F to the pool temperature limit. These margins are in excess of the 60 psi and 2°F "sufficient margin" criteria defined in TLTR Appendix L.

Based on the margin criteria and justification provided in the TLTR and the analyses performed by GE and the available margin for peak ATWS parameters, the NRC staff finds the licensee's evaluation to be acceptable, and concurs that CNS meets the ATWS rule requirements specified in 10 CFR 50.62.

3.6.3.6 <u>References for Section 3.6</u>

1. Letter from S. B. Minahan (NPPD) to USNRC, “License Amendment Request to Revise Technical Specifications - Appendix K Measurement Uncertainty Recapture Power Uprate,” November 19, 2007.

2. Letter from S.B. Minahan (NPPD) to USNRC, “Response to Request for

Additional Information Regarding LAR request to revise TS - App. K MUR," March 6, 2008.

3. Caldon Inc., Engineering Report ER-I 57P, "Supplement to Topical Report ER-80P: Basis for a Power Uprate with the LEFM or LEFM CheckPlus™ System," Revision 5, October 2001.

4. Letter from US NRC to M. A. Krupa (Entergy Operations, Inc.), 'Waterford Steam Electric Station, Unit 3; River Bend Station and Grand Gulf Nuclear Station - Review of Caldon, Inc., Engineering Report ER-157P," December 20, 2001 (ADAMS Accession No. ML01350256).

5. GE-Hitachi Nuclear Energy Safety Analysis Report for Cooper Nuclear Station Thermal Power Optimization, NEDC-33385P, November 2007.

6. General Electric, "Generic Guidelines and Evaluations for General Electric Boiling Water Reactor Thermal Power Optimization (TLTR)," Licensing Topical Report NEDC-32938P, July 2000.

7. NEDO-32465-A, "Reactor Stability Detect and Suppress Solutions Licensing Basis Methodology for Reload Applications," August 1996.

8. General Electric, "Generic Evaluation of Boiling Water Reactor Power Uprate," NEDC-31984P, Volume I, July 1991.

9. GE-NE-0000-0037-8293-RO, "NPPD Cooper Nuclear Station SAFERIGESTR Loss-of-Coolant Accident ECCS Analysis with Elimination of 1600 F Upper Bound PCT Limit," September 2005.

3.7 Thermal-Hydraulic Aspects of the LEFM CheckPlus UFM System

The introduction and regulatory evaluation for Section 3.7 are provided in Sections 3.6.1 and 3.6.2 above. References for Section 3.7 are listed in Section 3.7.4.

3.7.1 Background

In the licensee's application (Reference 1), the licensee noted the NRC staff original evaluations (References 2 and 3) of the Caldon Ultrasonic Flow Meter (UFM), and the NRC staff's re-evaluation of hydraulic issues (Reference 4). In Reference 4, the NRC staff provided the following information and conclusions:

- A theoretical description of UFM operation, which showed that flatness ratio, defined as the ratio of the measured average axial velocity at the outside chords to the average axial velocity at the inside chords, can be correlated to the UFM correction factor or calibration coefficient.[1]

[1] This does not apply if the CheckPlus is located too close to a flow perturbation such as an elbow.

- Substantiation that the uncalibrated CheckPlus is typically within a fraction of a percent of the flow rate measured at Alden Research Laboratory (ARL). The average correction factor for the uncalibrated Seabrook Station CheckPlus for a series of five ARL tests with swirl < 2.0 percent was +0.28 percent.

- Substantiation that the CheckPlus is typically relatively unaffected by flow profile distortion and swirl and, further, that the CheckPlus will provide an approximation of the flow profile[2].

- Flatness ratio can be used for correlation of the calibration coefficient so that reliance on a Reynolds Number extrapolation is not necessary to apply ARL test results to plant applications.[3]

- Generically, uncertainty associated with the CheckPlus calibration coefficient is ±0.25 percent.

- ARL flow rate uncertainty for the Seabrook Station CheckPlus calibration was ±0.088 percent.

- The NRC staff finds that the hydraulic aspects of Check and CheckPlus systems have been accurately described in applicable Caldon documentation, that there is a firm theoretical and operational understanding of behavior, and, with one exception, there is no further need to re-examine the hydraulic bases for use of the Check and CheckPlus systems in nuclear power plant feedwater applications. The exception, which should be followed up by Caldon for generic purposes, is to establish the effect of transducer replacement on the Check and CheckPlus system uncertainties.

The applicability of Reference 4 to the licensee's application is addressed in Section 3.7.2 below.

3.7.2 Technical Evaluation

3.7.2.1 Installation

The licensee describes in References 1 and 5 the CheckPlus UFMs as to be installed in accordance with the requirements of References 2 and 3. Each CheckPlus is to be installed in a pipe in the Feedwater Pump Room downstream of a reducer in a short section of straight pipe that is followed by an elbow and other hardware. Upstream of each reducer, the flow splits from a short pipe section that in turn receives water from each of the two feedwater trains that may differ in both flow rate and temperature. Thus, the CheckPlus may operate in a region where the flow profile is poorly developed and temperature may not be uniform in a plane perpendicular to the pipe centerline. These aspects are addressed in Section 3.7.2.4 below.

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[2]This conclusion does not apply if the flow profile consists of multiple individual flow paths such as may exist immediately downstream of a tubular flow straightener or if certain distortion of the flow profiles occurs.
[3] See Footnote 1.

3.7.2.2 CheckPlus Inoperability

To operate above the presently licensed power of 2381 MWt, the licensee proposes that the CheckPlus cannot have been out-of-service for more than 72 hours and there cannot have been any power changes that exceed 10 percent during the 72 hours. Power during the 72 hours without an operational CheckPlus will be monitored using the existing flow rate instrumentation that has been recalibrated to agree with CheckPlus before the failure. The licensee justifies the operation on the basis that there is not a significant uncertainty associated with using the existing flow rate instrumentation for 72 hours as long as the plant is essentially operated in a steady state condition and, if a power change in excess of 10 percent should occur during the 72 hours, then the plant thermal power will be reduced to the presently licensed 2381 MWt and the flow rate will be determined using the present, pre-uprate flow rate instrumentation calibration. Stated differently, after 72 hours without an operable CheckPlus, or if core thermal power changes by more than 10 percent while the CheckPlus is inoperable, the plant will be operated as though the CheckPlus was never installed and the power uprate was not in effect. These actions are to be covered by Technical Requirements Manual T3.3.5. The NRC staff finds that operation with an inoperable CheckPlus has been acceptably addressed.

3.7.2.3 Transducer Replacement

The Reference 4 qualification to establish the effect of transducer replacement on the Check and CheckPlus system uncertainties has been addressed in References 6 and 7. A number of tests were conducted in which the transducers were removed and replaced for each test. Each of the tests consisted of a statistically meaningful number of individual determinations of the calibration factor. The calibration factors, and uncertainty associated with each calibration factor, were provided. Calibration factor variation was shown to be bounded by changes in the fourth significant figure. The bounding uncertainty due to transducer installation variability was incorporated into the overall CheckPlus uncertainty calculation in Reference 8. This added an uncertainty that was not addressed in the older References 2, 3, and 9. Further, there are significant differences between the bounding uncertainties in the older references and in the licensee's submittal. Consequently, the NRC staff assessed these differences and found them to be justified and acceptable. Therefore, the NRC staff finds that transducer installation variability has been acceptably addressed.

3.7.2.4 CheckPlus Calibration

CheckPlus calibration was accomplished at ARL. Reference 8 covers the ARL test configuration. The NRC staff compared the test configuration to drawings and information in Reference 5 and noted the following:

- The two 18 inch plant feedwater pipes approach each other on the same pipe axis and join in a 24 inch common pipe section after each bends 90 degrees over a 3 foot length to become parallel. The ARL test does not simulate this configuration. Instead, there is a short pipe section downstream of the ARL pump discharge manifold that ends in either a flow straightener or an eccentric orifice followed by a reducer that increases pipe diameter from 18 inches to 24 inches.

- The plant common pipe section then splits into two parallel 24 inch pipes. Each of these pipes has a 24 inch to 18 inch reducer followed by a CheckPlus a short distance downstream of the reducers. The ARL test has a similar configuration with a CheckPlus located in one 18 inch pipe and the other provides a bypass to simulate unequal flow into the two downstream feedwater pipes following the exit from the flow split.

Both configurations will generate a non-symmetric velocity distribution at the CheckPlus location that will change as a function of individual train flow rates. Non-uniform flow rate in the plant will generate a skewed distribution at the entrance to the common pipe and a skewed but not identical distribution will also be generated during the eccentric orifice tests.

Calibration tests were conducted with the CheckPlus UFMs with flow rates ranging from all flow through the CheckPlus to a 50 / 50 split through the CheckPlus and the bypass, use of the eccentric orifice, and a change in CheckPlus orientation. Each test consisted of a statistically meaningful number of individual tests with multiple CheckPlus flow rate indications per individual test. The maximum change in calibration factor was a fraction of a percent and was observed during the high swirl rate tests. As was the case for previous reviews involving use of the CheckPlus, the test temperature was room temperature in contrast to the licensee's plant feedwater temperature of 367 ^{o}F. A correlation of flatness ratio with calibration factor was not used to extrapolate the test results to plant operating conditions, as was used for some previous CheckPlus applications and which was discussed in Reference 4, because of the distorted flow profile. Consequently, a straight line fit to a function of Reynolds number was used to extrapolate the test data. This increased the test correction factor. The licensee accounted for this in its plant correction factor.

Operation with unequal heating from the feedwater heaters could introduce thermal stratification in feedwater passing through the CheckPlus. The NRC staff addressed this situation in Reference 4 by noting that, "where feedwater enters a common header upstream of the UFM, there is a possibility that temperature will not be uniform at the UFM location. In some cases with off-normal feedwater heater operation, a temperature difference of as much as 30°F or 40°F may occur." The CheckPlus, by measuring transit time in both directions in 8 paths, will provide the average velocity of sound along those eight paths. The sound velocity will, in turn, provide temperatures. Therefore, the NRC staff concluded in Reference 4 "that CheckPlus would continue to measure bulk average feedwater flowrate within its design basis accuracy because the sound velocity is integrated over the pipe cross section." The use of CheckPlus to determine temperature and the associated temperature uncertainty is addressed by Caldon in Appendix A.2 of Reference 10, and the licensee confirmed in Reference 5 that the CheckPlus will be used to determine feedwater temperature. The NRC staff finds that the feedwater temperature will be determined consistent with the stated uncertainty.

3.7.3 Conclusion

The proposed license amendment is based on the use of the Caldon LEFM CheckPlus UFM system that would decrease the uncertainty in the FW flow rate, thereby decreasing the power level measurement uncertainty.

The NRC staff finds that the hydraulic aspects of the Caldon LEFM CheckPlus UFM system have been accurately described in applicable documentation and that there is a firm theoretical

and operational understanding of behavior. The NRC staff further finds that the calibration accomplished at ARL is appropriate for CheckPlus installation at CNS and is acceptable. Therefore, the NRC staff concluded, based on the considerations discussed above, that the proposed changes are acceptable with respect to the hydraulic aspects of the CheckPlus UFM when installed at CNS.

3.7.4 References for Section 3.7

1. Letter from S. B. Minahan (NPPD) to USNRC, "License Amendment Request to Revise Technical Specifications - Appendix K Measurement Uncertainty Recapture Power Uprate," ML073300571, November 19, 2007.

2. "Improving Thermal Power Accuracy and Plant Safety While Increasing Operating Power Level Using the LEFM Check™ System," Caldon Topical Report ER-80P, Rev 0, March 1997.

3. "Supplement to Caldon Topical Report ER-80P: Basis for Power Uprates with an LEFM Check or an LEFM CheckPlus," Caldon Engineering Report ER-157P, Revision 5, October, 2001.

4. Letter to Ernest M. Hauser (Caldon, Inc.) from Brian E. Thomas (USNRC), "Evaluation of the Hydraulic Aspects of the Caldon Leading Edge Flow Measurement (LEFM) Check and CheckPlus™ Ultrasonic Flow Meters (UFMs) (TAC No. MC6424)," ADAMS Accession No. ML061700222, July 5, 2006.

5. Letter from S. B. Minahan (NPPD) to USNRC, "Response to Request for Additional Information for License Amendment Request to Revise Technical Specifications - Appendix K Measurement Uncertainty Recapture Power Uprate," ADAMS Accession No. ML080990523, April 4, 2008.

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

7. "Flow Measurement Uncertainty Due To Transducer (Re)placement in Caldon® LEFM Check and CheckPlus Systems," Caldon Ultrasonics, PR-612P Rev. 0, ADAMS Accession No. ML070870441 (proprietary), ADAMS Accession No. ML070870435 (non-proprietary), March 15, 2007.

8. "LEFM✔ + Meter Factor Calculation and Accuracy Assessment for Cooper NPPD," Caldon Ultrasonics, ER-614 Revision 1, (proprietary), September, 2007. (Enclosure 5 to Reference 1.)

9. "Supplement to Topical Report ER-80P: Basis for a Power Uprate with the LEFM System," Caldon Engineering Report ER-160P, May 2000.

10. "Engineering Report: ER-477, Bounding Uncertainty Analysis for Thermal Power Determination at Cooper NPPD using the LEFM✔+ System," Caldon Ultrasonics, ER-592 Rev. 2, (proprietary), September, 2007. (Enclosure 4 to Reference 1.)

3.8 Vessel and Internals Integrity

The NRC staff review in the area of reactor pressure vessel (RPV) and RPV internals integrity for boiling water reactors focuses on the impact of the proposed MUR power uprate on adjusted reference temperature (ART) calculations, fluence evaluations, pressure-temperature (P-T) limit curves, upper-shelf energy (USE), surveillance capsule withdrawal schedules, and RPV internals. The review is conducted to verify that the results of licensee's analyses related to these areas continue to meet the requirements of 10 CFR 50.60, 10 CFR 50.55a, and 10 CFR Part 50, Appendices G and H following implementation of the proposed MUR power uprate. The guidance contained in RIS 2002-03 has been used by the NRC staff to conduct the review.

3.8.1 RPV Material Surveillance Program

3.8.1.1 Regulatory Evaluation

The RPV material surveillance program provides a means for determining and monitoring the fracture toughness of the RPV beltline materials to support analyses for ensuring the structural integrity of the ferritic components of the RPV. Appendix H of 10 CFR Part 50 provides the staff requirements for the design and implementation of the RPV material surveillance program. The NRC staff review primarily focused on the effects of the proposed MUR power uprate on the licensee's RPV surveillance capsule withdrawal schedule.

3.8.1.2 Technical Evaluation

Regarding the RPV surveillance program and capsule withdrawal schedule, the licensee concluded in Section 3.2.1(e) of Enclosure 1 to the submittal that, "TPO [thermal power optimization] has no effect on the existing surveillance schedule."

The licensee's RPV material surveillance program is an integrated surveillance program (ISP) designed by the Boiling Water Reactor Vessel and Internals Project (BWRVIP) for operating BWR plants. The ISP is documented in BWRVIP-78, "BWR Vessel and Internals Project, BWR Integrated Surveillance Program Plan," and BWRVIP-86A, "BWR Vessel and Internals Project, Updated BWR Integrated Surveillance Program (ISP) Implementation Plan." Both were reviewed and approved by the NRC in a SE dated February 1, 2002. The implementation of the BWRVIP ISP at CNS was approved in an SE dated October 31, 2003, which concluded that the BWRVIP ISP can be implemented for CNS as the basis for demonstrating the facility's continued compliance with the requirements of Appendix H to 10 CFR Part 50.

Table 3-1 of Enclosure 3 to the submittal reported the peak end-of-license (EOL), i.e., 32 effective full power years (EFPY), inside diameter (ID) fluence as 1.23×10^{18} n/cm^2 (E>1.0 MeV) for the unit's lower shell plates and 1.68×10^{18} n/cm^2 (E>1.0 MeV) for the unit's lower-intermediate shell plates and all welds. The NRC staff confirmed that these fluence values are negligibly greater than the values (1.22×10^{18} n/cm^2 (E>1.0 MeV) and 1.67×10^{18} n/cm^2 (E>1.0 MeV)) used in the most recent approved P-T limits that were evaluated in an NRC staff SE dated January 24, 2006. Based on the above, the NRC staff determined that the negligible change of the EOL ID fluence will have essentially no impact on the EOL transition temperature shift values and, therefore, on the capsule withdrawal schedule of BWRVIP ISP, currently in the CNS licensing basis. Therefore, the NRC staff determined that the CNS RPV surveillance program

would continue to meet the requirements of 10 CFR Part 50, Appendix H under the MUR power uprate condition.

3.8.2 Pressure-Temperature Limits and Upper-Shelf Energy

3.8.2.1 Regulatory Evaluation

10 CFR Part 50, Appendix G provides fracture toughness requirements for ferritic (low alloy steel or carbon steel) materials in the reactor coolant pressure boundary (RCPB), including requirements on the USE values used for assessing the safety margins of the RPV materials against ductile tearing and for calculating P-T limits for the plant. The P-T limits are established to ensure the structural integrity of the ferritic components of the RCPB during any condition of normal operation, including anticipated operational occurrences and hydrostatic tests. The NRC staff review of the USE assessments covered the impact of the MUR power uprate on the neutron fluence values for the RPV beltline materials and the USE values for the RPV materials through the end of the current licensed operating period. The NRC staff P-T limits review covered the P-T limits methodology and the calculations for the number of the EFPY specified for the proposed MUR power uprate, considering neutron embrittlement effects.

3.8.2.2 Technical Evaluation

Regarding the topic of the RPV P-T limits, the licensee concluded in Section 3.2.1(c) of Enclosure 3 to the application that, "[t]he current P-T curves bound TPO operation up to 28 EFPY. CNS will revise the curves for operation beyond 28 EFPY."

The current CNS TS contain P-T limit curves identified as being applicable through 30 EFPY. These curves are based on a projected peak RPV ID fluence of 1.57 x 10^{18} n/cm^2 (E>1.0 MeV), as evaluated in the SE dated January 24, 2006. The MUR power uprate projected fluence at 32 EFPY is 1.68 x 10^{18} n/cm^2 (E>1.0 MeV), or approximately 1.575 x 10^{18} n/cm^2 (E>1.0 MeV) at 30 EFPY according to the staff calculation. Since the difference is so small (1.575 x 10^{18} n/cm^2 versus 1.57 x 10^{18} n/cm^2), the NRC staff issued a supplemental RAI letter asking the licensee to confirm the need to revise the CNS P-T limit curves to 28 EFPY. The licensee provided in its letter dated March 6, 2008, the number of EFPYs (29.9 EFPY) for which the current P-T limit curves are good considering the MUR power uprate, and concluded that the P-T limit curves should be limited to 28 EFPY. The NRC staff considered the measure very conservative, but acceptable to the NRC staff. Hence, the NRC staff confirmed that the CNS RPV materials would continue to meet the requirements of 10 CFR Part 50, Appendix G under the MUR power uprate condition. It is worthwhile to mention that Regulatory Guide (RG) 1.99, Revision 2 provides no requirements on ART values for the RPV materials of operating plants. The adequacy of ART values are reflected in the P-T limit evaluation.

Regarding the topic of the RPV USE, the licensee concluded in Section 3.2.1(a) of Enclosure 3 to the application that:

> The upper shelf energy (USE) for the plate materials remains greater than 50 ft-lb for the design life of the vessel and maintains the margin requirements of 10 CFR 50 Appendix G. Equivalent margin for the weld material remains greater than the required 35 ft-lb per Code Case N-512. The maximum % decrease for the plate beltline materials is 21% of 32 EFPY and the USE is 43.67 ft-lb for the limiting weld.

The NRC staff evaluated the information provided by the licensee in the submittal as well as information regarding the equivalent margins analysis (EMA) contained in NEDO-32205-A, "10 CFR 50 Appendix G Equivalent Margin Analysis for Low Upper Shelf Energy in BWR/2 through BWR/6 Vessels." For plates which can be evaluated by the RG 1.99, Revision 2 method, the NRC staff found that the EOL USE values for the plate materials remain greater than 50 ft-lb, with an USE value of 60 ft-lb for limiting lower shell plate G-2803-3. For welds which require an EMA because of their lack of initial USE values, the staff found that the NEDO-32205-A EMA was based on the BWR fleet bounding weld of 0.35% copper (Cu) and 2.4 x 10^{18} n/cm^2 (E>1.0 MeV) fluence at a quarter wall thickness (1/4T) from the RPV ID. The corresponding values for the limiting CNS RPV weld are 0.27% Cu and 1.22 x 10^{18} n/cm^2 (E>1.0 MeV) fluence at 1/4T. Therefore, the NRC staff confirmed that the CNS RPV materials are bounded by the NEDO-32205-A EMA and continue to meet the USE criteria requirements of 10 CFR Part 50, Appendix G under the MUR power uprate condition.

Therefore, the NRC staff finds the proposed MUR power uprate acceptable with respect to the P-T limits and USE.

3.8.3 Reactor Internal and Core Support Materials

The reactor internals and core supports include structures, systems, and components (SSCs) that perform safety functions or whose failure could affect safety functions performed by other SSCs. The safety functions include reactivity monitoring and control, core cooling, and fission product confinement within both the fuel cladding and the reactor coolant system (RCS).

3.8.3.1 Regulatory Evaluation

The NRC staff review covered the materials' specifications and mechanical properties, welds, weld controls, nondestructive examination procedures, corrosion resistance, and susceptibility to degradation. The NRC acceptance criteria for reactor internal and core support materials is based on draft GDC-1 and 10 CFR 50.55a for material specifications, controls on welding, and inspection of reactor internals and core supports. Specific review criteria are contained in NUREG-0800, Section 4.5.2, BWRVIP-26-A, "BWR Top Guide Inspection and Flaw Evaluation Guidelines," RIS 2002-03, and Matrix 1 of RS-001, "Review Standard for Extended Power Uprates."

3.8.3.2 Technical Evaluation

Reactor internals and core support materials are subject to the following degradation mechanisms:

- Cumulative fatigue damage
- Crack initiation and growth due to flow induced vibration
- Crack initiation and growth due to stress corrosion cracking (SCC), intergranular stress corrosion cracking (IGSCC) and irradiation assisted stress corrosion cracking (IASCC)
- Loss of fracture toughness due to thermal aging and neutron embrittlement

Cumulative fatigue damage and crack initiation and growth due to flow induced vibration are discussed in Section 3.5 of this evaluation. Crack initiation and growth due to SCC and loss of fracture toughness due to thermal aging and neutron embrittlement are managed through the inservice inspection (ISI) program that conforms to the requirements of 10 CFR 50.55a and the BWRVIP. The BWRVIP supplements the ISI program required by 10 CFR 50.55a. The BWRVIP program was reviewed and approved by the NRC.

The licensee belongs to the BWRVIP organization and implementation of the procedurally-controlled program is consistent with the BWRVIP-issued documents. The inspection strategies recommended by the BWRVIP consider the effects of fluence on the applicable components and are based on component configuration and field experience. The RPV internals integrity is maintained consistent with the BWRVIP and established industry guidelines, except where technical justifications in accordance with the BWRVIP-94 report, "Program Implementation Guide," have been documented.

Note 1 in Matrix 1 of Section 2.1 of RS-001 indicates that guidance on the neutron irradiation-related threshold for inspection for IASCC in BWRs is provided in the BWRVIP-26 report. The *Final License Renewal SER for BWRVIP-26*, dated December 7, 2000, states that the threshold fluence level for IASCC is 5×10^{20} n/cm^2 (E > 1 MeV). The NRC staff, in past extended power uprate reviews for units such as Browns Ferry, Unit 1 and the Susquehanna units, identified five RPV internal components that may have EOL fluence values exceeding the threshold fluence level for IASCC (the top guide, core shroud, core plate, and incore instrumentation dry tubes and guide tubes). These components will be inspected and managed using the guidance in the relevant BWRVIP reports. It should be mentioned, however, that the past extended power uprate applications were for power uprates several times greater than that of the proposed MUR power uprate for CNS.

The proposed MUR power uprate only increases the RPV EOL fluence insignificantly (1.575×10^{18} n/cm^2 versus 1.57×10^{18} n/cm^2). It does not affect the current licensing basis regarding the inspection and evaluation program for monitoring IASCC for CNS RPV internal components. Further, the main driving force for IASCC, i.e., the governing stresses for all RPV internal components under the MUR power uprate condition remain bounded by the current design basis, as indicated in Section 3.3.2 of the submittal.

The NRC found in past power uprate reviews that the inspection and evaluation guidelines in BWRVIP-76, "BWR Vessel and Internal Project BWR Core Shroud Inspection and Flaw Evaluation Guidelines," BWRVIP-25, "BWR Vessel and Internal Project BWR Core Plate Inspection and Flaw Evaluation Guidelines," and BWRVIP-47, "Boiling Water Reactor Lower Plenum Inspection and Flaw Evaluation Guidelines," are acceptable to manage IASCC in core shroud, core plate, and incore instrumentation dry tubes and guide tubes. The only issue unresolved is related to the top guide. The BWRVIP-26-A states that there is no safety consequence resulting from a failure at a single beam intersection and that a large number of complete separations would need to occur before control rod insertion would be affected.

The NRC staff position regarding this issue is that multiple failures of the top guide beams are possible when the threshold fluence for IASCC is exceeded. The staff expects to resolve this issue on a generic basis, as the BWRVIP is working with the NRC staff to resolve this issue. Since the fluence increase due to the MUR power uprate is insignificant, and crack initiation and

growth due to SCC and loss of fracture toughness due to thermal aging and neutron embrittlement are managed through the licensee's ISI program that conforms to the requirements of 10 CFR 50.55a and the BWRVIP, the staff determined that IASCC will not be a concern for the top guide for the MUR power uprate.

3.8.3.3 Conclusion

The NRC staff reviewed the licensee's evaluation of the effects of the proposed MUR power uprate on the susceptibility of reactor internal and core support materials to known degradation mechanisms and concludes that the proposed MUR power uprate only increases the RPV EOL fluence insignificantly and does not affect the acceptability of the inspection and evaluation program for monitoring IASCC in CNS RPV internal components. Hence, the NRC staff concludes that the reactor internal and core support materials will continue to be acceptable and will continue to meet the requirements of draft GDC-1 and 10 CFR 50.55a with respect to material specifications, welding controls, and inspection following implementation of the proposed MUR power uprate.

3.8.4 Overall Conclusion for Section 3.8

The NRC staff reviewed the licensee's evaluation of the effects of the proposed MUR power uprate on the RPV and RPV internals integrity and the susceptibility of reactor internal and core support materials to known degradation mechanisms and concludes that the licensee addressed surveillance capsule withdrawal schedule, pressure-temperature limit curves and upper-shelf energy, and the reactor core support structures and vessel internals degradation satisfactorily. Hence, the NRC staff determined that the changes identified in the proposed LAR will not impact the remaining safety margins required for the above-mentioned structural integrity assessments.

3.9 Electrical Systems

3.9.1 Regulatory Evaluation

The licensee developed its license amendment request (LAR) to be consistent with the guidelines in RIS 2002-03.

The 1967 Proposed GDC as described in the USAR, Appendix F, are the licensing basis for CNS; however, the NRC staff concluded in its 1973 Safety Evaluation Report for CNS that the intent of the 1971 Final Rule for 10 CFR Part 50, Appendix A, had also been met. The NRC staff reviewed the application to verify that the intent of the following GDC continued to be met:

- General Design Criterion (GDC) 17, “Electric power systems,” of 10 CFR Part 50, Appendix A, which requires that an onsite power system and an offsite electrical power system be provided with sufficient capacity and capability to permit functioning of structures, systems, and components important to safety;

The regulatory requirements which the NRC staff applied in the review of the application included:

- Section 50.63 of 10 CFR, which requires that all nuclear plants have the capability to withstand a loss of all alternating current (AC) power (station blackout (SBO)) for an established period of time, and to recover therefrom; and

- Section 50.49 of 10 CFR, “Environmental Qualification (EQ) of Electric Equipment Important to Safety for Nuclear Power Plants,” which requires licensees to establish programs to qualify electric equipment important to safety.

3.9.2 Technical Evaluation

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

- AC Distribution System
- Power Block Equipment (Generator, Exciter, Transformers, Isolated-phase bus duct, Generator circuit breaker)
- Direct current (DC) system
- Emergency Diesel Generators (EDGs)
- Switchyard
- Grid Stability
- SBO
- Equipment Qualification Program

3.9.2.1 AC Distribution System

The AC Distribution System at CNS is the source of power to station auxiliaries and critical service loads. The AC Distribution system consists of the 4160 volt (V), 480 V, 115/230 V and 120/240 V systems. The licensee stated that the onsite power distribution loads were reviewed under normal and emergency operating scenarios. At the uprated power level, the loads are expected to operate at or below the nameplate rating for running kilowatts (kW) or brake horsepower (BHP) under both normal and emergency conditions. The licensee further stated that there are negligible changes to the load, voltage drop, and short circuit current values.

At the uprated power level, the condensate and condensate booster pumps will experience increased flow and pressure but will remain bounded by the existing design. The licensee stated that no increase in flow or pressure is required of the emergency core cooling system (ECCS) equipment. This provides an indication that the current emergency power system remains adequate.

Based on this information, the NRC staff finds that the existing AC distribution system will be able to support the loading for uprated conditions.

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

As a result of the power uprate, CNS rated thermal power will increase to 2419 MWt from the previously analyzed core power level of 2381 MWt. The generator is rated at 983 megavolt ampere (MVA) with a 0.85 power factor (pf). Currently, the generator outputs 815 MWe at 0.85 pf. At uprated conditions, CNS will produce 830.4 MWe at 0.85 pf, with administrative limits of

835.5 MWe and 150 megavolt ampere reactive (MVAR). The increase in electrical output remains bounded by the design ratings of the generator. In its supplemental letter dated March 6, 2008, the licensee stated that generator operation at uprated conditions remains bounded by the station design analysis and operating procedures. Based on this information, the NRC staff finds that the generator is capable of operation at uprated conditions.

Since the main generator will continue to operate within the exiting rating following the uprate, the existing isophase bus continuous current rating will not be challenged. Based on this, the NRC staff finds that the licensee's existing analyses that establish the fault and continuous ratings for the isophase bus remain bounding.

The normal station service transformers (NSSTs) provide power to the onsite distribution system during normal operations. At uprated conditions, the NSSTs will be operating below their design limit as shown in the March 6, 2008 letter. The startup station service transformer (SSST) supplies power to the onsite distribution system when the main generator is offline. In its March 6, 2008, supplemental letter, the licensee indicated that the SSST will be operating below its MWe design value and as such, operation at uprated conditions is bounded by design analyses. The emergency station service transformers (ESSTs) provide an additional source of power to the Class 1E buses. As stated in the licensee's March 6, 2008, supplemental letter, the load increase on the ESSTs is negligible since they only supply power to the Class 1E buses and therefore, the analyses for the ESSTs bound the MUR power uprate conditions. Based on this information, the NRC staff finds that the analyses for the transformers bound the MUR power uprate conditions.

The main generator voltage is stepped up to 345 kilovolts (kV) by the main transformer bank. The main transformer bank has three single phase units, each unit rated at 336 MVA and thus, the main transformer bank is rated at 1008 MVA. At uprated conditions, the maximum generator output is 983 MVA, which is below the rating of the main transformer bank. Therefore, the NRC staff finds that the main transformer bank is capable of operation at uprated conditions.

The small increase in generator output (15.4 MWe) does not cause overloading of the generator, iso-phase bus ducts or the transformers. Therefore, the ratings of the CNS transformers would not be impacted by MUR power uprate conditions.

3.9.2.3 <u>DC System</u>

The station 125 VDC and 250 VDC systems are each comprised of two batteries, three battery chargers, and distribution equipment that supply DC power for startup, operation, shutdown, and safety-related loads. Additionally, there are two 24 VDC systems, with each system consisting of two batteries, two battery chargers and associated distribution equipment. The purpose of the 24 VDC systems is to provide uninterruptible DC power to neutron monitoring and process radiation monitoring instrumentation.

The licensee stated that the DC system loads will continue to operate at or below the nameplate rating for running kW or BHP under both normal and emergency conditions following the uprate. The NRC staff reviewed the LAR and CNS's Updated Safety Analysis Report (USAR) and confirmed that the power uprate does not impact DC system loads. Therefore, the NRC staff finds that the analyses for DC system bound MUR power uprate conditions.

3.9.2.4 Emergency Diesel Generators

The EDG system provides a safety-related source of AC power to sequentially energize and restart loads necessary to shutdown the reactor safely, to maintain the reactor in a safe shutdown condition, and operate all auxiliaries necessary for safety. The EDG system is capable of performing this function during a loss of offsite power. There are two EDG sets of identical design, each dedicated to one of the 4160 V critical service buses (Class 1E), which supply power to critical loads required during abnormal operational transients and accidents.

The USAR discusses the loading of the EDGs for the worst case condition, a loss of coolant accident (LOCA) with a loss of offsite power. According to the licensee, the emergency core cooling pumps will continue to operate at or below the nameplate rating and within the calculated BHP following the power uprate. Hence, the EDG system has adequate capacity and capability to power the safety-related loads at MUR power uprate conditions.

Based on the above, the NRC staff, after reviewing the LAR and USAR, finds that the power uprate does not impact EDG system loads. Therefore, the NRC staff finds that the analyses for the EDG system bound MUR power uprate conditions.

3.9.2.5 Switchyard

The switchyard equipment and associated components are classified as non-safety related. The primary function of the CNS 161 kV and 345 kV switchyards and distribution system is to distribute the generated power to the transmission grid. In addition, the switchyards provide AC power for station startup and shutdown. The 345 kV switchyard has five incoming transmission lines whereas the 161 kV switchyard has one transmission line. In addition, a 69 kV transmission line can provide AC power in the event of an emergency shutdown. The 345 kV switchyard supplies the onsite distribution system through the NSST while the 161 kV switchyard supplies power via the SSST. The licensee stated that there are no modifications required for the power uprate that would increase the electrical loads beyond those levels previously analyzed.

The NRC staff confirmed that the small increase in plant output will not significantly impact the switchyard equipment. Therefore, the NRC staff finds that the capability of the high-voltage switchyard to support the transmission lines and supply power to various breakers and other equipment in the switchyard would not be adversely impacted by the MUR power uprate.

3.9.2.6 Grid Stability

The grid stability impact of the power uprate is discussed in Section 6 of Enclosure 3 of the LAR, and the licensee concludes that there is no significant effect on grid stability or reliability. The staff requested additional information on the grid stability study, specifically asking the licensee for the assumptions, methodology, and cases studied to support the conclusion that the uprate does not impact grid stability. In its March 6, 2008, supplemental letter, the licensee stated that a sensitivity analysis was performed to determine the impacts of different generator output levels on its LOCA analysis. The licensee further stated that the pre-LOCA generation levels had negligible impact on the post-LOCA voltage levels at the CNS critical buses. In its April 4, 2008 letter, the licensee stated that the stability sensitivity study evaluated all worst case disturbances associated with CNS operating at uprated conditions. This included evaluating the impact of

transmission line outages, the loss of CNS, as well as the loss of other generating units. The licensee stated that the study demonstrated a stable system response to all disturbances.

In its March 6, 2008, supplemental letter, the licensee stated that the power uprate should not affect the MVAR support needed to maintain post-trip loads and minimum voltage levels. Presently, the ESST has a 5.4 MVAR capacitor bank to maintain pre-accident voltage levels. The licensee stated that the SSST does not have any active MVAR support and an administrative limit of 150 MVARs has been established for the main generator. Furthermore, the 161 kV system maintains a pre-accident voltage at or above 167.5 kV. Based on this information, the NRC staff finds that the MVAR support is adequate to maintain post-trip loads and minimum voltage levels.

The NRC staff reviewed the grid stability study, and finds that the CNS MUR power uprate allows for continued stable and reliable grid operation.

3.9.2.7 Station Blackout

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

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

The licensee stated that the evaluation for SBO included the adequacy of condensate/reactor coolant inventory, the capacity of the Class IE batteries, the SBO compressed nitrogen requirements, the ability to maintain containment integrity, and the effect of loss of ventilation on rooms that contain equipment essential for plant response to an SBO event. The proposed MUR has no affect on CNS's station battery capacity as the MUR does not increase loads. Currently, CNS has adequate margin for condensate inventory as well as containment peak temperature. The licensee stated that CNS has margins of 31,568 gallons to the available condensate storage inventory and 13 degrees Fahrenheit (°F) to the containment peak temperature. Based on this information, the NRC staff finds that the MUR power uprate will have no impact on CNS's SBO coping duration. Therefore, the NRC staff finds that CNS will continue to meet the requirements of 10 CFR 50.63 under power uprate conditions.

3.9.2.8 Equipment Qualification Program

In the licensee's LAR, the licensee stated that the EQ of electrical equipment was performed at a core power level of ≥ 102% of 2381 MWt, which bounds the MUR operating conditions. The MUR power uprate causes the system operating temperatures and pressures to change slightly. Specifically, the feedwater lines near the pump discharge experience an increase of < 2°F and < 5 pounds per square inch (psi), and the recirculation lines increase < 1°F and < 1 psi. Although the radiation levels may increase slightly due to the power uprate, the licensee stated that the environmental envelope for radiation is not exceeded. Thus, there is adequate margin in the EQ envelopes to accommodate the small changes in temperature, pressure, radiation,

and humidity due to the MUR. Based on this information, the NRC staff finds that the current EQ parameters remain bounding for the MUR power uprate. Therefore, the NRC staff finds that the MUR power uprate will have no impact on CNS's EQ Program and continue to meet the requirements of 10 CFR 50.49.

3.9.3 Conclusion for Section 3.9

Based on the technical evaluation provided above, the NRC staff finds that CNS will continue to meet 10 CFR 50.63, 10 CFR 50.49, and the intent of GDC 17. Therefore, the NRC staff finds the MUR power uprate acceptable.

3.10 Instrumentation & Controls (I&C)

3.10.1 Introduction

The licensee's request to increase the core thermal power rating of CNS by 1.62 percent from 2381 MWt to 2419 MWt is based on a reduced measurement uncertainty of core thermal power due to the installation of a Caldon LEFM CheckPlus™ (√+™) system to measure feedwater (FW) flow at CNS. The licensee's application referenced Caldon Topical Report ER-80P, "Improving Thermal Power Accuracy and Plant Safety While Increasing Operating Power Level Using the System," and its supplement ER-157P, "Supplement to Topical Report ER-80P: Basis for a Power Uprate with the LEFM √ ™ [Check ™] or LEFM √+™ System." These two reports together provide a generic basis for the proposed 1.62 percent power uprate. The Caldon Topical Report ER-80P and its Supplement ER-157P were approved by letter from J. Hannon (NRC) to C. Terry (TU Electric), "Staff Acceptance of Caldon Topical Report ER-80P: Improving Thermal Power Accuracy While Increasing Power Level Using the LEFM System," dated March 8, 1999, and letter from S. Richards (NRC) to Michael A. Krupa (Entergy), "Review of Caldon, Inc. Engineering Report ER-157P," dated December 20, 2001, respectively.

The plant specific basis for the proposed uprate is provided in Cameron Engineering Reports ER-592, Rev. 2 (Proprietary), "Bounding Uncertainty Analysis For Thermal Power Determination at Cooper NPPD Using LEFM √+ System," and ER-614, Rev. 1 (Proprietary), "LEFM √+ Meter Factor Calculation and Accuracy Assessment for Cooper NPPD," dated September 2007.

The proposed TS changes include a flow biased neutron flux-high trip setpoint allowable value change in TS Table 3.3.1.1-1. The setpoint calculation methodology is provided in CNS's document, NEDC 98-024, Rev. 5C1, "APRM – RBM Setpoint Calculation" as Enclosure 4 of the licensee's response dated April 4, 2008, to a RAI from the NRC staff.

3.10.2 Regulatory Evaluation

Nuclear power plants are licensed to operate at a specified core thermal power. In this regard, Appendix K to 10 CFR Part 50 requires LOCA and ECCS analyses to assume "that the reactor has been operating continuously at a power level at least 102 percent of the licensed thermal power level to allow for instrumentation uncertainties. Alternately, Appendix K allows assuming lower than the specified 102 percent, but not less than the licensed thermal power level, "provided the proposed alternative value has been demonstrated to account for uncertainties due to power level instrumentation error." The allowance provides licensees an option of justifying a power uprate with reduced margin between the licensed power level and the power

level assumed in the ECCS analysis by using more accurate instrumentation to calculate the reactor thermal power. Because the maximum power level of a nuclear plant is a licensed limit, a proposal to raise the licensed power level must be reviewed and approved under the NRC's license amendment process. The LAR should include a justification for the reduced power measurement uncertainty to support the proposed power uprate.

The Caldon Topical Report ER-80P and the supplement ER-157P describe the LEFM CheckPlus™ System for the measurement of feedwater flow and provide a basis for the proposed 1.62 percent uprate of the licensed reactor power. The NRC staff also considered the guidance of RIS 2002-03 in its review of the licensee's application for the proposed power uprate request.

The LEFM CheckPlus™ System does not perform any safety function and is not used directly to control any plant system. However, adjustment of reactor power nuclear instrumentation (NI) is based on the LEFM CheckPlus™ System calorimetric calculations, which are considered important to safety.

3.10.3 Technical Evaluation

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

The instrumentation for measuring feedwater flow rate typically is a venturi. The device generates a differential pressure proportional to the feedwater velocity in the pipe. Due to the high cost of calibration of the venturi and the need to improve flow instrumentation measurement uncertainty, the industry assessed other flow measurement techniques and found LEFM Check™ and the LEFM CheckPlus™ System ultrasonic flow meters (UFMs) to be a viable alternative.

Both systems use the transit time methodology to measure fluid velocity. The basis of the transit time methodology to measure fluid velocity and temperature is that ultrasonic pulses transmitted into a fluid stream travel faster in the direction of the fluid flow than opposite the flow. The difference in the upstream and downstream traversing times of the ultrasonic pulses is proportional to the fluid velocity in the pipe and the temperature is determined using a pre-established correlation between the mean propagation velocity of the ultrasound pulses in the fluid and the fluid pressure.

Both UFMs use multiple diagonal acoustic paths, instead of a single diagonal path, so that velocities measured along each path can be numerically integrated over the pipe cross section to determine the average fluid velocity in the pipe. The fluid velocity is multiplied by a velocity profile correction factor, the pipe cross section area, and the fluid density to determine the feedwater mass flow rate in the piping. The mean fluid density may be obtained using the measured pressure and the derived mean fluid temperature as an input to a table of thermodynamic properties of water. The velocity profile correction factor is derived from

calibration testing of the LEFM in a plant-specific piping model at a calibration laboratory.

The LEFM Check ™ System, as described in Topical Report ER-80P, consists of a spool piece with eight transducer assemblies forming the four chordal acoustic paths in one plane of the spool piece. The system includes an electronics unit with hardware and software installed to provide flow and temperature measurements and an on-line verification of these measurements. An LEFM CheckPlus™ System, both hydraulically and electronically, is made up of two LEFM Check ™ Systems in a single spool piece. This layout has two sets of four chordal acoustic paths in two planes of the spool piece which are perpendicular to each other. The electronics for the two subsystems, while electrically separated, are housed in a single cabinet. To ensure independence, the two measurement planes of an LEFM CheckPlus™ System have independent clocks for measuring transit times of the ultrasound pulses.

Currently, the instrumentation used for measuring FW flow rate at CNS is a venturi. NPPD intends to install the Caldon LEMF CheckPlus™ ultrasonic FW flow element to reduce the uncertainty in the FW flow measurement. The licensee stated that this reduced uncertainty, in combination with other uncertainties, will result in an overall power level measurement uncertainty of 0.31 percent of reactor thermal power (RTP). The remaining margin of RTP forms the basis for the proposed MUR power uprate of 1.62 percent RTP. The licensee also confirmed that the LEMF CheckPlus™ mass flow uncertainty used in the total thermal power uncertainty determination included the uncertainty for the actual location of the transducers within the housing.

The NRC staff review in the area of Instrumentation & Controls (I&C) covers the proposed plant-specific implementation of the feedwater flow measurement technique and the power increase gained as a result of implementing this technique in accordance with the guidelines (A thru H) provided in Section I of Attachment 1 to RIS 2002-03. The NRC staff review was conducted to confirm that the licensee's implementation of the proposed feedwater flow measurement device was consistent with the staff-approved Caldon Topical Reports ER-80P and ER-157P and adequately addressed the four additional requirements listed in the staff safety evaluation reports (SERs). The NRC staff also reviewed the power measurement uncertainty calculations to ensure that (1) the conservatively proposed uncertainty value of 0.31 percent correctly accounted for all uncertainties due to power level instrumentation errors, and (2) the uncertainty calculations met the relevant requirements of Appendix K to 10 CFR Part 50 as described in the section above. Additionally, the NRC staff reviewed the proposed Limiting Condition for Operation (LCO), Surveillance Requirement (SR), and the Limiting Safety System Setting (LSSS) setpoint changes for compliance with the requirements of 10 CFR 50.36.

<u>Items A through C in Section I of Attachment 1 to RIS 2002-03</u>:

The licensee’s application provided the following information regarding the LEFM CheckPlus™ System FW flow measurement technique and its implementation at CNS.

The FW flow measurement system to be installed at CNS is a Caldon LEFM CheckPlus™ ultrasonic multi-path transit time flow meter as described in Caldon Topical Report ER-157P. The LEFM CheckPlus™ System at CNS will consist of a flow element to be installed in each of the two FW inlet lines just downstream of the mixing pipe in the FW pump room, and an electronics cabinet installed in the Turbine Building basement. The installation of each of the flow elements will conform to the requirements in Caldon Topical Reports ER-80P and ER-157P.

The system will utilize continuous calorimetric power determination by direct serial link with the plant computer, and will incorporate self-verification features. These features ensure that performance is consistent with the design basis.

Based on the review of NPPD's application as reflected in the above information, the NRC staff finds that the licensee sufficiently addressed the plant-specific implementation of the LEFM CheckPlus™ UFM System topical report guidelines, and that the licensee's description of the FW flow measurement technique and the MUR power uprate due to implementing this technique adequately addresses the guidance in Items A through C in Section I of Attachment 1 to RIS 2002-03.

Items D, G and H in Section I of Attachment 1 to RIS 2002-03:

The NRC staff SER on Caldon Topical Report ER-80P included four additional criteria to be addressed by a licensee referencing this topical report to support a MUR power uprate. In its LAR and supplements, NPPD addressed each of the four criteria as follows:

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

NPPD stated that calibration and maintenance of the LEFM CheckPlus™ System will be performed using site procedures developed from the Caldon LEFM CheckPlus™ System technical manuals. Ultrasonic signal verification and alignment is performed automatically with the LEFM CheckPlus™ System. Signal verification is possible by review of signal quality measurements performed and displayed by the LEFM CheckPlus™ System. Routine preventive maintenance procedures include physical inspections, power supply checks, back-up battery replacements, and internal oscillator frequency verification.

NPPD stated that work on the CNS LEFM CheckPlus™ System will be performed by site I&C personnel qualified per the CNS I&C Training Program, and who will have been formally trained on the LEFM CheckPlus™ System by Caldon. Work will be performed in accordance with site work control procedures. The CNS LEFM CheckPlus™ is under Caldon's Verification and Validation (V&V) Program, and procedures are maintained for user notification of important deficiencies.

NPPD stated that procedures governing normal operation, emergency operation, and off-normal operation, as well as equipment changes that may be affected by the power uprate, will be identified in the design change process and revised prior to the implementation of power uprate. Appropriate personnel will receive training on the Caldon LEFM CheckPlus™ System, as well as on the affected procedures.

NPPD stated that if the LEFM CheckPlus™ System becomes inoperable, control room operators are promptly alerted by control room computer indications. Feedwater flow input to the core thermal power calculation would then be provided by the existing venturi-based flow measurements. The venturi is continuously calibrated to the last validated good data from the LEFM CheckPlus™ System. When the LEFM is found inoperable, plant operation at 2419 MWt is allowed for up to 72 hours, provided no downward power change in excess of 10 percent

occurs during the 72 hours. If the instrumentation cannot be repaired within 72 hours, then power must be reduced to and maintained at no higher than 2381 MWt (the current Licensed Thermal Power) until the instrumentation is repaired.

In response to the NRC staff RAI, NPPD further indicated that the venturi transmitter drift data showed the transmitter drift on the power calorimetric during the 72 hour AOT is ±0.0177 percent. This uncertainty is considered to be insignificant for a 72 hour period.

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

NPPD stated that this criterion is not applicable to CNS, because CNS currently uses a venturi-based FW flow measurement system to obtain the daily calorimetric heat balance measurements.

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

NPPD stated that CNS uses a core thermal power uncertainty calculation approach consistent with ASME PTC-19.1 (1985), "Measurement Uncertainty;" ISA 67.04.02-2000, "Methodologies for the Determination of Set Points for Nuclear Safety-Related Instrumentation;" and Caldon's Topical Report ER-80P, as supplemented by ER-157P. The combination of errors within instrument loops is accomplished in accordance with the NRC-approved GE Setpoint Methodology as described in NEDC-31336P, "General Electric Instrument Setpoint Methodology," dated September 1996.

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

NPPD stated that criterion 4 does not apply to CNS. The calibration factor for the CNS spool pieces was established by tests of these spools at Alden Research Laboratory in August of 2007. A full-scale model of the CNS hydraulic geometry was used for these tests.

The calibration factor used for the LEFM CheckPlus™ System at CNS is based on Cameron Engineering Report ER-614 (Enclosure 5 to the licensee's application). The NRC staff reviewed the test configuration and the meter factor uncertainty calculation provided by the licensee in the report and found the results for the calibration factor acceptable.

NPPD also stated that final acceptance of the site-specific uncertainty analyses will occur after the completion of the commissioning process which is expected to be completed in May of 2008. The commissioning process will verify bounding calibration test data by Caldon, Inc. This step provides final positive confirmation that actual performance in the field meets the uncertainty bounds established for the instrumentation.

Based on the above listed responses provided by NPPD to the four criteria, the NRC staff finds that NPPD has fully addressed the four criteria specified in the staff's SER of topical reports ER-80P. Therefore, NPPD has adequately addressed the guidance in Items D, G, and H in Section I of Attachment 1 to RIS 2002-03.

Item E in Section I of Attachment 1 to RIS 2002-03:

To address Item E, NPPD provided a summary of the CNS core thermal power measurement uncertainty in a table format listing uncertainty values from the Cameron Engineering Report ER-592, Rev. 2, which provided a detailed calculation of the uncertainties. NPPD also provided the design calculations sheet for the reactor thermal power uncertainty in NEDC 06-035, Rev. 0, "Reactor Core Thermal Power Uncertainty Calculation." The licensee stated that the values in the uncertainty column of the table and the total power uncertainty determination are bounding values.

By auditing ER-592, the NRC staff found that the calculations determined individual measurement uncertainties of all parameters contributing to the core thermal power measurement uncertainty, and those uncertainties were then combined using square root of sum of squares methodology, which conforms to Regulatory Guide (RG) 1.105 and Instrument Society of America (ISA)-67.04.01-2000.

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

Item F in Section I of Attachment 1 to RIS 2002-03:

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

i) Maintaining Calibration

NPPD stated in its application that calibration and maintenance of the LEFM CheckPlus™ System will be performed using site procedures developed from the Caldon LEFM CheckPlus™ System technical manuals. Ultrasonic signal verification and alignment is performed automatically with the LEFM CheckPlus™ System. Signal verification is possible by review of

signal quality measurements performed and displayed by the LEFM CheckPlus™ System. Routine preventive maintenance procedures include physical inspections, power supply checks, back-up battery replacements, and internal oscillator frequency verification. Work will be planned and executed in accordance with established CNS work control procedures.

ii) Controlling Hardware and Software Configuration

NPPD stated that the LEFM CheckPlus™ System is designed and manufactured in accordance with Caldon's 10 CFR Part 50, Appendix B, Quality Assurance Program and its V&V program. Caldon's V&V program fulfills the requirements of ANSI/IEEE-ANS Std. 7-4.3.2, 1993, "IEEE Standard Criteria for Digital Computers in Safety Systems of Nuclear Power Generating Stations," Annex E, and ASME NQA-2a- 1990, "Quality Assurance Requirements for Nuclear Facility Applications." In addition, the program is consistent with guidance for software V&V in EPRI TR-103291s, "Handbook for Verification and Validation of Digital Systems," December 1994. The CNS LEFM CheckPlus™ is under Caldon's V&V Program and work on CNS LEFM CheckPlus™ System will be performed by site I&C personnel qualified per the CNS I&C Training Program and formally trained on the LEFM CheckPlus™ System by Caldon. Work will be planned and executed in accordance with established CNS work control procedures. The software falls under the CNS Software Quality Assurance Program.

iii) Performing Corrective Actions

The licensee stated that the LEFM CheckPlus™ System will be included in the preventive maintenance program and the CNS Quality Assurance Program. Conditions that are adverse to quality are documented under the CNS Corrective Action Program.

iv) Reporting Deficiencies to the Manufacturer

The licensee stated that the CNS LEFM CheckPlus™ System is under Caldon's V&V Program, and procedures are maintained for user notification of important deficiencies.

v) Receiving and Addressing Manufacturer Deficiency Reports

The licensee stated that vendor notifications are controlled in the CNS Operating Experience Program. Those vendor notifications considered applicable are entered into the Corrective Action Program for disposition. The equipment manuals are also included in the CNS vendor manual program.

Based on the information provided by NPPD, the NRC staff finds that NPPD addressed the calibration and maintenance aspects of the LEFM CheckPlus™ UFM system and all other instruments affecting the power calorimetric, and thus complied with the guidance in item F in Section I of Attachment 1 to RIS 2002-03.

Limiting Trip Setpoint (LSP) Calculations

The licensee stated that the only setpoint change involving a LSSS is as follows:

TS Table 3.3.1.1-1, Average Power Range Monitors ALLOWABLE VALUE of FUNCTION 2.b, Neutron Flux-High (Flow Biased), page 3.3-16, referenced by

LCO 3.4.1c (Recirculation Loops Operating), is revised from "≤ 0.66 W + 71.5% RTP[(b)]" to "≤ 0.75 W + 62.0% RTP[(b)]." Footnote (b) is revised from "≤ 0.66 W + 71.5% - 0.66 ΔW" to "≤ 0.75 W + 62.0% - 0.75 ΔW." Where W is the two loop recirculation flow rate in percent of rated flow providing 100% core flow at 100% power.

FUNCTION 2.b, "Neutron Flux-High (Flow Biased)," monitors neutron flux to approximate the thermal power being transferred to the reactor coolant. This trip setpoint varies as a function of recirculation flow. The APRM Neutron Flux-High (Flow Biased) Function is not specifically credited in the safety analyses, but is intended to provide protection against a transient where thermal power increases slowly, and to provide protection against power oscillations. The function is required to be operable in Mode 1 when there is the possibility of generating excessive thermal power.

In its letter to NRC dated April 4, 2008, NPPD provided its setpoint calculation for the Average Power Range Monitor (APRM) Neutron Flux-High (Flow Biased) in CNS document NEDC 98-024, Rev. 5C1, "APRM – RBM Setpoint Calculation," dated November 13, 2007. This document calculated the Neutron Flux-High instrumentation limiting trip setpoint (LSP) based on the total loop uncertainty per GE Instrument Setpoint Methodology, with CNS plant-specific calculations and plant-specific data.

The licensee also calculated as-found and as-left setpoint tolerances to establish the allowable values (AVs) and operating setpoint (OSP). The GE Instrument Setpoint Methodology, which was approved by the NRC by letter dated February 9, 1993, conforms to the guidelines of RG 1.105, Rev. 2, and Instrument Society of America (ISA) ISA-S67.04-1982.

The licensee stated that the calculation of LSP uses Analytical Limits defined by the inputs to the deign basis analyses, and the AVs from CNS TSs, with consideration of appropriate uncertainties. Operability of instrument channels is established by meeting the SRs and AVs specified in the CNS TSs. In RIS 2006-17, "NRC Staff Position on the Requirements of 10 CFR 50.36, "Technical Specifications," Regarding Limiting Safety System Settings During Periodic Testing and Calibration of Instrument Channels," the NRC staff presented an acceptable method of identifying LSSS values in the TSs that would meet the requirements of 10 CFR 50.36. 10 CFR 50.36(d)(1)(ii)(A) requires that LSSS values be in the TSs. The LSSS values are the settings for "automatic protective devices related to those variables having significant safety functions." Also, per 10 CFR 50.36, the instrumentation associated with these LSSS values is required to be operable (through limiting conditions for operation, LCOs) and have surveillances to demonstrate the instrumentation performs its safety function within the LSSS value (through surveillance requirements, SRs). The licensee proposed to add two footnotes to TS Table 3.3.1.1-1 for the APRM Neutron Flux-High (Flow Biased) trip setpoint verification surveillance. The addition of the two notes is in accordance with RIS 2006-17 and meets the NRC regulatory requirements of 10 CFR 50.36 by (1) demonstrating that the actual setting, with the specified actions in the notes, is the equivalent of an LSSS, and (2) specifying as requirements in the notes the actions needed to use the AV as a reference for determining the operability of the instrumentation in its safety function. The two footnotes are to be inserted at the bottom of TS page 3.3-6 as follows:

(c) If the as-found setpoint is outside its predefined as-found tolerance, then the

channel shall be evaluated to verify that it is functioning as required before returning the channel to service.

(d) The instrument channel setpoint shall be reset to a value that is within the as-left tolerance around the Nominal Trip Setpoint (NTSP) at the completion of the surveillance; otherwise the channel shall be declared inoperable. Setpoints more conservative than the NTSP are acceptable provided that the as-found and as-left tolerances apply to the actual setpoint implemented in the Surveillance procedures to confirm the channel performance. The NTSP and the methodologies used to determine the as-found and as-left tolerances are specified in station procedures implementing the GE Setpoint Methodology per NEDC-31336P-A approved in TS Amendment 178 SER, Section III.G.2.

The NRC staff review concludes that the calculated setpoint tolerances have sufficient margin to the AVs and, therefore, are acceptable.

3.10.4 Conclusion for Section 3.10

The NRC staff reviewed the licensee's proposed plant-specific implementation of the FW flow measurement device and the power uncertainty calculations and determined that the licensee's proposed changes are consistent with the NRC-approved Topical Report ER-80P, and its supplement ER-157P. The NRC staff also determined that the licensee adequately accounted for all instrumentation uncertainties in the reactor thermal power measurement uncertainty calculations and demonstrated that the calculations meet the relevant requirements of 10 CFR Part 50, Appendix K. Therefore, the NRC staff finds the proposed 1.62 percent thermal power uprate acceptable.

3.11 Plant Systems

3.11.1 Regulatory Evaluation

The NRC staff's review in the area of plant systems covers the impact of the proposed MUR power uprate on nuclear steam supply system (NSSS) interface systems, containment systems, safety-related cooling water systems, spent fuel pool (SFP) storage and cooling, radioactive waste systems, and engineered safety feature (ESF) heating, ventilation, and air conditioning (HVAC) systems. The staff's review is based on the guidance in SRP Chapters 3, 6, 9, 10, and 11, and RIS 2002-03, Attachment 1, Sections II, III, and VI. The licensee evaluated the effect of the MUR on the plant systems. This evaluation is reflected in Enclosure 1 of the licensee's application dated November 19, 2007.

3.11.2 Technical Evaluation

NSSS Interface Systems

NSSS interface systems include the reactor recirculation system (RRS), main steam line flow restrictors, main steam isolation valves (MSIVs), reactor core isolation cooling system (RCIC), residual heat removal system (RHR), and reactor water cleanup system (RWCU). The staff's review of RWCU is documented in Section 3.4 of this SE, and the reviews of RRS, RCIC, and RHR are documented in Section 3.6. In addition, the NRC staff's review of the structural and

pressure boundary integrity of NSSS systems and components is documented in Section 3.5 of this SE.

The licensee stated that requirements for the main steam line flow restrictors remain unchanged for uprate conditions. Since the operating pressure remains the same, there is not change in steam line break flow rate. The generic evaluation provided in General Electric Licensing Topical Report NEDC-32938P-A, “Generic Guidelines and Evaluations for General Electric Boiling Water Reactor Thermal Power Optimization (TLTR),” is applicable to CNS. Therefore, the safety and operational aspects of the main steam line flow restrictors remain within the current licensed thermal power analysis.

The licensee stated that the requirements for the MSIVs remain unchanged for uprate conditions. The generic evaluation provided in the TLTR is applicable; therefore, the safety and operational aspects of the MSIVs remain within the current licensed thermal power (CLTP) analysis.

The staff reviewed the licensee’s evaluation and concurs with the results. The licensee determined that there is no adverse impact on the NSSS interface systems from the MUR power uprate. The staff does not anticipate that an MUR power uprate will challenge the NSSS interface systems, and all systems have been shown to be operating within design limits. Therefore, the staff finds that the NSSS systems are acceptable for the MUR uprate.

Containment Systems

The NRC staff's evaluation of the impact of the proposed uprate on the plant-specific-provisions of GLs 89-10, 95-07, and 96-06 is documented in Section 3.5 of this SE. Since the current analysis for the containment systems is based on 102 percent of CLTP, the analysis bounds the proposed uprate conditions. Therefore, the NRC staff finds the containment systems acceptable for the MUR uprate.

Safety-Related Cooling Water Systems

The ultimate heat sink (UHS) for CNS is the Missouri River. The safety-related Service Water (SW) system provides water from the UHS for equipment cooling throughout the plant. The SW system provides cooling water during and following a design-basis-accident. The safety-related performance of the SW system during and following a LOCA, the most demanding design basis event for the SW system, does not change because the current LOCA analysis was based on 102 percent of CLTP. Therefore, the UHS and the SW system are acceptable for the MUR uprate.

SFP Storage and Cooling

The principal function of the SFP storage and cooling systems is to provide storage and cooling of spent fuel. The primary impact of a power uprate would be to the decay heat of the fuel recently discharged from the core. The licensee concluded that the SFP cooling system is not impacted by the MUR power uprate. The SFP cooling adequacy is determined by calculating the heat load generated by a full core discharge plus remaining SFP spaces filled with used fuel discharged at regular intervals. The analysis assumes the current 18-month fuel cycle length as the basis. The existing analyses and the licensee's continued compliance with pool design limits

by controlling the rate of fuel offload to the SFP confirm the ability of the SFP cooling system to maintain adequate SFP cooling for uprate conditions. Based on the licensee's offload-specific evaluation, the NRC staff finds that the SFP storage and cooling will not be impacted by the proposed uprate.

Radioactive Waste (Radwaste) Systems

The Standby Gas Treatment system (SGTS) minimizes the offsite and control room dose rates during venting and purging of the containment atmosphere under abnormal circumstances. The capability of the SGTS is not changed by the uprate conditions. The licensee stated that the SGTS can accommodate design-basis-accident conditions at 102 percent of CLTP. Therefore, the SGTS remains capable of performing its safety function for the uprate conditions.

The liquid radwaste system collects, monitors, processes, stores, and returns processed radwaste to the plant for reuse, discharge, or shipment. The activated corrosion products in the radwaste stream are expected to increase proportionally to the uprate. However, the licensee states that the total volume of processed waste is not expected to increase appreciably because the only significant increase in processed waste is due to more frequent backwashes of the condensate demineralizers and RWCU filter demineralizers. Therefore, the radiological limits of 10 CFR 20 and 10 CFR 50, Appendix I, continue to be met, and the uprate does not adversely effect the processing of liquid radwaste.

The gaseous waste systems, including the offgas system and the various building ventilation systems, collect, control, process, and dispose of gaseous radwaste. The activity of airborne effluents does not increase significantly due to the uprate, and the release limit is administratively controlled and is not a function of core power. The expected flow through the offgas system will increase slightly due to the uprate, but it remains well within the capacity of the system. The offgas system radiological release rate is administratively controlled, and is not a function of core power. Therefore, the uprate does not affect the offgas system design or operation.

The NRC staff reviewed the licensee's assessment. The staff does not expect the MUR uprate to result in a significant change to the operation of the radwaste systems; therefore, based on the licensee's assessment, the staff finds that the radwaste systems will function adequately for the proposed change.

ESF HVAC Systems

The main control room atmosphere control system maintains control room habitability following a postulated accident. The main control room atmosphere is unaffected by the uprate. The main control room atmosphere control system was previously evaluated at 102 percent of CLTP, which bounds the uprate. Therefore, the system remains capable of performing its safety function at uprate conditions.

The Combustible Gas Control System (CGCS) maintains the post-LOCA concentration of oxygen or hydrogen in the containment atmosphere below the flammable limit. The system operation was previously analyzed at 102 percent of CLTP; therefore, the analysis bounds the uprate conditions.

The power-dependent HVAC systems that are potentially affected by the uprate consist mainly of heating and cooling supply, exhaust, and recirculation units in the turbine building, reactor building (including steam tunnel and drywell), and control building. The licensee stated that the uprate results in a minor increase in the heat load in the turbine building caused by the slightly higher feedwater temperature (about 2°F). In the reactor building, there is an increase in heat load due to a slight SFP cooling process temperature increase. The slight increases in heat load are well within the margin of the area coolers, and are therefore acceptable.

Power Conversion Systems

Power conversion systems include the turbine generator, turbine steam bypass system, and the main feedwater (FW) and condensate systems. These systems are not safety-related, but the operation of these systems can affect safety-related systems.

The turbine generator was evaluated for the potential to generate missiles with the potential to affect safety-related components due to turbine overspeed. The licensee performed calculations to determine the MUR power uprate turbine steam path conditions, and found that these operating conditions were bounded by the previous analysis of the turbine and generator stationary and rotating components. Thus, the increased loadings, pressure drops, thrusts, stresses, overspeed capability and other design considerations resulting from operation at MUR power uprate conditions are within existing design limits. The existing rotor missile analysis was performed at design overspeed conditions, 120 percent of rated speed. The low-pressure turbine casing is designed to prevent rupture due to disc failure at design overspeed conditions. Therefore, the MUR power uprate does not change the potential for turbine missile generation.

The turbine steam bypass system was originally designed for a steam flow capacity of approximately 33 percent of the rated steam flow at the current licensed thermal power. While the bypass capacity as a percent of rated steam flow is reduced to 32 percent of rated steam flow at MUR power uprate conditions, the actual steam bypass capacity is unchanged. The transient analyses that credit the turbine bypass system use a bypass capacity that is less than the actual capacity. Therefore, the turbine bypass capacity remains adequate because the actual capacity continues to bound the value used in the analyses.

The FW and condensate systems are designed to provide FW at the temperature, pressure, quality, and flow rate required by the reactor. These systems are not safety-related; however, their performance may have an effect on plant availability and the capability to operate reliably at the MUR power uprate condition. The licensee reviewed the FW heaters, heater drains, condensate demineralizers, and the pumps (FW and condensate) and determined that the components are capable of performing in the proper design range to provide the slightly higher FW flow rate at the necessary temperature and pressure.

The staff reviewed the licensee's evaluation of the power conversion systems and concurs with the results. The licensee determined that there is no adverse impact on the power conversion systems from the MUR power uprate. The staff does not anticipate that an MUR power uprate will challenge the power conversion systems, and the systems most affected by the MUR power uprate have been shown to be operating within design limits. Therefore, the staff finds that the power conversion systems are acceptable for the MUR uprate.

3.11.3 Conclusion

The NRC staff has reviewed NPPD's analyses of the impact of the proposed MUR power uprate on NSSS interface systems, containment systems, safety-related cooling water systems, SFP storage and cooling, radioactive waste systems, ESF HVAC systems, and power conversion systems. The NRC staff has determined that the results of NPPD's analyses related to these areas will continue to meet the applicable acceptance criteria following implementation of the MUR power uprate. Therefore, the NRC staff finds the proposed MUR power uprate to be acceptable with respect to the plant systems review.

4.0 STATE CONSULTATION

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

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding published January 29, 2008 (73 FR 5224). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

6.0 CONCLUSION

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

7.0 LICENSEE COMMITMENTS

The following table identifies regulatory commitments made by the licensee in the application dated November 19, 2007.

COMMITMENT	COMMITMENT NUMBER	COMMITTED DATE OR OUTAGE
Final acceptance of the CNS uncertainty analysis and verification of bounding calibration test data will occur.	NLS2007069-01	During power ascension and commissioning process following RE24

Procedure changes governing normal operation, emergency operation, and off-normal operation, as well as equipment changes that may be affected by power uprate, will be made.	NLS2007069-02	Prior to implementation of uprated power
Technical Requirements Manual will be revised to include CheckPlus System out-of-service administrative controls.	NLS2007069-03	Prior to implementation of uprated power
Core power from Average Power Range Monitors (APRMs) will be rescaled to the uprated power level and any necessary adjustments of APRM alarm and trip settings will be made.	NLS2007069-04	Prior to exceeding Current Licensed Thermal Power (CLTP) level
Demonstration of an acceptable fuel thermal margin will be performed at each of the following steady-state heat balance power levels: 95% and 100% of CLTP, and 100% of uprated power level.	NLS2007069-05	Prior to and during power ascension to 100% of uprated power level
Routine measurements of reactor and system pressures, flows, and selected major rotating equipment vibration will be taken near 95% and 100% of CLTP, and at 100% of uprated power level.	NLS2007069-06	Prior to and during power ascension to 100% of uprated power level
Operational aspect of the uprate will be demonstrated by performing turbine pressure regulator controller and feedwater controller testing, and reactor pressure control system testing. During this testing, a water level change of ± 3 inches, and pressure setpoint changes of ± 3 psi will be used. If necessary, controllers and actuator elements will be adjusted. • Performance of feedwater level control system will be recorded at 95% and 100% of CLTP, and confirmed at uprated power level. • Turbine pressure controller setpoint will be readjusted at 95% and 100% CLTP level and held constant prior to recording baseline power ascension data.	NLS2007069-07	Prior to and during power ascension to 100% of uprated power level
Ensure compliance with the methodology contained in Reg. Guide 1.20 for vibration assessment.	NLS2007069-08	Prior to exceeding CLTP and ascension to uprated power level

Appropriate personnel will receive training on Caldon LEFM CheckPlus System, and on affected procedures.	NLS2007069-09	Prior to operation at uprated power.
Simulator changes and validation for power uprate will be completed in accordance with ANSI/ANS 3.5-1985.	NLS2007069-10	Prior to implementation of the requested license amendment
A Startup Test Report will be submitted.	NLS2007069-11	Within 90 days following resumption of power operation following RE24
A process will be implemented to use the LEFM CheckPlus System feedwater mass flow and temperature to adjust or calibrate the existing feedwater flow nozzle-based signals.	NLS2007069-12	Following power ascension to 100% of uprated power level.

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