UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

July 22, 2009

Mr. James A. Spina, Vice President
Calvert Cliffs Nuclear Power Plant, Inc.
Calvert Cliffs Nuclear Power Plant
1650 Calvert Cliffs Parkway
Lusby, MD 20657-4702

SUBJECT: CALVERT CLIFFS NUCLEAR POWER PLANT, UNIT NOS. 1 AND 2 - AMENDMENT RE: MEASUREMENT UNCERTAINTY RECAPTURE POWER UPRATE (TAC NOS. MD9554 AND MD9555)

Dear Mr. Spina:

The Commission has issued the enclosed Amendment No. 291 to Renewed Facility Operating License No. DPR-53 and Amendment No. 267 to Renewed Facility Operating License No. DPR-69 for the Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2. These amendments consist of changes to the Technical Specifications (TSs) in response to your application transmitted by letter dated August 29, 2008, as supplemented by letters dated December 3, two letters dated December 29, December 30, 2008, February 17, February 18, March 10, May 7, and June 11, 2009.

These amendments revise the license and TSs to reflect an increase in the rated thermal power from 2700 megawatts thermal (MWt) to 2737 MWt (1.38 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 the related Safety Evaluation is enclosed. A Notice of Issuance will be included in the Commission's next regular biweekly *Federal Register* notice.

Sincerely,

Douglas V. Pickett

Douglas V. Pickett, Senior Project Manager
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-317 and 50-318

Enclosures:
1. Amendment No. 291 to DPR-53
2. Amendment No. 267 to DPR-69
3. Safety Evaluation

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

CALVERT CLIFFS NUCLEAR POWER PLANT, INC.

DOCKET NO. 50-317

CALVERT CLIFFS NUCLEAR POWER PLANT, UNIT NO. 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 291
Renewed License No. DPR-53

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

   A. The application for amendment by Calvert Cliffs Nuclear Power Plant, Inc. (the licensee) dated August 29, 2008, as supplemented by letters dated December 3, two letters dated December 29, December 30, 2008, February 17, February 18, March 10, May 7, and June 11, 2009, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.2. of Renewed Facility Operating License No. DPR-53 is hereby amended to read as follows:

2. Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 291, are hereby incorporated into the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance and shall be implemented within 180 days following completion of the 2010 refueling outage.

FOR THE NUCLEAR REGULATORY COMMISSION

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

Attachment:
Changes to the License and Technical Specifications

Date of Issuance: July 22, 2009

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

CALVERT CLIFFS NUCLEAR POWER PLANT, INC.

DOCKET NO. 50-318

CALVERT CLIFFS NUCLEAR POWER PLANT, UNIT NO. 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 267
Renewed License No. DPR-69

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

   A. The application for amendment by Calvert Cliffs Nuclear Power Plant, Inc. (the licensee) dated August 29, 2008, as supplemented by letters dated December 3, two letters dated December 29, December 30, 2008, February 17, February 18, March 10, May 7, and June 11, 2009, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.2. of Renewed Facility Operating License No. DPR-69 is hereby amended to read as follows:

2. Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 267, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance and shall be implemented within 180 days.

FOR THE NUCLEAR REGULATORY COMMISSION

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

Attachment:
Changes to the License and Technical Specifications

Date of Issuance: July 22, 2009

ATTACHMENT TO LICENSE AMENDMENTS

AMENDMENT NO. 291 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-53

AMENDMENT NO. 267 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-69

DOCKET NOS. 50-317 AND 50-318

Replace the following page of the Facility Operating License with the attached revised page. The revised page is identified by amendment number and contains marginal lines indicating the areas of change.

Remove Page	Insert Page
3	3

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

Remove Page	Insert Page
1.1-5	1.1-5

rules, regulations, and orders of the Commission, now or hereafter applicable; and is subject to the additional conditions specified and 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 2737 megawatts-thermal in accordance with the conditions specified herein.

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 291, are hereby incorporated into this license. The licensee shall operate the facility in accordance with the Technical Specifications.

(a) For Surveillance Requirements (SRs) that are new, in Amendment 227 to Facility Operating License No. DPR-53, the first performance is due at the end of the first surveillance interval that begins at implementation of Amendment 227. For SRs that existed prior to Amendment 227, including SRs with modified acceptance criteria and SRs whose frequency of performance is being extended, the first performance is due at the end of the first surveillance interval that begins on the date the Surveillance was last performed prior to implementation of Amendment 227.

(3) Additional Conditions

The Additional Conditions contained in Appendix C as revised through Amendment No. 267 are hereby incorporated into this license. Calvert Cliffs Nuclear Power Plant, Inc. shall operate the facility in accordance with the Additional Conditions.

(4) Secondary Water Chemistry Monitoring Program

The Calvert Cliffs Nuclear Power Plant, Inc., shall implement a secondary water chemistry monitoring program to inhibit steam generator tube degradation. This program shall include:

a. Identification of a sampling schedule for the critical parameters and control points for these parameters;

b. Identification of the procedures used to quantify parameters that are critical to control points;

C. This license is deemed to contain and is subject to the conditions set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act, and the rules, regulations, and orders of the Commission, now and hereafter applicable; and is subject to the additional conditions specified and incorporated below:

(1) Maximum Power Level

The licensee is authorized to operate the facility at reactor steady-state core power levels not in excess of 2737 megawatts-thermal in accordance with the conditions specified herein.

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 267 are hereby incorporated into this license. The licensee shall operate the facility in accordnace with the Technical Specifications.

(a) For Surveillance Requirements (SRs) that are new, in Amendment 201 to Facility Operating License No. DPR-69, the first performance is due at the end of the first surveillance interval that begins at implementation of Amendment 201. For SRs that existed prior to Amendment 201, including SRs with modified acceptance criteria and SRs whose frequency of performance is being extended, the first performance is due at the end of the first surveillance interval that begins on the date the Surveillance was last performed prior to implementation of Amendment 201.

(3) Less Than Four Pump Operation

The licensee shall not operate the reactor at power levels in excess of five (5) percent of rated thermal power with less than four (4) reactor coolant pumps in operation. This condition shall remain in effect until the licensee has submitted safety analyses for less than four pump operation, and approval for such operation has been granted by the Commission by amendment of this license.

(4) Environmental Monitoring Program

If harmful effects or evidence of irreversible damage are detected by the biological monitoring program, hydrological monitoring program, and the radiological monitoring program specified in the Appendix B Technical Specifications, the licensee will provide to the staff a detailed analysis of the problem and a program of remedial action to be taken to eliminate or significantly reduce the detrimental effects or damage.

1.1 Definitions

OPERABLE-OPERABILITY	A system, subsystem, train, component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified safety function(s) and when all necessary attendant instrumentation, controls, normal or emergency electrical power, cooling and seal water, lubrication, and other auxiliary equipment that are required for the system, subsystem, train, component, or device to perform its specified safety function(s) are also capable of performing their related support function(s).
PHYSICS TESTS	PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation. These tests are: a. Described in Chapter 13, Initial Tests and Operation of the Updated Final Safety Analysis Report; b. Authorized under the provisions of 10 CFR 50.59; or c. Otherwise approved by the Nuclear Regulatory Commission.
RATED THERMAL POWER (RTP)	RTP shall be a total reactor core heat transfer rate to the reactor coolant of 2737 MWt.
REACTOR PROTECTIVE SYSTEM (RPS) RESPONSE TIME	The RPS RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its RPS trip setpoint at the channel sensor until electrical power to the CEAs drive mechanism is interrupted. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and methodology for

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SAFETY EVALUATION BY THE OFFICE NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 291 TO RENEWED

FACILITY OPERATING LICENSE NO. DPR-53

AND AMENDMENT NO. 267 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-69

CALVERT CLIFFS NUCLEAR POWER PLANT, INC.

CALVERT CLIFFS NUCLEAR POWER PLANT, UNIT NOS. 1 AND 2

DOCKET NOS. 50-317 AND 50-318

1.0 INTRODUCTION

By letter dated August 29, 2008 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML082470623), as supplemented by letters dated December 3 (ADAMS Accession No. ML083430009), two letters dated December 29, (ADAMS Accession Nos. ML083650053 and ML090020382), December 30, 2008, (ADAMS Accession No. ML091240106), February 17 (ADAMS Accession No. ML090500398), February 18 (ADAMS Accession No. ML090630750), March 10 (ADAMS Accession No. ML090700310), May 7 (ADAMS Accession No. ML091310169), and June 11, 2009 (ADAMS Accession No. ML091660293), Calvert Cliffs Nuclear Power Plant, Inc. (the licensee) submitted a request for changes to the Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2 (CCNPP) Technical Specifications (TSs).

The supplements dated December 3, two letters dated December 29, December 30, 2008, February 17, February 18, March 10, May 7, and June 11, 2009, 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 November 4, 2008 (73 FR 65688).

The proposed changes revise the license and TSs to reflect an increase in the rated thermal power from 2700 to 2737 megawatts thermal (MWt) (1.38 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 Nuclear Regulatory Commission (NRC) Regulatory Issue Summary (RIS) 2002-03, "Guidance on the Content of Measurement Uncertainty Recapture Power Uprate Applications."

Specifically, the licensee proposes the following changes:

- Paragraphs 2.C.(1) in Facility Operating License Nos. DPR-53 and DPR-69 (page 3) are revised to authorize operation at a steady state reactor core thermal power level not in excess of 2737 MWt.
- The definition of RATED THERMAL POWER (RTP) in TS 1.1, page 1.1-5, is revised to reflect an increase from 2700 MWt to 2737 MWt.

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. This calculation is called "secondary calorimetric" for a pressurized-water reactor (PWR). 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.

Feedwater flow rate is typically measured using a venturi. This device generates a differential pressure proportional to the square of the feedwater velocity in the pipe. Because of the high cost of calibrating the venturi and the need to improve flow instrumentation measurement uncertainty, the industry evaluated other flow measurement techniques and found the Caldon LEFM Check and LEFM CheckPlus ultrasonic flow meters to be a viable alternative.

This power uprate is based on a reduced measurement uncertainty of core thermal power resulting from the installation of a Cameron (formerly Caldon) LEFM CheckPlus system to measure feedwater flow and temperature at CCNPP. The licensee's submittal referenced Cameron Topical Report ER-80P, Revision 0, "Improving Thermal Power Accuracy and Plant Safety While Increasing Operating Power Level Using the System," issued in March 1997 (ADAMS Accession No. 9807210146), and its supplement, Topical Report ER-157P, Revision 5, "Supplement to Topical Report ER-80P: Basis for a Power Uprate with the LEFM √™ or LEFM CheckPlus™ System," issued in October 2001 (ADAMS Accession No. ML013440078).

Topical Report ER-80P describes the LEFM technology, includes calculations of power measurement uncertainty using a Caldon LEFM Check system in a typical two-loop PWR or two-feedwater-line boiling-water reactor (BWR), and provides guidelines and equations for determining the plant-specific power calorimetric uncertainties. ER-80P was approved by the NRC staff on March 8, 1999 (ADAMS Accession No. 9903190053) for use in justification of MUR power uprates up to 1 percent. Its supplement, Topical Report ER-157P, describes the Caldon LEFM CheckPlus system and lists nonproprietary results of a typical PWR or BWR thermal power measurement uncertainty calculation using either the Caldon LEFM Check or LEFM CheckPlus system. ER-157P was approved by the NRC staff on December 20, 2001 (ADAMS Accession No. ML013540256), for use in justifying MUR power uprates up to 1.7 percent. Together, these two reports provide a generic basis and guidelines for power uprate.

Cameron Engineering Report ER-507, Revision 2, "Bounding Uncertainty Analysis for Thermal Power Determination at Calvert Cliffs Using the LEFM CheckPlus System," issued in January 2009, and CCNPP document CA06945, Revision 1, "Calorimetric Uncertainty Using the Caldon LEFM CheckPlus Flow Measurement System," provide the plant-specific basis for the proposed

uprate at CCNPP. These reports were included in the licensee's supplemental letter dated February 18, 2009.

The NRC has recently issued similar MUR power uprate license amendments for Crystal River, Unit 3 on December 26, 2007 (ADAMS Accession No. ML073600419), Vogtle Electric Generating Plant, Units 1 & 2 on February 27, 2008 (ADAMS Accession No. ML080350347), Cooper Nuclear Station on June 30, 2008 (ADAMS Accession No. ML081540280) and for Davis Besse Nuclear Power Station, Unit 1 on June 30, 2008 (ADAMS Accession No. ML081410652).

2.0 BACKGROUND

Nuclear power plants are licensed to operate at a specified maximum core thermal power, often called rated thermal power (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 assume 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. CCNPP is currently licensed to operate at a maximum power level of 2700 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 Part 50, Appendix K. Currently, with the RTP of 2700 MWt, an analytical power level of 2754 MWt (102 percent of 2700 MWt) is used in the safety analysis. With a requested revised RTP of 2737 MWt and a revised uncertainty, the analytical power level is unchanged at 2754 MWt.

The desired MUR power uprate will be accomplished by increasing the electrical demand on the turbine-generator. As a result of this demand increase, steam flow will increase and the resultant steam pressure will decrease. The reactor coolant system (RCS) nominal cold leg temperature will remain constant while the hot let temperature will increase slightly in response to the increased steam flow demand. As a result, the RCS average temperature will increase slightly.

The NRC staff finds that the LEFM-assisted core thermal power measurement uncertainty is limited to 0.62 percent of actual reactor thermal power and, therefore, can support the proposed

1.38 percent power uprate. This results in the proposed increase of 1.38 percent in the CCNPP license power level using current NRC approved methodologies.

3.0 TECHNICAL EVALUATION

3.1 Human Factors

3.1.1 Regulatory Evaluation

The area of human factors deals with programs, procedures, training, and plant design features related to human performance during normal and accident conditions. The NRC staff's human factors evaluation was conducted to ensure that human performance would not be adversely affected as a result of system and procedure changes made to implement the proposed MUR power uprate. The staff's review covered the licensee's evaluation of changes to operator actions, human-system interfaces, and procedures and training needed for the proposed MUR power uprate. The staff's review criteria are contained in NUREG-0800 (Rev. 1), Standard Review Plan (SRP), Sections 13.2.1, "Reactor Operator Requalification Program; Reactor Operator Training," 13.2.2, "Non-Licensed Plant Staff Training," 13.5.2.1, "Operating and Emergency Operating Procedures," and Chapter 18.0, "Human Factors Engineering."

3.1.2 Technical Evaluation

The NRC staff has developed a standard set of human factors questions for review of proposed MUR power uprate license amendment requests (LARs) (RIS 2002-03, Attachment 1, Section VII, items 1 through 4). The following sections evaluate the licensee's response to these questions in the LAR.

3.1.2.1 Operator Actions

The licensee stated in its submittal that no new operator actions or changes to existing operator actions will be required for the emergency operating procedures (EOPs) or abnormal operating procedures (AOPs) as a result of the proposed MUR power uprate. Existing operator actions credited in the CCNPP Updated Final Safety Analysis Report (UFSAR) were also reviewed for potential changes due to the proposed MUR. The existing operator actions and the corresponding response times credited in the UFSAR Chapter 14 events were found to be unaffected by the proposed increase in power level. More specifically, the analysis of UFSAR Chapter 14 events used a postulated maximum power level of 2754 MWt (i.e., 102 percent licensed power), which bounds the 2737 MWt power level proposed in this LAR.

The NRC staff concludes that the proposed MUR power uprate will have no adverse impact on existing operator actions and little, or no, impact on time available to execute those actions. The only new operator actions are those associated with: 1) specifying the Caldon LEFM CheckPlus system as the data source to be used by the plant computer when performing a calorimetric; 2) response to the new control room annunciator associated with the Caldon LEFM CheckPlus system; and finally, 3) actions to be taken when the Caldon LEFM CheckPlus system is degraded or out-of-service. None of these actions affect EOPs or AOPs, nor are these actions credited in the current UFSAR. Administrative and procedural controls will be established to provide guidance to plant operators for all three of these new types of actions.

The NRC staff concludes that the licensee has satisfied Section VII.1 of Attachment 1 to RIS 2002-03.

3.1.2.2 Emergency and Abnormal Operating Procedures

The licensee reviewed the EOPs and AOPs for potential changes related to the proposed MUR power uprate. The licensee concluded that no new operator actions will be required in the EOPs and AOPs due to the proposed MUR power uprate. Power level changes in the EOPs and AOPs that are identified during the design change process will be addressed by revising the affected procedures. Training on the changes will be incorporated in the normal operator training cycles prior to the implementation of the MUR power uprate.

The NRC staff has reviewed the licensee's evaluation and concludes that the proposed changes needed to satisfy the MUR power uprate conditions will not present any adverse impacts on the EOPs and AOPs. This conclusion is based on the licensee making required changes to the EOPs and AOPs and completing the associated operator training prior to implementing the MUR power uprate. The NRC staff concludes that the licensee has satisfied Sections VII.2.A of Attachment 1 to RIS 2002-03.

3.1.2.3 Control Room Controls, Displays, and Alarms

In its submittal dated August 29, 2008, the licensee described changes to control room controls, displays, and alarms related to the proposed MUR power uprate as follows:

> There are no LEFM CheckPlus System controls available in the Control Room. All control functions reside locally at the LEFM CheckPlus system cabinets located in the Turbine Building.
>
> Control Room operators can select the LEFM CheckPlus System output as the source of input data for the Plant Computer calculation of calorimetric calculation via a control room display interface. The results of the calorimetric calculation are displayed on the Plant Computer to Control Room operators. System alarms trigger an alarm resulting in control room annunciation. There are no hardwired alarms from the LEFM CheckPlus System cabinet to the Control Room. The following conditions trigger the alarm:
>
> - LEFM CheckPlus System Meter Status Not Normal. . . .
> - Loss of Communication. . . .
> - LEFM CheckPlus System Cabinet High Temperature. . . .
>
> Guidance will be provided to identify the actions to be taken by the Control Room staff upon alarm annunciation.
>
> . . . .
>
> Also, a review of plant systems has indicated that only minor modifications are necessary (e.g., software modification that redefines the new 100% RTP, rescaling of plant indications to reflect the new 100% RTP). Calvert Cliffs follows

the established engineering procedures to ensure the necessary minor modifications are installed prior to implementing the proposed power uprate.

The NRC staff has reviewed the licensee's evaluation of the proposed changes to the control room and concludes that the proposed changes are minor and do not present any adverse effects to the operators' functions in the control room. This conclusion is based upon the licensee description above and its declaration that all modifications to the control room, including the software modifications, along with operator training on these changes will be made prior to MUR power uprate implementation. Therefore, the staff concludes that the licensee has satisfied Sections VII.2.B and VII.3 of Attachment 1 to RIS 2002-03.

3.1.2.4 Control Room Plant Reference Simulator and Operator Training Program

The plant simulator will be physically modified from the original design to reflect the MUR power uprate. Prior to the implementation of the MUR power uprate, the licensee will also modify the operator training program to address the changes made to the EOPs and AOPs, control room components, and plant simulator modifications. The licensee also plans to develop and complete operational training on the new Caldon LEFM system for the operators prior to MUR power uprate implementation.

The MUR power uprate is being implemented under the administrative controls of the design change process. As part of this process, any necessary changes to the simulator, such as modifications of displays, controls, and alarms to mimic the actual control room, will be identified during the design change review process, and will be completed prior to implementing the MUR power uprate. The NRC staff has reviewed the licensee's proposed changes related to the MUR power uprate and concludes that the changes will be appropriately addressed and do not present any adverse effects on the plant simulator or the operator training program. This conclusion is based on the licensee's description of the changes and its intention of making the changes to the plant simulator and operator training program prior to implementing the MUR power uprate. The staff concludes that the licensee has satisfied Sections VII.2.C and VII.2.D of Attachment 1 to RIS 2002-03.

3.1.3 Conclusion

The NRC staff has reviewed the licensee's evaluation of the proposed changes related to the human factors area and concludes that the licensee has adequately considered the impact of the proposed MUR power uprate on changes to operator actions, procedures, control room components, plant simulator and operator 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

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

record that do not bound the proposed uprated power level, the licensee should provide a detailed discussion of the reanalysis.

This safety evaluation (SE) documents the NRC staff review of the impact of the proposed changes on analyzed DBA radiological consequences. In CCNPP Amendments 281 and 258 to Units 1 and 2, respectively, which were issued on August 29, 2007 (ML072130521), the NRC approved implementation of a full-scope alternative source term in accordance with 10 CFR 50.67, and following the guidance of Regulatory Guide (RG) 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors." Therefore, the staff conducted this evaluation to verify that the results of the licensee's DBA radiological dose consequence analyses continue to meet the dose acceptance criteria given in 10 CFR 50.67 for offsite doses and 10 CFR Part 50, Appendix A, General Design Criterion (GDC) 19 (or equivalent for plants licensed before the GDC were in existence) with respect to control room habitability. The applicable acceptance criteria are 5 rem Total Effective Dose Equivalent (TEDE) in the control room, 25 rem TEDE at the exclusion area boundary, and 25 rem TEDE at the outer boundary of the low population zone. The staff utilized the regulatory guidance provided in applicable sections of RG 1.183, NUREG-0800, Standard Review Plan (SRP) Section 6.4, "Control Room Habitability System," for control room habitability, and CCNPP UFSAR Chapter 14, for DBAs, in performing this review.

3.2.2 Technical Evaluation

The NRC staff reviewed the regulatory and technical analyses performed by the licensee in support of its proposed MUR power uprate license amendment, as they relate to the radiological consequences of DBA analyses. Information regarding these analyses was provided by the licensee in Attachment 2 to the August 29, 2008, submittal. The findings of this SE are based on the descriptions and results of the licensee's analyses and other supporting information docketed by the licensee.

The NRC staff reviewed the impact of the proposed 1.38 percent MUR power uprate on DBA radiological consequence analyses, as documented in Chapter 14 of the UFSAR. The specific DBA analyses that were reviewed were as follows:

- Loss-of-Coolant Accident (LOCA)
- Fuel-Handling Accident (FHA)
- Main Steam Line Break (MSLB) Accident
- Seized Rotor Event (SRE) / Locked Rotor Accident (LRA)
- Control Element Assembly Ejection Accident (CEAEA) / Control Rod Ejection Accident (CREA)

The CCNPP Steam Generator Tube Rupture DBA was not reviewed because its licensing basis accident model assumes no fuel failure. Only coolant activity contributes to the dose consequence associated with this accident.

In the LAR submittal, the licensee stated that each of the current DBA dose analyses of record for CCNPP which depend on core power level, were performed at 2754 MWt, or 102 percent of the currently licensed thermal power of 2700 MWt. Therefore, the current analyses bound any analyses that would be performed at the proposed MUR uprated power level of 2737 MWt, as

the currently analyzed power is 100.62 percent of the proposed uprated power. This margin is within the assumed uncertainty associated with advanced flow measurement techniques, including use of the Caldon LEFM CheckPlus™ system credited by the licensee.

Using the licensing basis documentation, as contained in the current CCNPP UFSAR, in addition to information in the August 29, 2008, LAR submittal letter, the staff verified that the existing CCNPP UFSAR Chapter 14 radiological analyses source term and release assumptions bound the conditions for the proposed 1.38 percent power uprate to 2737 MWt, considering the higher accuracy of the proposed feedwater flow measurement instrumentation.

3.2.3 Conclusion

As described above, the NRC staff reviewed the assumptions, inputs, and methods used by the licensee to reassess the radiological consequences of the postulated DBA with the proposed uprated power level. The staff finds that the licensee will continue to meet the applicable dose acceptance criteria, as identified in Section 2.0 of this evaluation, following implementation of the proposed 1.38 percent MUR power uprate. The staff further finds reasonable assurance that CCNPP, as modified by this approved license amendment, will continue to provide sufficient safety margins, with adequate defense-in-depth, to address unanticipated events and to compensate for uncertainties in accident progression, analysis assumptions, and input parameters. Therefore, the staff concludes that the proposed license amendment is acceptable with respect to the radiological dose consequences of the 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's 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 a 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 (1) 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; (2) 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 (3) 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.

3.3.2 Technical Evaluation

The NRC staff reviewed the licensee's commitment to 10 CFR 50.48, "Fire protection" (i.e., approved fire protection program). The staff's review also covered the impact of the proposed

MUR power uprate on the results of the safe-shutdown fire analysis as noted in RIS 2002-03, Attachment 1, Sections II and III. The review focused on the effects of the MUR power uprate on the post-fire safe-shutdown capability and increase in decay heat generation following plant trips.

The NRC staff requested the licensee to verify that the overall temperature changes in the primary and secondary systems are "very small" (i.e., provide the values) and, at these higher temperatures, Appendix R equipment and plant operators are "unaffected," thereby remaining in compliance with 10 CFR Part 50, Appendix R. Further, the staff requested the licensee to verify that additional heat in the plant environment from the MUR power uprate will not prevent required post-fire operator manual actions, as identified in the CCNPP fire protection program from being performed at their designated time.

In a letter to the NRC dated December 30, 2008, the licensee stated that as a result of the MUR power uprate, the major changes potentially affecting existing heat loads are due to the increase in process temperatures in the RCS and feedwater (FW) system. RCS T_{hot} will increase 0.8 °F from 595.1 °F to 595.9 °F and FW temperature will increase 2.1 °F from 431.5 °F to 433.6 °F. Further, the licensee concluded that the effect of the increased temperatures in the RCS hot leg and in the condensate/FW trains have no meaningful impact on the containment, auxiliary building and turbine building environments under normal, accident and Appendix R plant conditions and scenarios.

The NRC staff requested the licensee to verify that the plant can meet the 72-hour safe shutdown requirements found in 10 CFR Part 50, Appendix R, Sections III.G.1.b and III.L, with the increased decay heat at MUR power uprate conditions.

In a letter to the NRC dated December 30, 2008, the licensee stated that all Appendix R calculations have been verified to use core decay heat generation rates which bound those calculated for the MUR core power level of 2737 MWt. In addition, no other Appendix R Safe Shutdown requirements (such as condensate requirements, 72-hour cold shutdown requirements, or the repair requirements) have been impacted. As a result, CCNPP remains in compliance with the requirements in 10 CFR Part 50, Appendix R, Sections III.G.1.b and III.L.

The NRC staff concludes that the information provided in the LAR, as supplemented by the response to the staff's request for additional information (RAI), satisfactorily demonstrates that the licensee will remain in compliance with the requirements of 10 CFR Part 50, Appendix R, Sections III.G.1.b and III.L following implementation of the MUR power uprate. The MUR power uprate does not change the equipment necessary for post-fire safe-shutdown nor does it require reroute of essential cables or relocation of essential components/equipment credited for post-fire safe-shutdown. The licensee has made no changes to the plant configuration or combustible loading as a result of modifications necessary to implement the MUR power uprate that affect the CCNPP fire protection program. The staff concurs with the licensee's conclusion that the proposed 1.38 percent power uprate will not have an effect on post-fire safe-shutdown capability of the plant.

3.3.3 Conclusion

The NRC staff has reviewed the licensee's fire-related safe-shutdown assessment and concludes that the licensee has adequately accounted for the effects of the 1.38 percent increase in decay heat on the ability of the required systems to achieve and maintain safe-shutdown conditions. The staff finds this aspect of the capability of the associated SSCs to perform their design basis functions at the increased core power level of 2737 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 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. Material loss rates due to FAC depend on flow velocity, fluid temperature, steam quality, oxygen content, and pH. During plant operation, control of these parameters is limited and the optimum conditions for minimizing FAC often cannot be achieved. Therefore, loss of material by FAC is likely to occur. The staff reviewed the LAR for potential effects on FAC and the adequacy of the licensee's FAC program to predict loss rates so that repair or replacement of affected components could be made before reaching critical minimum thickness. 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

The licensee stated that the impact of the proposed MUR power uprate on the FAC program resulted in some long-term impacts, such as increased inspection scope and possibly some increased replacement scope prior to end-of-plant life expectancy, and no short-term impacts. The licensee stated that the system most susceptible to long-term impacts is the feedwater system. Since there is an expected increase in flow velocity, the CHECWORKS computer model will be updated to account for the increase in flow rate. Based on the results from this update, areas that are predicted to experience high wear are identified, sorted by wear rate and time until reaching code minimum wall thickness, and then scheduled for pre-uprate and follow-up inspections to evaluate the actual wear once the power uprate has been implemented.

In the licensee's License Renewal Application dated April 8, 1998 (ADAMS Accession No. 9804100416), the licensee stated that inspection locations are also determined through evaluations of site-specific-data and failures at other plant sites. CCNPP is a member of the CHECWORKS Users group, which is an industry organization that shares industry information and provides training on methods and technology. Once the inspection is complete and the components are determined to be experiencing wear due to FAC, components are trended and evaluated for time until reaching code minimum values and replacements are then scheduled at an outage prior to reaching this minimum allowable value. The licensee stated that wear rate changes from the MUR may be undetectable using measurement techniques. This is due to the

fact that velocity changes are predicted to be minimal, thereby causing little change in wear rates experienced by the systems.

3.4.1.3 Conclusion

On the basis of its review, the NRC staff concludes that the FAC program is acceptable for MUR power uprate operating conditions because the effect on FAC rates is expected to be small and will be adequately controlled by procedures in the existing FAC program. Therefore, the staff finds the proposed LAR acceptable with respect to the FAC program.

3.4.2 Coatings

3.4.2.1 Regulatory Evaluation

Protective coatings (paints) inside containment are used to protect equipment and structures from corrosion and radionuclide contamination and provide wear protection during plant operation and maintenance activities. The coatings are subject to 10 CFR Part 50, Appendix B. The NRC staff review focused on whether the pressure and temperature conditions under the proposed power uprate continue to be bounded by the conditions to which the coatings were qualified.

3.4.2.2 Technical Evaluation

Equipment and structures inside containment are protected from the environment during normal operating and accident conditions by protective coating systems (paints). The licensee stated that the coatings within the containment will not be impacted by the MUR power uprate since mass and energy values are not changed from previously analyzed conditions. In response to an RAI, the licensee stated that the Service Level I coatings were originally qualified to temperature-pressure curves that bound CCNPP's temperature-pressure curve for the DBA. The licensee concluded that the coatings are still qualified for operation under power uprate conditions.

3.4.2.3 Conclusion

On the basis of its review, the NRC staff concludes that the coatings will not be adversely impacted by the MUR power uprate. The temperature and pressure limits under power uprate conditions continue to be bounded by the conditions to which the coatings were qualified. Therefore, the staff finds the proposed MUR power uprate acceptable with respect to coatings.

3.4.3 Steam Generator Program

3.4.3.1 Regulatory Evaluation

Steam generator (SG) tubes constitute a large part of the reactor coolant pressure boundary (RCPB). The NRC staff reviewed the effects of changes in operating parameters (e.g., pressure, temperature, and flow velocities) resulting from the proposed power uprate on the design and operation of the SGs. Specifically, the staff evaluated whether changes to these

parameters continue to be bounded by those considered in the plant design and licensing basis (i.e., the TS plugging limits).

3.4.3.2 Technical Evaluation

The licensee replaced the SGs during the 2002 (Unit 1) and 2003 (Unit 2) refueling outages. The replacement SGs were manufactured by Babcock and Wilcox Canada and each contains 8471 thermally-treated Alloy 690 tubes. The tubes have an outside diameter of 0.750 inches and a nominal wall thickness of 0.042 inches. The tubes are expanded for the full depth of the tubesheet in both the hot-leg and cold-leg. The licensee stated that the SG manufacturer performed a thermo-hydraulic and structural evaluation of a 1.7 percent power uprate. Feedwater flow, operating temperature, and differential pressure across the tubes will change under power uprate conditions. The licensee expects a marginal increase in the stress corrosion cracking susceptibility due to a temperature increase of 0.8 °F. Alloy-690 tubing is more resistant to stress corrosion cracking than the Alloy-600 mill annealed tubing used in the original SGs. In response to an RAI regarding whether the SG will satisfy all original design criteria under power uprate conditions, the licensee stated that the pressure differential across the tubes between the primary and secondary side will experience a negligible increase under MUR power uprate conditions. The new differential pressure is bounded by the original specification design pressure of 2500 psia (maximum) for the primary side (unchanged for the power uprate) and 850 psia (minimum) for the secondary side (estimated at 860.3 psia at end-of-life under power uprate conditions). As for the temperature, the expected increase of 0.8 °F to a maximum of 595.9 °F is still bounded by the original specification design temperature of 650 °F.

In addition, the licensee stated that the manufacturer performed a flow-induced vibration (FIV) analysis at 1.7 percent MUR power uprate end-of-life conditions. The licensee stated that the results showed that a 1.7 percent MUR power uprate will have an insignificant impact on the FIV response of the critical U-bend and bundle entrance tubes. Furthermore, the licensee stated that because of the minimal increase in the bundle entrance and riser flow velocity as a result of the power uprate, the existing FIV calculation for other internal components will not be affected by operation under power uprate conditions.

The licensee also reviewed the current condition of the SGs. Given that the SGs were recently put into service, very few tubes have been plugged. The licensee stated that the current plug and tube stabilizer design parameters bound the power uprate conditions. Loose parts left in the SGs were reviewed and it was determined that the power uprate will have no effect on the loose parts due to location of the parts, and that the tubes in the area were preemptively plugged and stabilized to ensure pressure boundary for the unit.

In response to an RAI regarding whether the plugging limit is still appropriate according to RG 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes," the licensee stated that the RG 1.121 analysis was reviewed and found to still be appropriate for the power uprate conditions, with no changes. The analysis utilized a combination of 1400 psi differential for normal operating conditions, which bounds the MUR PU conditions of 1390 psi differential maximum.

3.4.3.3 Conclusion

On the basis of its review, the NRC staff concludes that the power uprate is acceptable from a SG design and inservice inspection perspective because the licensee's evaluations of thermal-hydraulic performance, their structural evaluation, and their FIV analysis have shown that the power uprate is expected to introduce only negligible changes in the SG parameters, which will not significantly affect the performance of the SGs, and it will continue to operate within its design limits under uprate conditions. The licensee has confirmed that the plugging limit continues to be appropriate for power uprate conditions according to the guidance in RG 1.121. Therefore, the NRC staff finds the proposed MUR power uprate acceptable with respect to the SGs.

3.4.4 Chemical and Volume Control System (CVCS)

3.4.4.1 Regulatory Evaluation

As discussed in SRP Section 9.3.4, "Chemical and Volume Control System (PWR)" the CVCS provides a means for (1) maintaining water inventory and quality in the RCS, (2) supplying seal-water flow to the reactor coolant pumps and pressurizer auxiliary spray, (3) controlling the boron neutron absorber concentration in the reactor coolant, (4) controlling the primary-water chemistry, (5) reducing coolant radioactivity level, (6) supplying recycled coolant for demineralized water makeup for normal operation, and (7) providing high-pressure injection flow to the ECCS in the event of a postulated accident. The NRC staff has reviewed the safety-related functional performance characteristics of CVCS components to ensure they are not affected by the MUR power uprate. The acceptance criteria are based on GDC 14, "Reactor coolant pressure boundary," which requires the RCPB to be designed to have an extremely low probability of abnormal leakage, of rapidly propagating fracture, and of gross rupture, and GDC 29, "Protection against anticipated operational occurrences," which requires the reactivity control systems be designed to assure an extremely high probability of accomplishing their safety functions in the event of anticipated operational occurrences. Specific review criteria are contained in SRP Section 9.3.4.

3.4.4.2 Technical Evaluation

During plant operation, reactor coolant letdown flow originates from the cold-leg on the suction side of the reactor coolant pump (RCP). The flow progresses through the tube side of the regenerative heat exchanger, the letdown flow control valves, a letdown heat exchanger, and the letdown pressure regulating valves. The regenerative heat exchanger reduces the temperature of the reactor coolant and the letdown flow control valves limit the flow rate through the CVCS system. The letdown heat exchanger further reduces the reactor coolant temperature and the letdown pressure regulating valves maintain pressure on the coolant to prevent it from flashing to steam. Flow continues through purification filters and ion exchangers, where suspended solids and ionic impurities are removed, thus keeping the reactor coolant activity within design limits. The reactor coolant then passes through the letdown filter and enters the volume control tank (VCT). The charging pumps take suction from the VCT and return the coolant through the shell side of the regenerative heat exchanger to the RCS in the cold-leg, downstream of the RCP.

Under the MUR power uprate conditions, the licensee indicated that because the cold-leg temperature will not change, there will be no impact on the thermal performance or requirements of the CVCS system, as it is supplied from the cold-leg.

The licensee concluded that there is a slight increase of N-16 activity, as a result of the MUR power uprate conditions, but this will have a negligible effect on the decay time requirements of the letdown line and no changes to the letdown and makeup requirements are required. In addition, the licensee stated that the small increase in the average coolant temperature (as a result of the 0.8 °F increase in hot-leg temperature) causes a small increase in the makeup requirement for coolant shrinkage during RCS cooldown, but that this effect is considered insignificant and remains bounded by the design conditions. Therefore, the CVCS system is capable of supporting the MUR power uprate.

3.4.4.3 Conclusion

Based on its review, the NRC staff concludes that the CVCS is adequate because the proposed MUR power uprate will introduce negligible changes in the CVCS operating parameters, which will not affect satisfactory performance of the CVCS intended functions, and the CVCS will continue to operate within its design limits under the uprate conditions. Therefore, the staff finds the proposed MUR power uprate acceptable with respect to the CVCS.

3.4.5 Steam Generator Blowdown System (SGBS)

3.4.5.1 Regulatory Evaluation

Control of secondary-side water chemistry is important for preventing degradation of SG tubes. The SGBS provides a means for removing SG secondary-side impurities, and thus assists in maintaining acceptable secondary-side water chemistry in the SGs. The design basis of the SGBS includes consideration of expected design flows for all modes of operation. The NRC staff reviewed the ability of the SGBS to remove particulate and dissolved impurities from the SG secondary-side during normal operation, including condenser in-leakage and primary-to-secondary leakage. The NRC's acceptance criteria for the SGBS are based on GDC 14 which requires the RCPB to be designed to have an extremely low probability of abnormal leakage, of rapidly propagating fracture, and of gross rupture. Specific review criteria are contained in SRP Section 10.4.8, "Steam Generator Blowdown System (PWR)."

3.4.5.2 Technical Evaluation

The SGBS is designed to extract water containing particulates and dissolved solids from the secondary side of the SGs, as a means of controlling SG water chemistry. The water collected from the SG is piped to the blowdown tank, which is vented to the atmosphere and drains to the service water system. The SGBS also provides a means for sampling the secondary side water in the SG. These samples are used for monitoring water chemistry and for detecting the amount of radioactive primary coolant leakage through the SG tubes. Proper control of SG secondary-side chemistry reduces the probability of secondary-side-initiated SG tube degradation.

The licensee indicated that the blowdown flow rates required during plant operation are based on chemistry control and tubesheet sweeping requirements to control the buildup of solids. Since the variables that influence the required blowdown flow rates (i.e., allowable condenser in-leakage, total dissolved solids level in the plant service water system, the amount of corrosion products generated by FAC, and allowable primary-to-secondary leakage) are not changed by the MUR power uprate, the blowdown flow rates required for maintaining chemistry control and tubesheet sweeping will not be affected.

The licensee also stated that since the no-load and full-load SG steam pressures (1,106 psia and 825 psia respectively) are not changing with the MUR power uprate, there will be no impact on blowdown flow control (since inlet pressure to the SGBS varies with SG operating pressure).

3.4.5.3 Conclusion

Based on its review, the NRC staff concludes that the SGBS remains adequate for power uprate conditions because the blowdown flow rate, the SG secondary-side water chemistry, and the blowdown pressures and temperatures remain within the original system design. Therefore, the staff finds the proposed MUR power uprate acceptable with respect to the SGBS.

3.4.6 Overall Chemical Engineering Conclusion

In the areas of SGs and chemical engineering, the NRC staff concludes that the licensee has adequately addressed (1) the changes in the plant operating conditions for the FAC program, (2) the effects on protective coatings, (3) the changes in the SG operating parameters, the effects on the SGs and the determination that the SG tube integrity will continue to be maintained, (4) the changes of the reactor coolant and their effect on the CVCS, and (5) the changes in the system flow and impurity levels, and their effects on the SGBS.

3.5 Mechanical and Civil Engineering

3.5.1 Regulatory Evaluation

The NRC staff's review in the areas of mechanical and civil engineering covers the structural and pressure boundary integrity of the nuclear steam supply system (NSSS) and balance-of-plant (BOP) systems and components. This 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) the pressure retaining portions of the control element drive mechanisms (CEDMs); (5) the replacement steam generators (RSGs) and their supports; (6) the pressure retaining portions of the RCPs and their supports; (7) the pressurizer and its supports; (8) the reactor vessel internals (RVIs); (9) safety-related valves; and (10) safety-related pumps. Technical areas covered by this review include stresses, cumulative usage factors (CUFs), FIV, high-energy line break (HELB) locations, and jet impingement and thrust forces.

The 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 Code), Section III, the United States of America Standards (USAS) B31.1 Code for Power Piping, and the USAS B31.7 Code for Nuclear Power

Piping. The NRC staff's evaluation considered draft GDC 1, 2, 9, 33, 34, 35, and 51 which are located in Appendix 1C of the CCNPP UFSAR. The staff review focused on verifying that the licensee has provided reasonable assurance of the structural and functional integrity of piping systems, components, component internals and their supports under normal and vibratory loadings, including those due to fluid flow, postulated accidents, and natural phenomena such as earthquakes.

The acceptance criteria are based on continued conformance with the requirements of the following regulations: (1) 10 CFR 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 RCPB being designed and constructed to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture; (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; (5) draft GDC 35 as it relates to the prevention of a brittle fracture of the RCPB and (6) 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.

3.5.2 Technical Evaluation

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

The proposed 1.38 percent power uprate will increase the RTP level from 2700 MWt to 2737 MWt at CCNPP. The power uprate will be achieved by an increase in demand to the turbine-generator. In turn, an increase in steam flow will occur due to the increased demand on the secondary side of the plant. In addition, there will be an increased temperature difference across the core with the RCS pressure remaining the same.

Table IV-1 of Attachment 2 to the LAR dated August 29, 2008, shows the pertinent temperatures, pressures, and flow rates for the current and projected uprated conditions. At full power, the hot-leg temperature increases from 595.1 to 595.9 degrees Fahrenheit (°F) while the cold-leg temperature remains constant at 548 °F. The RSG pressures decrease from 888.0 to 886.5 psia and the steam flow increases from 5.9 to 5.999 million pounds per hour (Mlbm/hr). The FW temperature increases from 431.5 to 433.6 °F and the FW flow increases from 5.9 to 5.999 Mlbm/hr (same flow rate increase as steam flow). The design parameters for the primary and secondary systems at CCNPP are found in Tables 4-1 and 10-1, respectively, of the CCNPP UFSAR. The RCS components are designed to 650 °F (except the pressurizer, which is designed to 700 °F) and 2,500 psia. The FW system design temperature is 460 °F with a design pressure of 1515 psia. The main steam (MS) system design temperature is 580 °F with a design pressure of 1015 psia.

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.

3.5.2.1 Reactor Pressure Vessel

The Code of record for the RPV, nozzles, and supports is the ASME Code, Section III, 1965 Edition with Winter 1967 Addenda. The Code of record for the replacement reactor vessel closure head (RVCH) is the ASME Code, Section III, 1995 Edition with 1996 Addenda. The licensee compared the expected temperatures and pressures for the proposed power uprate condition against the analyses of record. The licensee confirmed in its submittal that there is no change in RCS design or operating pressure, and the effects of operating temperature changes for cold and hot legs are within design limits. The licensee confirmed that MUR power uprate conditions are bounded by the design basis analyses for the RPV and the operating transients continue to bound the uprate conditions and no additional transients have been proposed.

With regards to the RVIs, the licensee stated that structural and mechanical evaluations were performed on these components to determine any effects on the RVIs due to the uprated conditions. The power uprate will not affect the design basis for the seismic and LOCA loads for CCNPP, negating the need to re-assess the structural integrity of the RVIs with regards to the LOCA-induced hydraulic and dynamic loads and seismic loads. In addition, since there is a negligible decrease in the primary side fluid density and the RCS primary design flow rate remains unchanged, the licensee indicated that flow and pump induced vibration effects on the RVIs resulting from the uprate are negligible. Due to the potential increase in thermal loadings on the RVIs resulting from the power uprate, the licensee re-analyzed the thermal stresses on the most vulnerable portion of the RVIs; the core shroud. The stress analysis on the core shroud performed by the licensee determined that the maximum primary-plus-secondary stress value exceeded the ASME Code allowable value for the shroud. A subsequent elastic-plastic stress analysis performed in accordance with ASME acceptance criteria found the maximum primary-plus-secondary stress intensity value of 37,927 psi acceptable and within the ASME Code allowable value of 43,800 psi. Additionally, the CUF for the shroud, which was determined to be 0.375 for the MUR conditions, is well below the ASME Code allowable value of 1.00.

The existing loads, stresses, and fatigue CUF values for reactor vessel and internals remain valid for the proposed power uprate. The NRC staff concurs with the licensee's assessment that the RPV and internals are acceptable for operation at the uprated power level given that the current design basis analyses remain valid for these components at the uprated conditions.

3.5.2.2 Control Element Drive Mechanisms

The Code of record for the pressure retaining components of the CEDMs is the ASME Code, Section III, 1998 Edition with 2000 Addenda. The CEDMs are affected by the RCS pressure, hot leg temperature, and hot leg design transients, of which only the hot leg temperature changes as a result of the power uprate (0.8 °F increase to 595.5 °F). Based on this information, the licensee confirmed in its submittal that the CEDM design analyses continue to be bound by the conditions at the proposed uprated power level and all critical margins on these components will be maintained. In addition, the operating transients continue to bound the

uprate conditions and no additional transients have been proposed. The NRC staff concurs with the licensee's assessment that the CEDMs are acceptable for operation at the uprated power level due to the bounding nature of the current CEDM design analyses.

3.5.2.3 Reactor Coolant Piping and Components

The RCS piping was designed to ASME Code, Section III, 1965 Edition with Winter 1967 Addenda and the USAS B31.7 Code for Nuclear Power Piping. The licensee reviewed the revised design conditions for impact on the existing design basis analyses for the reactor coolant piping and supports. It was stated that there is no change in RCS design or operating pressure, and the effects of the increased operating temperature for the hot legs are within design limits. The MUR power uprate conditions are bounded by the design conditions and the RCS piping remains within the allowable stress limits provided by the ASME Code. 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 RCS piping and supports remain valid for the proposed power uprate.

The RSGs are designed to two editions of Section III of the ASME Code. The upper portions of the current RSGs (the steam drums vessels) which remain from the previous SGs were designed to the 1965 Edition with Winter 1967 Addenda, while the lower assembly portions of the SGs which were replaced at CCNPP were designed to the 1989 edition of the code. The licensee reviewed the revised design conditions for impact on the existing design basis analyses for the RSGs including the RSG tubes, secondary side internal support structures, shell and nozzles. There is a no change in RCS mass flow rate and the RCS temperatures and pressures used in the design continue to bound the uprate conditions. At the uprate conditions, there is an increase in the steam flow and FW flow. In response to a staff RAI, the licensee confirmed that the steam and FW flow rates used in the design of the RSGs continue to bound the expected uprate conditions since the analyses performed to support the power uprate were completed at a power level which bounds the proposed uprate conditions. In addition, the operating transients continue to bound the uprate conditions and no additional transients have been proposed. Additionally, flow-induced vibration was considered due to the secondary side flow increase in the RSGs. The licensee stated that turbulent induced vibrations will increase 4 percent due to the steam flow increase. However, this will not create any adverse effects since the uprate does not cause a shift in the vibration excitation frequency. The existing loads, stresses, and fatigue CUF values for the RSGs remain valid for the proposed MUR uprate.

The pressure retaining parts of the four RCPs at CCNPP were designed in accordance with the ASME Code, Section III, 1965 Edition with Winter 1967 Addenda. The licensee reviewed the revised design conditions to determine the impact on the existing design basis analyses for the RCPs. Due to the arrangement of the CCNPP RCS, the RCP loading conditions and thermal transients are only affected by changes in the RSG outlet temperature, i.e. the cold leg temperature. As previously mentioned, there is no change in the cold leg temperature at CCNPP as a result of the proposed MUR power uprate. Therefore, it was stated that the existing design basis analyses and the existing loads, stresses, and fatigue CUF values for the RCPs remain valid for the MUR power uprate. In addition, the operating transients continue to bound the uprate conditions and no additional transients have been proposed.

The Code of record for the pressurizer, including the nozzles, is the ASME Code, Section III, 1965 Edition with Winter 1967 Addenda. The licensee reviewed the revised design conditions to determine the impact on the existing design basis analyses for the pressurizer. The licensee stated that the temperature changes due to the MUR power uprate are bounded by those used in the existing analyses. The licensee indicated that some of the thermal transients were affected by the uprate. Subsequently, the licensee re-evaluated critical locations on the pressurizer to determine any effects these transients may have on the pressurizer components. It was found that the operating transients continue to bound the uprate conditions. Additionally, no additional transients have been proposed. The existing loads, stresses, and fatigue CUF values for the pressurizer remain valid for the proposed power uprate.

The NRC staff concurs with the licensee's conclusion that the design of the reactor coolant piping and components, including the RSGs, RCPs, and pressurizer, and their supports, is adequate to maintain the structural and pressure boundary integrity of the reactor coolant loop since the design basis analyses of these components remain bounding for the proposed 1.38 percent power uprate condition.

3.5.2.4 High Energy Line Break Locations

The licensee stated that the current HELB analysis for CCNPP was reviewed in support of the proposed MUR power uprate. The licensee stated that the changes in the secondary side steam properties were negligible with respect to their effects on the current HELB analysis, which continues to bound the MUR uprated conditions. In addition, no new piping was added, no postulated break locations were changed, no changes were made to the assumed blowdown from the current postulated break locations, and there are no new systems that qualify as HELB systems as a result of the uprate. Based on this information, the licensee concluded that the current CCNPP HELB analysis remains unaffected by the uprate. The NRC staff agrees with the licensee's conclusion regarding HELBs given that the current analysis remains bounding and there are no changes to the analysis required in support of the uprate.

3.5.2.5 BOP Piping Systems

The licensee evaluated the BOP piping systems by comparing the conditions for the proposed power uprate with the analyses of record conditions and the current operating conditions. The BOP piping systems evaluated in support of the proposed power uprate include the main steam, FW, extraction steam, moisture separator drains, condensate, and heater drain piping. As previously mentioned, in response to a staff RAI regarding the increase in steam flow and its effects on the main steam system piping, the licensee indicated that the analyses performed in support of the uprate utilized a power level increase of 1.7 percent, which bounds the proposed increase of 1.38 percent. With respect to the steam hammer loads corresponding to the increase in steam flow, the licensee indicated that the current design steam hammer loads continue to bound the MUR uprate conditions. Also as previously mentioned, the licensee indicated in its RAI response that turbulent induced vibrations will increase 4 percent due to the steam flow increase. However, this will not create any adverse effects as the uprate does not cause a shift in the vibration excitation frequency of the piping system and associated components.

The changes in the operating temperatures and flow rates due to the MUR power uprate have been evaluated for the aforementioned piping systems and were determined by the licensee to have a negligible effect on the existing design basis analyses. 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 systems remain valid for the proposed power uprate.

The licensee concluded that the CCNPP BOP piping systems remain acceptable for operation at the uprated conditions. Based on the above, the NRC staff agrees with the licensee's conclusion that the proposed 1.38 percent power uprate will not have adverse effects on BOP system piping.

3.5.2.6 Safety-Related Valves

The NRC staff reviewed 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 also provided by the plant-specific evaluations of 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," 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."

The licensee reviewed the impact of the proposed MUR power uprate conditions on the existing design basis analyses for the safety-related valves. The evaluation showed that the temperature changes due to MUR power uprate are insignificant and bounded by those used in the existing analyses. The analyses also confirmed that the installed capacities and lift setpoints of the RCS and main steam safety valves remain valid for the MUR power uprate 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 power uprate. The licensee stated that the plant uprate accident analysis required flows are not changing, the specific nuclear class valve response times are not changing, and the nuclear grade valve components are not physically changing to support the MUR power uprate; and, therefore, the inservice testing program for safety-related valves will not be affected.

The licensee also stated that systems that have valves maintained within the air-operated valve program, the GL 89-10 motor-operated valve program, and the GL 95-07 pressure locking/thermal binding program were reviewed. The review concluded that the MUR power uprate does not impact program valves since the operating temperature and pressure ranges are bounded by the original design parameters and the MUR power uprated accident analysis required flows are not changing. Therefore, the NRC staff finds the performance of existing safety-related valves acceptable with respect to the MUR power uprate.

3.5.2.7 Safety-Related Pumps

The NRC staff reviewed the licensee's safety-related pumps analysis. The NRC's acceptance criteria for reviewing the safety-related pumps analysis are based on 10 CFR 50.55a, "Codes and standards."

The licensee reviewed the impact of the proposed MUR power uprate conditions on the existing design basis analyses for the safety-related pumps. The evaluation showed that the operating temperature and pressure ranges for the pumps due to the MUR power uprate are bounded by the original design parameters. Also, the original design transients for the safety-related pumps bound the transients associated with the MUR power uprate. The licensee stated that the MUR power uprate accident analysis required flows are not changing, and therefore the inservice testing program for safety-related pumps will not be affected. Therefore, the NRC staff finds the performance of existing safety-related pumps acceptable with respect to the MUR power uprate.

3.5.3 Conclusion

The NRC staff has reviewed the licensee's assessment of the impact of the proposed MUR power uprate on NSSS and BOP systems and components with regard to stresses, CUFs, flow-induced vibration, HELB locations, and jet impingement and thrust forces. On the basis of this review described above, the staff concludes that the proposed MUR power uprate will not have an adverse impact on the structural integrity of the piping systems, components, their supports, RVIs, CEDMs, or BOP piping.

3.6 Reactor Systems

3.6.1 Regulatory Evaluation

The Reactor Systems review includes the thermal hydraulic aspects of the CheckPlus UFM, including those aspects of the transducers that may influence the perceived flow profile, and consideration of the associated uncertainty. This review also includes evaluating certain aspects related to the effects of the power uprate on reactor pressure vessel integrity. These aspects were related to the reactor vessel neutron fluence determinations that were used in calculations supporting this request. The NRC staff evaluated the licensee's fluence calculations using the guidance contained in RG 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence." Finally, this review includes the CCNPP UFSAR accident and transient analyses.

The licensee developed its license amendment request consistent with the guidelines in NRC Regulatory Issue Summary (RIS) 2002-03, "Guidance on the Content of Measurement Uncertainty Recapture Power Uprate Applications."

3.6.2 Technical Evaluation

The NRC staff reviewed the thermal-hydraulic aspects of the LEFM CheckPlus system installation, including its laboratory calibration, the effects of system changes such as transducer replacement, and the impact the system installation will have, if any, on the applicable plant safety analyses.

3.6.2.1 Feedwater Flow Measurement Device Installation

The Caldon LEFM CheckPlus Systems at CPNPP Unit Nos. 1 and 2 consist of two measurement section/spool pieces located in the 16-inch feedwater header for each steam generator. The CheckPlus UFMs are to be installed in accordance with approved Caldon

Topical Reports ER-80P and ER-157P. The measurement sections are located upstream of the existing feedwater flow venturis in the turbine building and auxiliary building and downstream of the feedwater regulating valves.

In Unit 1, Loop A, 11 Feedwater Header, the flow meter is located approximately 5 feet 4 inches below the feedwater regulating valve and 3 feet 3 inches from the exit of a 90 degree elbow located upstream of the flowmeter spool piece. In Unit 1, Loop B, 12 Feedwater Header, the flowmeter is located approximately 7 feet from the exit of a 90 degree elbow located upstream of the flow meter spool piece.

In Unit 2, Loop A, 21 Feedwater Header, the flowmeter is located approximately 7 feet 4 inches below the feedwater regulating valve and 3 feet 3 inches from the exit of a 90 degree elbow located upstream of the flowmeter spool piece. In Unit 2, Loop B, 22 Feedwater Header, the flowmeter is located approximately 7 feet 4 inches below the feedwater regulating valve and 10 feet 2 inches from the exit of a 45 degree elbow located upstream of the flowmeter spool piece.

It can be seen from the description of the installations that the CheckPlus may operate in regions 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 3.6.2.4 below.

3.6.2.2 CheckPlus Non-Functionality

To operate above the presently licensed power of 2700 MWt, the licensee proposes that the CheckPlus can be out-of-service for up to 72 hours provided that the plant computer remains available to perform the secondary calorimetric calculation, and the plant exhibits steady-state conditions. The licensee defines steady-state conditions as power changes that do not exceed 10 percent of the initial power level when the system is declared out-of-service.

Power level during the 72 hours without an operational CheckPlus will be monitored using existing plant instrumentation, such as the feedwater venturis, currently being used to calculate secondary calorimetric power. The licensee justifies this operation on the basis that (1) alternate plant instrumentation exists to calculate calorimetric power, (2) plant computer calculations normalize the alternate input for feedwater flow, (3) Calvert Cliffs has not had a history of venturi nozzle fouling or defouling, (4) the instrument drift is negligible, and (5) Calvert Cliffs' 1.38 percent uprate is conservative when compared to the 1.6 percent or 1.7 percent uprates the CheckPlus system supports.

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 2700 MWt. Stated differently, after 72 hours without an operable CheckPlus, or if core thermal power changes by more than 10 percent while the CheckPlus is non-functional, 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 in the Technical Requirements Manual. The NRC staff finds that operation with a non-functional CheckPlus has been acceptably addressed.

3.6.2.3 Transducer Replacement

The effect of transducer replacement on the CheckPlus system uncertainties has been addressed in "Caldon Ultrasonics, Engineering Report: ER-551P Rev.1, LEFM✔ + Transducer Installation Sensitivity." The original uncertainty calculations included a 0.16 percent transducer replacement uncertainty per LEFM CheckPlus meter based upon the results of "Caldon Ultrasonics, Engineering Report: ER-551P Rev.3, LEFM✔ + Transducer Installation Sensitivity." Since the uncertainty of 0.16 percent per LEFM CheckPlus meter is incorporated in the original uncertainty calculation and no additional uncertainty terms need to be applied whenever a transducer is replaced, the NRC staff finds that transducer installation variability has been acceptably addressed.

3.6.2.4 CheckPlus Calibration

The CheckPlus calibration was accomplished at Alden Laboratories. The licensee's supplemental letter dated February 18, 2009, included the test configuration and the Alden Test Plans. The NRC staff compared the test configuration to drawings and information in the licensee's supplemental letter and noted the following:

- An inconsistency was noted between the laboratory calibration setup and the piping run in that there are nearby elbows downstream of the in-situ ultrasonic flow meter installation which are absent in the laboratory calibration setup.

Significant variations in piping configuration between the in-situ LEFM installation and the experimental calibration facility could adversely affect the LEFM calibration. The licensee's letter discussed the use of a mitered 90-degree elbow in lieu of a feedwater regulating valve, a difference which was compensated by use of different flow orifices and flow straighteners installed in the pipe. According to the Alden Test Plans, different flow orifices were used for a parametric study of the effects of the associated flow restriction on the flatness ratio and meter factor. This is a difference that the NRC staff finds acceptable in light of the equipment that the mitered elbow and flow orifices are intended to simulate.

The NRC staff reviewed drawings and schematics provided by the licensee and confirmed that, insofar as upstream configuration is concerned, the laboratory configuration largely matched the in-situ configuration. However, several of the provided diagrams indicated that key geometric components located downstream of certain chordal meter exit regions were not included in the test configuration.

Presumably, the effects of downstream equipment need not be considered in this type of installation. It would be expected that, at the turbulent flow regimes anticipated when the LEFM system is operating and the plant is at or near full power and full feedwater flow conditions, perturbation from the downstream piping would not propagate any significant length upstream. In some cases, however, the CCNPP in-situ installation contains piping elbows that begin to curve 15 inches downstream of the LEFM exit. This distance is narrowly less than a single piping diameter.

The NRC staff performed a simple comparison of the in-situ piping installation with downstream piping to that tested in the laboratory setting. To perform this comparison, the staff used a

computational fluid dynamics model of the two pipe runs to observe potential differences in the static pressure profile across the section of piping immediately at the exit of the LEFM. Based on the staff's comparison, some minor pressure differences were observed, and the staff requested that the licensee account for the difference in downstream piping between the in-situ installation and the laboratory testing configuration.

In the licensee's supplemental letter dated May 7, 2009, the licensee explained why the close proximity of a downstream piping difference between in-situ and laboratory installations still results in an acceptable calibration testing configuration. The licensee clarified that, although the spool piece exit region was 15 inches from the in-situ downstream elbow, the chordal paths terminate upstream of the spool piece's end, meaning that the chordal paths are actually 2.7 diameters upstream of the entrance to the piping bend. The NRC staff evaluated the licensee's assertions and accepts the licensee's arguments regarding the actual location of the CCNPP flow meters because operating experience and experimental testing have shown that installations with pipe bends located closer downstream of the UFM chordal paths have little effect on the UFM's meter factor.

In addition, the licensee indicated that previous UFM calibrations and installations had confirmed that pipe bends in downstream locations closer than the CCNPP in-situ installations had an insignificant effect on the meter factor. Based on the installation location of the CCNPP flow meters, and on experience with other flowmeters installed upstream of piping bends, the NRC staff finds that the licensee's laboratory calibration was sufficiently fabricated to provide meaningful data based on the modeling of piping geometry upstream of the UFM.

3.6.2.5 Nuclear Steam Supply System

The Nuclear Steam Supply System (NSSS) design parameters provide the RCS and secondary system conditions (pressures, temperatures, and flow) that are used as the basis for the design transients and for systems, components, accidents and transient analyses and evaluations. The parameters are established using conservative assumptions to provide bounding conditions to be used in the NSSS analyses.

In all safety analyses except where noted in Table 3.6.1, the assumed initial power level was 102 percent of original RTP. The sections following Table 3.6.1 discuss those safety analyses not performed assuming 102 percent RTP as an initial condition, and explain why the NRC staff found those analyses acceptable.

In all analyses, the licensee referenced the current analysis of record, which used previously NRC-approved computer codes and methodologies for each accident and transient analysis. Unless noted, the analyzed core power level was 2754 MWt, which is 2.0 percent greater than the current RTP of 2700 MWt and 0.62 percent greater than the MUR core power level of 2737 MWt. The NRC staff reviewed and approved the licensee's transient and accident analyses at 2754 MWt conditions assumed for normal operations confirming that the acceptance criteria were still met under these conditions.

Results of the NRC staff's review are summarized in Table 3.6.1 below. In the sections that follow, discussion is provided for those analyses that were not performed at 102 percent of RTP.

Table 3.6.1 Accident and Transient Analyses

Accident/Transient	Analyzed Core Power Level	Analysis of Record Bounds MUR Uprate	NRC Staff Conclusion/Discussion
Control Element Assembly Drop Event	2754	Yes	Acceptable
Asymmetric Steam Generator Event	2754	Yes	Acceptable
Control Element Assembly Ejection	2754	Yes	Acceptable
Steam Line Break Event	2754	Yes	Acceptable
Steam Generator Tube Rupture	2754	Yes	Acceptable
Seized Rotor Event	2754	Yes	Acceptable
Loss of Coolant Accident	2754	Yes	See Section 3.7.1
Fuel Handling Incident	2754	Yes	Acceptable
Turbine-Generator Overspeed Incident	---		See Section 3.7.2
Containment Response	2754	Yes	Acceptable
Hydrogen Accumulation Inside Containment	N/A	N/A	See Section 3.7.3
Waste Gas Incident	2754	Yes	Acceptable
Waste Processing System Incident	2754	Yes	Acceptable
Maximum Hypothetical Accident	2754	Yes	Acceptable
Excessive Charging Event	---		See Section 3.7.4
Feed Line Break Incident	2754	Yes	Acceptable

3.6.2.5.1 Loss-of-Coolant Accident

The licensee's LOCA analyses were performed using NRC-approved methods specifically applicable to Combustion Engineering Nuclear Steam Supply Systems. The licensee confirmed that the current analyses demonstrate compliance with the acceptance criteria set forth in 10 CFR 50.46(b), regarding peak fuel cladding temperature, hydrogen generation, cladding oxidation, long-term core cooling and coolable geometry requirements. The licensee stated that the LOCA analyses assume that the core is operating at 102 percent of RTP, consistent with original requirements established in Paragraph I.A of 10 CFR Part 50, Appendix K.

The licensee noted that the evaluation models used for the LOCA analyses are specific to these requirements of Appendix K in that they require the assumption that the core is operating at 102 percent of RTP. As is the intent of the measurement uncertainty recapture, the licensee is reducing the value of the uncertainty associated with this requirement. Therefore, the licensee is also requesting to implement the NRC-approved LOCA analytic methodology that is currently in place at CCNPP in a manner slightly inconsistent with its description in that the LOCA analyses will assume the same bounding power level, which is representative of a reduction in the feedwater flow uncertainty.

The NRC staff finds this exception acceptable on the basis that the new feedwater flow instrumentation has the demonstrated capability to perform within the bounds of the reduced uncertainty value, such that the LOCA analyses at the power level of 2754 MWt continue to bound operation of the plant at the uprated power level of 2737 MWt.

3.6.2.5.2 Turbine Generator Overspeed Incident

The licensee stated that the turbine-generator overspeed incident is an analyzed event based on the failure of rotating elements of the steam-turbines and generators. The licensee noted that the event is neither a design basis event nor an anticipated operational occurrence. The licensee also stated that the requested thermal power increase does not impact the results of the analysis, and that the analyzed incident bounds operation at the uprated power levels.

The NRC staff reviewed the event, which is described in Section 5.3.1.2 of the CCNPP UFSAR. The event is discussed from the perspective of the generation of turbine missiles. From a reactor safety perspective, this event could possibly cause a decrease in main steam header pressure, resulting in increased flow from the steam generators, or it would be caused by some increase in turbine header pressure that would be indicative of an increase in steam flow from the steam generators. In either case, the increase in secondary steam flow would be bounded by a more limiting transient, the main steam line break.

The NRC staff concludes that a more severe transient, i.e., the main steam line break, would bound the reactor systems consequences of this transient. In addition, the event is not considered as a DBA or anticipated operational occurrence, and on this basis, the staff concludes that it need not be considered for the requested power uprate.

3.6.2.5.3 Hydrogen Accumulation Inside Containment

The licensee stated that this analysis has been deleted from the UFSAR per License Amendment Numbers 262 and 239 for the CCNPP Units 1 and 2 Facility Operating Licenses. On this basis, this event is not part of the facility licensing basis and need not be considered for the requested power uprate.

3.6.2.5.4 Excessive Charging Event

The licensee stated that the analysis of record for the excessive charging event is analyzed to verify compliance with the limits of TS 3.4.4, and to provide the basis for associated alarms and setpoints. The analysis of record verifies that operator action no sooner than 15 minutes following receipt of a pressurizer high level alarm suffices to terminate this event without violating limits on pressurizer level. The licensee also stated that the analysis is based on reactor coolant system volumes and chemical and volume control system flow rates, and is unaffected by reactor power level.

Because this event is largely unaffected by the reactor power level, the NRC staff accepts the licensee's disposition regarding the excessive charging event. Cycle-specific re-confirmation of the 15-minute limit on operator action to terminate this event will demonstrate adequate performance for this transient at the uprated power levels.

3.6.2.6 Reactor Vessel Neutron Fluence

The fuel loading associated with the requested power uprate has the possibility to change the neutron fluence, thus affecting the reactor vessel integrity. The licensee addresses these issues in Section IV.5 of its submittal;. This section evaluates the acceptability of the reactor vessel neutron fluence calculations used to support its reactor vessel integrity conclusions. The reactor vessel integrity is discussed in Section 3.5 of this SE.

The licensee provided supplemental information in its letter dated February 17, 2009, clarifying that fluence calculations used to support the power uprate request were performed using methods and techniques that met the intent, although, "not the specific details," of RG 1.190. The licensee also provided references to the methodologies used to determine the reactor vessel neutron fluence.

The licensee provided sufficient information to determine that the numerical approach used in the fluence calculations was adherent to RG 1.190, but the licensee did not provide information concerning the nuclear data used to calculate the reactor vessel fluence. The NRC staff reviewed the licensee's surveillance capsule dosimetry reports and determined that the nuclear data file supporting the fluence calculations was developed using Version 5 of Brookhaven National Laboratory's Evaluated Nuclear Data File (ENDF/B-V). Particular concerns regarding the use of ENDF/B-V-based nuclear data for neutron transport calculations involving reactor vessel materials are enumerated in RG 1.190; the calculations are generally unacceptable unless justified.

The licensee also stated that new fluence calculations would be performed for CCNPP in concert with its surveillance capsule withdrawal, which includes the extraction of surveillance capsules in 2010 and 2011 for Units 1 and 2, respectively. These fluence calculations would precipitate the calculation of updated reactor vessel integrity-related parameter limits.

The specific concern regarding the use of nuclear data from ENDF/B-V is that differences in the iron cross sections between ENDF/B-V and ENDF/B-VI could result in a 20-percent under prediction of vessel inner wall fluence. Therefore, the licensee multiplied the peak fluence values expected at the time of surveillance capsule withdrawal by a factor of 1.2 (i.e., 20 percent) and compared them to the end-of-extended life fluence values to ensure that the projections for end-of-extended-life fluence values were bounding of the expected fluence values at the time of surveillance capsule withdrawal when multiplied by this safety factor. The NRC staff finds this approach acceptable because the licensee showed that current limiting fluence values contain adequate safety margin to envelope plant conditions until such time as more acceptable fluence calculations can be provided.

Based on the considerations discussed above, the NRC staff finds that the licensee's fluence projections are acceptable. In summary, the licensee's calculations are nearly adherent to all guidance contained in RG 1.190. The only exception is acceptable because the effects of use of an inadequate nuclear data file are bounded with sufficient safety margin, as discussed above.

3.6.3 Conclusion

The NRC staff reviewed the reactor systems and thermal-hydraulic aspects of the proposed LAR in support of implementation of a measurement uncertainty recapture power uprate. Based on the considerations discussed above, the staff determined that the results of the licensee's analyses related to these areas continue to meet applicable acceptance criteria following implementation of the MUR. Most of the current analyses of record are based on operation at 2754 MWt, which includes 2.0 percent measurement uncertainty. The proposed amendment is based on the use of a Caldon LEFM Check Plus system that would decrease the uncertainty in the feedwater flow, thereby decreasing the power level measurement uncertainty from 2.0 percent to 0.62 percent. In these cases, the proposed MUR rated thermal power of 2737 MWt is bounded by the current analyses of record.

3.7 Reactor Vessel and Internals Integrity

The NRC staff's review in the area of RV and RV internals integrity focuses on the impact of the proposed MUR power uprate on pressurized thermal shock (PTS) calculations, heatup and cooldown pressure-temperature (P-T) limit curves, low-temperature overpressure protection (LTOP) system setpoints and enable temperature, upper shelf energy (USE), RV surveillance capsule withdrawal schedules, and the integrity of the pressurizer shell and RV internals. This review is conducted to verify that the results of licensee analyses related to these areas continue to meet the requirements of 10 CFR 50.55a, 10 CFR 50.60, 10 CFR 50.61, and Appendices G and H to 10 CFR Part 50, following implementation of the proposed MUR power uprate.

3.7.1 Pressurized Thermal Shock

3.7.1.1 Regulatory Evaluation

The NRC staff's requirements, methods of evaluation, and safety criteria for PTS assessments are provided in 10 CFR 50.61. 10 CFR 50.61 requires that all RV beltline materials maintain pressurized thermal shock temperature (RT_{PTS}) values below the screening criteria provided in the regulation. Specifically, RV axial welds, plates, and forgings must maintain RT_{PTS} values below 270 °F, and RV circumferential welds must maintain RT_{PTS} values below 300 °F.

3.7.1.2 Technical Evaluation

The licensee provided the RT_{PTS} values for the limiting beltline material of the CCNPP, Unit Nos. 1 and 2 RVs in Section IV.5 of Attachment 2 of the LAR dated August 29, 2008, and concluded:

> The highest RT_{PTS} value for Calvert Cliffs Unit 1 at the end of the extended license was determined to be 255 °F which is associated with the RV lower shell course axial weld seams. This is based on a projected fluence of 5.11×10^{19} n/cm^2, E>1MeV. The highest RT_{PTS} value for Calvert Cliffs Unit 2 at the end of the extended license was determined to be 199 °F which is associated with the RV lower shell course plate D8906-1. This is based on a projected fluence of 5.79×10^{19} n/cm^2, E>1MeV. In both cases the projected value of RT_{PTS} is less than the

> pressurized thermal shock screening criterion of 270 °F such that the planned uprate does not result in exceeding the screening criterion.

The NRC staff's review covered the PTS methodology and the calculation of the reference temperature of RV beltline materials for pressurized thermal shock (RT_{PTS}) at the expiration of the renewed operating license, considering neutron embrittlement effects. The staff independently verified the RT_{PTS} calculations with the known RV material properties from the NRC's Reactor Vessel Integrity Database (RVID) and the uprated fluence projections from the LAR and found the results to be consistent with the licensee's evaluation. As these results are within the acceptable limits given in 10 CFR 50.61, the staff confirmed that the CCNPP, Units Nos. 1 and 2 RV materials will continue to meet the PTS screening criteria of 10 CFR 50.61 under MUR conditions.

3.7.2 P-T Limits and LTOP

3.7.2.1 Regulatory Evaluation

Under 10 CFR Part 50, Appendix G, limits on P-T during core critical, core non-critical, and system hydrostatic testing operation to provide adequate fracture toughness of ferritic (low alloy or carbon steel) materials in the RV are prescribed. These 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. 10 CFR Part 50, Appendix G requires the facility P-T limit shall be at least as conservative as those established by following the procedures of ASME Code, Section XI, Appendix G and the minimum temperature requirements in the rule.

3.7.2.2 Technical Evaluation

The licensee stated in IV.5 of Enclosure 1 to its August 29, 2008, submittal that:

> There are no significant changes to the values used to establish the Appendix G normal operating limits. The 0.05×10^{19} n/cm^2 [E > 1.0 MeV] increase in fluence results in less than 0.3 °F change to the adjusted reference temperature at the one-quarter thickness location. The low temperature overpressure protection limits for the MUR power uprate conditions are unchanged for those same reasons.

The NRC staff confirmed that the existing CCNPP, Unit Nos. 1 and 2 P-T limit curves and LTOP system setpoints were reviewed and approved by the NRC by letter dated December 9, 2003. The staff found that the limiting material adjusted reference temperature (ART) value upon which the current CCNPP, Unit Nos. 1 and 2 P-T limits are based are 254 °F and 198 °F, respectively (Unit 1 3-203 A/B/C welds and Unit 2 D-8906-1 plate). The staff confirmed that the licensee's stated increase in neutron fluence for this material would result in an increase in ART of 0.3 °F. An ART increase of this magnitude has a negligible impact on the facility's P-T limit curves and LTOP system setpoints. Therefore, the staff concludes, that the existing CCNPP, Unit Nos. 1 and 2 P-T limit curves and LTOP system setpoints remain valid and in accordance with 10 CFR Part 50, Appendix G after consideration of the effects of the MUR.

3.7.3 Upper Shelf Energy (USE)

3.7.3.1 Regulatory Evaluation

10 CFR Part 50, Appendix G provides fracture toughness requirements for ferritic (low alloy steel or carbon steel) materials in the RCPB, including requirements on the USE values used for assessing the safety margins of the RV materials against ductile tearing. Specifically, 10 CFR Part 50, Appendix G requires that RV beltline materials must have a Charpy USE of no less than 50 ft-lb throughout the licensed operating period to ensure adequate safety margins of the RV materials against ductile tearing.

3.7.3.2 Technical Evaluation

The licensee stated in Section IV.5 of Attachment 2 of the August 29, 2008, LAR:

> For Calvert Cliffs, Units 1 and 2, the upper shelf energy values at the end of the current license were determined to range from 52 ft-lb to 85 ft-lb for the RV beltline plates and welds. This demonstrates that all the beltline materials will exceed the upper shelf energy screening criteria.

The NRC staff's review of the USE assessment covered the impact of the MUR power uprate on the neutron fluence values for the RV beltline materials and the USE values for the RV beltline materials through the end of the current licensed operating period. The staff independently verified the licensee calculations of USE values to verify that no beltline RV materials will exceed the regulatory limits. The staff calculations were performed using the methods of RG 1.99 Rev. 2, "Radiation Embrittlement of Reactor Vessel Materials," known RV material properties from the RVID, and the projected uprated neutron fluence values provided by the licensee. The staff-calculated uprated USE values are all greater than 50 ft-lbs with a limiting USE value of 51 ft-lb (intermediate shell plate D-8906-3, heat ID A-4463-2). Therefore, the staff finds that the RV beltline materials will continue to satisfy the requirements of 10 CFR Part 50, Appendix G after consideration of the effects of the MUR.

3.7.4 Surveillance Capsule Withdrawal Schedule

3.7.4.1 Regulatory Evaluation

The regulatory requirements of implementing the RV material surveillance program are given in 10 CFR Part 50, Appendix H by reference to American Society for Testing and Materials (ASTM) Standard Practice E185, "Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels." 10 CFR Part 50, Appendix H requires that licensees establish a RV material surveillance program which meets the requirements of the edition of ASTM E185 which was current on the issue date of the ASME Code to which the RV was purchased. Later editions of the ASTM E1985 may also be used up to the 1982 Edition (E185-82). The version of ASTM E185 that was current on the purchase date of CCNPP RVs was the 1970 edition (ASTM E185-70). The licensee stated that their RV surveillance program complies with the requirements of ASTM E185-82.

3.7.4.2 Technical Evaluation

As noted in the licensee's August 29, 2008 LAR, the first two RV surveillance capsules have already been withdrawn from each CCNPP unit. In a separate submittal dated July 29, 2008 (ADAMS Accession No. ML082110562), and approved by the NRC on February 3, 2009 (ADAMS Accession No. ML090270206), the licensee established the current RV surveillance capsule withdrawal schedule for each unit. This withdrawal schedule is shown in Table 1 with uprated neutron fluence values added:

CCNPP, Unit 1					
Capsule Azimuthal Position (degrees)	Target Fast Neutron Fluence (xE19 n/cm2)		Projected Fast Neutron Fluence (xE19 n/cm2)		
	Reference Submitted Values	Values with MUR Power Uprate Included	Reference Submitted Values	Values with MUR Power Uprate Included	Projected End of Cycle Date
263	0.62	N/A	0.62	N/A	Removed – 1979
97	2.64	N/A	2.64	N/A	Removed 1992
104	3.06	3.08	3.12	3.12	2010
83	5.26	5.28	5.33	5.34	2020
277	6.59	6.62	6.59	6.62	2032
284	Standby	Standby	Standby	Standby	
CCNPP, Unit 2					
Capsule Azimuthal Position (degrees)	Target Fast Neutron Fluence (xE19 n/cm2)		Projected Fast Neutron Fluence (xE19 n/cm2)		
	Reference Submitted Values	Values with MUR Power Uprate Included	Reference Submitted Values	Values with MUR Power Uprate Included	Projected End of Cycle Date
263	0.806	N/A	0.806	N/A	Removed – 1982
97	1.85	N/A	1.85	N/A	Removed 1993
104	3.24	3.27	3.23	3.23	2011
83	6.16	6.21	6.30	6.64	2025
277	7.46	7.50	7.46	7.50	2033
284	Standby	Standby	Standby	Standby	

Table 1. Updated Capsule Withdrawal Schedule

The licensee stated that:

> The vessel fluence is predicted to increase only 0.04×10^{19} n/cm^2, E>1MeV, as a result of the planned uprate. Therefore, the updated surveillance capsule

withdrawal schedule is also applicable under conditions including the MUR power uprate.

The NRC staff has reviewed the impact of the licensee's stated increase in neutron fluence as a result of the MUR on the units' RV surveillance program. The staff has concluded that the program given in Table 1, as approved by the NRC on February 3, 2009 (ADAMS Accession No. ML090270206), continues to meet the requirements of ASTM E185-82 for CCNPP, Unit Nos. 1 and 2. Therefore, the licensee's current RV surveillance program will continue to meet the requirements of 10 CFR Part 50, Appendix H after consideration of the effects of the MUR.

3.7.5 Pressure Vessel Shell and Reactor Vessel Internals Materials Aging and Degradation Issues

3.7.5.1 Regulatory Evaluation

Pressurizer Surge Line Thermal Stratification

NRC Bulletin 88-11, "Pressurizer Surge Line Thermal Stratification," addresses the issue of pressurizer surge line thermal stratification. Differences in temperature between the hot leg and the pressurizer may cause thermal stratification in the surge line. Stratification occurs due to a temperature gradient within the surge line that may cause differential thermal expansion that can cause the pipe to deflect significantly. This deflection may cause high piping stress that exceeds design limits for fatigue and stresses. If this effect occurs where a pipe is restrained, the induced stresses may be even higher resulting in high local stresses, low cycle fatigue and functional impairment of the line. NRC Bulletin 88-11 requires licensees to demonstrate that the pressurizer meets applicable design codes for the licensed life of the plant considering the phenomenon of thermal stratification and thermal striping in the fatigue and stress evaluations and to update their analyses based on plant-specific data.

RV Internals and Supports

The NRC's acceptance criteria for RV internals and core support materials are based on 10 CFR Part 50, Appendix A, GDC 1 and 10 CFR 50.55a for material specifications, controls on welding, and inspection of RV internals and core supports. Table Matrix-1 of NRC RS-001, Revision 0, "Review Standard for Extended Power Uprates," (hereafter Matrix-1) (ADAMS Accession No. ML033640024) provides the staff's basis for evaluating the potential for power uprates to induce aging effects on RV internals. The industry has established, among other aging issue guidelines, the neutron irradiation-related thresholds for irradiation-assisted stress-corrosion cracking (IASCC) for various PWR RV internal components. Compliance with industry criteria and guidelines on this issue, when they are approved by the NRC, fulfill the requirements of RS-001.

3.7.5.2 Technical Evaluation

Pressurizer Thermal Stratification

The licensee reviewed the parameters associated with the MUR uprate that may affect pressurizer surge line piping including the effects of thermal stratification. In its LAR of August 29, 2008, the licensee stated:

> T_{hot} has increased slightly for the MUR power uprate [0.38%]. This change has a negligible effect on the stratification AOR [analysis of record], since it only results in a slight reduction in the delta-T between the pressurizer and the hot leg during steady-state normal operation. Therefore, the stratification temperature ranges developed in the AOR bound the new operating conditions.

The NRC staff reviewed the requirements and concerns, detailed in NRC Bulletin 88-11 relevant to thermal stratification, and determined that the change in T_{hot} is minor and would not result in a significant increase in thermal stratification and/or fatigue damage in the pressurizer surge line. Therefore, the staff concludes that the licensee has adequately addressed thermal stratification concerns.

RV Internals and Supports

The licensee discussed the aging and degradation of RV internals in its LAR supplement dated December 29, 2008, and included the following:

> Materials Reliability Program-227, Revision 0, PWR Internals Inspection and Evaluation Guidelines, is scheduled for approval December 2008 and will provide guidelines for the development of a RV internals inspection plan. An industry Material Degradation Management Program (Nuclear Energy Institute 03-08) supported requirement to develop a RV Internals Inspection Program Plan, meeting the MRP-227 guidelines is also expected. Guidelines from these industry documents will be incorporated into the development of Calvert Cliffs RV internal inspection program. The development of the RV internal inspection program is included as part of Calvert Cliffs license renewal implementation program plans.

The information provided by the licensee regarding their development of a RV internals aging management program in accordance with industry guidelines, consistent with the provisions under which CCNPP Unit Nos. 1 and 2 were issued a renewed operating license on March 3, 2000, is consistent with Matrix-1 of RS-001. Therefore, the NRC staff concludes that the licensee's program to address aging of RV internals at CCNPP, Unit Nos. 1 and 2 is consistent with NRC staff guidance and, therefore, is acceptable.

3.7.6 Conclusion

The NRC staff has reviewed the licensee's evaluation of the effects of the proposed MUR power uprate on the RV and RV internals integrity and the susceptibility of reactor internal and core support materials to known degradation mechanisms. The staff concludes that the licensee has addressed PTS calculations, P-T limit curves, LTOP system setpoints, USE, RV surveillance

capsule withdrawal schedules, thermal stratification in the pressurizer surge line, and RV internals in this context. The staff concludes that the licensee has established that adequate safety margins and procedures are in place as required. Therefore, the staff has determined that the changes identified in the above areas with respect to the MUR power uprate are acceptable.

3.8 Electrical Systems

3.8.1 Regulatory Evaluation

GDC 17, "Electric power systems," of 10 CFR Part 50, Appendix A, 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.

10 CFR 50.63 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.

10 CFR 50.49, "Environmental qualification [EQ] of electric equipment important to safety for nuclear power plants," requires licensees to establish programs to qualify electric equipment important to safety.

3.8.2 Technical Evaluation

3.8.2.1 AC Distribution System

The AC distribution system at CCNPP is the source of power to plant auxiliaries during normal operations and accident conditions. The AC distribution system consists of the 13.8 kilo volt (kV), 4160 volt (V), 480 V, and 120/208 V systems. The onsite power distribution loads were reviewed under normal and emergency operating scenarios. At the uprated power level, the loads are expected to operate within their design ratings.

At the uprated power level, the non-Class 1E loads (condensate pumps, condensate booster pumps, and heater drain pumps) will experience increased flow and pressure but will remain bounded by the existing design ratings of the non-Class 1E equipment. The licensee stated that the electrical motors and supporting equipment are sized for maximum accident load requirements and no changes are required for Class 1E 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.

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

As a result of the power uprate, CCNPP rated thermal power will increase to 2737 MWt from the previously analyzed core power level of 2700 MWt. The Unit 1 generator is rated at 1020 megavolt ampere (MVA) at 25 kV with a 0.9 lagging power factor (pf). The Unit 2 generator is

rated at 1012 MVA at 22 kV with a 0.9 pf. Currently, the Unit 1 generator outputs 896 MWe, and the Unit 2 generator produces 885 MWe. At uprated conditions, CCNPP Unit 1 will produce 908 MWe with a gross reactive power capability of 367 MVAR lagging and -50 MVAR leading. At uprated conditions, Unit 2 will produce 897 MWe, with a gross reactive power capability of 350 MVAR lagging and -50 MVAR leading. The increase in electrical output (12 MWe) remains bounded by the design ratings of the generator. The licensee stated in the LAR that the new operating points of the generators are within the design rating of both machines. Based on this information, the NRC staff finds that the generator is capable of operation at uprated conditions.

Since the main generators will continue to operate within the existing ratings 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 plant service transformers serve as the plant auxiliary sources of power and are fed from separate 500 kV switchyard busses and a 13 kV line. The ratings of the plant service transformers are 500 kV/14 kV, 3 phase, 60 Hertz, 100 MVA, as stated in the licensee's December 29, 2008, letter. The licensee further stated that the transformers and associated 13 kV and 4 kV electrical systems are designed such that the entire service load from both units can be aligned through one plant service transformer. At uprated conditions, the maximum calculated load is expected to increase from 96.7 MVA to 96.87 MVA, which is within the design rating of the plant service transformer of 100 MVA.

The main generator voltage is stepped up to 500 kV by the main power transformers (two per unit), which are designed to carry the maximum generator output. Each of the paralleled transformers is rated for 810 MVA at 65 °C rise. At uprated conditions, the maximum generator output is 1020 MVA for Unit 1 and 1012 MVA for Unit 2, which is below the rating of the paralleled transformers (1620 MVA). Therefore, the NRC staff finds that the main power transformers are capable of operation at uprated conditions.

The small increase in generator output (12 MWe) does not cause overloading of the generators, iso-phase bus ducts or the transformers. As a result, the generators, iso-phase bus ducts and transformers are still within their design ratings. Therefore, the NRC staff concludes that the ratings of the CCNP transformers would not be impacted by MUR power uprate conditions.

3.8.2.3 DC System

The station 125 V DC systems (four channels) are each comprised of one battery, two battery chargers, and distribution equipment that supply DC power for plant vital instrumentation and control systems. The safety-related function of the DC onsite power system is to provide reliable continuous 125 V DC power to the plant protection system and other loads for safe operation of the reactor. Additionally, a reserve 125 V DC system, consisting of one battery, one battery charger, and associated DC switching equipment, serves to replace any one of the 125 V DC batteries.

The licensee stated that the DC distribution system will experience minor load variations due to the power uprate; however, the resulting electrical loads remain within the ratings of the existing distribution system. The NRC staff reviewed the LAR and UFSAR and confirmed that the power

uprate does not impact DC system loads. Therefore, the staff finds that the analyses for DC system bound MUR power uprate conditions.

3.8.2.4 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. Each EDG (two per unit) is dedicated to one of the 4160 V engineered safety feature (ESF) buses (Class IE), which supply power to critical loads required during abnormal operational transients and accidents.

The UFSAR states that the loading of the EDGs for a large-break LOCA, small-break LOCA and main steam line break is less than 3000 kW. According to the licensee, there are no load changes to the safety-related buses and EDGs due to the MUR, and thus, the existing accident analyses remain bounding. Hence, the EDG system has adequate capacity and capability to power the safety-related loads at MUR power uprate conditions.

Based on the above, after reviewing the LAR and UFSAR, the NRC staff finds that the power uprate does not impact EDG system loads. Therefore, the staff finds that the analyses for the EDG system bound MUR power uprate conditions, and the onsite power system will continue to meet the requirements of GDC 17.

3.8.2.5 Switchyard

The switchyard equipment and associated components are classified as non-safety related. The primary function of the 500 kV switchyard and distribution system is to distribute the generated power to the transmission grid. In addition, the switchyard provides the required AC power for station startup and shutdown. The 500 kV switchyard has three incoming transmission lines. In addition, a 69 kV transmission line (SMECO transmission line) can provide AC power to maintain Units 1 and 2 in safe shutdown. The 500 kV switchyard supplies the onsite distribution system through the plant service transformers while the SMECO line supplies power to the 13 kV Bus 11 or 21 as required.

The NRC staff confirmed that the small increase in plant output will not significantly impact the switchyard equipment. Therefore, the staff concludes 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.

3.8.2.6 Grid Stability

The impact on grid stability is discussed in Section V.4 of Attachment 2 of the LAR and the licensee concludes that there is no significant effect on grid stability or reliability. The NRC 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 December 29, 2008, supplemental letter, the licensee stated that an interim impact study and final impact study was performed by PJM Interconnection to determine the impacts of the expected increase in MWe. The final impact study in Enclosure 2

of letter dated December 29, 2008, indicates that the study evaluated an increase of 55 MWe to both units and evaluated for compliance with reliability criteria for summer peak conditions in 2009. Thus, the impact study bounds the increase in 12 MWe of each CCNPP unit from the power uprate.

Attachment 1 of Enclosure 1 of the licensee's supplemental letter dated December 29, 2008, provides the fault cases evaluated. The range of contingencies evaluated were those necessary to demonstrate compliance with the associated reliability criteria. Enclosure 4 of this letter is the PJM Manual 14B: PJM Region Transmission Planning Process, which describes the stability analysis and states that PJM ensures generator and system stability. Generator stability is evaluated for critical system conditions, which includes both light load and three phase faults with normal clearing and single line to ground faults with delayed clearing. This included evaluating the impact of transmission line outages, the loss of CCNPP, as well as the loss of other generating units, as stated in Enclosure 3 of the letter dated December 29, 2008. The final impact study concluded that there were no problems identified in the stability analysis or for multiple facility contingencies (tower line outages).

In its supplemental letter dated December 29, 2008, the licensee stated that the maximum MVAR capability is within the generator's ratings. Furthermore, the final impact study indicates that Unit 1 can receive a maximum increase of 35 MWe if the reactive capability is maintained at 367 MVAR and Unit 2 can receive a 20 MWe increase. 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 CCNPP MUR power uprate allows for continued stable and reliable grid operation.

3.8.2.7 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.

CCNPP's SBO coping duration is 4 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 RG 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.

CCNPP has a SBO diesel generator which is designed to provide sufficient power to any of the Class 1E busses to safely shutdown one unit and maintain it in a safe shutdown condition during a SBO event. In its LAR, the licensee stated that the MUR has no impact on the existing SBO analyses. As such, the SBO diesel generator will continue to adequately supply the necessary equipment to mitigate an SBO event. The NRC concurs with the licensee's analysis and finds that the CCNPP will continue to meet the requirements of 10 CFR 50.63 under power uprate conditions.

3.8.2.8 Equipment Qualification (EQ) Program

In its LAR, the licensee stated that the EQ of electrical equipment was performed at a core power level of ≥ 102 percent of 2700 MWt, which bounds the MUR operating conditions (1.38 percent increase). The MUR power uprate causes the system operating temperatures and pressures to change slightly. In its LAR, the licensee stated that the current analyses for containment LOCA and MSLB temperature and pressure will remain bounding for the power uprate. Although the radiation levels may increase slightly due to the power uprate, the licensee stated that environmental qualification of equipment was reevaluated against revised accident radiation doses and confirmed to remain environmentally qualified. The licensee evaluated the equipment in the Containment and Auxiliary Buildings and determined that the heating, ventilation, and air conditioning (HVAC) systems will continue to operate within their design ranges in these buildings. 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 concurs that the current EQ parameters remain bounding for the MUR power uprate. Therefore, the staff finds that the MUR power uprate will have no impact on CCNPP's EQ Program and continue to meet the requirements of 10 CFR 50.49.

3.8.3 Overall Electrical Systems

Based on the technical evaluation provided above, the NRC staff finds that CCNPP will continue to meet GDC 17, 10 CFR 50.63, and 10 CFR 50.49.

3.9 Instrumentation & Controls

3.9.1 Regulatory Evaluation

Topical Report ER-80P and its supplement, Topical Report ER-157P, describe the Caldon LEFM CheckPlus System for the measurement of feedwater flow and provide a basis for the proposed 1.38 percent MUR uprate of the licensed reactor thermal power. The NRC staff also considered guidance of Regulatory Issue Summary (RIS) 2002-03, "Guidance on the Content of Measurement Uncertainty Recapture Power Uprate Applications," in its review of the licensee's submittals for the proposed power uprate request.

3.9.2 LEFM Technology and Measurement

Both the Caldon LEFM Check and LEFM CheckPlus systems use transit time methodology to measure fluid velocity. The basis of the transit time methodology for measuring fluid velocity and temperature is that ultrasonic pulses transmitted through 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 pulse 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 systems use multiple diagonal acoustic paths, instead of a single diagonal path, allowing velocities measured along each path to be numerically integrated over the pipe cross-section to determine the average fluid velocity in the pipe. This 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 Caldon LEFM Check system consists of a spool piece with eight transducers, two on each of the four acoustic paths in a single plane of the spool piece. The velocity measured by any one of the four acoustic paths is the vector sum of the axial and the transverse components of fluid velocity as projected onto the path. The Caldon LEFM CheckPlus system uses 16 transducers, 8 each in two orthogonal planes of the spool piece. In the Caldon LEFM CheckPlus system, when the fluid velocity measured by an acoustic path in one plane is averaged with the fluid velocity measured by its companion path in the second plane, the transverse components of the two velocities are canceled and the result reflects only the axial velocity of the fluid. This makes the numerical integration of four pairs of averaged axial velocities and computation of volumetric flow inherently more accurate than a result obtained using four acoustic paths in a single plane. Also, since there are twice as many acoustic paths and there are two independent clocks to measure the transit times, errors associated with uncertainties in path length and transit time measurements are reduced.

The NRC staff's review in the area of instrumentation and control 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 through H) provided in Section I of Attachment 1 to RIS 2002-03. The staff conducted its review to confirm that (1) the licensee's implementation of the proposed feedwater flow measurement device is consistent with the staff-approved Topical Reports ER-80P and ER-157P and (2) the licensee adequately addressed the four additional requirements listed in the staff's SE. The staff also reviewed the power measurement uncertainty calculations to ensure that (1) the conservatively proposed uncertainty value of 0.62 percent correctly accounted for all uncertainties associated with power level instrumentation errors and (2) the uncertainty calculations met the relevant requirements of Appendix K to 10 CFR Part 50, as described in Section 2 of this SE.

The Caldon LEFM CheckPlus system at CCNPP includes an electronic cabinet in the turbine building and two spool piece measurement sections per unit with one spool piece installed in the 16-inch feedwater header for each steam generator. Each spool piece consists of 16 transducers, arranged in two planes with four pairs of transducers in each plane. The transducers are located in wells; thus, a transducer may be removed at power without disturbing the pressure boundary of the spool piece. The installation of the LEFM CheckPlus system will conform to the requirements of Cameron Topical Reports ER-80P and ER-157P. In addition, the licensee will install the measurement sections in accordance with approved CCNPP procedures and work control processes to achieve installation tolerances within the bounds stated in the Cameron uncertainty analysis as described in Cameron Engineering Report ER-507.

The LEFM flow meters were calibrated at the Alden Laboratories using the site-specific model with all calibration standards traceable to National Institute of Standards and Technology standards. The calibrated data provide a quantitative measure of the Caldon LEFM CheckPlus

meter factor versus the actual velocity profile encountered and determines the meter uncertainty to be used in the overall calorimetric uncertainty.

The licensee will install two feedwater pressure transmitters (Model: Rosemount 3051CG5) in each feedwater header in the vicinity of the spool pieces. The pressure transmitters provide input of feedwater pressure to the electronic unit for the calculation of feedwater mass flow. In response to the NRC staff's request regarding the total loop uncertainty calculation of feedwater pressure transmitter, the licensee provided Baltimore Gas & Electric Company (BG&E) calculation CA07018, Revision 1, "Main Feedwater Pressure Input Uncertainty to Caldon CheckPlus LEFM for Calvert Cliffs Nuclear Power Plant Units 1 & 2," dated April 15, 2009. The staff reviewed this document and concluded that the total loop uncertainty for the feedwater pressure transmitters on the Caldon LEFM CheckPlus system and the plant computer are bounded within the assumption limit for the total power measurement uncertainty calculation. Therefore, the total loop uncertainty calculation for these feedwater pressure transmitters is acceptable.

The Caldon LEFM CheckPlus system first determines feedwater mass flow and feedwater temperature and then combines feedwater pressure for continuous calculation of secondary calorimetric power. The measured feedwater parameters are communicated to the plant computer and data acquisition system (DAS) over the plant data network for use in the calorimetric power algorithm.

Each Caldon LEFM CheckPlus system has self-verification features to ensure that the system continues to operate within the design-basis uncertainty analysis. Diagnostic and signal quality data are communicated to the DAS to allow monitoring of degradation of the Caldon LEFM CheckPlus system. The system triggers control room annunciation by means of the plant computer when conditions reach a state that could impact the flow measurement uncertainty.

3.9.3 LAR Compliance to RIS 2002-03, Attachment 1, Section I Guidance A through H

Items A through C

Items A, B, and C in Section I of Attachment 1 to RIS 2002-03, respectively, guide licensees to identify the approved topical reports, provide references to the NRC's approval of the measurement technique, and provide a discussion of the plant-specific implementation of the guidelines in the topical report and the NRC staff's approval of the feedwater flow measurement technique.

In this LAR, the licensee identified Topical Reports ER-80P and ER-157P as applicable to the Caldon LEFM CheckPlus system. The licensee also referenced NRC SEs for Topical Reports ER-80P, dated March 8,1999, and ER-157P, dated December 20, 2001. The licensee stated that it will install the Caldon LEFM CheckPlus system at CCNPP in accordance with the requirements of Topical Reports ER-80P and ER-157P.

Based on its review of the licensee's submittals as reflected in the above discussion, the NRC staff finds that the licensee has sufficiently addressed the plant-specific implementation of the Caldon LEFM CheckPlus system using proper topical report guidelines. Therefore, the licensee's description of the feedwater flow measurement technique and implementation of the

power uprate using this technique meets the guidance in Items A through C of Section I of Attachment 1 to RIS 2002-03.

Item D

Item D in Section I of Attachment 1 to RIS 2002-03 guides licensees to provide dispositions of the four criteria that the NRC stated should be addressed when implementing the feedwater flow measurement uncertainty technique.

The NRC staff SE on Topical Report ER-80P included four additional criteria to be addressed by a licensee referencing this topical report for power uprate. The licensee's submittal addressed each of the four criteria as follows:

Criterion 1

> Discuss maintenance and calibration procedures that will be implemented with the incorporation of the LEFM, including processes and contingencies for inoperable LEFM instrumentation and the effect on thermal power measurements and plant operation.

The licensee will incorporate the vendor's maintenance and calibration requirements into plant procedures prior to declaring the Caldon LEFM CheckPlus system operable and raising the power level above 2700 MWt. Site procedures will be amended using Caldon LEFM CheckPlus technical manuals and work instructions.

Calibration and maintenance activities will be performed by qualified Calvert Cliffs maintenance personnel that have participated in formal training on system operation and maintenance. Training will be conducted in accordance with approved site procedures by qualified Calvert Cliffs personnel. The system will not be declared operable until all necessary personnel training is completed.

The MUR power uprate is based on the use of the Caldon LEFM CheckPlus flowmeter. The NRC staff finds that it is appropriate to derate to the current license power level (2700 MWt) if the Caldon LEFM CheckPlus flowmeter is out-of-service beyond the allowed outage time. After the allowed outage time has expired and 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.

Criterion 2

> For plants that currently have LEFMs installed, provide an evaluation of the operational and maintenance history of the installed instrumentation and confirmation that the installed instrumentation is representative of the LEFM system and bounds the analyses and assumptions set forth in Reference 5.

The licensee stated that the LEFM CheckPlus System is not currently installed at CCNPP. Therefore, the NRC staff concludes that Criterion 2 is not applicable.

Criterion 3

> Confirm that the methodology used to calculate the uncertainty of the LEFM in comparison to the current feedwater instrumentation is based on accepted plant setpoint methodology (with regard to the development of instrument uncertainty). If an alternative approach is used, the application should be justified and applied to both venturi and ultrasonic flow measurement instrumentation installations for comparison.

Calvert Cliffs uses a methodology consistent with the approved Topical Reports ER-80P and ER-157P to calculate the uncertainty of the Caldon LEFM CheckPlus system. Uncertainties for the parameters that are not statistically independent are arithmetically summed to produce groups that are independent of each other, which can be statistically combined. Then, all independent parameters/groups that contribute to the power measurement uncertainty are combined using a square root of sum of squares approach. Finally, systematic biases are added to the result to determine the overall power measurement uncertainty.

Criterion 4

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

The licensee plans to have the Caldon LEFM CheckPlus system calibrated at Alden Laboratories using site-specific piping configurations. The testing will encompass a wide range of hydraulic test conditions intended to encompass the expected hydraulic conditions of installation location. The testing will be observed by Calvert Cliffs personnel. The results of the test will verify that Calvert Cliffs has conservatively determined the calorimetric uncertainty when using the Caldon LEFM CheckPlus flow measurement system.

Final acceptance of the Calvert Cliffs specific uncertainty analysis will occur after the completion of the commissioning process. The commissioning process verifies that in-situ test data is bounded by the calibration test data. This step provides the final positive confirmation that actual performance in the field meets the uncertainty bounds established for the instrumentation. Final commissioning is expected to be completed following the Unit 2 refueling outage in the summer 2009 and the Unit 1 refueling outage in spring 2010.

Based on its review of the licensee's responses, the NRC staff determined that the licensee has addressed the four criteria specified in the staff's evaluation of Topical Reports ER-80P and ER-175P, provided the calibration at Alden Laboratories is satisfactory, and it is consistent with the guidelines of RIS 2002-03.

Item E

Item E in Section I of Attachment 1 to RIS 2002-03 provides guidance to submit a plant-specific total power measurement uncertainty calculation, explicitly identifying all parameters and their individual contribution to the power uncertainty.

To address Item E of RIS 2002-03, the licensee provided two documents: (1) Cameron Engineering Report ER-507, which provides the analysis of the uncertainty contribution of the Caldon LEFM CheckPlus system to the overall thermal power uncertainty of CCNPP, and (2) BG&E Calculation CA06945, which derives the uncertainties of secondary calorimetric calculation in both normal mode and degraded mode by using the Caldon LEFM CheckPlus system. The licensee confirmed that the uncertainties in the as-built dimensions lie within the bounding values used in the bounding analysis. The NRC staff audited Engineering Report ER-507 and Calculation CA06945 and found that the licensee identified all parameters contributing to the thermal power measurement uncertainty, determined individual measurement uncertainties, and provided the overall thermal power uncertainty calculation.

Using site standards, the licensee arithmetically summed uncertainties for parameters that are not statistically independent, and statistically combined with other parameters. The licensee combined random uncertainties using the square root sum of squares approach and added systematic biases to the result to determine the overall uncertainty. This methodology is consistent with the vendor determination of Caldon LEFM CheckPlus system uncertainty, as described in the topical reports, and is consistent with the guidelines in RG 1.105.

The NRC staff finds that the licensee has provided calculations of the total power measurement uncertainty at the plant, explicitly identifying all parameters and their individual contribution to the power uncertainty. Therefore, the staff concludes that the licensee has adequately addressed guidance in Item E of Section I of Attachment 1 to RIS 2002-03. The licensee committed to confirm that the time measurement uncertainties are within the bounding values used in the analysis during the commissioning test following the installation of the Caldon LEFM CheckPlus system at the plant.

Item F

Item F in Section I of Attachment 1 to RIS 2002-03 provides guidance to licensees to address the specified aspects of the calibration and maintenance procedures related to all instruments that affect the power calorimetric. In the LAR, the licensee addressed each of the five aspects of the calibration and maintenance procedures listed in Item F of RIS 2002-03 related to instruments that affect the power calorimetric as follows:

1. Maintaining Calibration

This aspect is discussed under item D, Criterion 1 above.

2. Controlling Hardware and Software Configuration

The Caldon LEFM CheckPlus system is designed and manufactured in accordance with the vendor's quality assurance program, which meets the requirements of Appendix B, "Quality

Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to 10 CFR Part 50, and its verification and validation program. The licensee committed that, after installation, software and hardware configuration will be controlled in accordance with site procedures and processes for software configuration control. The licensee will evaluate proposed changes to the software and hardware configuration for all components that provide input to the calorimetric calculation in accordance with the approved engineering change process.

3. Performing Corrective Actions

Licensee engineering personnel will monitor the reliability of the Caldon LEFM CheckPlus system and other calorimetric instrumentation. The licensee documents and resolves adverse performance trends, failed preventive maintenance, or other observed equipment deficiencies in accordance with the site's corrective action process. Qualified licensee maintenance personnel will perform any needed corrective maintenance.

4. Reporting Deficiencies to the Manufacturer

Corrective action procedures include instructions for notification of deficiencies and error reporting. The licensee notifies equipment manufacturers as required to correct the deficiency.

5. Receiving and Addressing Manufacturer Deficiency Reports

The licensee reviews and dispositions manufacturer deficiency reports in accordance with the site's corrective action program. In addition, site personnel will review incoming operating experience from the Institute of Nuclear Power Operations for its applicability to CCNPP. Plant personnel document those deficiencies applicable to CCNPP under the corrective action process.

The NRC staff's review of the above statements conclude that the licensee addressed the calibration and maintenance aspects of the Caldon LEFM CheckPlus system and all other instruments affecting power calorimetric. Thus, the staff concludes that the licensee meets the guidance included in Item F of Section I of Attachment 1 to RIS 2002-03.

Items G and H

Items G and H in Section I of Attachment 1 to RIS 2002-03 guide licensees to provide a proposed allowed outage time (AOT) for the instrument and to propose actions to reduce power if the AOT is exceeded.

There are no Caldon LEFM CheckPlus system controls in the control room at CCNPP. All control functions reside locally at the LEFM CheckPlus system cabinets located in the turbine building and its outputs are provided to the plant computer by means of the plant data network and DAS for the calculation of calorimetric power. The Caldon LEFM CheckPlus system meter status includes Normal, Alert, and Failed status. Programmed logic in the DAS and plant computer will alert control room operators when the system is in out-of-service (OOS) condition. The LEFM CheckPlus is considered to be OOS for the following conditions:

- The meter status shows either an Alert or Failed status for:

- failure of one or more tranducer paths,
- velocity profile out of limits,
- analog input out of limits,
- system uncertainty out of limits.

- Loss of communication from the LEFM CheckPlus system to the plant computer.
- Cabinet temperature exceeds its high temperature limit.

CCNPP proposed two AOTs according to the licensee's letter dated June 11, 2009:

- If the LEFM CheckPlus System is OOS and the plant computer is available to perform the secondary calorimetric calculation, the allowable outage time is 72 hours, provided steady-state conditions exist. Steady-state conditions are defined as power variations of less than 10 percent from the initial power level when the system is declared OOS.

- If the plant computer is unavailable or if another input (main steam pressure) to the secondary calorimetric calculation fails (other than the LEFM CheckPlus System), the allowable outage time is less than or equal to 24 hours.

During the 72-hour AOT of LEFM CheckPlus system, feedwater flow will be measured by the current flow instrument (venturi) which has continuously been calibrated to the LEFM CheckPlus measurements. In this regard, the licensee submitted Hurst Technologies Engineering Report CCN-IR-08001, Revision 0, "Allowed Outage Time (AOT) Justification Report Appendix K Uprate Project," dated November 12, 2008. This report indicated that the venturi transmitter drift data showed the transmitter drift on the power calorimetric during the 72 hour AOT is ±0.0206 percent span of the instrument. The NRC staff considers the venturi transmitter drift uncertainty to be insignificant for a 72 hour period.

The 24-hour AOT is based on the minimum frequency for the calibration of the power range nuclear instrumentation according to TS Surveillance Requirement 3.3.1.2. Per TS Surveillance Requirement 3.3.1.2, the power range nuclear instruments are adjusted every 24 hours based on the reactor thermal power calculation.

For each of these two AOTs, if necessary repairs are not completed within the AOT window, then the power must be reduced and limited to the original licensed thermal power (no higher than 2700 MWt). One additional restraint on maximum power operation will be placed whenever a unit is within the 72 hour outage window due to the Caldon LEFM CheckPlus system being OOS. In this situation, if the plant experiences a power change of more than 10 percent power, the maximum thermal power limit will be limited to the pre-uprate licensed power limit of 2700 MWt. This conservative action ensures that a plant transient does not adversely impact the accuracy of the alternate calorimetric instrumentation.

CCNPP will provide guidance to identify the actions to be taken by the control room staff upon alarm annunciation. The licensee will document, in the site's technical requirements manual, the necessary operator actions to address the instances when the LEFM CheckPlus system is not available to provide the feedwater flow inputs for the calorimetric power measurement, as well as actions to be taken if these inputs are not restored within the allowed time.

The NRC staff reviewed the licensee's submittals and found that the licensee had provided sufficient justification for the proposed AOTs and the proposed actions to reduce power level if the AOT is exceeded. Therefore, the staff concludes that the licensee has met the guidance in Items G and H of Section I of Attachment 1 to RIS 2002-03.

3.9.4 Conclusion

The NRC staff reviewed the licensee's proposed plant-specific implementation of the feedwater flow measurement device and the power uncertainty calculations and determined that the licensee's proposed amendment is consistent with the staff's approved Topical Reports ER-80P and its supplement ER-157P. The staff has 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 as described in Section 2 of this SE.

3.10 Plant Systems

3.10.1 Regulatory Evaluation

The NRC staff's review in the area of plant systems covers the impact of the proposed MUR power uprate on the NSSS interface systems, containment systems, safety-related cooling water systems, spent fuel pool (SFP) storage and cooling, radioactive waste systems, and ESF 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 Attachment (2) of the licensee's application dated August 29, 2008.

3.10.2 Technical Evaluation

NSSS Interface Systems

The NSSS interface systems include the main steam (MS) system, the FW and condensate systems, the extraction steam/heater drain system, the circulating water system, the auxiliary feedwater system (AFW), and the shutdown cooling system (SDC).

The MS system provides isolation of the SGs after a steam line failure, provides overpressure relief and/or decay heat removal during accidents, and provides steam to the AFW system. For the MS system, the licensee stated that, following the MUR power uprate, there will be a slight increase (about 2 percent) in steam flow but MS system operating temperature and pressure will not change. The licensee stated that the relief capacity of the main steam safety valves (MSSVs) is greater than the steam flow at the MUR power uprate conditions. The licensee stated that the MS isolation valves are not impacted by the MUR power uprate because the MS system operating pressure will remain the same. Therefore, the MS system will continue to operate within its design parameters.

The FW and condensate systems provide FW to the SG from the condenser hotwell during normal operation. The FW system isolates during accidents. The MUR power uprate results in approximately a 2 percent increase in both FW and condensate flow. The licensee performed

hydraulic calculations and determined that both the FW and condensate systems are capable of providing sufficient flow to the SGs under the MUR power uprate conditions.

The extraction steam/heater drain system uses extraction steam to heat condensate and FW. The licensee determined that the MUR power uprate will result in an approximately 2 percent increase in heater drain flow and a corresponding temperature increase. The licensee performed a system evaluation and concluded that the equipment will operate at the MUR power uprate conditions and that the design temperature and pressure ratings for Units 1 and 2 extraction steam drain trip air-operated and motor-operated valves bound the MUR power uprate conditions. Therefore, the extraction steam/heater drain system is capable of supporting the MUR power uprate.

The circulating water system contains one condenser for each unit with an operational limit of 12 °F temperature rise in the circulating water across the condensers. After the MUR power uprate, the licensee expects the temperature rise to go from the current value of 11.6 °F to 11.8 °F. The licensee determined that the condenser vacuum in-balance is not adversely affected by the MUR power uprate. Therefore, the circulating water system is capable of supporting the MUR power uprate.

The AFW system provides FW to the SGs when the FW or condensate systems are unavailable in order to cool the primary system to 300 °F. The AFW analysis is based on 102 percent of 2700 MWt or 2754 MWt. The licensee stated that the analytical power level, including revised uncertainty, with the MUR power uprate remains unchanged at 2754 MWt. Therefore, the AFW system is capable of supporting the MUR power uprate.

SDC is designed to remove sensible and decay heat from the RCS during plant cooldown. The licensee stated that the SDC system was previously determined to be capable of supporting the removal of decay heat at 102 percent of 2700 MWt or 2754 MWt. After the MUR power uprate, the analytical power level including revised uncertainty will remain at 2754 MWt. Therefore, the SDC is capable of supporting the MUR power uprate.

The NRC staff reviewed the licensee's evaluations and concurs with the results. The licensee determined that there is no adverse impact on the NSSS interface systems from the MUR power uprate because there is sufficient operating margin to produce an additional 1.38 percent power, and all equipment will be operated within its design limits. 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. Therefore, the staff finds that the NSSS systems are acceptable for the MUR uprate.

Containment Systems

The containment systems include the containment building spray system, penetrations, and hatches. The spray system removes fission products from the post-accident containment atmosphere and assists in post-accident temperature and pressure control. The penetrations and hatches maintain structural integrity. As discussed in Section 3.6, Reactor Systems, of this SE, the containment response analyses to both LOCA and MSLB were evaluated using mass and energy release based on 102 percent of current RTP. These analyses are bounding for the

MUR power uprate. Therefore, the NRC staff finds the containment systems acceptable for the MUR power uprate.

Safety-Related Cooling Water Systems

The safety-related cooling water systems include the service water (SRW) system, the salt water (SW) system, and the component cooling water (CCW) system.

The SRW system removes heat from auxiliary plant systems through heat exchangers cooled by the SW system. The licensee determined that the SRW system does not see significant impact from the MUR power uprate. The licensee determined that the SRW has adequate margin to perform its design functions within its design parameters and is, therefore, capable of supporting the MUR power uprate.

The SW system provides cooling water to the SRW and has two phases following a LOCA, both before and after recirculation actuation. The licensee determined that since the LOCA analysis has been performed at 102 percent of 2700 MWt, the SW system is capable of supporting the MUR power uprate.

The CCW system removes heat from auxiliary plant systems through heat exchangers cooled by the SW system. The licensee determined that the MUR power uprate results in a change to the CCW system heat loads. The licensee evaluated the most limiting mode of CCW operation at the analytical power level of 2754 MWt and determined that the increased decay heat levels associated with the MUR power uprate had only a small impact on the cooling water temperature; therefore, the CCW system is capable of supporting the proposed MUR power uprate.

The NRC staff reviewed the licensee's evaluation of safety-related cooling water systems. Based upon the licensee's determination that the existing analyses for these systems were evaluated for 102 percent RTP, the staff finds there is reasonable assurance that the systems are acceptable for the MUR power uprate.

SFP Cooling (SFPC) System

The principal function of the SFPC system is to provide storage and cooling of the 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 stated that the decay heat source term used in the evaluation of the SFPC system was conservative for the MUR power uprate conditions; therefore, the SFPC system is capable of supporting the MUR power uprate. The NRC staff does not expect that the MUR power uprate will result in a significant change to the operation of the SFPC system. Therefore, the staff concurs with the licensee's conclusion and finds that the SFPC system will not be impacted by the power uprate.

Radioactive Waste Systems

The waste processing systems provide the means to sample, collect, process, store/hold, re-use, and/or release gaseous and liquid low-level effluents. The gaseous waste disposal system provides post-accident containment isolation and venting of excess gas from the reactor

building. The system was designed to operate with 1 percent failed fuel, therefore, the licensee determined that the MUR power uprate does not represent a significant challenge to this system. The NRC staff reviewed the licensee's assessment. The staff does not expect a 1.38 percent increase in power to result in a significant change to the operation of the radioactive waste systems. Therefore, based on the licensee's assessment, the staff finds that the radioactive waste systems will function adequately for the MUR power uprate.

ESF HVAC Systems

The licensee evaluated the ESF HVAC systems. The heat load from the primary systems increases only marginally as a result of the minor change in T_{HOT}. The heat load from the FW piping in the containment, auxiliary building (steam tunnel) and turbine building were evaluated to account for a < 2 °F increase in FW process fluid temperature. The licensee stated that the remaining balance of plant piping temperatures do not change appreciably.

The licensee examined the ESF HVAC systems in the containment, MS penetration rooms, turbine building, AFW pump room, auxiliary building, and the control room. The licensee concluded that the safety functions of these systems are not impacted by the power uprate. The NRC staff reviewed the licensee's evaluation and, based on the licensee's assessment, the staff concludes that the ESF HVAC systems are acceptable for the MUR power uprate.

3.10.3 Conclusion

In summary, the licensee reviewed the design and operation of the plant systems and determined that the proposed MUR power uprate does not adversely impact any of the systems. For the reasons noted above, the NRC staff concurs with the licensee's conclusion and finds that the plant systems will be acceptable for the MUR power uprate.

3.11 Changes to Renewed Facility Operating License and TSs

3.11.1 Regulatory Evaluation

This LAR revises the licensed power level identified in Condition 2.C of the Renewed Facility Operating Licenses and the TS definition of reactor thermal power (RTP).

Licensees may revise the TS content provided that plant-specific review supports a finding of continued adequate safety because: (1) the change is editorial, administrative or provides clarification (i.e., no requirements are materially altered), (2) the change is more restrictive than the licensee's current requirement, or (3) the change is less restrictive than the licensee's current requirement, but nonetheless still affords adequate assurance of safety when judged against current regulatory standards. The detailed application of this general framework, and additional specialized guidance, are discussed in this section in the context of the specific proposed changes.

3.11.2 Technical Evaluation

The submittal includes TS requirements that would demonstrate compliance with 10 CFR 50.36, "Technical specifications," for plant operating conditions related to the requested authorization

for a power level increase. The plant modifications will improve the accuracy of the plant power calorimetric measurement based on the Caldon LEFM Checkplus ™ System (ultrasonic flow meter) instrumentation.

3.11.2.1 Renewed Facility Operating License and Definitions – Rated Thermal Power

The licensee proposed to revise paragraph 2.C of the Renewed Facility Operating License and TS 1.3, Definitions – RTP to reflect the authorized power level increase. The TS RTP will limit the maximum reactor core heat transfer rate to the reactor coolant to 2737 MWt. The NRC staff finds that this change meets 10 CFR 50.36 and is acceptable because the TS limit for operation is derived from the analyses and evaluation included in the safety analysis report (SAR) as accepted by the SE for the requested power level increase discussed herein.

4.0 STATE CONSULTATION

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

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change 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 amendments involve 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 amendments involve no significant hazards consideration, and there has been no public comment on such finding (73 FR 65688). Accordingly, the amendments meet 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 amendments.

6.0 CONCLUSION

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

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Date: July 22, 2009

July 22, 2009

Mr. James A. Spina, Vice President
Calvert Cliffs Nuclear Power Plant, Inc.
Calvert Cliffs Nuclear Power Plant
1650 Calvert Cliffs Parkway
Lusby, MD 20657-4702

SUBJECT: CALVERT CLIFFS NUCLEAR POWER PLANT, UNIT NOS. 1 AND 2 - AMENDMENT RE: MEASUREMENT UNCERTAINTY RECAPTURE POWER UPRATE (TAC NOS. MD9554 AND MD9555)

Dear Mr. Spina:

The Commission has issued the enclosed Amendment No. 291 to Renewed Facility Operating License No. DPR-53 and Amendment No. 267 to Renewed Facility Operating License No. DPR-69 for the Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2. These amendments consist of changes to the Technical Specifications (TSs) in response to your application transmitted by letter dated August 29, 2008, as supplemented by letters dated December 3, two letters dated December 29, December 30, 2008, February 17, February 18, March 10, May 7, and June 11, 2009.

These amendments revise the license and TSs to reflect an increase in the rated thermal power from 2700 megawatts thermal (MWt) to 2737 MWt (1.38 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 the related Safety Evaluation is enclosed. A Notice of Issuance will be included in the Commission's next regular biweekly *Federal Register* notice.

Sincerely,
/ra/
Douglas V. Pickett, Senior Project Manager
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-317 and 50-318
Enclosures:
1. Amendment No. 291 to DPR-53
2. Amendment No. 267 to DPR-69
3. Safety Evaluation

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