Calvert Cliffs Nuclear Power Plant

1650 Calvert Cliffs Parkway
Lusby, Maryland 20657

Constellation Energy
Nuclear Generation Group

August 29, 2008

U. S. Nuclear Regulatory Commission
Washington, DC 20555

ATTENTION: Document Control Desk

SUBJECT: Calvert Cliffs Nuclear Power Plant
Unit Nos. 1 & 2; Docket Nos. 50-317 & 50-318
License Amendment Request: Appendix K Measurement Uncertainty Recapture – Power Uprate Request

REFERENCES:

(a) Nuclear Regulatory Commission Regulatory Issue Summary (RIS) 2002-03, "Guidance on the Content of Measurement Uncertainty Recapture Power Uprate Applications," dated January 31, 2002

(b) SECY-04-0104, Status Report on Power Uprates

Pursuant to 10 CFR 50.90, the Calvert Cliffs Nuclear Power Plant, Inc. hereby requests an amendment to the Renewed Operating License Nos. DPR-53 and DPR-69 to increase the licensed core power. Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2 are currently licensed for a Rated Thermal Power of 2700 MWt. Based on the implementation of more accurate feedwater flow measurement instrumentation, approval is sought to increase the core power by 1.38 percent to 2737 MWt.

The approach used in this amendment request follows that outlined in Reference (a). Reference (a) provides guidance on the scope and detail of the information that should be provided to the Nuclear Regulatory Commission for the review of measurement uncertainty recapture power uprate applications.

The significant hazards discussion and the technical basis for this proposed change are provided in Attachment (1). Attachment (2) provides the information delineated in Reference (a). Marked up pages of the affected Operating Licenses and Technical Specifications are provided in Attachment (3). The Technical Specification Bases will not require any changes to be made.

Based on expected Nuclear Regulatory Commission review timeframes as expressed in Reference (b), we request approval of this proposed change by March 1, 2009. Although this requested approval date does not impact continued operation of the Units at our current allowed power level (2700 MWt) it is needed to allow implementation of the proposed amendment following Unit 2 expected return to operation date following its 2009 refueling outage. We also request a 180 day implementation period for the approved amendment to allow sufficient time to implement procedure changes and operator training associated with this change.

A001
NRR

Should you have questions regarding this matter, please contact Mr. Jay S. Gaines at (410) 495-5219.

Very truly yours,

STATE OF MARYLAND :
: **TO WIT:**
COUNTY OF CALVERT :

I, Douglas R. Bauder, being duly sworn, state that I am Plant General Manager - Calvert Cliffs Nuclear Power Plant, Inc. (CCNPP), and that I am duly authorized to execute and file this License Amendment Request on behalf of CCNPP. To the best of my knowledge and belief, the statements contained in this document are true and correct. To the extent that these statements are not based on my personal knowledge, they are based upon information provided by other CCNPP employees and/or consultants. Such information has been reviewed in accordance with company practice and I believe it to be reliable.

Subscribed and sworn before me, a Notary Public in and for the State of Maryland and County of Calvert, this 29 day of August, 2008.

WITNESS my Hand and Notarial Seal:

Notary Public
Wendy L. Hunter
NOTARY PUBLIC
Calvert County, Maryland
My Commission Expires 01/01/10

My Commission Expires:

Date

DRB/KLG/bjd

Attachments: (1) Technical Basis and No Significant Hazards Consideration
(2) Summary of Calvert Cliffs Nuclear Power Plant Measurement Uncertainty Recapture Evaluation
Enclosure (1) CA006945, Revision 0000, Calorimetric Uncertainty Using the LEFM CheckPlus Flow Measurement System
(3) Marked up Technical Specification Pages

cc: D. V. Pickett, NRC
S. J. Collins, NRC
Resident Inspector, NRC
S. Gray, DNR

ATTACHMENT (1)

TECHNICAL BASIS AND

NO SIGNIFICANT HAZARDS CONSIDERATION

TABLE OF CONTENTS

1.0 SUMMARY DESCRIPTION

This letter requests an amendment to Renewed Operating License DPR-53 and DPR-69 for Calvert Cliffs Nuclear Power Plant (Calvert Cliffs) Unit Nos. 1 and 2, including Appendix A, Technical Specifications, to increase the licensed core power. Calvert Cliffs Units 1 and 2 are currently licensed for a Rated Thermal Power (RTP) of 2700 MWt. Through the use of more accurate feedwater flow measurement equipment, approval is sought to increase this core power by 1.38 percent to 2737 MWt. The power uprate is based on the use of the Caldon Leading Edge Flow Measurement (LEFM) CheckPlus system for determination of main feedwater flow and the associated determination of reactor power through the performance of the power calorimetric calculation currently required by Calvert Cliffs Technical Specifications.

2.0 DETAILED DESCRIPTION

This proposed license amendment would revise the Calvert Cliffs Nuclear Power Plant Operating Licenses and Technical Specifications to increase the licensed power level to 2737 MWt, or 1.38 percent greater than the current level of 2700 MWt. The proposed changes, which are indicated on the marked up pages in Attachment (3), are described below:

1. Paragraph 2.C.(1) in Renewed Operating License Nos. DPR-53 and DPR-69 is revised to authorize operation at a steady-state reactor core power level not in excess of 2737 megawatts-thermal (100 percent power).
2. The definition of RATED THERMAL POWER (RTP) in Technical Specification 1.1 is revised to reflect the increase from 2700 MWt to 2737 MWt.

Calvert Cliffs Units 1 and 2 are presently licensed for an RTP of 2700 MWt. Through the use of more accurate feedwater flow measurement equipment, approval is sought to increase this core power by 1.38 percent to 2737 MWt.

The approach used in this amendment request follows that outlined in Reference (1). Reference (1) provides guidance on the scope and detail of the information that should be provided to the Nuclear Regulatory Commission (NRC) for the review of measurement uncertainty recapture (MUR) power uprate applications.

The 1.38 percent core power MUR uprate for Calvert Cliffs is based on eliminating unnecessary analytical margin originally required for Emergency Core Cooling System (ECCS) evaluation models performed in accordance with the requirements set forth in Reference (2). The NRC has approved a change to the requirements of Reference (2) as described in the Federal Register (65 FR 34913, June 1, 2000). The change provides licensees with the option of maintaining the two percent power margin between the licensed power level and the assumed power level for the ECCS evaluation, or applying a reduced margin for ECCS evaluation. For the reduced margin for ECCS evaluation case, the proposed alternative reduced margin must account for uncertainties due to power level instrumentation error. Based on the proposed use of the Caldon LEFM CheckPlus instrumentation with a power measurement uncertainty of less than 0.6 percent, it is proposed to reduce the licensed power uncertainty required by Reference (2). This results in the proposed increase of 1.38 percent (2737 MWt) in the Calvert Cliffs licensed power level using current NRC-approved methodologies. The Caldon LEFM CheckPlus instrumentation provides a more accurate indication of feedwater flow (and correspondingly reactor thermal power) than assumed during the original development of Reference (2) requirements. The improved thermal power measurement accuracy eliminates the need for the full two percent power margin assumed in Reference (2), thereby increasing the thermal power available for electrical generation.

3.0 TECHNICAL EVALUATION

The impact of the proposed power uprate on applicable systems, components, and safety analyses has been evaluated. Attachment (2) summarizes the results of the comprehensive engineering review performed to evaluate the increase in the licensed core power from 2700 MWt to 2737 MWt. The results of this evaluation are provided in a format consistent with regulatory guidance provided in Reference (1).

As discussed in Attachment (2), the evaluations and analyses have been completed to support an increase in RTP from 2700 MWt to 2737 MWt. In many cases an RTP of 2746 MWt (or a target power uprate of 1.7 percent) was used in order to provide bounding input for these evaluations. 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.

4.0 REGULATORY EVALUATION

4.1 Applicable Regulatory Requirements/Criteria

The requirements for the ECCS Evaluation Models are set forth in 10 CFR Part 50, Appendix K. The NRC-approved a change to these requirements (Federal Register Notice 65 FR 34913, June 1, 2000) that provides licensee with the option of maintaining the two percent power measurement uncertainty in the ECCS analyses or of using a lower value provided the proposed alternative value has been demonstrated to account for the uncertainties due to power level instrumentation error.

Calvert Cliffs proposes to increase RTP from 2700 MWt to 2737 MWt on both Units 1 and 2. The proposed change justifies use of an alternate power level other than 102% of RTP in the ECCS analysis based on the installation of a high accuracy feedwater flow instrumentation system (Caldon LEFM CheckPlus system) which results in a reduction of uncertainty in the power level measurement. This resultant increase in RTP level is referred to as MUR. The analysis and detailed review conducted to support this requested power increase conforms to the guidance specified in NRCs Regulatory Issue Summary 2002-03.

4.2 Precedent

Below is a list of other facilities that have been granted approval for MUR power uprates involving the use of Caldon LEFM CheckPlus feedwater flow instrumentation includes:

Facility	Amendment #(s)	Approval Date
Crystal River, Unit 3	228	December 26, 2007
Vogtle Electric Generating Plant, Units 1 & 2	149/129	February 27, 2008
Cooper Nuclear Station	231	June 30, 2008
Davis Besse Nuclear Power Station, Unit 1	278	June 30, 2008

4.3 Significant Hazards Consideration

Calvert Cliffs is proposing an amendment to the Facility Operating License and Technical Specifications that will increase the licensed power level from 2700 MWt to 2737 MWt based on the use of more accurate feedwater flow measurement equipment. Use of the Caldon LEFM CheckPlus feedwater flow instrumentation reduces measurement uncertainty in the measurement system for determination of main feedwater flow and the associated determination of reactor power through the performance of the power calorimetric calculation currently required by Calvert Cliffs Technical Specifications. The proposed changes have been evaluated against the standards in 10 CFR 50.92 and have been determined to not

involve a significant hazards consideration in the operation of the facility in accordance with the proposed amendment.

1. *Operation of the facility in accordance with the proposed amendment would not involve a significant increase in the probability or consequences of an accident previously evaluated.*

   In support of this measurement uncertainty recapture (MUR) power uprate, a comprehensive evaluation was performed for Nuclear Steam Supply System (NSSS), balance of plant systems and components, and analyses that could be affected by this change. A power calorimetric uncertainty calculation was performed, and the impact of increasing plant power by 1.38 percent on the plant's design and licensing basis was evaluated. The result of these evaluations is that structures, systems, and components required to mitigate transients will continue to be capable of performing their design function at an uprated core power of 2737 MWt. In addition, an evaluation of the accident analyses demonstrates that applicable analysis acceptance criteria continue to be met. No accident initiators are affected by this uprate and no challenges to any plant safety barriers are created by this change. Therefore, operation of the facility in accordance with the proposed change will not involve a significant increase in the probability of an accident previously evaluated.

   The proposed change does not affect the radiological release paths, the frequency of release, or the source-term for release for any accidents previously evaluated in the Updated Final Safety Analysis Report. Structures, systems, and components required to mitigate transients remain capable of performing their design functions, and thus were found acceptable. The reduced uncertainty in the feedwater flow input to the power calorimetric measurement ensures that applicable accident analyses acceptance criteria continue to be met in support of operation at a core power of 2737 MWt. Analyses performed to assess the effects of mass and energy remain valid. The source-terms used to assess radiological consequences have been reviewed and determined to bound operation at the uprated condition. Therefore, operation of the facility in accordance with the proposed change will not involve a significant increase in the consequences of an accident previously evaluated.

   Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. *Operation of the facility in accordance with the proposed amendment would not create the possibility of a new or different kind of accident from any accident previously evaluated.*

   No new accident scenarios, failure mechanisms, or single-failures are introduced as a result of the proposed changes. The installation of the Caldon LEFM CheckPlus feedwater flow instrumentation system has been analyzed, and failures of this system will have no adverse effect on any safety-related system or any structures, systems, and components required for transient mitigation. All structures, systems and components previously required for the mitigation of a transient remain capable of fulfilling their intended design functions. The proposed changes have no adverse effects on any safety-related system or component and do not challenge the performance or integrity of any safety-related system.

   This change does not adversely affect any current system interfaces or create any new interfaces that could result in an accident or malfunction of a different kind than was previously evaluated. Operating at a core power level of 2737 MWt does not create any new accident initiators or precursors. The reduced uncertainty in the feedwater flow input to the power calorimetric measurement ensures that applicable accident analyses acceptance criteria continue to be met to support operation at a core power of 2737 MWt. Credible malfunctions continue to be bounded by

the current accident analysis of record or evaluations that demonstrate that applicable acceptance criteria continue to be met.

Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. *Operation of the facility in accordance with the proposed amendment would not involve a significant reduction in a margin of safety.*

   The margins of safety associated with the MUR power uprate are those pertaining to core power. This includes those associated with the fuel cladding, Reactor Coolant System pressure boundary, and containment barriers. A comprehensive engineering review was performed to evaluate the 1.38 percent increase in the licensed core power from 2700 MWt to 2737 MWt. The 1.38 percent increase required that revised NSSS design thermal and hydraulic parameters be established, which then served as the basis for all of the NSSS analyses and evaluations. This engineering review concluded that no design modifications are required to accommodate the revised NSSS design conditions. The NSSS components were evaluated and it was concluded that the NSSS components have sufficient margin to accommodate the 1.38 percent power uprate. The NSSS accident analyses were evaluated for the 1.38 percent power uprate. In all cases, the evaluations demonstrate that the applicable analyses acceptance criteria continue to be met. As a result, the margins of safety continue to be bounded by the current analyses of record for this change.

   Therefore, the proposed change does not involve a significant reduction in the margin of safety.

Based on the above, Calvert Cliffs concludes that the proposed amendment does not involve a significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and accordingly, a finding of "no significant hazards consideration" is justified.

4.4 Conclusions

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

5.0 ENVIRONMENTAL CONSIDERATION

Implementation of the MUR power uprate is expected to result in a correspondingly small increase (no more than 1.38%) in general radiation levels and in the liquid and gaseous effluent releases. This small increase will have minimal impact as existing site processes and practices are adequate to maintain offsite release concentrations and individual doses within the limits of 10 CFR Part 20 and 10 CFR Part 50, Appendix I.

Calvert Cliffs, has determined that operation with the proposed amendment would not result in any significant change in the types, or significant increases in the amounts, of any effluents that may be released offsite, nor would it result in any significant increase in individual or cumulative occupational radiation exposure. Therefore, the proposed amendment is eligible for categorical exclusion as set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment is needed in connection with the approval of the proposed amendment.

6.0 REFERENCES

(1) Nuclear Regulatory Commission Regulatory Issue Summary 2002-03, "Guidance on the Content of Measurement Uncertainty Recapture Power Uprate Applications," dated January 31, 2002

(2) 10 CFR Part 50, Appendix K, ECCS Evaluation Models

ATTACHMENT (2)

SUMMARY OF CALVERT CLIFFS NUCLEAR POWER PLANT MEASUREMENT UNCERTAINTY RECAPTURE EVALUATION

Calvert Cliffs Nuclear Power Plant , Inc.
August 29, 2008

TABLE OF CONTENTS

LIST OF ACRONYMS

ABB	Asea Brown Boveri, Inc.
ABB-NV	Asea Brown Boveri, Inc.-Non-Turbo Vane
ABB-TV	Asea Brown Boveri, Inc.-Turbo Vane
AC	Alternating Current
ADV	Atmospheric Dump Valves
AFAS	Auxiliary Feedwater Actuation System
AFW	Auxiliary Feedwater
ALARA	As Low As Reasonably Achievable
ANSI	American National Standards Institute
AOO	Anticipated Operational Occurrence
AOP	Abnormal Operating Procedures
AOR	Analysis of Record / Analyses of Record
AOV	Air-Operated Valve
ASME	American Society of Mechanical Engineers
AST	Alternative Radiological Source Term
ATWS	Anticipated Transients Without SCRAM
BLPB	Branch Line Pipe Break
BOP	Balance of Plant
Calvert Cliffs	Calvert Cliffs Nuclear Power Plant, Inc.
CCW	Component Cooling Water
CE	Combustion Engineering
CEA	Control Element Assembly
CEDM	Control Element Drive Mechanism
CEOG	Combustion Engineering Owners Group
CFR	Code of Federal Regulations
COLR	Core Operating Limits Report
CUF	Cumulative Usage Factor
CVCS	Chemical and Volume Control System
DAS	Data Acquisition System
DBA	Design Basis Accident
DBE	Design Basis Event
DC	Direct Current
DNB	Departure from Nucleate Boiling
DNBR	Departure from Nucleate Boiling Ratio
D/P	Differential Pressure
ECCS	Emergency Core Cooling System
EM	Evaluation Model
EOP	Emergency Operating Procedures
EQ	Environmental Qualification
gpm	gallons per minute
HELB	High Energy Line Break
HFP	Hot Full Power
HVAC	Heating, Ventilation, and Air Conditioning
HZP	Hot Zero Power
ICI	Incore Instrumentation
IFBA	Integral Fuel Burnable Absorber

LIST OF ACRONYMS

ISI	Inservice Inspection
IST	Inservice Testing
K_e	plastic strain correction factor
LBB	Leak Before Break
LBLOCA	Large Break Loss-of-Coolant Accident
LCO	Limiting Condition for Operation
LEFM	Leading Edge Flow Measurement
LHR	Linear Heat Rate
LOCA	Loss-of-Coolant Accident
LOSP	Loss of Secondary Pressure
LPSI	Low Pressure Safety Injection
MCLB	Main Coolant Loop Break
MNSA	Mechanical Nozzle Seal Assembly
MOV	Motor-Operated Valve
MSIV	Main Steam Isolation Valve
MSLB	Main Steam Line Break
MSS	Main Steam System
MSSV	Main Steam Safety Valve
MTC	Moderator Temperature Coefficient
MUR	Measurement Uncertainty Recapture
MVA	MegaVolt Ampere
MVAR	MegaVolt Ampere Reactive
MWt	Megawatt Thermal
NRC	Nuclear Regulatory Commission
NSSS	Nuclear Steam Supply System
OBE	Operating Basis Earthquake
ODCM	Offsite Dose Calculation Manual
OOS	Out-of-Service
PLHGR	Peak Linear Heat Generation Rate
PWR	Pressurized Water Reactor
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RIS	Regulatory Issue Summary
RPS	Reactor Protective System
RTD	Resistance Temperature Detector
RTP	Rated Thermal Power
RV	Reactor Vessel
RVI	Reactor Vessel Internal
S2M	Supplement 2 to CENPD-137 Evaluation Model
SAFDL	Specified Acceptable Fuel Design Limits
SBLOCA	Small Break Loss-of-Coolant Accident
SDC	Shutdown Cooling
SER	Safety Evaluation Report
SFPC	Spent Fuel Pool Cooling
SG	Steam Generator
SI	Safety Injection

LIST OF ACRONYMS

SIT	Safety Injection Tank
SLB	Steam Line Break
S_m	Primary Membrane Stress
SRW	Service Water
SW	Saltwater
T_{ave}	Vessel Average Coolant Temperature
T_{cold}	Vessel/Core/Inlet Temperature
T_{hot}	Vessel Outlet Temperature
TM/LP	Thermal Margin/Low Pressure
TRM	Technical Requirements Manual
UF	Usage Factor
UFM	Ultrasonic Flow Measurement
UFSAR	Updated Final Safety Analysis Report
VAP	Value Added Fuel
ZrB_2	Zirconium Diboride

INTRODUCTION

BACKGROUND AND REASON FOR PROPOSED CHANGE

Calvert Cliffs Nuclear Power Plant (Calvert Cliffs) Units 1 and 2 are presently licensed for a Rated Thermal Power (RTP) of 2700 MWt. Through the use of more accurate feedwater flow measurement equipment, approval is sought to increase this core power by 1.38% to 2737 MWt. The impact of a 1.38% core power uprate for applicable systems, components, and safety analyses has been evaluated.

The analyses and evaluations were performed for both Calvert Cliffs Units 1 and 2. In some cases where cycle specific data is needed, the analyses/evaluations targeted Unit 1 as the lead unit for the Measurement Uncertainty Recapture (MUR) power uprate. However because of the timing involved, Unit 2 will likely be the first unit to implement the MUR power uprate. Confirmation of the applicability of the cycle specific analyses and evaluations on Unit 2 for this operating cycle, and all subsequent cycles of Units 1 and 2, are performed as part of the normal reload design process.

The approach used in this amendment request follows that outlined in Nuclear Regulatory Commission (NRC) Regulatory Issue Summary (RIS) 2002-03, "Guidance on the Content of Measurement Uncertainty Recapture Power Uprate Applications" dated January 31, 2002. Regulatory Issue Summary 2002-03 provided guidance on the scope and detail of the information that should be provided to the NRC for the review of MUR power uprate applications.

The 1.38% MUR power uprate for Calvert Cliffs is based on eliminating unnecessary analytical margin originally required for Emergency Core Cooling System (ECCS) evaluation models (EMs) performed in accordance with the requirements set forth in the Code of Federal Regulations (CFR), 10 CFR Part 50, Appendix K (ECCS).

As discussed in detail in Section II, the evaluations and analyses described have been completed to support an increase in RTP from 2700 MWt to 2737 MWt. In many cases an RTP of 2746 MWt (or a target power uprate of 1.7%) was used in order to provide bounding input for these evaluations. Currently, with the RTP of 2700 MWt, the analytical power level of 2754 MWt (102% of 2700 MWt) is used in the safety analysis. With a revised RTP of 2737 MWt and a revised uncertainty, the analytical power level is unchanged at 2754 MWt.

The NRC has approved a change to the requirements of 10 CFR Part 50, Appendix K [65 FR 34913, June 1, 2000]. The change provides licensees with the option of maintaining the two-percent power margin between the licensed power level and the assumed power level for the ECCS evaluation, or applying a reduced margin for ECCS evaluation. For the reduced margin for ECCS evaluation case, the proposed alternative reduced margin must account for uncertainties due to power level instrumentation error. Based on the proposed use of the Caldon Leading Edge Flow Measurement (LEFM) CheckPlus instrumentation with a power measurement uncertainty of less than 0.6%, it is proposed to reduce the licensed power uncertainty required by 10 CFR Part 50, Appendix K. This results in the proposed increase of 1.38% in the Calvert Cliffs licensed power level using current NRC approved methodologies.

The Caldon LEFM CheckPlus instrumentation provides a more accurate indication of feedwater flow (and correspondingly reactor thermal power) than assumed during the development of Appendix K requirements. Technical support for this conclusion is discussed in detail in the Caldon LEFM CheckPlus Topical Reports (References I-1 and I-2). The improved thermal power measurement accuracy eliminates the need for the full 2% power margin assumed in Appendix K, thereby increasing the thermal power available for electrical generation.

The desired power increase of 1.38% 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 leg temperature will increase slightly in response to the increased steam flow demand. As a result, the RCS average temperature will increase slightly.

Procedures for maintenance and calibration of the Caldon LEFM CheckPlus system will be enhanced per the design control process based on the vendor's recommendations. Should the Caldon LEFM CheckPlus system be unavailable, the main steam or feedwater flow venturis can be used to measure flow rate in the feedwater system, as is done currently. If the Caldon LEFM CheckPlus system is not functional, the power limit will be administratively controlled at a level consistent with the accuracy of the available instrumentation as described in this amendment request. The power limit reduction requirement for the Caldon LEFM CheckPlus system out-of-service (OOS) will be incorporated into the Calvert Cliffs Technical Requirements Manual (TRM).

DESCRIPTION OF PROPOSED CHANGE

The proposed license amendment would revise the Calvert Cliffs Operating Licenses and Technical Specifications to reflect an increase in core power level by 1.38% to 2737 MWt. The power uprate is based on the use of the Caldon LEFM CheckPlus system for determination of main feedwater flow and the associated determination of reactor power through the performance of the power calorimetric calculation currently required by Calvert Cliffs Technical Specifications. The proposed changes are identified on the markups of the current Calvert Cliffs Operating Licenses and Technical Specification pages.

Calvert Cliffs notes that various Combustion Engineering (CE) topical reports that are part of the Calvert Cliffs licensing basis (Technical Specification 5.6.5), consistent with 10 CFR Part 50, Appendix K may have included explicit references to their use of "102% of licensed core power levels." These topical reports describe the NRC-approved methodologies which support the Calvert Cliffs safety analyses, including the small break and large break loss-of-coolant accident (LOCA) analyses. Along with the proposal to increase the reactor thermal power to 2737 MWt, Calvert Cliffs requests continued use of these topical reports. Calvert Cliffs does not consider that these topical reports require revision to reflect this requested power uprate. Rather, it will be understood that those statements refer to the Appendix K margin and the original licensed power level. Calvert Cliffs proposes that these topical reports be approved for use consistent with this license amendment request, and further, the NRC acknowledges that the change in the power uncertainty does not constitute a significant change, as defined in 10 CFR 50.46 and 10 CFR Part 50, Appendix K, to these topical reports.

GENERAL LICENSING APPROACH FOR PLANT ANALYSIS USING PLANT POWER LEVEL

The MUR power uprate program for Calvert Cliffs as described addresses Nuclear Steam Supply System (NSSS) performance parameters, design transients, systems, components, accidents, and nuclear fuel as well as interfaces between the NSSS and Balance of Plant (BOP) systems. No new analytical techniques have been used to support the MUR power uprate project. The key points include the use of:

- Well-defined analysis input assumptions/parameter values
- Currently approved analytical techniques
- Applicable licensing criteria and standards

The evaluations and analyses described have been completed in order to bound an increase in RTP from 2700 MWt to 2737 MWt, or a 1.38% increase. The RTP used for many evaluations targeted a bounding power uprate of 1.7% with MUR power uprate, or 2746 MWt. Currently, with the RTP of 2700 MWt, the analytical power level of 2754 MWt (102% of 2700 MWt) is used in the safety analysis. With a revised RTP of 2737 MWt and a revised uncertainty, the analytical power level is unchanged at 2754 MWt.

Section I provides a description of the feedwater flow measurement system that will be installed on both units. Section I also provides a summary of the overall thermal power measurement uncertainty.

Section II provides the results of the accident analyses and evaluations performed for the LOCA and non-LOCA transients. Section II also summarizes the containment accident analyses and evaluations and the radiological consequence evaluations.

Section III provides results for accidents and transients for which the existing analyses of record (AOR) do not bound plant operation at the proposed uprated power level.

Section IV of this report discusses the revised NSSS design thermal and hydraulic parameters that were modified as a result of the MUR power uprate and that serve as the basis for all of the NSSS analyses and evaluations. In addition this section discusses the effect of the power uprate on the structural integrity of major plant components. Section IV also contains the results of the fuel-related analyses.

Section V provides an analysis of the effects of the power uprate on the Calvert Cliffs electrical power systems.

Section VI presents information on the impact of the proposed power uprate on the system design [e.g., safety injection (SI), shutdown cooling (SDC), and control systems] and components [e.g., reactor vessel (RV), pressurizer, Reactor Coolant Pumps (RCPs), steam generator (SG), and NSSS auxiliary equipment] and the evaluations completed for the revised design conditions. Section VI also summarizes the effects of the uprate on the BOP (secondary) systems based upon a heat balance evaluation.

Section VII evaluates the impact of the power uprate on various other areas including the impact on plant operations, the impact on the environment and the impact on occupational radiation exposure.

Section VIII presents information on changes to Technical Specifications, protection system settings, and emergency system settings as a result of the proposed power uprate.

The results of all of the analyses and evaluations performed demonstrate that all acceptance criteria continue to be met.

I. FEEDWATER FLOW MEASUREMENT TECHNIQUE AND POWER MEASUREMENT UNCERTAINTY

I.1 APPROVED TOPICAL REPORTS ON FEEDWATER FLOW MEASUREMENT TECHNIQUE

The reference Topical Reports associated with the Caldon LEFM CheckPlus feedwater flow measurement system are as follows:

1. ER-80P, "Improving Thermal Power Accuracy and Plant Safety While Increasing Operating Power Level using the LEFM Check System," dated March 1997 (Reference I-1)
2. ER-157P, "Supplement to Topical Report ER-80P: Basis for a Power Uprate with the LEFM Check or CheckPlus System" dated October 2001 (Reference I-2)

I.2 NRC APPROVAL OF FEEDWATER FLOW MEASUREMENT TECHNIQUE

The NRC approved the subject Topical Reports listed in Section 1.1 above on the following dates:

1. Reference I-1, NRC Safety Evaluation Report (SER) dated March 8, 1999
2. Reference I-2, NRC SER dated December 20, 2001

I.3 CALDON LEFM CHECKPLUS SYSTEM

The feedwater flow measurement system to be installed at Calvert Cliffs is the Caldon LEFM CheckPlus ultrasonic multi-path transit time flow meter. The installation of this system will conform to the requirements of References I-1 and I-2. Subsequent reviews by the NRC, in Reference I-3 found that the performance of the CheckPlus system was consistent with the topical reports, with one exception to further evaluate the effects of transducer replacement on system uncertainty. The exception was subsequently addressed in Reference I-4 and disseminated to the industry via Reference I-5. The installation at Calvert Cliffs will include the additional uncertainty associated with transducer replacement, described in References I-4 and I-5.

The Caldon LEFM CheckPlus System to be installed at Calvert Cliffs consists of two spool piece measurement sections per unit with one spool piece installed in the 16" feedwater header for each SG. 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, such that a transducer may be removed at power without disturbing the pressure boundary of the spool piece. The installation locations conform to the requirements of References I-1 and I-2. The measurement sections will be installed in accordance with approved Calvert Cliffs procedures and work controls processes to achieve installation tolerances within the bounds stated in the Caldon uncertainty analysis.

A cabinet mounted Caldon LEFM CheckPlus electronic unit will also be installed in the Turbine Building, in the vicinity of the measurement sections. One cabinet will be installed per unit. The cabinets contain an integral air conditioning unit to maintain an acceptable operating environment.

Two pressure transmitters meeting the uncertainty requirements of the Caldon Topical Reports will be installed 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 flow.

The Caldon LEFM CheckPlus Systems determine feedwater parameters for feedwater mass flow, feedwater temperature, and feedwater pressure to be used for the 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 system incorporates self-verification features to ensure that the system continually operates within the design basis uncertainty analysis. Diagnostic and signal quality data is communicated to the DAS to allow monitoring of degradation of the Caldon LEFM CheckPlus System. The system triggers control room annunciation via the Plant Computer when conditions are at a state which could impact the flow measurement uncertainty.

The LEFM CheckPlus measurement sections are calibrated in a site-specific model test at Alden Research Laboratories with all calibration standards traceable to National Institute of Standards and Technology standards. The site-specific test plan provides meter factor calibration data over a wide range of hydraulic test conditions intended to envelope the expected hydraulic conditions at the installation locations. The tests include plant piping modeling and parametric variations of those models, straight pipe testing, and inducement of extreme swirl conditions. The meter factor data, determined by comparing the Alden Lab reference standard to the flow as measured by the Caldon LEFM CheckPlus System, is collected for each piping configuration at various flow rates. Measurement of the hydraulic profile, called the flatness ratio, is also collected at each flow rate. The meter factor versus flatness ratio is plotted for all conditions and all flow rates and compared to analytically derived expected performance curves for quality control purposes. These data provide a quantitative measure of the Caldon LEFM CheckPlus Meter Factor verses the actual velocity profile encountered and determines the meter uncertainty to be used in the overall calorimetric uncertainty.

LEFM CheckPlus System Controls, Displays, and Alarms

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 – the system and meter status (Normal, Alert, Failed) are communicated to the DAS. An Alert or Failed condition indicates a condition that may adversely affect the uncertainty of the LEFM CheckPlus System mass flow rate determination and triggers the Plant Computer alarm and control room annunciation. Upon receipt of this alarm, the LEFM CheckPlus System is considered either in a degraded status or OOS.
- Loss of Communication – a communications failure from the LEFM CheckPlus System to the Plant Computer triggers the Plant Computer alarm and control room annunciation. Upon receipt of this alarm, the LEFM CheckPlus System is considered OOS.
- LEFM CheckPlus System Cabinet High Temperature – a cabinet high temperature condition also triggers the Plant Computer alarm and control room annunciation. If the maximum temperature limit is exceeded, the LEFM CheckPlus System is considered OOS.

Guidance will be provided to identify the actions to be taken by the Control Room staff upon alarm annunciation. Detailed LEFM CheckPlus System process and diagnostic data, communicated to the

DAS, is also be available for use by operations staff for diagnosis of system alarms. The process and diagnostic data is also available locally at the LEFM CheckPlus cabinet.

Refer to Sections I.7 and I.8 for additional information regarding operation in a degraded or OOS condition.

I.4 COMPLIANCE WITH NUCLEAR REGULATORY COMMISSION SAFETY EVALUATION REPORT

The installation of the Caldon LEFM CheckPlus flow measurement system at Calvert Cliffs is consistent with References I-1 and I-2. In addition to the installation requirements, the NRC identified in Reference I-6, the following four criteria that must be addressed by licensees requesting a license amendment based on the Topical Reports. Calvert Cliffs meets the four criteria as described below.

Criterion 1

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

Response to Criterion 1

Implementation of the power uprate license amendment includes developing the necessary procedures and documents required for operation, maintenance, calibration, testing, and training at the uprated power level with the new Caldon LEFM CheckPlus System. Plant procedures will be revised to incorporate the vendor's maintenance and calibration requirements prior to declaring the Caldon LEFM CheckPlus System operational and raising reactor core power above 2700 MWt (98.6% of proposed RTP). The incorporation of, and continued adherence to, these requirements assure that the Caldon LEFM CheckPlus System is properly maintained and calibrated. Calibration and maintenance are further discussed in Section I.6 below.

Administrative and procedural controls will be established to provide guidance to operators in the event that Caldon LEFM CheckPlus system is unavailable. Contingency plans for operation of the plant with the Caldon LEFM CheckPlus degraded or OOS are described in detail in Sections I.7 and I.8 below.

Criterion 2

For a plant that currently has Caldon LEFM CheckPlus system 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 bound the analysis and assumptions set forth in Reference I-1.

Response to Criterion 2

The Caldon LEFM CheckPlus system is not currently installed at Calvert Cliffs.

Criterion 3

Confirm that the methodology used to calculate the uncertainty of the Caldon LEFM CheckPlus system in comparison to 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 (UFM) instrumentation installations for comparison.

Response to Criterion 3

The methodology used to calculate the uncertainty of the Caldon LEFM CheckPlus system is consistent with the approved Topical Reports. The Topical Reports have been reviewed by site personnel and found to be consistent with Calvert Cliffs engineering standards, derived from Reference I-7 and consistent with Reference I-8. An alternative methodology is not used.

Using site standards, uncertainties for parameters that are not statistically independent are arithmetically summed, then statistically combined with other parameters. Random uncertainties are combined using the Square Root Sum of Squares approach. Systematic biases are then added to the result to determine the overall uncertainty. This methodology is consistent with the vendor determination of LEFM CheckPlus System uncertainty, described in the topical reports.

Criterion 4

For plants where the ultrasonic meter (including Caldon LEFM CheckPlus) 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 profile 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 UFM installation and calibration assumptions.

Response to Criterion 4

The Caldon LEFM CheckPlus System will be calibrated using a site-specific piping configuration at Alden Research Laboratories. Testing will be witnessed by Calvert Cliffs personnel. The site-specific test plan provides meter factor calibration data over a wide range of hydraulic test conditions intended to envelope the expected hydraulic conditions at the installation locations. The results of the tests will be used as the basis for the vendor uncertainty reports and will be provided to Calvert Cliffs. The calibration meter factor and the uncertainty in the calibration factor are based upon these reports.

Since the calibration of the Caldon LEFM CheckPlus measurement sections has not been completed, a flow measurement uncertainty of +/- 0.5% flow has been assumed to support the requested uprate. Furthermore, the calculation is based on using +/- 1.88°F uncertainty using the existing feedwater resistance temperature detectors (RTDs) for feedwater enthalpy and not the more precise temperature measurement available using the LEFM CheckPlus System. These assumptions are very conservative as the Caldon LEFM CheckPlus System is capable of a flow measurement uncertainty on the order of +/- 0.3% with a temperature measurement uncertainty of +/- 0.6°F.

Final acceptance of the Calvert Cliffs specific uncertainty analysis occurs after completion of the commissioning process. The commissioning process verifies that the in-situ test data is bounded by the calibration test data (see Appendix F of Reference I-1). This step provides final positive confirmation that actual performance is within the bounds established for the instrumentation. Final commissioning of the Caldon LEFM CheckPlus Systems is expected to be completed following the 2009 Unit 2 spring refueling outage and the 2010 Unit 1 spring refueling outage.

I.5 THERMAL POWER MEASUREMENT UNCERTAINTY

The impact of the LEFM CheckPlus system on the overall thermal power measurement uncertainty is presented in Enclosure (1). Since the calibration of the LEFM CheckPlus measurement sections has not been completed, conservative assumptions for flow and temperature measurement uncertainty (as detailed in the Response to Criterion 4 section above) are used in calculating the overall thermal power measurements. These assumptions will be confirmed during acceptance of the vendor uncertainty reports by Calvert Cliffs.

Upon receipt of the vendor calibration reports, the calorimetric uncertainty assessment will be revised, if necessary, to determine the available margin at the uprated power of 2737 MWt. The vendor's site-specific uncertainty analysis includes uncertainty associated with transducer replacement as required by Reference I-3 and described in References I-4 and I-5.

Tables I-1 and I-2 summarize the core thermal power measurement uncertainty in percentage of the proposed uprated power of 2737 MWt for Calvert Cliffs for each input to the calorimetric calculation. The parameter uncertainties in Table I-1 are based upon the instrumentation uncertainties listed in Table I-2.

For each random input in Table I-1, an effective contribution is also listed to permit the algebraic summing of the bias inputs with the random contribution to develop the combined uncertainty of each input. As shown in Table I-1, the sum of the effective contributions is equivalent to the square root of the sum of the squares of the random inputs.

Table I-1
Process Parameter Inputs to Secondary Calorimetric Calculation

INPUT	Random Inputs to Uncertainty, MBTU/hr	Effective Random Contribution, MBTU/hr	Bias Inputs to Uncertainty, MBTU/hr	Combined Uncertainty, MBTU/hr	Combined Uncertainty, % RTP	Percent Contribution to Uncertainty
Feedwater Flow	-29.1498	-25.3725		-25.3725	-0.2717%	61.679%
Blowdown Flow	-4.5779	-0.6258		-0.6258	-0.0067%	1.521%
Feedwater Enthalpy:	15.2863	-6.9774		-6.9774	-0.0747%	16.962%
Feedwater Temperature	*-15.2682*	*-6.9610*		*-6.9610*	*-0.0745%*	*16.922%*
Feedwater Pressure	*-0.0884*	*-0.0002*		*-0.0002*	*0.0000%*	*0.001%*
Plant Computer Calculation of Sub-cooled Liquid Enthalpy	*-0.7370*	*-0.0162*		*-0.0162*	*-0.0002%*	*0.039%*
Steam Enthalpy:	4.1484	-0.5139	-1.1812	-1.6951	-0.0182%	4.121%
Steam Pressure	*-4.0837*	*-0.4980*	*-1.1812*	*-1.6792*	*-0.0180%*	*4.082%*
Plant Computer Calculation of Saturated Vapor Enthalpy	*-0.7295*	*-0.0159*		*-0.0159*	*-0.0002%*	*0.039%*
Plant Computer Calculation of Saturated Liquid Enthalpy	*-0.0091*	*0.0000*		*0.0000*	*0.0000%*	*0.000%*
Calorimetric Constants (1)			-6.4657	-6.4657	-0.0692%	15.718%
Totals	-33.4895 (2)	-33.4895 (3)	-7.6469 (3)	-41.1365 (3)	-0.4405% (3)	100.00%

(1) Adjustments for miscellaneous heat addition and heat removal terms from the RCS, such as input from pressurizer heaters
(2) Square Root Sum of Squares
(3) Algebraic Sum

Table I-2
Uncertainties of Inputs to Secondary Calorimetric Calculation

Input	Random Uncertainty	Bias Uncertainty
Feedwater flow (assumed), % Flow	+/- 0.50%	
Feedwater pressure (assumed), psi	+/- 15.00	
Feedwater temperature (assumed), °F	+/- 1.88	
Steam Pressure, psi	+/- 19.80	+ 3.40
Total Blowdown Flow, klbm/hr	+/- 8.1	
Plant Computer Calculation of Enthalpies, BTU/lbm	+/- 0.10	
Calorimetric Constants, MBTU/hr [1]		-6.4657

[1] Adjustments for miscellaneous heat addition and heat removal terms from the RCS, such as input from pressurizer heaters

I.6 CALIBRATION AND MAINTENANCE

A. Maintaining Calibration:

Calibration and maintenance is performed by qualified Calvert Cliffs maintenance personnel using site procedures. The site procedures will be enhanced using the Caldon LEFM CheckPlus technical manuals and work instructions. All work is performed in accordance with site work control procedures.

Formal training on system operation and maintenance will be provided to the appropriate Calvert Cliffs personnel. Operations training is conducted by qualified Calvert Cliffs personnel in accordance with approved site procedures for the performance of training. All necessary training will be completed prior to commissioning of the Caldon LEFM CheckPlus System.

Routine maintenance activities for the Caldon LEFM CheckPlus System include:

- physical inspections of system components,
- power supply checks,
- analog input checks,
- acoustic processor unit checks,
- watchdog timer checks,
- communications checks,
- transducer cable checks,
- dimensional checks, and
- calibration of pressure transmitters for feedwater pressure input to the cabinet.

Other instruments which provide input to the secondary calorimetric are periodically calibrated in accordance with approved site procedures to ensure reliable operation that satisfies the requirements of the calorimetric uncertainty calculation.

B. Controlling Software and Hardware Configuration:

The Caldon LEFM CheckPlus System is designed and manufactured in accordance with the vendor's 10 CFR Part 50, Appendix B, Quality Assurance Program and its Verification and Validation program. The vendor's Verification and Validation program satisfies the

requirements of References I-9 and I-10. In addition the program is consistent with guidance for software Verification and Validation in Reference I-11.

After installation, software and hardware configuration is controlled in accordance with site procedures and processes for software configuration control. Proposed changes to the software and hardware configuration for all components that provide input to the calorimetric calculation are evaluated in accordance with the approved engineering change process.

C. Performing Corrective Actions:

Reliability of the Caldon LEFM CheckPlus system and other calorimetric instrumentation is monitored by Calvert Cliffs Plant Engineering personnel. Adverse performance trends, failed preventive maintenance, or other observed equipment deficiencies are documented and resolved in accordance with the site's corrective action process.

Any needed corrective maintenance is performed by qualified Calvert Cliffs maintenance personnel.

D. Reporting Deficiencies to the Manufacturer:

Corrective action procedures include instructions for notification of deficiencies and error reporting. Equipment manufacturers are contacted as required to correct the deficiency.

E. Receiving and Addressing Manufacturer Deficiency Reports:

Manufacturer deficiency reports are reviewed and dispositioned in accordance with the site's corrective action program. In addition, incoming Institute of Nuclear Power Operations Operating Experience are reviewed by site personnel for applicability. Those deficiencies applicable to Calvert Cliffs are documented under the site's corrective action process.

I.7 OUTAGE TIME

Each of the Caldon LEFM CheckPlus Systems to be installed will consist of two measurement sections. One measurement section is installed in the feedwater header to each SG. Each measurement section consists of two planes of transducers with four pairs of transducers in each plane, as described in Reference I-2. The transducers provide input to the electronic unit cabinet, which consists of two subsystems of electronics hardware. Each subsystem receives input from one plane of the measurement sections. Outputs from the electronic unit are provided to the Plant Computer via the Plant Data Network and DAS for the calculation of calorimetric power. Programmed logic in the DAS and Plant Computer, alert operators when the system is in a degraded or OOS condition. The following conditions trigger the Plant Computer alarm:

- LEFM CheckPlus System Meter Status Not Normal – the meter status (Normal, Alert, Failed) is communicated to the DAS and Plant Computer. A meter status of other than normal triggers the Plant Computer alarm. The meter status is determined from a series of on-line self-diagnostics to verify that the system is operating within its design basis uncertainty limits. The following conditions result in a meter status of other than normal:
  - failure of one or more transducer 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 limit.

Guidance will be provided to identify the actions to be taken by the Control Room staff upon alarm annunciation. If the system is degraded or OOS, time accrues against the allowable outage times. Upon reaching the limit for the allowable outage time, the maximum power limit will be reduced to the pre-uprate licensed power limit of 2700 MWt (98.6% proposed RTP). Power is adjusted, as required, to ensure the pre-uprate licensed power limit is not exceeded.

Three outage times are proposed:

- If the LEFM CheckPlus System is in a degraded condition with the Plant Computer available to perform the secondary calorimetric calculation, the allowable outage time is 30 days.
- If the LEFM CheckPlus System is OOS with the Plant Computer 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% from the initial power level when the system is declared OOS.
- If the Plant Computer is unavailable or if another input to the secondary calorimetric calculation fails (other than the LEFM CheckPlus System), the allowable outage time is less than or equal to 24 hours.

Allowable outage times will be described in the TRM. If the site-specific uncertainty analysis for the LEFM CheckPlus System does not support operation in a degraded condition, the 30-day outage time will not be adopted.

LEFM CheckPlus System Degraded, Plant Computer Available

A 30-day outage time is proposed if the LEFM CheckPlus System is degraded but the Plant Computer is available to perform the secondary calorimetric calculation. The system is considered to be degraded when an alert condition is detected and reported by the system, resulting in control room annunciation. The site-specific uncertainty calculation for the LEFM CheckPlus System includes uncertainty with the system in an alert condition. If the resultant calorimetric uncertainty supports the proposed uprate, operation in the degraded condition can theoretically continue indefinitely, although at a reduced margin. However, operation in a degraded condition is limited to 30 days to ensure that the system is restored to a fully operational status. If an alert condition is detected, an operator verifies the cause of the alarm and determines if the system can continue to be operated in the degraded status.

As described in Reference I-2, the Caldon LEFM CheckPlus System consists of subsystems of electronic hardware. An alert condition basically informs the operator of the malfunction of a single subsystem, resulting in a slight increase to calorimetric uncertainty. In this condition, the system basically operates as the LEFM Check System described in References I-1 and I-2, capable of supporting uprates on the order of the requested 1.38% uprate. However, if the site-specific uncertainty analysis for the LEFM CheckPlus System does not support the uprate, the 30-day outage time will not be adopted.

If Calvert Cliffs is unable to restore the LEFM CheckPlus system to full operation within the 30-day outage window, operators take action as indicated in Section I.8 below.

LEFM CheckPlus System OOS, Plant Computer Available

A 72 hour outage time is proposed if the LEFM CheckPlus System is OOS but the Plant Computer is available to perform the secondary calorimetric calculation. The system is considered to be OOS when either a fail condition is detected and reported by the system failure or when communication with the system is lost, resulting in control room annunciation. The 72 hour outage time is based upon:

- Calculation of calorimetric power using the Plant Computer from alternate plant instrumentation. The operator can select an alternate set of parameters in lieu of the output of the Caldon LEFM CheckPlus System to calculate calorimetric power for feedwater flow, temperature, and pressure. Existing plant instrumentation, such as the feedwater venturis, currently being used to calculate secondary calorimetric power, is used for the alternate set of parameters.
- Normalizing the alternate input for feedwater flow and temperature to the Caldon LEFM CheckPlus feedwater flow and temperature. A rolling average of the ratio of the LEFM CheckPlus input to the alternate input is calculated on the Plant Computer. When the alternate set of parameters is selected, the last known good value of the average ratios will be applied such that the output of the calorimetric calculation using the alternate parameters closely matches the output of the calculation using the Caldon LEFM CheckPlus System. As shown in Table I-1, the calorimetric calculation is not sensitive to changes in feedwater pressure, such that no correction is necessary to feedwater pressure.
- Unlikely occurrence of venturi nozzle fouling or defouling. Calvert Cliffs does not have a history of venturi nozzle fouling and subsequent defouling. Therefore, no change in calorimetric output from fouling or defouling is anticipated during the 72 hour OOS time. With the LEFM CheckPlus System OOS, alternate indications such as turbine first stage pressure and feedwater temperature, will be used to ensure that plant power is not adjusted to account for venturi nozzle defouling, in the unlikely event fouling exists. Adjustments based on nozzle fouling, should it occur, would result in a conservative adjustment to calorimetric power.
- Negligible instrument drift. Instrument drift over a 72 hour period is negligible and can be verified using alternate plant instrumentation such as turbine first stage pressure.
- Anticipated margin. The assumed values for feedwater flow uncertainty and feedwater temperature uncertainty to support the requested 1.38% uprate are more conservative than typical values for the LEFM CheckPlus System, which can be used to support uprates on the order of 1.6% to 1.7%. When the calorimetric uncertainty assessment is revised to incorporate the vendor calibration reports, the calorimetric uncertainty is reduced, increasing the available margin.

Most repairs to the Caldon LEFM CheckPlus System are expected to be completed within a shift. The 72 hours gives plant personnel sufficient time to diagnose, plan, implement, and verify repairs to the system. If repairs are not completed within the 72 hour window, operators take action as indicated in Section I.8 below.

Plant Computer Unavailable

An outage time less than or equal to 24 hour is proposed if the Plant Computer is unavailable or if another input to the secondary calorimetric calculation fails, regardless of the status of the Caldon LEFM CheckPlus System. The outage time is based upon:

- The minimum frequency for the calibration of the power range nuclear instrumentation in accordance with Technical Specification Surveillance Requirement 3.3.1.2. Per Technical Specification Surveillance Requirement 3.3.1.2, the power range nuclear instruments are adjusted every 24 hours based on the reactor thermal power calculation. Therefore, the actual duration of

the allowable outage time is determined from the next required adjustment of the power range nuclear instruments after the failure is identified.

- The precision of the Plant Computer calculation is required to support the increased power level. Without the Plant Computer, the uncertainty of alternate indications that may be used to calculate calorimetric power exceeds the uncertainty required to support the power uprate. Additionally, averaging of the calorimetric calculation is no longer available.
- The failure of shared inputs to the calorimetric calculation. Alternate inputs are available only for feedwater flow, temperature, and pressure. Other inputs, such as steam pressure, do not have alternate inputs. If a shared input fails, calorimetric power cannot be calculated on the Plant Computer.

Occasional bad quality data is expected and would not result in entrance into the OOS time unless the bad quality data resulted in bad quality for the four hour averaged calorimetric power calculation.

If Calvert Cliffs is unable to restore the Plant Computer to normal operation within the 24 hour window, operators take action as indicated in Section I.8 below.

I.8 OPERATOR ACTION TO REDUCE POWER

For each of the three outage times indicated in Section I-7, if necessary repairs are not completed within the allowed outage time window, operators take action to limit the maximum thermal power limit to the pre-uprate licensed power limit of 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% power, the maximum thermal power limit will be limited to the pre-uprate licensed power limit of 2700 MWt. Although power changes have not been shown as having a significant effect on the alternate calorimetric instrumentation, this conservative action ensures that a plant transient does not adversely impact the accuracy of the alternate calorimetric instrumentation.

Calvert Cliffs intends to document, within the site's TRM, necessary operator actions to address the instances when the Caldon LEFM CheckPlus System is not available to provide the feedwater flow element inputs to the heat balanced calorimetric algorithm power measurement, as well as actions to be taken if these inputs are not restored in the allowed time. Operator actions are captured in the TRM vice the Technical Specifications as the feedwater flow element inputs to the heat balance calorimetric algorithm do not meet the criteria of 10 CFR 50.36(d)(2)(ii) for establishing a Technical Specification Limiting Condition for Operation (LCO) as indicated below.

Criterion 1

The Caldon LEFM CheckPlus feedwater flow element inputs are not used to detect and indicate abnormal degradation of the reactor coolant pressure boundary.

Criterion 2

The Caldon LEFM CheckPlus feedwater flow element inputs are not initial conditions of a design basis accident (DBA) or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

Criterion 3

The Caldon LEFM CheckPlus feedwater flow element inputs are not part of the primary success path and do not function or actuate to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

Criterion 4

In the event of the Caldon LEFM CheckPlus ultrasonic feedwater flow element inputs not being available for the heat balance calorimetric algorithm, the inputs will be determined by alternate instrumentation thus, the Caldon LEFM CheckPlus ultrasonic feedwater flow element inputs are not significant to public health and safety.

It is therefore concluded that an LCO is not required to be included in the Technical Specifications in accordance with 10 CFR 50.36(d)(2)(ii) to address the functional requirements for the Caldon LEFM CheckPlus feedwater flow element inputs to the heat balance calorimetric algorithm.

I.9 REFERENCES

I-1 ER-80P, "Improving Thermal Power Accuracy and Plant Safety While Increasing Operating Power Level Using the LEFM Check System," dated March 1997 approved by NRC SER, dated March 8, 1999

I-2 ER-157P, "Supplement to Topical Report ER-80P: Basis for a Power Uprate with the LEFM Check or CheckPlus System," Revision 5, dated October 2001, approved by NRC SER, dated December 20, 2001

I-3 Letter from B.E. Thomas (NRC) to Mr. E.M. Hauser (Caldon, Inc.), dated July 5, 2006, "Evaluation of the Hydraulic Aspects of the Caldon Leading Edge Flow Measurement (LEFM) Check and CheckPlus Ultrasonic Flow Meters (UFMs) (TAC No. MC6424)," Project No. 1311

I-4 ER-551P, "LEFM CheckPlus Transducer Installation Sensitivity," Revision 3, dated April 2008

I-5 Customer Information Bulletin CIB125, Transducer (Re)Placement Uncertainty, dated April 23, 2007

1-6 NRC Regulatory Issue Summary 2002-03, Guidance on the Content of Measurement Uncertainty Recapture Power Uprate Applications, dated January 31, 2002

I-7 Instrument Society of America (ISA) S67.04

I-8 NRC Regulatory Guide 1.105, Setpoints for Safety Related Instrumentation, Revision 3, dated December 1999

I-9 ANSI/IEEE-ANS Std. 7-4.3.2. 1993, "IEEE Standard Criteria for Digital Computers in Safety Systems of Nuclear Power Generating Stations"

1-10 ASME NQA-2a-1990, "Quality Assurance Requirements for Nuclear Facility Applications"

1-11 EPRI TR-103291s, "Handbook for Verification and Validation of Digital Systems," December 1994

II. ACCIDENTS AND TRANSIENTS FOR WHICH THE EXISTING ANALYSES OF RECORD BOUND PLANT OPERATION AT THE PROPOSED INCREASED POWER LEVEL

INTRODUCTION

The reactor core and/or NSSS thermal power are used as inputs to most plant safety, component, and system analyses. These analyses generally model the core and/or NSSS thermal power in one of three ways.

First, some Calvert Cliffs analyses apply an explicit 2% increase to the initial condition power level to account solely for the power measurement uncertainty. These analyses have not been re-performed for the requested MUR power uprate conditions because the sum of increased core power level and the decreased power measurement uncertainty falls within the previously analyzed conditions.

The power calorimetric uncertainty calculation described in Section I indicates that with the Caldon LEFM CheckPlus devices installed, the power measurement uncertainty (based on a 95% probability at a 95% confidence interval) is less than 0.6%. Therefore, these analyses only need to reflect a 0.6% power measurement uncertainty. Currently with the RTP of 2700 MWt, the analytical power level of 2754 MWt (102% of 2700 MWt) is used in the safety analysis. With a revised RTP of 2737 MWt and a revised uncertainty, the analytical power level is unchanged at 2754 MWt.

Second, some Calvert Cliffs analyses employ a nominal initial condition power level. These analyses have been evaluated for the increased power level with the MUR power uprate. The results demonstrate that the applicable analysis acceptance criteria continue to be met at the MUR power uprate conditions.

Third, some of the Calvert Cliffs analyses are performed at zero power initial conditions or do not actually model the core power level. Consequently, these analyses have not been re-performed for the proposed MUR power uprate since they are unaffected by the core power-level.

II.1 NUCLEAR STEAM SUPPLY SYSTEM ACCIDENT EVALUATION

The analyses referenced in Table II-1 are the AOR for Calvert Cliffs Units 1 and 2. These analyses do not change, that is, they continue to remain valid for the MUR power uprate.

The information in the table is organized to comply with Reference II-1. The first column contains the applicable Updated Final Safety Analysis Report (UFSAR) section. The second column identifies the transient, and columns three through six contain power and uncertainty information from the AOR, as well as confirmation that the AOR remains bounding with the MUR power uprate. Column seven provides the reference for the NRC's previous approval of the AOR, as well as an indication of type of approval. Approval types are either NRC SER or performed under 10 CFR 50.59. The final column elaborates briefly on the impact of the power uprate on the AOR.

The sections that follow provide details of the safety analyses.

TABLE II-1
Impact of Power Uprate on the UFSAR Chapter 14 Accident Analyses

UFSAR SECTION/EVENT		AOR ASSUMPTIONS AND REFERENCES						NOTES
		RTP (MWt)	Uncert. (%)	Total Core Power (MWt)	Bounds MUR?	Reference Last NRC Approval	Reference 50.59/ AOR	
RIS 2002-03 Rqmnt→	A	B,C	B,C	B,C	B,C	D	D	
14.2	Control Element Assembly Withdrawal Event	2700	±2	2754	Yes	II-2	II-3, II-4	Re-analyzed for thermal margin credits seen with TURBO fuel. MUR has no impact.
14.3	Boron Dilution Event	{ _ }	{ _ }	{ _ }	{ _ }	**	II-3	Not effected by an increase RTP. Analysis based on boron concentrations and RCS volumes which are unchanged for power uprate.
14.4	Excess Load Event	2700	±2	2754	Yes	II-5	II-3, II-6	Re-analyzed for thermal margin credits seen with TURBO fuel. MUR has no impact.
14.5	Loss of Load Event	2700	±2	2754	Yes	II-2	II-7, II-8	Evaluated for impact of MUR. Existing AOR plus uncertainty bounds the MUR total core power.
14.6	Loss of Feedwater Flow Event	2700	±2	2754	Yes	II-9	II-7, II-10	Evaluated for impact of MUR. Existing AOR plus uncertainty bounds the MUR total core power.
14.7	Excess Feedwater Heat Removal Event	2700	±2	2754	Yes	II-5	II-3, II-6	Re-analyzed as a sub-set of the Excess Load event. MUR has no impact.
14.8	Reactor Coolant System Depressurization	2700	±2	2754	Yes	II-5	II-7, II-11	Evaluated for impact of MUR. Existing AOR plus uncertainty bounds the MUR total core power.
14.9	Loss-of-Coolant Flow Event	2700	±2	2755	Yes	II-2	II-3, II-12	Re-analyzed for thermal margin credits seen with TURBO fuel. MUR has no impact.
14.10	Loss-of-Non-Emergency AC Power	2700	±2	2754	Yes	II-5	II-7, II-13	Evaluated for impact of MUR. Existing AOR plus uncertainty bounds the MUR total core power.

TABLE II-1
Impact of Power Uprate on the UFSAR Chapter 14 Accident Analyses

UFSAR SECTION/EVENT		AOR ASSUMPTIONS AND REFERENCES						NOTES
		RTP (MWt)	Uncert. (%)	Total Core Power (MWt)	Bounds MUR?	Reference Last NRC Approval	Reference 50.59/ AOR	
RIS 2002-03 Rqmnt→	A	B,C	B,C	B,C	B,C	D	D	
14.11	Control Element Assembly Drop Event	2700	±2	2754	Yes	II-5	II-3, II-14	Re-analyzed for thermal margin credits seen with TURBO fuel. MUR has no impact.
14.12	Asymmetric Steam Generator Event	2700	±2	2754	Yes	II-5	II-3, II-7, II-15	Re-analyzed for thermal margin credits seen with TURBO fuel. MUR has no impact.
14.13	Control Element Assembly Ejection	2700	±2	2754	Yes	II-5	II-3, II-16	Evaluated for impact of MUR. Existing AOR plus uncertainty bounds the MUR total core power.
14.14	Steam Line Break Event	2700	±2	2754	Yes	II-2	II-3, II-15, II-17, II-18, II-19, II-20, II-21	Pre-trip portion re-analyzed for thermal margin credits seen with TURBO fuel. Post-trip re-analyzed for cycle specific credits. MUR has no impact on either portion of the event.
14.15	Steam Generator Tube Rupture Event	2700	±2	2754	Yes	II-2	II-15, II-22	Evaluated for impact of MUR. Existing AOR plus uncertainty bounds the MUR total core power.
14.16	Seized Rotor Event	2700	±2	2754	Yes	II-5	II-3, II-23	Re-analyzed for thermal margin credits seen with TURBO fuel. MUR has no impact.
14.17	Loss-of-Coolant Accident	2700	±2	2754	Yes	II-24	II-3, II-7, II-25, II-26	Evaluated for impact of MUR. Existing AOR plus uncertainty bounds the MUR total core power.
14.17.2	Large Break LOCA	2700	±2	2754	Yes	II-24	II-3, II-7, II-26	
14.17.3	Small Break LOCA	2700	±2	2754	Yes	II-24	II-3, II-7, II-25	

TABLE II-1
Impact of Power Uprate on the UFSAR Chapter 14 Accident Analyses

UFSAR SECTION/EVENT		AOR ASSUMPTIONS AND REFERENCES						NOTES
		RTP (MWt)	Uncert. (%)	Total Core Power (MWt)	Bounds MUR?	Reference Last NRC Approval	Reference 50.59/ AOR	
RIS 2002-03 Rqmnt→	A	B,C	B,C	B,C	B,C	D	D	
14.18	Fuel Handling Incident	2700	±2	2754	Yes	**	II-27, II-28	Evaluated for impact of MUR. Radionuclide inventories based upon 2754 MWt. Existing AOR plus uncertainty bounds the MUR total core power.
5.3.1.2	Turbine-Generator Overspeed Incident	{ _ }	{ _ }	{ _ }	{ _ }	**	II-29, II-30	Not effected by power increase. Analysis based on pitching turbine blades.
14.20	Containment Response	2700	±2	2754	Yes	**	II-31	Evaluated for impact of MUR. Existing AOR plus uncertainty bounds the MUR total core power.
14.21	Hydrogen Accumulation in Containment	{ _ }	{ _ }	{ _ }	{ _ }	**	No Longer Analyzed for Chapter 14	A change to the Calvert Cliffs Technical Specifications removed this incident.
14.22	Waste Gas Incident	2700	±2	2754	Yes	**	II-32	Evaluated for impact of MUR. Radionuclide inventories based upon 2754 MWt. Existing AOR plus uncertainty bounds the MUR total core power.
14.23	Waste Processing System Incident	2700	±2	2754	Yes	**	II-32	Evaluated for impact of MUR. Radionuclide inventories based upon 2754 MWt. Existing AOR plus uncertainty bounds the MUR total core power.
14.24	Maximum Hypothetical Accident	2700	±2	2754	Yes	**	II-33	Evaluated for impact of MUR. Radionuclide inventories based upon 2754 MWt. Existing AOR plus uncertainty bounds the MUR total core power.

TABLE II-1
Impact of Power Uprate on the UFSAR Chapter 14 Accident Analyses

UFSAR SECTION/EVENT		AOR ASSUMPTIONS AND REFERENCES						NOTES
		RTP (MWt)	Uncert. (%)	Total Core Power (MWt)	Bounds MUR?	Reference Last NRC Approval	Reference 50.59/ AOR	
RIS 2002-03 Rqmnt→	A	B,C	B,C	B,C	B,C	D	D	
14.25	Excessive Charging Event	{ _ }	{ _ }	{ _ }	{ _ }	**	II-15, II-34	Not affected by power increase. Evaluated to assure that the operator has at least 15 minutes from initiation of high pressure level alarm to take corrective action and terminate the event prior to filling the pressurizer solid.
14.26	Feed line Break Event	2700	±2	2754	Yes	II-2	II-3, II-35	Evaluated for impact of MUR. Existing AOR plus uncertainty bounds the MUR total core power.

** - Not applicable for reference to previous NRC review

II.2 NON-LOSS-OF-COOLANT ACCIDENT/TRANSIENT ANALYSES

All of the UFSAR Chapter 14 non-LOCA transient analyses were evaluated for increase in RTP due to the MUR power uprate. The analyses include the NSSS response with replacement SGs (References II-7 and II-11). Replacement SGs decreased the number of plugged SG tubes, which in turn increased RCS flow. For some events, the original SG results are reported in the UFSAR because they are representative of the replacement SG results.

Many of the events were reanalyzed for thermal margin credit associated with TURBO fuel and these events included a target power uprate of 2746 MWt (1.7%), which bounds the proposed change in RTP to 2737 MWt. The events that use the target of 2746 MWt include a 0.3% uncertainty. The uprated RTP with uncertainty is equivalent to the pre-uprate total core power, which is 2754 MWt.

In the evaluation of the remaining events (those not reanalyzed for TURBO fuel), the existing assumption on core power plus uncertainty bounds the MUR power uprate. For all events, no changes to the Reactor Protective System (RPS) or Engineering Safety Features were assumed or were necessary.

The evaluation of the UFSAR Chapter 14 non-LOCA transient analyses concludes that the current analyses are applicable for Calvert Cliffs with the MUR power uprate.

II.2.1 Control Element Assembly Withdrawal Event (UFSAR 14.2)

A failure in either the Control Element Assembly (CEA) Drive Mechanism Control System or the Reactor Regulating System may initiate a sequential bank withdrawal, inserting positive reactivity and causing increases in reactor power, RCS temperature, and RCS pressure. The event is terminated by either the Variable High Power Trip, the High Pressurizer Pressure Trip, the Thermal Margin/Low Pressure (TM/LP) Trip, or the insertion of negative reactivity due to Doppler and negative Moderator Temperature Coefficient (MTC) feedbacks.

The current AOR for the CEA Withdrawal Event is analyzed and documented in Reference II-4. In support of the MUR power uprate, this referenced analysis is performed with the assumption of a rated power of 2746 MWt plus uncertainties, which bounds the MUR power uprate power level of 2737 MWt plus uncertainties. This re-analysis also implements the Asea Brown Boveri, Inc.-Turbo Vane (ABB-TV) correlation for critical heat flux (approved in Reference II-36), and makes all appropriate input and assumption adjustments associated with both ABB-TV and the MUR power uprate. Approved methodologies and codes (References II-2, II-37, II-38, and II-39) were used, along with approved associated limits/constraints and acceptance criteria. As with all applicable UFSAR Chapter 14 analyses, associated with implementation of the ABB-TV critical heat flux correlation was a change in the departure from nucleate boiling (DNB) specified acceptable fuel design limits (SAFDL) to a value of 1.24, determined by application of extended statistical combination of uncertainties (Reference II-40). This value is acceptable in relation to the NRC-approved minimum departure from nucleate boiling ratio (DNBR) value of 1.13 associated with the approved methodologies of this analysis. Assuming a rated full power level of 2746 MWt plus uncertainties and implementing ABB-TV, all acceptance criteria are met with respect to DNBR, peak linear heat generation rate (PLHGR), maximum primary and secondary pressure and radiological consequence. The analysis for CEA withdrawal is acceptable relative to applicable SERs and bounds operation at the proposed MUR power uprate power level of 2737 MWt plus uncertainties.

II.2.2 Boron Dilution Event (UFSAR 14.3)

A Boron Dilution Event is defined as any event caused by a malfunction or an inadvertent operation of the Chemical and Volume Control System (CVCS) that results in a dilution of the active portion of the

RCS. The analysis of this event covers all six modes of operation, each mode being associated with a required minimum time to lose required shutdown margin. This analysis, most recently documented in Reference II-3 for the current operating conditions, is unaffected by the proposed MUR power uprate. The analysis is based on RCS and CVCS volumes, along with the boron concentration, to show that operator action within the required minimum time period will terminate the dilution prior to violating the assumed parameters for shutdown margin. The boron dilution event assumes boron concentration levels associated with operating modes, which continues to bound the MUR power uprate power level of 2737 MWt plus uncertainties.

II.2.3 Excess Load Event (UFSAR 14.4)

An Excess Load Event, as documented in the Calvert Cliffs UFSAR, is a rapid uncontrolled increase in SG steam flow not caused by a Steam Line Break (SLB). In the assumed presence of a negative MTC and Fuel Temperature Coefficient, positive reactivity addition leads to an increase in core power level, decreasing DNBR and linear heat rate (LHR) margin. The transient continues until the Variable High Power Trip is reached on neutron flux or core temperature differential (ΔT), terminating the event. The limiting scenario is most likely to be caused by a full opening of the turbine control valves, atmospheric dump valves (ADVs), or turbine bypass valves during steady-state operation. Limiting cases are determined at both Hot Full Power (HFP) and Hot Zero Power (HZP).

The current AOR, as documented in Reference II-6, bounds operation at the MUR power uprate power level of 2737 MWt plus uncertainties. That AOR also has been verified to use approved methodologies and codes, along with all associated limits and conditions as prescribed by associated SERs (References II-5, II-36, and II-39). The current AOR at HFP assumes an initial reactor thermal power of 2754.2 MWt, including uncertainties. This thermal power level bounds the proposed MUR power uprate power level of 2737 MWt, plus uncertainties. All criteria for acceptance are met with respect to DNBR, PLHGR, pressure limits, and radiological consequence. The Excess Load Event AOR bounds operation at the proposed MUR power uprate power level of 2737 MWt plus uncertainties.

II.2.4 Loss of Load Event (UFSAR 14.5)

As defined in UFSAR Section 14.5, a Loss of Load Event is defined as any event that results in a reduction in the SGs' heat removal capacity through a loss of secondary steam flow. Such an event could be caused by a closure of all main steam isolation valves (MSIVs), turbine stop valves, or turbine control valves along the steam flow path between the SGs and the high pressure turbine. The most limiting Loss of Load Event is a turbine trip without concurrent reactor trip, or an inadvertent closure of the turbine stop valves at HFP.

Reference II-8 provides a bounding AOR for both Calvert Cliffs Units 1 and 2. The assumed power level for transient initiation at HFP is 2771 MWt, which includes a 2.0% instrument uncertainty and a conservative assumption of an additional 17 MWt for RCP heat. This assumed power level in the analysis of 2771 MWt bounds the proposed operation at an MUR power uprate power level of 2737 MWt and the power measurement uncertainty. All assumptions and methodologies associated with and documented in the AOR are consistent with previously approved analyses and associated SERs and limitations/conditions for application (Reference II-39 for CESEC-III). All acceptance criteria were found to be met for the bounding analysis with respect to DNBR, fuel performance, peak pressures (RCS and secondary), and radiological consequence. This analysis, having been performed at HFP with a thermal power of 2771 MWt (including uncertainty and RCP heat), bounds operation at the proposed MUR power uprate power level of 2737 MWt plus uncertainties.

II.2.5 Loss of Feedwater Flow Event (UFSAR 14.6)

A Loss of Feedwater Flow Event is defined as a reduction or loss of feedwater to the SGs without a corresponding reduction in steam flow from the SGs. The most limiting Loss of Feedwater initiating event is determined to be an inadvertent instantaneous closure of the feedwater regulating valves, which results in the largest steam and feedwater flow mismatch and the most rapid reduction in SG inventory. The transient causes an increase in primary and secondary pressures and is ultimately terminated by the High Pressurizer Pressure Trip or the Low SG Level Trip to ensure that all acceptance criteria are met.

The current AOR for this event was documented in References II-7 and II-10 and was reviewed and accepted by the NRC as documented in Reference II-9. The assumed initial power for the transient is 2771 MWt (including uncertainties and RCP energy), which bounds operation at the MUR power uprate power level of 2737 MWt plus uncertainties. All acceptance criteria were met with respect to DNBR, PLHGR, peak pressures, SG inventory, and radiological consequences. The current AOR for Loss of Feedwater Flow bounds operation at the proposed MUR power uprate power level of 2737 MWt plus uncertainties.

II.2.6 Excess Feedwater Heat Removal Event (UFSAR 14.7)

The Excess Feedwater Heat Removal Event results from an extraction of excessive heat from the RCS through the SGs caused by a reduction in SG feedwater temperature without a corresponding reduction in steam flow from the SGs. The limiting circumstance of a loss of both high pressure feedwater heaters, coupled with the presence of a conservatively negative MTC and Fuel Temperature Coefficient, results in a core power increase due to the corresponding decrease in RCS temperature. This reactor power increase causes the system to approach the SAFDLs, and is ultimately mitigated by the Variable High Power Trip.

This analysis is documented as an Appendix to the Excess Load analysis of Reference II-6 and discussed in Reference II-3. This analysis is bounded by the inputs and results of the AOR for the Excess Load Event. As the Excess Load Event has already been determined to bound operation at the proposed MUR power uprate power level of 2737 MWt, the Excess Feedwater Heat Removal Event is also bounded by the current AOR. Bounding inputs with respect to initial reactor core power level (2754.2 MWt, including uncertainties), and associated methodologies, are identical to those discussed for the Excess Load Event. As such, the current AOR for the Excess Feedwater Heat Removal Event bounds operation at the proposed MUR power uprate power level of 2737 MWt plus uncertainties.

II.2.7 Reactor Coolant System Depressurization Event (UFSAR 14.8)

The RCS Depressurization Event is considered an Anticipated Operational Occurrence (AOO) for which action of the RPS is required to prevent SAFDL violation. The event is initiated by assuming the inadvertent opening of both power-operated relief valves, resulting in a rapid depressurization of the RCS. The analysis shows that action of the RPS by way of the TM/LP trip prevents exceeding the associated SAFDLs, particularly DNBR.

As stated in the AOR, the assumed initial core power does not affect the results of the event. However, the documented AOR (Reference II-11, justifying results of Reference II-41 with replacement SGs) is performed with an assumed initial reactor core power level of 2771 MWt (including uncertainties and RCP energy), which bounds operation at the proposed MUR power uprate power level of 2737 MWt plus uncertainties.

II.2.8 Loss of Coolant Flow Event (UFSAR 14.9)

The Loss of Coolant Flow Event is classified as an AOO for which RPS trips and/or sufficient initial steady-state thermal margin, maintained by the applicable Technical Specifications, are necessary to prevent exceeding acceptable limits. This transient event is initiated from a HFP condition and modeled to envelope the occurrence of two separate postulated scenarios for losing power to the RCPs: a complete loss of alternating current (AC) to the plant, and a failure of the fast transfer breakers to close following an assumed trip of the main generator. The intermediate system response to the RCP coast down is a rapid decrease in coolant mass flow rate through the reactor core, causing a rise in enthalpy across the core in the direction of coolant flow. A relatively slight power increase results, due to the assumed presence of a positive MTC. The main concern with respect to SAFDLs for this event is DNBR, which is met in the analyses (Reference II-12) by ensuring that initial steady-state margin is built into the DNB design operating limit such that, in conjunction with crediting of the low flow trip function, the DNBR SAFDL is not exceeded.

The Loss of Coolant Flow Event AOR (Reference II-12) credits the thermal margin gains associated with implementation of TURBO fuel. For this analysis, the maximum core power with uncertainties applied was 2755 MWt, (2746 MWt plus uncertainties). The analyzed maximum power level of 2755 MWt bounds the proposed MUR power uprate power level of 2737 MWt, plus uncertainties and rounded up. Methodologies associated with this analysis were verified to be consistent and within the limitations and conditions of associated SERs (References II-5, II-36, II-42, II-43, and II-44) and previously NRC-approved analyses (Reference II-2).

All acceptance criteria were met with respect to DNBR, PLHGR, peak pressures, and radiological consequence. The maximum analyzed power level of 2755 MWt (including uncertainties) and assumed RTP of 2746 MWt bound operation at the proposed MUR power uprate power level of 2737 MWt plus uncertainties.

II.2.9 Loss of Non-Emergency AC Power Event (UFSAR 14.10)

The Loss of Non-Emergency AC Power Event involves a loss of electrical power to RCPs, resulting in an RCS flow coast down that challenges SAFDLs and yields an increased steam release to the atmosphere via the main steam safety valves (MSSVs) and ADVs. With respect to DNBR and PLHGR, this event is bounded by the Loss of Coolant Flow Event described above and documented in References II-7 and II-12. Loss of Coolant Flow has been verified to bound operation at the MUR power uprate power level, with use of applicable approved codes, methodologies, and limitations/constraints.

The Loss of Non-Emergency AC Power was evaluated and documented as an AOR in Reference II-13. An explicit analytical calculation was not performed for the reanalysis, but the documented AOR justifies the results of the previous AOR for operation with the replacement SGs. The analysis was performed at an initial power level of 2754 MWt, including uncertainties, which bounds the proposed MUR power uprate power level of 2737 MWt, plus uncertainties. As previously stated, all SAFDL limits, including DNBR, are bounded by the Loss of Coolant Flow Event. Additionally, the peak pressures associated with Loss of Non-Emergency AC are bounded by the results of the Loss of Load Event (also discussed above). Results of the AOR (Reference II-13) meet all applicable criteria and are verified to be produced by NRC-approved methodologies in accordance with applicable SERs. Current analysis for the Loss of Non-Emergency AC Power bounds operation at the proposed MUR power uprate power level of 2737 MWt plus uncertainties.

II.2.10 Control Element Assembly Drop Event (UFSAR 14.11)

The CEA Drop Event entails the drop of a single full length CEA into the core, reducing fission power in the vicinity of the dropped CEA and adding negative reactivity core-wide. A prompt drop in core power and heat flux results from the negative reactivity insertion, the magnitude of which depends on the reactivity worth of the dropped CEA. Assuming an inoperable turbine runback circuit, the resulting power mismatch between the primary and secondary systems leads to a cooldown of the RCS and a subsequent positive reactivity addition due to the effects of a negative MTC. Doppler reactivity and moderator feedbacks ultimately terminate the reactivity excursion, producing a re-stabilized core condition with an asymmetric power distribution and correspondingly higher peaking factors. Criteria with respect to DNB, PLHGR and radiological consequence must be shown analytically to be met.

A new AOR was established for both Units 1 and 2 with References II-3 and II-14. The AOR implements the methodologies associated with TURBO fuel and ensures bounding inputs and results for the anticipated MUR power uprate. Rated power for this event is assumed to be 2746 MWt, and the maximum initial power including uncertainties is 2754.2 MWt, which bounds operation at the proposed MUR power uprate power level of 2737 MWt plus uncertainties. The performance of this analysis has been verified by the vendor and Calvert Cliffs to have been done in accordance with all applicable SERs (References II-5, II-36, and II-39) and limitations/conditions. All results are shown to be acceptable with respect to the acceptance criteria for DNBR, PLHGR, peak pressures, and radiological consequence. The current AOR for the CEA Drop Event bounds operation at the proposed MUR power uprate power level of 2737 MWt plus uncertainties.

II.2.11 Asymmetric Steam Generator Event (UFSAR 14.12)

The Asymmetric SG Event is classified as an AOO, described as a rapid imbalance in heat transfer between the two SGs, initiated by one of the following: a loss of load to one SG, excessive increase in load to one SG, loss of feedwater to one SG, or excessive feedwater flow increase to one SG. The limiting cause evaluated for the current AOR at Calvert Cliffs is a loss of load to one SG, caused by instantaneous closure of one of two MSIVs. This circumstance produces the most rapid temperature tilt across the core, resulting in a limiting approach to the DNBR SAFDL for this analysis.

The current bounding AOR for Units 1 and 2 is documented in Reference II-45. This revision to the AOR explicitly addresses the implementation of TURBO fuel, ABB-TV critical heat flux correlation, and the MUR power uprate. Rated power for this analysis is assumed to be 2746 MWt, and the maximum initial power, including uncertainties, for the analysis is assumed to be 2754.2 MWt. Methodologies and codes associated with this analysis are verified to be consistent and within the limitations and conditions of associated SERs (References II-5, II-36, and II-39) and previously NRC-approved analyses (Reference II-5). All acceptance criteria were met with respect to DNBR, PLHGR, peak pressures, and radiological consequence. As such, the current AOR for the Asymmetric SG Event bounds operation at the proposed MUR power uprate power level of 2737 MWt plus uncertainties.

II.2.12 Control Element Assembly Ejection Event (UFSAR 14.13)

The CEA Ejection Event results from a postulated complete circumferential break of the control element drive mechanism (CEDM) housing or of the CEDM nozzle on the RV head. The analysis is performed from postulated HFP and HZP initial conditions, each resulting in a rapid core power increase for a brief period of time. Doppler reactivity feedback inhibits the core reactivity and power rise, and the reactor is ultimately shutdown by a high power level trip, thereby terminating the transient. The core is protected from fuel damage by CEA insertion limits associated with various power levels (Power-Dependent Insertion Limit of the Technical Specifications) and the high power trip. Being a postulated event, a

small fraction of fuel failure is permitted in the analysis within the restrictions of criteria for acceptance placed on deposited energy limits and offsite radiological consequence.

The current bounding AOR for Units 1 and 2 at Calvert Cliffs is documented in Reference II-16. This revision to the AOR explicitly addresses the implementation of Zirconium Diboride (ZrB_2) fuel with axial blankets, as well as ZIRLO™ cladding and encompasses the MUR power uprate. Rated power for this analysis is assumed to be 2746 MWt, and the maximum initial power, including uncertainties, for the analysis is assumed to be 2754 MWt. Methodologies and codes associated with this analysis are verified to be consistent and within the limitations and conditions of associated SERs (References II-42, II-46, II-47, and II-48) and the NRC-approved analysis (Reference II-5). All acceptance criteria were met with respect to fuel clad failure and radiological consequence. As such, the current bounding AOR for the CEA Ejection Event bounds operation at the proposed MUR power uprate power level of 2737 MWt plus uncertainties.

II.2.13 Steam Line Break Event (UFSAR 14.14)

A SLB Event is defined as a breach in the Main Steam piping that carries steam from the SGs to the turbine-generator and other equipment. That breach in the main steam piping produces an increase in heat extraction by the SGs, causing a cooldown of the RCS. In the presence of an assumed negative MTC, that RCS cooldown leads to an addition of positive reactivity to the RCS. The transient is terminated by a reactor trip associated with the severe decrease in SG pressure, and the MSIVs in the main steam line close to isolate steam flow from the affected SG. The SLB Event is divided analytically into two separate phases, pre-trip and post-trip for separate safety concerns and associated evaluation against respective acceptance criteria. The primary concern in the pre-trip SLB analysis is the power excursion related to the RCS cooldown and the assumed negative MTC. A loss of power coincident with reactor trip is also assumed. A limiting combination of break size and MTC is determined parametrically for SLBs both inside and outside Containment during the pre-trip SLB analysis. The primary concern associated with the post-trip analysis is a return-to-power in the vicinity of an assumed stuck control rod. Limiting scenarios with respect to DNBR are determined parametrically for HFP and HZP initial conditions, both with and without loss of power.

The critical heat flux correlation utilized in the SLB analysis is the MACBETH correlation, NRC-approved in Reference II-49. Associated with that documented SER is a minimum DNBR limit of 1.30 for the MACBETH critical heat flux correlation. Additional applicable SERs for the SLB and this discussion are References II-39 and II-50.

The current AOR for the pre-trip SLB was established in References II-15 and II-19. The maximum initial power level at event initiation assumed in that analysis is 2754.2 MWt including uncertainties, which bounds the proposed MUR power uprate power level of 2737 MWt (plus uncertainties) at Calvert Cliffs. This AOR for the pre-trip SLB credits the thermal margin gains associated with TURBO fuel and the ABB-TV critical heat flux correlation, and bounds the proposed MUR power uprate operation. Acceptance criteria with respect to DNBR, PLHGR, peak pressures and radiological consequence are all met. The current AOR for the pre-trip SLB bounds operation at the proposed MUR power uprate power level of 2737 MWt plus uncertainties.

The post-trip SLB Event is currently analyzed separately for each operating cycle to credit the cycle-specific physics input to the analysis. The current AORs for Units 1 and 2 are documented in References II-20 and II-21, respectively. The AORs for the two operating units employ the MACBETH critical heat flux correlation (design DNBR $\geq$ 1.30, SER Reference II-49). Each current AOR for the post-trip SLB is also performed with an assumed rated power level of 2746 MWt and a maximum initial total power of 2754 MWt including uncertainties. This power level input assumption bounds the

proposed operation at the MUR power uprate power level of 2737 MWt. The AOR for Units 1 and 2 (References II-20 and II-21) also bound operation with ZrB_2 Integral Fuel Burnable Absorber (IFBA) in conjunction with axial blankets. All applicable restrictions, limits and conditions associated with the respective methodologies, codes, and correlations are met within the bounds of respective appropriate SER references. The current AOR for the post-trip SLB bounds operation at the proposed MUR power uprate power level of 2737 MWt plus uncertainties.

II.2.14 Steam Generator Tube Rupture Event (UFSAR 14.15)

The SG Tube Rupture Event is a breach of the barrier between the RCS and the main steam system (MSS), resulting in mass transfer between the primary and secondary systems and, more consequentially, a radiological release to the environment through the MSSVs and the ADVs.

The current bounding AOR for the SG Tube Rupture Event is documented in Reference II-22. As the primary concern associated with this analysis is radiological consequence, a reanalysis was not explicitly performed for TURBO fuel implementation. The AOR is documented as bounding in terms of affected neutronic parameters (e.g., Scram curves) for implementation of ZrB_2 IFBAs in conjunction with axial blankets (Reference II-42). The AOR is also supported by Reference II-51 with regard to justifying parameter assumptions related to proportional and backup heater nominal heat rates, MSSV setpoints, and charging pump flow. The assumed maximum power level at initiation of the transient from HFP conditions in the AOR is 2754 MWt, which bounds operation at the MUR power uprate power level of 2737 MWt, plus uncertainties. The SG Tube Rupture event as documented in the current AOR bounds operation at the proposed MUR power uprate power level including uncertainties, and meets the requirements, limitations and conditions associated with all applicable SERs.

II.2.15 Seized Rotor Event (UFSAR 14.16)

The Seized Rotor Event is classified as a postulated event, for which a limited amount of fuel failure is permitted within the bounds of associated acceptance criteria. The transient event is caused by an instantaneous seizure of a RCP shaft, postulated to occur as a result of mechanical failure or a loss of component cooling water to the RCP shaft seals. The flow rate rapidly reduces to a value corresponding to three RCPs, as opposed to four. The corresponding reduction in RCS flow rate causes a reactor trip on low RCS flow. The reduction of RCS flow rate results in a degradation of DNBR with respect to the SAFDL.

Reference II-23 documents the current AOR for the Seized Rotor Event. The AOR credits the thermal margin benefits of TURBO fuel, as realized by application of the ABB-TV critical heat flux correlation in conjunction with CETOP-D (References II-36 and II-38). The effects of implementation of ZrB_2 fuel in conjunction with axial blankets are evaluated in Reference II-52. The assumed maximum power level for the currently bounding AOR is 2754 MWt, which bounds the proposed operation following MUR power uprate of 2737 MWt, plus uncertainties. As all acceptance criteria with respect to DNBR, PLHGR, peak pressures, and radiological consequence are met within the restrictions, limitations, and constraints of NRC-approved methodologies and codes, the Seized Rotor Event as currently analyzed bounds operation at the proposed MUR power uprate power level of 2737 MWt plus uncertainties.

II.2.16 Fuel Handling Incident (UFSAR 14.18)

The Fuel Handling Incident analysis assumes that a fuel assembly is dropped during fuel handling, either in the Containment or in the Spent Fuel Pool. The results of this analysis are dependent upon the radionuclide inventory assumed for the dropped fuel assembly. The inventories associated with this analysis have been generated based on an assumption of core operating power of 2754 MWt, and source term values are based on the TID-14844 methodology in accordance with Regulatory Guide 1.25.

Therefore, the current analysis relating to the Fuel Handling Incident bounds operation at the proposed MUR power uprate power level of 2737 MWt plus uncertainties.

Reference II-66 approves Technical Specification changes requested in Reference II-67 associated with the implementation of the alternative radiological source term (AST). The AST methodology replaces the existing accident radiological source term that is described in TID-14844. The Fuel Handling Incident was reanalyzed using AST and the AOR, documented in Reference II-68, assumed the core isotopic inventory is based upon a maximum full power operation of 254 MWt. Calvert Cliffs expects to switch to the AST methodology for the Fuel Handling Incident in the year 2010, and since the reanalysis of the Fuel Handling Incident was performed assuming operation at 2754 MWt, the reanalysis bounds operation at the proposed MUR power uprate power level of 2737 MWt plus uncertainties.

II.2.17 Turbine-Generator Overspeed Incident (UFSAR 5.3.1.2)

The Turbine-Generator Overspeed Incident is an analyzed event based on the failure of rotating elements of the steam-turbines and generators. This analysis is not a Design Basis Event (DBE) or AOO and is documented in detail in UFSAR Section 5.3.1.2. The thermal power increase related to the MUR power uprate does not impact the results of this analysis. As such, the Turbine-Generator Overspeed Incident bounds operation at the proposed MUR power uprate power level of 2737 MWt plus uncertainties.

II.2.18 Hydrogen Accumulation in Containment

This analysis has been deleted from the UFSAR per License Amendment Nos. 262/239. Reanalysis is not required to verify that the analysis bounds operation at the proposed MUR power uprate power level of 2737 MWt plus uncertainties.

II.2.19 Waste Gas Incident (UFSAR 14.22)

The limiting Waste Gas Incident analyzed for UFSAR Chapter 14 is an uncontrolled and unexpected release to the atmosphere of radioactive xenon and krypton fission gases stored in one waste decay tank. The assumed maximum activity, in accordance with Reference II-32, is determined based on conditions in the waste gas decay tank shortly after plant heatup and startup after cold shutdown conditions near the end of a 24-month operating cycle. Associated limiting activity levels are calculated in Reference II-32 with the assumption of constant full-power operation at 2754 MWt. Radiological consequence limits are met. The Waste Gas Incident bounds operation at the proposed MUR power uprate power level of 2737 MWt plus uncertainties.

II.2.20 Waste Processing System Incident (UFSAR 14.23)

The Waste Processing System Incident assumes a seismically-induced failure of the reactor coolant Waste Processing System whereby the contents of the system are released. Reference II-32, as discussed in Section II.2.19, contains the analysis for this event. As previously mentioned, the depletion calculations for generating radio-isotopic inventories for these analyses is performed at a core thermal power level of 2754 MWt. Therefore, the Waste Processing System Incident analysis bounds operation at the proposed MUR power uprate power level of 2737 MWt plus uncertainties.

II.2.21 Maximum Hypothetical Accident (UFSAR 14.24)

The results of this analysis demonstrate bounding compliance with the guidelines of 10 CFR Part 100. As stated in UFSAR Section 14.24, the pre-accident thermal power for the Maximum Hypothetical Accident is 2754 MWt. All methodologies and results are consistent with approved methodologies and previously submitted analyses. The documented Maximum Hypothetical Accident bounds operation at the proposed MUR power uprate power level of 2737 MWt plus uncertainties.

II.2.22 Excessive Charging Event (UFSAR 14.25)

The Excessive Charging Event is analyzed to verify compliance with the limits of Technical Specification 3.4.4, and to provide the basis for associated alarm setpoints. Specifically, the AOR, Reference II-34, verifies that operator action no sooner than 15 minutes following receipt of pressurizer high level alarm suffices to terminate the event without violating limits on pressurizer level. The associated analysis is based on RCS volumes and CVCS flow rates (letdown and charging). Reactor power level does not affect the results. The current AOR is bounding and acceptable with respect to the plant configuration (charging pump flows, installed pressurizer level setpoints, etc.), and remains valid for the power level associated with the proposed MUR power uprate, including uncertainties.

II.2.23 Feedline Break Event (UFSAR 14.26)

The Feedline Break Event is a postulated accident whereby a piping failure occurs downstream of the check valves between the SG and Containment. The affected SG empties, causing elevated temperatures in that SG and the RCS. A reactor trip occurs on either loss of SG Level or High Pressurizer Pressure, terminating the pressure transient in combination with the opening action of the pressurizer safety valves and MSSVs.

The AOR for the Feedline Break Event is contained in Reference II-35, and described in Reference II-52, and bounds operation under current and proposed MUR power uprate power levels. The maximum core power level assumed in the analysis as an input condition is 2771 MWt, including rated power plus uncertainties and RCP energy. All acceptance criteria for the event with regard to DNBR, peak RCS and secondary pressure limits, radiological consequence, and long-term cooling capability are verified to have been met. Additionally, all methodologies and code implementation are consistent with the most recent NRC-reviewed analysis documented in Reference II-2. Compliance with applicable SERs is verified for use of CESEC-III (Reference II-39). As all results and methodologies are acceptable, all results meet associated acceptance criteria, and the maximum initial power level exceeds the proposed MUR power uprate plus uncertainties, the results of the current Feedline Break Event AOR bound the proposed MUR power uprate power level of 2737 MWt plus uncertainties.

II.3 EMERGENCY CORE COOLING SYSTEM PERFORMANCE

The Calvert Cliffs Units 1 and 2 ECCS performance analysis consists of a large break loss-of-coolant accident (LBLOCA) and a small break loss-of-coolant accident (SBLOCA) analysis. Both analyses were performed at a core power level of 2754 MWt. Consistent with the original requirement of Paragraph I.A of Appendix K to 10 CFR Part 50, 2754 MWt is equal to 102% of the current licensed core power level, i.e., RTP of 2700 MWt.

The Calvert Cliffs Units 1 and 2 LBLOCA and SBLOCA analyses were performed with the 1999 Evaluation Model (EM) (Reference II-53) and Supplement 2 to CENPD-137 Evaluation Model (S2M) (Reference II-54) versions of the Westinghouse ECCS EMs for CE pressurized water reactors (PWRs). The SERs for the 1999 EM (Reference II-55) and the S2M (Reference II-56) generically approved the EMs for referencing in licensing applications for CE designed PWRs. The two EMs were specifically accepted for Calvert Cliffs Units 1 and 2 as allowed analytical methods for use in determining core operating limits in Reference II-57. A summary of the Calvert Cliffs LBLOCA and SBLOCA analyses using the 1999 EM and the S2M was provided to the NRC in Reference II-58. Detailed descriptions of the analyses are contained in Calvert Cliffs UFSAR Section 14.17.

As allowed by Paragraph I.A of Appendix K, Calvert Cliffs Nuclear Power Plant proposes to increase the licensed core power level and decrease the power measurement uncertainty such that the analytical core

power level, after accounting for the new power measurement uncertainty, remains equal to 2754 MWt. Since the Calvert Cliffs Units 1 and 2 ECCS performance analysis was performed at an analytical core power level of 2754 MWt, it complies with Paragraph I.A of Appendix K for the proposed values for the licensed core power level and power measurement uncertainty.

A review of the impact that the proposed increase in licensed core power level (2737 MWt) has on the Calvert Cliffs Unit 1 values for plant data used in the Calvert Cliffs Units 1 and 2 ECCS performance analyses concluded that the increase in power does not affect the applicability of the analysis to Calvert Cliffs Unit 1 under the MUR power uprate conditions.

The analyses and evaluations were performed for Calvert Cliffs Units 1 and 2. In some cases where cycle specific data is needed the analyses/evaluations targeted Unit 1 as the lead unit for the MUR power uprate. Consequently, for Calvert Cliffs Unit 1, there are no changes to the peak cladding temperature or any other result of the Calvert Cliffs Units 1 and 2 ECCS performance analyses as a consequence of the proposed changes to the licensed core power level and power measurement uncertainty. Confirmation of the applicability of the analyses and evaluations on future cycles of Unit 2, and subsequent cycles of Unit 1, will be performed as part of the normal reload design process.

The 1999 EM and the S2M EMs consist, in part, of topical reports that were written prior to the revision to Paragraph I.A of Appendix K. Some of those earlier topical reports contain statements that the analyses will use 102% of the licensed core power level. For example, Section III.A of CENPD-132P (Reference II-59) states that *"The reactor will be assumed to be operating at a power level of 102% of the maximum licensed power."* Subsequent to the revision to Paragraph I.A of Appendix K, the topical reports that comprise the LBLOCA and SBLOCA EMs were not amended to reflect the revision to Appendix K; i.e., sentences like the above were not revised. As identified in the Introduction Section, Calvert Cliffs requests that approval of this license amendment request constitutes approval to apply the EMs at the proposed core power level and power measurement uncertainty.

II.3.1 Loss-of-Coolant Accident (UFSAR 14.17)

The LOCA Analyses are performed in order to provide confirmation of the ECCS performance within the criteria listed in 10 CFR 50.46. The following two subsections address the AOR for both large break and small break LOCA with respect to the projected MUR power uprate power level of 2737 MWt.

II.3.1.1 LBLOCA

The current AOR bounding operation for Units 1 and 2, are found in Reference II-26. The methodology was generically approved by the NRC and documented in Reference II-24. The results of that analysis are applicable to the following plant configuration conditions:

- RTP (including measurement uncertainty) ≤ 2754 MWt
- Maximum integrated radial peaking factor, $F_{r,\ max}$, Core Operating Limits Report (COLR) limit of 1.65 (full power, all rods out operation)
- Full core representation of the TURBO fuel assembly design
- Value added fuel (VAP), ZIRLO™ clad, ZrB_2 IFBA, and UO_2 fuel rod designs operating at a PLHGR of 14.5 kw/ft with 2x6-inch low-enriched axial blankets with annular pellets
- Once-burned VAP ZIRLO™ clad Erbia fuel rod designs operating at 14.0 kW/ft PLHGR
- SGs with ≤ 10% tube plugging

This bounding analysis employs the "1999 EM" version of Westinghouse's LBLOCA ECCS Performance Evaluation Model for Combustion Engineering designed Pressurized Water Reactors

(PWRs), as documented in Reference II-53 and NRC-approved in Reference II-55, conforming to the requirements associated with ZIRLO™ (SER Reference II-46) and ZrB_2 (SER Reference II-42).

The ECCS acceptance criteria of 10 CFR 50.46 are compared to the calculated results for the bounding LBLOCA analysis for any Calvert Cliffs operating cycle that meets the aforementioned applicability criteria.

Parameter	Criterion	Result
Peak Cladding Temperature	≤ 2200°F	2057°F
Maximum Cladding Oxidation	≤ 17%	9.95%
Maximum Core-Wide Oxidation	≤ 1%	< 0.99%
Coolable Geometry	Yes	Yes

All results for the bounding LBLOCA analysis are acceptable with respect to acceptance criteria applied by 10 CFR 50.46. The LBLOCA, as evinced by the foregoing discussion, is performed according to all applicable SERs and bounds operation at the proposed MUR power uprate power level of 2737 MWt plus uncertainties.

II.3.1.2 SBLOCA

The current AOR for SBLOCA applicable to Units 1 and 2 and future applicable Calvert Cliffs operating cycles is documented in Reference II-25 and discussed in Reference II-3. The results of Reference II-25 are applicable to the following plant configuration conditions:

- RTP (including measurement uncertainty) ≤ 2754 MWt
- TURBO fuel assembly design
- VAP, ZIRLO™ and Zircaloy-4 clad UO_2 fuel, with and without Erbia IFBA
- VAP, ZIRLO™ clad, ZrB_2 IFBA, and UO_2 fuel rod designs with 2x6-inch low-enriched axial blankets with annular pellets
- SGs with ≤ 10% tube plugging
- PLHGR of 15.0 kW/ft

This SBLOCA ECCS performance analysis is performed with the NRC-accepted S2M version of the Westinghouse CE SBLOCA EM (Reference II-56). As documented above for the LBLOCA for both units, the bounding AOR for SBLOCA complies with all limitations and conditions of applicable SERs, such as those associated with ZIRLO™ and ZrB_2.

The results demonstrate conformance for a bounding SBLOCA analysis (within the conditions of applicability) with respect to acceptance criteria of 10 CFR 50.46 as follows.

Parameter	Criterion	Result
Peak Cladding Temperature	≤ 2200°F	1855°F
Maximum Cladding Oxidation	≤ 17%	7.20%
Maximum Core-Wide Oxidation	≤ 1%	< 0.60%
Coolable Geometry	Yes	Yes

As the bounding SBLOCA analysis is found to comply with all SER limitations and conditions, and all acceptance criteria for 10 CFR 50.46 are met, the associated bounding LBLOCA AOR bounds operation at the proposed MUR power uprate of 2737 MWt plus uncertainties.

II.4 ANTICIPATED TRANSIENTS WITHOUT SCRAM

As noted in Reference II-60, Calvert Cliffs has installed a Diverse Scram System. The NRC concluded that the Diverse Scram System met the requirements of 10 CFR 50.62 in Reference II-61. Reference II-62 stated that the installation of the Diverse Scram System, diverse turbine trip, and diverse Auxiliary Feedwater Actuation System (AFAS), maintain the probability and consequences of an Anticipated Transients Without Scram (ATWS) as low, and eliminate the need to consider an ATWS as a DBE. Therefore, the proposed MUR power uprate does not adversely impact ATWS.

II.5 CONTAINMENT RESPONSE

The mass and energy transfer data for the limiting LOCA DBA is based on three types of LOCA DBAs; hot leg LOCA with minimum SI, cold leg LOCA with minimum SI, and cold leg LOCA with maximum SI. The limiting LOCA DBA is the cold leg LOCA with maximum SI. The limiting LOCA DBA assumes an initial reactor power of 102% (2754 MWt).

The mass and energy for the Main Steam Line Break (MSLB) DBA includes a spectrum of core power levels to determine the most limiting mass and energy transfer for containment peak pressure and temperature including 0%, 50%, 75%, and 102% power levels. The most limiting for MSLB DBA corresponds to a 75% power level.

Note that all other events that challenge the containment integrity and are mentioned in other UFSAR sections are bounded by the limiting LOCA and MSLB DBA analyzed in Section 14.20 and discussed above.

II.5.1 Containment Response (UFSAR 14.20)

The Containment Response is a DBE, the analysis of which verifies the integrity of the containment structure under the adverse pressure and temperature conditions resulting from a postulated LOCA or MSLB Event. Parametric combinations of break size, break location, and power level are analyzed to determine the most limiting scenario with specific regard to containment response for both LOCA and MSLB. Design and acceptance criteria are placed on the limiting temperature and pressure results, which ensure the integrity of the containment structure under the conditions of the analyzed events.

Reference II-31 is the current bounding AOR for containment response, applicable to plant conditions with and without the replacement SGs, and valid beyond a rated power level of 2737 MWt (MUR power uprate). In support of the replacement SG installation, the bounding AOR (Reference II-63) was established. Reference II-64 provides the qualification of the GOTHIC computer code for modeling containment response at Calvert Cliffs. This methodology was implemented at Calvert Cliffs in accordance with the 10 CFR 50.59 process, as documented in Reference II-65. Limiting mass and energy releases are determined parametrically, and include power levels of 2754 MWt. Decay heat values following the modeled plant trip are calculated based on the NRC Branch Technical Position ASB 9-2 for LOCA. The MSLB results bound those of LOCA with respect to both peak pressure and peak temperature in Containment during the course of the analyzed limiting events. The limiting initial power level for the MSLB event is 75% RTP, however a power level of 2754 MWt, plus pump heat, was analyzed parametrically with various break sizes to determine the limiting contribution of mass and energy to the containment atmosphere through the break. All documented bounding results in Reference II-31 (AOR) bound operation at the MUR power uprate power level of 2737 MWt, plus uncertainties, and are found to be in compliance with the applicable qualification restraints of Reference II-64. Therefore, the current AOR for the Containment Response Analysis is appropriately applicable to, and bounds, operation at the MUR power uprate power level of 2737 MWt plus uncertainties.

II.6 STATION BLACKOUT EVENT

The proposed changes to the licensed core power level and power measurement uncertainty have no impact on the station blackout analysis. The initial portion of the station blackout transient (i.e., loss of AC power) was determined to be unaffected by the proposed MUR power uprate (see Table II-1). The small increase in decay heat as a result of the proposed MUR power uprate has a negligible impact on post-trip equipment (e.g., opening of MSSVs) or operator response.

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III. ACCIDENTS AND TRANSIENTS FOR WHICH THE EXISTING ANALYSES OF RECORD DO NOT BOUND PLANT OPERATION AT THE PROPOSED INCREASED POWER LEVEL

There are no accidents or transients that are not bounded by the existing AOR (see Table II-1). However, other related personnel and equipment concerns need to be addressed. Therefore, the potential effects of the MUR power uprate were evaluated for the following issues:

- Normal Operational Shielding and Personnel Exposure
- Radiological Environmental Qualification (EQ)
- Post-LOCA Access to Vital Areas

As discussed in the previous section, no Chapter 14 accidents or transients required additional analysis because the existing AOR remained bounding for plant operation at the proposed increased power level. Discussion on the impact of the proposed MUR power uprate on plant radioactive waste effluents is provided in Section VI under Radioactive Waste Systems.

Normal Operational Shielding and Personnel Exposure

The MUR power uprate is expected to cause a 1.38% increase in radiation levels. However, these increases will not affect radiation zoning or shielding requirements in the various areas of the plant. Individual worker exposures are maintained within acceptable limits by the site as low as reasonably achievable (ALARA) program that controls access to radiation areas. In addition, procedural controls may be used to compensate for increased radiation levels.

Radiological Environmental Qualification

In accordance with 10 CFR 50.49, safety-related electrical equipment must be qualified to survive the radiation environment at their specific location during normal operation and during an accident.

The Containment and Auxiliary Buildings are divided into various rooms for environmental zoning purposes. The radiological environmental conditions noted for these rooms are the maximum conditions expected to occur. The current normal operation values represent 40 years of operation, while the AOR post-accident radiation exposure levels are determined for a one-year period following an accident using Regulatory Guide 1.89 source-term assumptions and a core power level of 2700 MWt.

For the MUR power uprate, the EQ accident source-term was reanalyzed for a core power level that bounds the proposed MUR power uprate with the same release assumptions as before. The increased source-term was compared to the AOR to develop integrated energy ratios that were used to adjust the doses from various sources (airborne, sump, iodine filters, etc.) for each Containment and Auxiliary Building room. The normal operation contribution to the EQ dose is based on survey data. It was increased by 1.38% (MUR power uprate), as well as by a factor of 1.5 to account for the extended operation period of 60 years.

Post-LOCA Access to Vital Areas

Vital access dose considerations are described in NUREG-0737, Item II.B.2. Specifically, the design dose for personnel in a vital area should not exceed 5 rem whole body, or its equivalent to any part of the body, for the duration of DBAs. Updates of the dose analyses were performed to confirm that this requirement was met for a LOCA using Regulatory Guide 1.4 source-term assumptions and a core power level of 2737 MWt. The UFSAR time-dependent radiation dose rate maps that cover plant areas and access paths which may require occupancy during post-LOCA recovery operations will be updated to

reflect the proposed MUR power uprate. The MUR power uprate does not have an impact on vital area access requirements.

IV. MECHANICAL/STRUCTURAL/MATERIAL COMPONENT INTEGRITY AND DESIGN

IV.1 INTRODUCTION

The RCS component specifications define the frequency and severity of the design transients that must be considered in the fatigue evaluations of the components in accordance with the American Society of Mechanical Engineers (ASME) code. The design transients in the individual component specifications represent events that are expected to occur, or may occur, during the life of the plant. The design transients are characterized in terms of the type of transients, the frequency of occurrence, the initial design conditions, and the associated thermal-hydraulic conditions experienced by various systems and components as a result of the transients. This information is then used in fatigue evaluations for those systems and components. The design transients defined in the current component specifications were reviewed to determine the effect of the MUR power uprate.

With respect to the type of transients and frequency of occurrence, the implementation of the MUR power uprate does not create new types of transients nor change the original event frequencies for the design transients.

With respect to the initial conditions and the thermal-hydraulic response during the transients, some were found to be affected by the uprate and some were not. The transients which occur in the lower operating modes remain valid because the HZP (no load) plant conditions are unaffected by the MUR power uprate. Many of the transient responses remain valid because the original design hot and cold leg temperatures are higher than the increased operating point due to the MUR power uprate.

Where necessary, the design transients were re-analyzed quantitatively to assess the impact of the changes on existing design conditions due to the MUR power uprate. In these cases, the analyses simulated the transients under the increased conditions and produced thermal-hydraulic responses (pressures, temperatures, and flow rates) for use in the component-by-component fatigue evaluations described in this section.

This section also provides the results of the structural integrity evaluations for RCS components and supports at MUR power uprate conditions. Table IV-1 shows a comparison of current nominal operating parameters values versus the expected values following implementation of the MUR power uprate. The remaining portions of this section discuss the impact of the MUR power uprate RCS components, nuclear fuel and core thermal hydraulics, and various other components.

Table IV-1
Current NSSS Design and MUR Power Uprate
Nominal Operating Parameters for Calvert Cliffs

Parameter	Current Normal Operating Conditions	MUR Power Uprate Normal Operating Conditions	Percent Change
Core Power, MWt (input)	2700	2746 (6)	1.70% (6)
No. of Plugged Tubes per SG	≤ 10%	≤ 10%	-
Primary Bulk T_h, °F	595.1	595.9 (7)	0.13%
Primary T_c, °F	548	548	-
Primary ΔT, °F	48.4 (3)	49.2 (3)	1.65%
Primary Flow Rate, gpm (input)	370,000 - 422,250	370,000 - 422,250	-
Core Bypass Flow Rate, %	3.9	3.9	-
Primary Pressure, psia	2250	2250	-
Feedwater Temperature, °F	431.5 (5)	433.6 (5)	0.70%
Feedwater Enthalpy, Btu/lbm (input)	409.2 (1, 5)	410.8 (1, 5)	0.39%
Feedwater Flow Rate per SG, lbm/sec (input)	Same as Steam Flow	Same as Steam Flow	-
SG Blowdown Flow per SG, lbm/sec (input)	41.7 (max)(6)	41.7 (max)(6)	-
SG Steam Flow per SG, Mlbm/hr	5.900 (1)	5.999 (1, 5, 6)	1.68%
Steam Pressure, psia	888 (1, 2)	886.5 (1, 2)	-0.17%
	863 (1, 3)	860.3 (1, 3)	-0.31%
SG Total Mass, lbm	138,524 (1, 4)	138,024 (1, 4)	-0.36%
SG Liquid Mass (lbm)	128,130 (1, 4)	127,636 (1, 4)	-0.39%

(1) At 100% power
(2) No plugged tubes
(3) 10% plugged tubes
(4) SG level at 35.95 ft
(5) Based on best available data
(6) Bounding value selected for the evaluation
(7) A large portion of the MUR power uprate evaluation was completed using an estimated temperature increase for T_{hot} of 1.1°F. Further evaluations have since been finalized, predicting a 0.8°F increase for T_{hot}. Therefore, the original evaluation performed for MUR power uprate remains bounding.

IV.2 REACTOR COOLANT SYSTEM LOSS-OF-COOLANT ACCIDENT FORCES EVALUATION

The purpose of a LOCA hydraulic forces analysis is to generate the hydraulic forcing functions and blowdown loads that occur on RCS components as a result of a postulated LOCA. These forcing functions and loads act on the component's shell and internal structures.

The full set of RCS loadings considered in the structural analysis of a LOCA event consists of the internal forcing functions generated from the hydraulic forces analysis, the pipe tension release, and jet impingement forces acting at the break locations, and, where applicable, the external loads due to subcompartment pressurization effects that act on the components and their supports.

Except for the thimble support plate and selected RV internals components, the faulted loads and stresses in the current AORs are based on main coolant loop breaks (MCLBs) where thrust loadings were based on simplified (pressure x area) terms and where asymmetric blowdown loadings were calculated using

design setpoint parameters for 2700 MWt, where T_{hot}=604°F and T_{cold}=548°F. Since the MCLBs have been eliminated by leak before break (LBB) and replaced by branch line pipe breaks (BLPBs), loads and motions on NSSS components due to pipe breaks are greatly reduced. Furthermore, since the RV blowdown loads are primarily affected by changes in T_{cold}, and T_{cold} remains the same for the MUR power uprate, the effects of BLPBs at the MUR power uprate would not be significantly different from the effects of BLPBs under pre-MUR power uprate conditions. Therefore, the effects on NSSS components of BLPBs at the MUR power uprate are less severe than the effects of pipe breaks currently documented in the AORs.

Based on this conclusion, the design transient for blowdown loads at the MUR power uprate conditions remains the original design basis LOCA analyzed. Except where noted, the following structural evaluation discussions are based on the original design transient, and do not make direct use of the mitigating effects of LBB.

IV.3 REACTOR COOLANT SYSTEM MAJOR COMPONENT ASSESSMENTS

As noted in the introduction to this section, the majority of the NSSS design transients are demonstrated to be unaffected by the MUR power uprate. Transients with the potential to adversely affect the AOR results for particular RCS components were evaluated for their effects on the critical stress margins identified for the RCS components. The transients involved are listed below, on a component by component basis:

<u>RV, RCPs, RCS Piping and Fittings (except Surge Line), and Original Control Rod Drive Mechanism & Part Length Control Rod Drive Mechanisms</u>

- Reactor Trip – The rate of change in temperature for the MUR power uprate for this transient is slightly greater than that for the design basis.

<u>Surge Line and Fittings</u>

- Reactor Trip, Loss of Flow, Step Load Increase/Decrease, Plant Loading/Unloading – The change in temperature for the MUR power uprate for these transients is slightly greater than that for the design basis.

<u>Pressurizer</u>

- Step Load Increase - The rate of change in temperature for the MUR power uprate is greater than that for the design basis for this transient.

The above observations were used to help determine which MUR power uprate transients needed to be evaluated with respect to their effects on fatigue for limiting RCS components. Evaluations of these limiting components are discussed in the remainder of this section.

In another assessment, the Calvert Cliffs RCS loads and displacements due to normal operating thermal expansion effects under the MUR power uprate conditions were reconciled with the loads and displacements from the pre-uprate RCS thermal expansion analysis, where T_{hot} was set to 604°F and T_{cold} was set to 550°F. It was concluded that MUR power uprate does not cause any significant changes in thermal anchor motions, and that all previously documented thermal anchor motions for Calvert Cliffs remain valid. All RCS loads due to normal operating thermal expansion either decrease or change insignificantly due to the decrease in delta-T between T_{cold} and T_{hot} of the initial power rating design setpoint temperature and the MUR power uprate conditions stated in Table IV-1. The SG inlet nozzle

moment (the moment around the horizontal axis perpendicular to the hot leg axis) increases, but this moment is not a limiting load with respect to stress margins on either the hot leg or the SG inlet nozzle.

Based on the results of this normal operating thermal expansion evaluation, specified normal operating loads on NSSS component supports and nozzles, and the main loop piping, and normal operating displacements on RCS tributary nozzles do not need to be revised for the MUR power uprate. This conclusion is utilized in the AOR stress evaluations discussed in the remainder of this section.

IV.3.1 Reactor Vessel Structural Evaluation

This evaluation assesses the effects that the MUR power uprate has on the most limiting locations with regard to ranges of stress intensity and fatigue usage factors (UFs) in each of the vessel regions, as identified in the RV stress reports and addenda.

The nominal vessel outlet temperature increases to 595.9°F (597.2°F end-of-life), and the nominal vessel inlet temperature remains at the current value of 548.0°F as a result of the MUR power uprate (see Table IV-1 for a comparison of operating parameters). Therefore, the T_{hot} variation during normal plant loading and plant unloading increases while the T_{cold} variation remains unchanged.

As noted above, the nominal vessel inlet temperature associated with the MUR power uprate is the same as the nominal temperature for the current fuel cycle. The nominal vessel outlet temperature has increased slightly but is still less than the normal design vessel outlet temperature of 604°F that was originally used in the analysis of the RV outlet nozzles. Therefore, the effects of the plant loading and unloading transients on the inlet and outlet nozzles remain bounded by the stress AOR.

The RV main closure flange region and CEDM housings were originally evaluated for the effects of a higher vessel outlet temperature. Therefore, the effects of the MUR power uprate vessel outlet temperature on these regions are also bounded by the current design basis.

The remaining RV regions, including the inlet nozzles, vessel wall transition, core support guides, bottom head-to-shell juncture, and instrumentation nozzles are affected by the vessel inlet temperature, which is unchanged for the MUR power uprate. Therefore, the previously determined maximum stress intensity ranges and maximum cumulative fatigue UFs for these regions are valid.

The critical stress margin at the closure head studs remains unchanged because it is based on compression due to the bolt-up procedure, which is unchanged by the MUR power uprate. The critical margins at the vessel wall at the core stabilizer lugs remain unchanged because they are based on normal operating pressure and Operating Basis Earthquake (OBE), none of which are changed by the MUR power uprate. The margins on the load capability of the RV cold leg and hot leg horizontal supports due to MCLBs are significantly increased to non-critical margins due to the elimination of these breaks and their replacement with BLPBs.

None of the margins on the incore instrumentation (ICI) flange assembly for either unit are critical. The lowest margin on stress is 7.7%, which is a margin on the bearing stress at the nut-to-compression collar surface for the flange assembly on Unit 2. This margin is unchanged because the bearing stress is due to design pressure and OBE, neither of which are changed by the MUR power uprate.

None of the margins on the RV vent line are considered to be critical. The lowest margin on stress is for the primary-plus-secondary stress at the J-weld on the RV closure head. The controlling stress range is generated from the loss of secondary pressure (LOSP) and the normal heatup transients. These specified

design transients are not changed by the MUR power uprate. Therefore, the critical stress margins for this component remain valid for MUR power uprate conditions.

The critical stress margins for other RV components are discussed below.

RV Closure Head Instrument Nozzle Bimetallic Weld

The critical margin of 10.18% on the stress intensity is due to a design pressure of 2500 psia, and the allowable S_m is based on a design temperature of 650°F. The design pressure and temperature are not changed due to the MUR power uprate. Therefore, the stress margin for this component is unchanged and remains acceptable for the MUR power uprate.

Vessel Wall at RV Outlet Nozzle

The critical stress margin of 2.5% is for primary-membrane-plus-local stress. Per the AOR, the calculated stress is a function of design moments and forces on the pipe, and of a design pressure of 2500 psia. The design moments and forces are unchanged due to the MUR power uprate. In addition, the design pressure of 2500 psia is unchanged by the MUR power uprate. Therefore, this stress margin remains acceptable for the MUR power uprate.

RV Outlet Nozzle

As above, the critical stress margin for primary-membrane-plus-local stress is a function of design moments and forces on the pipe, and of a design pressure of 2500 psia. Therefore, the stress margin of 0.79% for this component is unchanged and remains acceptable for the MUR power uprate.

Vessel Wall Transition Part of Vessel Support

The critical stress margin of 0.64% is for the primary-membrane stress. Per the AOR, the calculated primary-membrane stress is based on a design pressure of 2500 psia. The design pressure of 2500 psia is not changed by the MUR power uprate. Therefore, the stress margin of 0.64% for this component is unchanged and remains acceptable for the MUR power uprate.

Taper between RV Dome and Closure Flange

The critical stress margin of 32.0% is for primary-membrane-plus-local stress. Per the AOR, the calculated stress is due to design pressure, flange bolt-up loads and core (i.e., vessel internals) loads. The specified flange bolt-up loads and core loads are unchanged due to the MUR power uprate. In addition, the design pressure of 2500 psia is not changed by the MUR power uprate. Therefore, this stress margin is unchanged and remains acceptable for the MUR power uprate.

Surveillance Holder and Brackets

The critical margin on the alternating stresses (S_{alt}) from peak stresses is 2.33%. Per the AOR, the calculated stress is based on design moments and forces and stress concentration factors. None of these parameters are changed as a result of the MUR power uprate. Therefore, the stress margin for this component is unchanged and remains acceptable for the MUR power uprate.

Vessel Wall at Core Stabilizer Lugs

The critical margin on the maximum stress (S_{max}) due to the lateral load on the shell at the lug attachment is 5.25%. Per the AOR, the calculated stress is a function of the lateral moment and the design pressure. The specified design moments and forces are unchanged due to the MUR power uprate. In addition, the

design pressure of 2500 psia is not changed by the MUR power uprate. Therefore, this stress margin is unchanged and remains acceptable for the MUR power uprate.

Vessel Shell and Bottom Head

The critical margin on primary-membrane stress is 0.64%. It is noted that the critical location coincides with the vessel wall transition part of vessel support location discussed above. Therefore, the stress margin for this component is unchanged and remains acceptable for the MUR power uprate.

Head Lift Rig

This evaluation pertains to the currently installed head lift rigs at Calvert Cliffs Units 1 and 2. Evaluation of the planned replacement lift rig will be performed prior to its installation after the MUR power uprate.

The vertical link in the head lift rig has a critical margin on stress of 2.3%. This small margin is due to tension stress during closure stud handling, which is not affected by the MUR power uprate. Therefore, this head lift rig subcomponent remains acceptable for the MUR power uprate.

The cooling duct cover plate has a critical margin on stress intensity of 0.9%, which is due to a combination of dead weight, seismic excitation, and flow loads. These flow loads are hot air flow loads across the cover plate for which the operating temperature under the MUR power uprate conditions remains lower than the design temperature. In addition, the dead weight and seismic loads are not affected by the MUR power uprate. Therefore, this critical head lift rig subcomponent also remains acceptable.

Conclusion

The RV evaluation for the MUR power uprate demonstrates that the maximum ranges of stress intensity remain within their applicable acceptance criteria, and the maximum cumulative fatigue UFs remain below the acceptance criterion of 1.0.

In addition, the faulted condition stress analyses for the Calvert Cliffs RV do not change as a result of the MUR power uprate, because the seismic loads are unchanged from the AOR, and the pipe break load input remains based on the original MCLBs. Therefore, no changes in the faulted condition RV/reactor internals interface loads or other faulted condition parameters are identified as a result of the MUR power uprate.

IV.3.2 Reactor Vessel Internals Evaluation

The reactor internals support the fuel and control rod assemblies, experience control rod assembly dynamic loads, and transmit these and other loads (e.g., deadweight, seismic vibration) to the RV. The internals also direct flow through the fuel assemblies, provide adequate cooling to various internals structures, and support ICI. The changes in the RCS design parameters identified previously in Table IV-1 produce insignificant changes in the boundary conditions experienced by the reactor internals components. This section describes the evaluation performed to demonstrate that the reactor internals can perform their intended design functions at the MUR power uprate conditions.

IV.3.2.1 Thermal-Hydraulic Systems Evaluations

The MUR power uprate can potentially affect such parameters as reactor vessel internal (RVI) component heating rates, coolant temperature levels, and their downstream impacts. A key area in evaluation of core performance is the determination of the hydraulic behavior of coolant flow and its effect within the reactor internals system. The core bypass flows are required to ensure reactor performance and adequate

RV head cooling. The hydraulic forces are critical in the assessment of the structural integrity of the reactor internals. The results of the thermal-hydraulic evaluations are provided below.

RVI Component Temperatures

The AOR on RVI component temperatures were reviewed to determine the component most affected by the MUR power uprate. The component selected from this review process is the core shroud. Component metal temperatures, and therefore, thermal stresses, are dependent on the core power level and coolant temperatures. Calvert Cliffs core shroud metal temperatures were re-evaluated for the MUR power uprate level. The resulting core shroud component temperatures were used to calculate thermal stresses, in order to evaluate the structural margins for the shroud. The structural evaluation demonstrated that there is adequate structural margin for the core shroud for the MUR power uprate; see Section IV.3.5

RVI Component Hydraulic Loads

A review of the AOR design hydraulic loads on the RVI components indicated that the current design loads are bounding for the MUR power uprate operation. Small increases in power level, such as the MUR power uprate, have minimal impact on the design hydraulic loads.

Core Bypass Flow Calculation

Bypass flow is the total amount of reactor coolant flow bypassing the core region and is, therefore, not considered effective in the core heat transfer process. The design core bypass flow limit is 3.90% of the total RV flow. This value is used in the thermal margin calculations. A lower bound value of 1.6% is used in the calculation of hydraulic loads since the higher core flow results in higher core pressure drops and, therefore, higher uplift and differential pressure (D/P) loads. The best-estimate core bypass flow is 3.51% of the RV flow.

Core bypass flow is negligibly affected by the MUR power uprate. The core pressure drop will tend to increase very slightly, due to the higher power level. This will have the effect of diverting very slightly more bypass flow through the various leakage flow paths. But the margin between the best estimate and design values of core bypass flow will readily accommodate the negligible increase in core bypass flow due to the uprate.

Therefore, the core bypass flow limit of 3.9% remains valid for the MUR power uprate.

CEA Drop Time Analyses

Calvert Cliffs Technical Specification Surveillance Requirement 3.1.4.6 requires that the drop times of all full-length CEAs from a fully withdrawn position, must be verified to be less than or equal to 3.1 seconds prior to the startup of each cycle.

Control element assembly drop times are explicitly confirmed to meet the times assumed in the accident analyses. An evaluation was performed for CE fleet plants to demonstrate continued compliance with the current Technical Specification requirements based on CE fleet's robust five finger silver tip CEA design, which has not shown failure to insert at any time in life through the end-of-life core burnup. The MUR power uprate conditions will slightly increase the power level in leading rodded fuel assemblies; however the projected burnup levels and fluences are substantially less than the values assumed in the design calculations. The assembly burnups and fluences are confirmed on a cycle specific bases to be within the values assumed in the CEA design analysis. In addition, the CEA drop time is measured prior to the startup of each reload cycle.

Finally, the fluid density has not increased for the MUR power uprate since T_{cold} has not changed and T_{hot} has increased only slightly. Therefore, CEA drop times are not adversely affected by the MUR power uprate.

Based on the above, the current limiting rod drop time requirements remain valid for the MUR power uprate conditions.

CEA and ICI Cooling Assessment

Cooling analyses for CEAs and incore instruments were performed for the MUR power uprate. These analyses indicate that the following design criteria are met:

- No coolant bulk boiling will prevail inside the CEA and ICI guide tubes.
- B_4C and AgInCd maximum temperatures will stay within the design limits of 2000°F and 1400°F, respectively.

IV.3.2.2 Mechanical Evaluations

As discussed previously, the MUR power uprate conditions do not affect the current design bases for seismic and LOCA loads. Therefore, it was not necessary to re-evaluate the structural effects from seismic OBE and safe shutdown earthquake loads, or from the LOCA hydraulic and dynamic loads. Furthermore, it is noted that the LOCA hydraulic and dynamic loads would be less severe if BLPBs were analyzed instead of the original design basis MCLBs.

With regard to flow and pump induced vibration, the current analysis uses a mechanical design flow that does not change for the revised design conditions (see Table IV-1). The MUR power uprate conditions alter the T_{hot} fluid density. However, this very small change in the T_{hot} fluid density has a negligible effect on the forces induced by flow. In addition, the MUR power uprate results in a negligible change in T_{ave}. Therefore, the mechanical loads are not affected by the MUR power uprate conditions.

IV.3.2.3 Structural Evaluations

As described in Section IV.3.3, the normal operating hydraulic loads used in the AOR for the structural evaluation of the RVI components are bounding for the MUR power uprate. Seismic and LOCA loads on the RVI components are unaffected by the MUR power uprate operation, and the primary stresses calculated in the AOR therefore remain applicable. The MUR power uprate can potentially increase thermal loadings and the resulting thermal stresses in the RVI components. Because this MUR power uprate is relatively small (~1.38%), it was concluded that potential adverse effects on the RVI structures would be confined to the core shroud, which is more sensitive than the other RVI components to minor variations in thermal loading.

To quantify these potential effects, the AOR for the calculation of thermal stresses in the core shroud was reviewed. Because of limitations in this AOR, the applicability of a more recent analysis, performed for a similar core shroud design, was investigated. The applicability of this analysis was confirmed, and the thermal stresses calculated therein were combined with the appropriate primary stress and evaluated against acceptance criteria. Elevated (> 800°F) temperature effects (reflecting the core shroud maximum temperature of 885°F) were considered in the determination of these acceptance criteria, and a fatigue evaluation was performed.

Under thermal loadings that encompass the MUR power uprate, the core shroud analysis determined that the maximum primary-plus-secondary stress intensity exceeded the allowable value. Therefore, a

simplified elastic-plastic analysis with attendant re-evaluation of fatigue usage was performed in accordance with the acceptance criteria.

The resulting maximum primary-plus-secondary stress intensity (excluding thermal bending stresses) was 37,927 psi, which is less than the 43,800 psi allowable. Furthermore, the cumulative usage factor (CUF) for the MUR power uprate conditions was determined to be 0.375, which is significantly less than the 1.00 allowable. Thus, the core shroud satisfies the acceptance criteria.

Increases in core thermal power will slightly increase nuclear heating rates in the RVIs, such as lower core support plate, fuel alignment plate, and core shroud. Evaluations have been performed verifying that the existing thermal-hydraulic AOR will support the MUR power uprate. Therefore, the calculated component lifetimes will envelop the component lifetimes associated with the MUR power uprate related increases in nuclear heating.

IV.3.3 Control Element Drive Mechanisms

The CEDMs are mounted on top of the Calvert Cliffs reactor head. These components are affected by the reactor coolant pressure, vessel outlet temperature, and hot leg NSSS design transients.

According to Table IV-1, the vessel outlet temperature for the MUR power uprate has increased slightly to 595.9°F. This small temperature increase remains well below the design operating temperature of 604°F. Therefore, no additional assessments of the impact of thermal loads on the CEDMs and CEDM nozzles are required. The reactor coolant operating pressure (2250 psia) for the MUR power uprate conditions remains the same as originally specified for the CEDMs so no additional assessment is required for pressure considerations.

Since all critical margins on the CEDMs are maintained for the MUR power uprate, these components remain acceptable.

IV.3.4 Nuclear Steam Supply System Piping and Pipe Whip

The reactor coolant main coolant loop piping system (including primary loop piping and pipe whip restraints, and tributary piping nozzles) was assessed for the MUR power uprate effects. It was concluded that these equipment designs remain acceptable and continue to satisfy design basis requirements in accordance with applicable design basis criteria, which include the criteria associated with the original design basis mechanistic LOCA breaks, when considering the operating temperature, operating pressure, and flow rate effects resulting from the MUR power uprate conditions. The primary piping and tributary nozzles remain within allowable stress limits in accordance with ASME Section III, 1965 Edition, up to and including the Winter 1967 Addendum [and in accordance with ASME Section III, 1986 Edition for components with a mechanical nozzle seal assembly (MNSA)].

Reconciliation of a number of critical locations on the Calvert Cliffs Units 1 and 2 RCS piping and fittings under the MUR power uprate conditions is summarized below.

Hot and Cold Leg Piping

The critical margins on the maximum primary-local-plus-bending stress intensity at the hot leg and cold leg elbows are 5.80% and 3.26%, respectively. The calculated stresses are based on the design moments from dead weight and seismic excitation, and the design pressure of 2500 psia. The specified design loads do not change for the MUR power uprate. Therefore, the stress margins of 5.80% and 3.26% are unchanged and remain acceptable for the MUR power uprate.

Pump/Pipe Junction at the Discharge Nozzle

The critical margin on the maximum primary-bending stress at the pipe-pump discharge nozzle juncture is 4.82%. The stress is a function of the design loads (including loads due to OBE) and the design pressure of 2500 psia. The specified design moments are not changed by the MUR power uprate. In addition, the design pressure is not changed by the MUR power uprate. Therefore the stress margin of 4.82% for the pipe/pump juncture is unchanged and remains acceptable for the MUR power uprate.

Pump/Pipe Junction at the Suction Nozzle

The CUF at this location is 0.836 (i.e., a 16.4% margin). Most of the fatigue usage (0.833 of 0.836) is due to the seismic transients. The rest of the fatigue usage (0.003) comes from the heatup transient. The specified normal operating loads and design seismic loads are unaffected by the MUR power uprate. Therefore, the effect of seismic on stress and fatigue is unchanged. The fatigue usage due to the heatup transient is also unchanged because this transient occurs at zero power and is not affected by the MUR power uprate. Therefore, this component remains acceptable for the MUR power uprate.

Safety Injection Nozzle

The critical margin on the maximum primary-plus-secondary stress, (Local Primary Membrane Stress (PL) plus Primary Bending Stress (PB) Secondary Stress (Q)) or the primary plus secondary stress intensity (PL+PB+Q), is 2.24%, and is due to the seismic load and heatup transient and the cooldown transient when combined with the effects of design pressure, dead weight, and seismic loads. The design pressure, dead weight, and seismic loads are not affected by the MUR power uprate. The heatup/cooldown transients occur at zero power and therefore are not affected by the MUR power uprate. Therefore, this critical margin on stress is unchanged for the MUR power uprate.

The highest CUF in the SI nozzle is only 0.1892, and is primarily due to an alternating stress from combinations of plant cooldown/seismic and plant heatup/cooldown transients. The plant heatup/cooldown transients and seismic excitations are not affected by the MUR power uprate. Therefore, the CUF for the SI nozzle is unchanged, and the nozzle remains acceptable for the MUR power uprate.

Hot Leg RTD and Pressure Differential Transmitter Nozzles with a MNSA

The critical margin on the maximum primary-plus-secondary stress, PL+PB+Q, is 1.68% for either the hot leg RTD or pressure differential transmitter nozzle with a MNSA installed. This limiting stress is based on the hydrostatic test and LOSP transients. These transients are not altered by the MUR power uprate. Therefore, this component remains acceptable for the MUR power uprate.

Hot Leg Drain Nozzle

The critical margin on primary-local-plus-bending stress is 1.25%. The calculated stress range is based on design moments from design pressure, dead weight, thermal and design seismic effects. These effects are changed by the MUR power uprate. Therefore the stress margin of 1.25% for this nozzle is unchanged and the nozzle remains acceptable for the MUR power uprate.

In conclusion, the Calvert Cliffs Units 1 and 2 primary piping and tributary nozzles remain within allowable stress limits in accordance with ASME Section III, 1965 Edition, up to and including the Winter 1967 Addendum (and in accordance with ASME Section III, 1986 Edition for components with MNSA). Furthermore, no piping or pipe restraint modifications are required as a result of the increased

power level, because conservatively determined LOCA loads due to MCLBs were used to design the pipe restraint systems.

IV.3.5 Steam Generators

The thermal-hydraulic performance of the SGs was analyzed for the MUR power uprate conditions. Given the new RCS input of 595.9°F for T_{hot}, with flow, and T_{cold} remaining the same as at 2700 MWt, the secondary side of the SG experiences a 1.7% flow rate increase with pressure and temperature decreasing 1.5 psi and 0.1°F. There is also a slight 0.36% decrease in SG inventory at 100% power. The new conditions were checked against the design, test and Level A, B, C and D stress levels specified in the ASME Code and found to be acceptable. The internals and flow induced vibration effects were found to be negligible. Structurally there is negligible effect. Therefore, the SGs remain fully qualified to operate at the MUR power uprate.

Steam Generator Upper and Lower Supports Structural Integrity

The Calvert Cliffs SG support system consists of the following components at each SG:

- Lower SG supports - a sliding base, with four vertical pad supports and two lower keys.
- Upper SG supports - two upper shear key supports and eight directing-acting hydraulic snubbers.

Even though the operating setpoint temperatures (T_{hot} and T_{cold}) for the MUR power uprate conditions are enveloped by the design setpoint temperatures used in the original design basis structural analyses of the RCS, an assessment was performed to determine the effects of the MUR power uprate condition operating temperatures on the RCS components and supports. This analysis concluded that the loads on the RCS supports, including the supports on the SGs, either decreased or changed insignificantly due to the decrease in delta-T between T_{cold} and T_{hot}, relative to the original design basis analyses. Therefore, the effect of RCS thermal expansion on SG support loads due to the MUR power uprate is insignificant.

Since the original seismic and LOCA loads are also unchanged for the MUR power uprate, the SG upper, and lower supports continue to be acceptable under the MUR power uprate conditions.

IV.3.6 Reactor Coolant Pumps and Motors

IV.3.6.1 RCP Structural Analysis

The four RCPs are installed in the cold legs of the reactor coolant loops. The RCPs are affected by the reactor coolant pressure, SG outlet temperature, and primary side cold leg NSSS design transients. The SG outlet temperature affects both the thermal expansion and thermal transient loads on the RCPs.

The nominal SG primary outlet temperature for the MUR power uprate (i.e., T_{cold} = 548.0°F) is the same as the current nominal and design basis temperature for the SG outlet, RCP suction and discharge and RV inlet. Consequently, RCP thermal expansion loadings for the MUR power uprate are bounded by the design condition.

The RCP supports are designed to carry loads due to normal operating conditions and seismic excitations, neither of which is changed by the MUR power uprate. Under LBLOCA, the RCP nozzles were shown to be capable of carrying the faulted condition loads that the RCP supports were not designed to carry. When LBLOCAs are replaced by BLPBs via LBB considerations, the faulted loads on the RCP supports and nozzles are significantly reduced, and the margin on the nozzle loads is increased.

The AOR also identified a critical margin on the horizontal strut load due to OBE. Since seismic excitations are unaffected by the MUR power uprate, this margin remains unchanged and acceptable.

In addition to the support system, other critical RCP components stress margins were addressed in the MUR power uprate assessments, as follows.

Casing Diffuser Vanes

Design Conditions - Critical margins exist for the primary-membrane and the primary-membrane-plus-local stress intensities at Vane 8. These margins are 2.74% and 4.81%, respectively. Per the AOR, the calculated membrane stress intensity is the average stress in the vane, and the calculated membrane plus local stress intensity represents the largest surface stress intensity, adjusted by removing the discontinuity bending stress. These stresses are a function of design moments and forces on the structure and the design pressure of 2500 psia. The specified design moments and forces are unchanged due to the MUR power uprate. In addition, the design pressure of 2500 psia is unchanged by the MUR power uprate. Therefore, these diffuser vane stress margins are unchanged and remain acceptable for the MUR power uprate.

Suction Nozzle

Design Conditions – The critical margin on the primary-membrane stress of the suction nozzle due to the design conditions is 2.0% and is only due to the design pressure 2500 psia. The design pressure, and therefore, primary-membrane stress, is not affected by the MUR power uprate.

Emergency Conditions - The critical margin in the suction nozzle, due to emergency conditions, involve primary-local membrane stresses. In the AOR, the overall emergency condition stresses exceeded the ASME code primary-general stress limit of 1.2 S_m. However the 1.2 S_m limit does not include local stress effects. The overall emergency condition stresses are, however, bounded by the primary-local stress ASME code limit of 1.8 S_m which does include local stress effects. The AOR concluded that the conservatism inherent in the more restrictive primary-general membrane stress allowable was unwarranted and the primary-local stress limit of 1.8 S_m was an acceptable bound for the suction nozzle stresses. This reasoning also is applicable to the stresses for the MUR power uprate.

Furthermore, since the specified external moments and forces, and the operating pressure loads are unchanged for the MUR power uprate conditions, the stress margins are unaffected and the suction nozzle design remains acceptable.

Discharge Nozzle

Design Conditions - Critical margins exist for the primary-membrane and the primary-membrane-plus-bending stress intensities in the crotch region of the discharge nozzle. These margins are 2.20% and 2.67%, respectively. Per the AOR, the acceptable primary-membrane stress margin of 2.20% was obtained after correcting the as-calculated stress analysis results for the as-cast thickness of the discharge nozzle and shell. Regarding the primary-membrane-plus-bending stress, the acceptable margin of 2.67% was obtained by removing the secondary bending stress from the greatest surface stress intensity in the crotch. The specified design moments and forces are unchanged due to the MUR power uprate. In addition, the design pressure of 2500 psia is unchanged by the MUR power uprate. Therefore, the stress margins of 2.20% and 2.67% for the discharge nozzle remain acceptable for the MUR power uprate.

Emergency Conditions - Critical margins exist for the primary-membrane and the primary-local-plus-bending stress intensities in the crotch region of the discharge nozzle. Primary-local-plus-bending results apply to the top half of the nozzle. These margins are 0.10% and 0.87%, respectively. Since the external

moments and forces, and the operating pressure loads are unchanged for the MUR power uprate conditions, the stress margins remain acceptable for the MUR power uprate.

Hanger Bracket

Design Conditions - The critical margin for this location, 2.67%, is associated with the primary-membrane-plus-bending stress intensity. The membrane stress, which is 1.046 S_m, is classified as primary-local stress, and therefore is well below the limit of 1.5 S_m. The specified design moments and forces are unchanged due to the MUR power uprate. In addition, the design pressure of 2500 psia is unchanged by the MUR power uprate. Therefore the stress margin of 2.67% for the hanger bracket remains acceptable for the MUR power uprate.

Volute, Lower Flange

Design Conditions - The critical margin for this location, 4.70%, is associated with the primary-membrane stress intensity. The AOR also states that no surface stress exceeds the 1.5 S_m limit. Consequently, the primary-membrane-plus-bending limits are also satisfied for this region of the structure. The specified design moments and forces are unchanged due to the MUR power uprate. In addition, the design pressure of 2500 psia is unchanged by the MUR power uprate. Therefore, the stress margin of 4.70% for the volute/lower flange region remains acceptable for the MUR power uprate.

Cover, Region 4

Design Condition - According to the AOR, the critical stress margin for this region of the cover (the inside corner of the cover between cooling holes) occurs under operating conditions (i.e., for operation between steady-state hot and steady-state cold conditions). In this case, the highest stress intensity range was determined from linearized surface stresses and compared to 3 S_m at operating temperature, resulting in the critical margin of 4.96%. Since the heatup and cooldown transients are not affected for the MUR power uprate, the critical margin of 4.96% is unchanged for the MUR power uprate conditions.

Based on the above discussions, it can be concluded that the existing RCP stress analyses are bounding and remain applicable for the pressure boundary components.

IV.3.6.2 RCP Motor Evaluation

Previous analyses determined that the RCP motors are acceptable for continuous operation with limiting hot loop and cold loop conditions under 2700 MWt. The RCP motors were determined to remain acceptable for operation at the MUR power uprate parameters based on the following:

- No-load T_{ave} is unchanged by the MUR power uprate. Therefore, the RCP hot start is not affected.
- Limiting RCP motor starting conditions occur during RCS cold loop conditions that are unchanged, and therefore not impacted by the MUR power uprate (i.e., T_{cold} remains at the design value of 548°F).
- The mechanical loads controlling RCP motor thrust bearing design are associated with seismic and LOCA conditions (i.e., RCP motor peak accelerations). Seismic loads are not affected by the MUR power uprate, and LOCA condition loads are reduced when BLPBs are invoked as the limiting design basis pipe breaks.
- The thermal transients on the drive motor are not affected by the MUR power uprate.

IV.3.7 Pressurizer

IV.3.7.1 Pressurizer Vessel

The conditions that could affect the primary-plus-secondary stresses, and the primary-plus-secondary-plus-peak stresses, are the changes in the RCS hot leg temperature (T_{hot}), the RCS cold leg temperature (T_{cold}), and the pressurizer transients. Table IV-1 indicates that T_{cold} is unchanged, and that the increase in T_{hot} is very small. A T_{hot} change of this magnitude is enveloped by the current stress analysis. Some of the calculated thermal transients, however, were affected by the MUR power uprate. Therefore, critical locations in the pressurizer were re-examined, as discussed below.

Pressurizer Upper, Bottom and Side RTD Nozzles, and Heater Sleeves

The maximum CUF for these pressurizer locations, after MNSA repairs, is 0.863. In all cases, the CUFs were entirely due to fatigue usage from plant heatup/cooldown and leak test transients. None of these transients are affected by the MUR power uprate; therefore the CUFs for these components are unchanged, and the components remain acceptable for the MUR power uprate.

Surge Nozzle at the Pressurizer End

The surge nozzle at the pressurizer end has a CUF of 0.764. Per the AOR, a UF of 0.716 (or greater than 94% of the CUF) is due to contributions from the normal plant variations at steady-state transient and from the step load increase transient.

The normal plant variations at steady-state transient (defined as ±100 psi and ±6°F) is unchanged by the MUR power uprate. The effect of the MUR power uprate on the step load increase transient was evaluated by calculating stress factors based on a comparison of the calculated transient based on the MUR power uprate setpoints vs. the originally specified transient. The evaluation showed that while the effect of the MUR power uprate on the step load transient increased the alternating stress significantly (by a factor of 2.5) at the nozzle, the original UF was calculated too conservatively. The number of occurrences used in the AOR to calculate the UF for this transient was 34,470 (which is the number of occurrences for heatup/cooldown) instead of the 2,000 occurrences specified for design for step load increases or decreases. By removing that conservatism and adding the MUR power uprate effect, the UF was reduced from 0.716 (pre-uprate conditions) to 0.333 (the MUR power uprate conditions), and the new CUF was reduced to 0.381.

As a result, this component remains acceptable for the MUR power uprate.

It is, therefore, concluded that all pressurizer components meet the stress and fatigue analysis requirement of Section III of the ASME Code 1965 Edition, up to and including the winter 1967 Addenda for plant operation at the MUR power uprate conditions.

IV.3.7.2 Pressurizer Surge Line Piping

Parameters associated with the MUR power uprate were reviewed for their impact on the design basis analysis for the pressurizer surge line piping including the effects of thermal stratification. Nuclear Steam Supply System design parameters, NSSS design transients, and changes at the reactor coolant loop auxiliary Class 1 branch nozzle connections due to deadweight, thermal, seismic, and LOCA loading conditions were considered.

Thermal stratification takes place during plant transients (e.g., during plant heatup), and the temperature ranges defined in the stratification AOR were conservatively based on plant operating data. T_{hot} has increased slightly for the MUR power uprate (see Table IV-1). This change has a negligible effect on the

stratification AOR, 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.

There is no impact on the deadweight analysis due to the MUR power uprate because there is no discernable change in the weight of the auxiliary Class 1 pressurizer surge line piping systems. Fluid weight changes due to the change in T_{hot} are very small, and their effect on the overall piping weight is insignificant. The seismic response spectra remain unchanged. Therefore, there is no impact on the seismic analysis. Although BLPBs could be invoked through LBB implementation, continuing to base the RCS structural analyses on the original design basis LOCA events is conservative and valid. Therefore, no change to the LOCA hydraulic forcing functions is required. In conclusion, the MUR power uprate has no impact on auxiliary Class 1 branch nozzle connection loads resulting from the deadweight, thermal, seismic, or LOCA input loading conditions.

It is noted in the introduction to this section, however, that some of the NSSS calculated thermal transients are affected by the MUR power uprate. The calculated transients refer to thermal transients re-calculated for the MUR power uprate conditions. Reconciliation between the design transients and re-calculated transients for the MUR power uprate was performed for critical locations on the RCS surge lines and fittings. These reconciliations are summarized below.

Surge Line Piping

The critical margin on the maximum primary-plus-secondary stress is 1.78%. Per the AOR, the maximum calculated stress intensity range is based on design pressure, dead weight, seismic loads, and specified normal operating transients for surge line piping.

The design pressure, dead weight, and seismic loads are not affected by the MUR power uprate. Per the AOR, the dominant stresses are from the plant loading transient and the plant unloading transient. The effect of the MUR power uprate on these transients was evaluated. Based on this evaluation, these transients as originally specified for design remain applicable for the MUR power uprate. Therefore, the stresses on the surge nozzle are also unchanged.

The critical CUF of 0.937 in the surge line occurs at the elbow under the pressurizer. This CUF is primarily due to stratified flow and striping, was developed for the Combustion Engineering Owners' Group (CEOG), and represents a bounding case for the combined effects of stratified flow and striping on the maximum CUF of any CE plant surge line.

The report to the CEOG (Reference IV-1) stated the following with respect to the calculated generic CUF of 0.937:

> *The actual usage factor for each specific plant is expected to be lower because 1) the loadings are generic and very conservative, 2) the assumptions made for material properties are conservative, and 3) the most highly stressed line (elastically) was used as the line for shakedown. The highest contribution to fatigue results from a load set which ranges between a non-stratified load state and a 340°F stratified flow load state. Virtually all of the cumulative usage includes load sets with a stratified flow load state. This indicates that the OBE and full flow thermal stresses contribute very little to the overall fatigue conclusions.*

Therefore, there is a much greater margin on the allowable CUF for the Calvert Cliffs surge lines, and the actual margin for the Calvert Cliffs surge lines is not considered critical. As documented in the piping specification, all transients affecting the RCS piping, including the surge line, are unchanged by the MUR

power uprate. Consequently, the Calvert Cliffs surge line piping remains acceptable for the MUR power uprate.

Surge Line Temperature Measurement Nozzle

The surge line temperature measuring nozzle is a RTD nozzle that has a pre-uprate CUF of 0.732, and the transient contributing the largest UF (0.333) is the LOSP transient in combination with heatup, neither of which is affected by the MUR power uprate conditions. The design transients contributing to the RTD nozzle fatigue usage that are also affected by the MUR power uprate are reactor trip, loss of flow, step load increase/decrease, and plant unloading. The step load increase/decrease transients under the MUR power uprate conditions will not increase the alternating stresses or UF on the RTD nozzle. Therefore, the delta UF calculation for the MUR power uprate includes only the effects from the reactor trip, loss of flow, and plant unloading transients. Using a conservative 35 years of additional operation to end-of-life in order to envelop plant operation under the MUR power uprate conditions, the CUF is increased by 0.109, from 0.732 to 0.841. This CUF continues to meet the acceptance criterion. The surge line temperature measuring RTD nozzle is therefore considered acceptable for the MUR power uprate.

Surge Line Sampling Nozzle

The surge line sampling nozzle has a pre-uprate CUF of 0.996, and the transient contributing the largest UF (0.263) is the LOSP transient in combination with heatup, neither of which is affected by the MUR power uprate conditions. In addition, this UF was very conservatively generated originally using the simplified elastic-plastic analysis (i.e., application of the K_e factor as defined in Paragraph NB-3228.5 of the ASME Code). A full elastic-plastic analysis reduces this UF from 0.263 to 0.044, thereby reducing the pre-uprate CUF to 0.777 (0.996 - 0.263 + 0.044).

The design transients that contribute to the sampling nozzle fatigue usage and that are also affected by the MUR power uprate conditions are reactor trip, loss of flow, step load increase/decrease and plant unloading. The step load increase/decrease transients under the MUR power uprate conditions will not increase the alternating stresses or UF on the sampling nozzle. Therefore, the delta UF calculation for the MUR power uprate includes only the effects from the reactor trip, loss of flow, and plant unloading transients. Using a conservative 35 years of additional operation to end-of-life in order to envelop plant operation under the MUR power uprate conditions, the CUF is increased by 0.207, from 0.777 to 0.984. This CUF continues to meet the acceptance criterion. The surge line sampling nozzle is, therefore, considered acceptable for the MUR power uprate.

Based on the above, the existing pressurizer surge line piping analysis remains valid.

IV.4 EFFECTS OF OPERATING POINT DATA VARIATIONS

The MUR power uprate operating point values shown in Table IV-1 represent a best estimate. In all probability, the MUR power uprate operating point may move slightly over time, resulting in small changes in the operating point parameters.

Regardless of these small anticipated changes, particularly in the operating temperatures and the resulting delta-T, the structural AOR performed for the Calvert Cliffs Units 1 and 2 RCS components remains bounding. The following discussion is based on the fact that the AOR considered T_{hot} and T_{cold} design values of 604°F and 548°F, respectively, with a resulting delta-T of 56°F.

IV.4.1 Reactor Coolant System Thermal Movements

The maximum thermal movements of various locations on the RCS (e.g., tributary nozzle ends) result from the change in RCS temperature from ambient conditions to operating conditions. The MUR power uprate thermal movements will be enveloped by the AOR results, since AOR results are based on ambient to operating condition nominal temperature ranges that bound the temperature ranges associated with the MUR power uprate conditions. This was demonstrated in an analysis comparing RCS thermal movements due to design operating setpoint temperatures to similar results determined at the MUR power uprate nominal setpoint temperatures. The MUR power uprate condition thermal movements either remained the same or decreased slightly, relative to the movements due to design operating setpoint temperatures. In general the decreases were on the order of 1 to 2%. Maximum decreases were 4 to 5%.

Furthermore, this conclusion will remain valid if the nominal values of T_{hot} and T_{cold} vary slightly after the MUR power uprate, because 1) there is sufficient margin between the MUR power uprate nominal T_{hot} value of 595.9°F and the design T_{hot} of 604°F, and 2) the T_{cold} value is anticipated to remain at the design value of 548°F, which has been the case for previous plant operation.

IV.4.2 Reactor Coolant System Loads

Reactor Coolant System component nozzle and primary piping thermal expansion loads are directly affected by delta-T, the temperature difference between T_{hot} and T_{cold}. Given the same RCS configuration and operating temperatures that are generally the same, lower delta-T values result in lower piping and nozzle loads, which in turn result in proportionally lower loads at intermediate component locations and at the component supports. This conclusion can be drawn because the general RCS characteristics of stiffness, mass and connectivity will not change for the MUR power uprate, thus resulting in an overall RCS load distribution for the MUR power uprate conditions that are very similar to the load distribution analyzed in the AOR.

The delta-T values associated with current and the MUR power uprate conditions are both less than the delta-T value used in the AOR. Therefore, even though delta-T increases slightly when going from the current to the MUR power uprate conditions (by ~1°F), the AOR piping, component and component support thermal expansion loads remain bounding, because they are associated with a higher value of delta-T.

Per Section IV.3, the majority of the AOR design thermal transients remain bounding for the MUR power uprate. Even those that do not remain bounding were demonstrated to have little effect on the AOR stress calculations (see detailed discussions in Section IV.10.3).

Original design basis RCS seismic analysis results are negligibly affected by the MUR power uprate, because small changes in temperature have virtually no effect on the material properties of the structure, and therefore, on the manner in which the structure responds to a given set of input loads. Furthermore, Section IV.2 concludes that it is valid to base the MUR power uprate LOCA evaluations on the original DBEs. Furthermore, since LBB can be used to mitigate any adverse effects from the MCLB load contributions, basing the MUR power uprate LOCA evaluations on the original DBEs is both valid and conservative.

Finally, since the RCS structure responds to the same design input loadings in essentially the same manner under the MUR power uprate conditions, the original design basis structural analysis results remain valid.

IV.4.3 Reactor Coolant System Stresses and Usage Factors

Since the AOR normal operating conditions, seismic, and LOCA structural analysis results remain bounding for the MUR power uprate, the only changes to the AOR design, emergency, and faulted condition load combinations used to calculate the stresses and fatigue UFs of record are related to the design thermal transients. As discussed above and throughout Section IV.3, the CUFs determined in the AOR were insensitive to the effects of the transient input changes associated with the MUR power uprate. It is safe to conclude that any further, even smaller, changes resulting from operating point drift will also be acceptable.

It is also noted that the ASME Code stress allowables used in the AOR are unaffected by small changes in operating temperatures, leading to the conclusion that the bounding stresses determined in the AOR will continue to remain below their corresponding ASME Code allowables. Consequently, the structural integrity of the RCS components is further confirmed for small variations in the MUR power uprate conditions, and the stress margins identified in the AOR calculations remain applicable.

IV.5 REACTOR VESSEL INTEGRITY

The factors influencing RV integrity are the initial properties of the materials and the neutron fluence incident on the materials. The MUR power uprate does not affect the initial material properties, but the neutron fluence can change. The effect of neutron fluence changes on vessel integrity are assessed below using 10 CFR Part 50, Appendices G and H, and 10 CFR 50.61.

Pressurized Thermal Shock - The screening criteria in 10 CFR 50.61 are 270°F for forgings, plates, and axial welds and 300°F for circumferential welds. 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.

Vessel Fluence Evaluation - The Units 1 and 2 extended end-of-life neutron fluence values were re-evaluated assuming a 1.4% MUR power uprate (which bounds the requested MUR power uprate of 1.38%).

The fluence results are listed in the following table.

	Unit 1 Fluence	Unit 1 Fluence	Unit 2 Fluence	Unit 2 Fluence
	Current	1.4% Power Uprate	Current	1.4% Power Uprate
Critical Weld	5.09E+19	5.11E+19	5.74E+19	5.79E+19
1/4 T Location	3.06E+19	3.08E+19	3.02E+19	3.05E+19
3/4 T Location	6.09E+18	6.12E+18	6.33E+18	6.38E+18

For completeness, the RT_{PTS} values for the RV welds and plates are listed in the following table. Note that the RT_{PTS} values are well below the 10 CFR 50.61 pressurized thermal shock screening criteria limits of 270°F for plates, forgings, and axial weld materials, and 300°F for circumferential weld materials.

	Unit 1	Unit 1	Unit 1	Unit 1
	Current	Current	1.4% Power Uprate	1.4% Power Uprate
Seam/Plate	Fluence n/cm^2	RT_{PTS} °F	Fluence n/cm^2	RT_{PTS} °F
WELDS				
2-203-A/B/C	5.09E+19	243.6	5.11E+19	243.7
3-203-A/B/C	5.09E+19	254.1	5.11E+19	254.2
9-203	5.09E+19	53.4	5.11E+19	53.5
PLATES				
D-7206-1	5.09E+19	158.0	5.11E+19	158.1
D-7206-2	5.09E+19	105.1	5.11E+19	105.2
D-7206-3	5.09E+19	145.1	5.11E+19	145.2
D-7207-1	5.09E+19	170.5	5.11E+19	170.6
D-7207-2	5.09E+19	139.3	5.11E+19	139.3
D-7207-3	5.09E+19	118.0	5.11E+19	118.1

	Unit 2	Unit 2	Unit 2	Unit 2
	Current	Current	1.4% Power Uprate	1.4% Power Uprate
Seam/Plate	Fluence n/cm^2	RT_{PTS} °F	Fluence n/cm^2	RT_{PTS} °F
WELDS				
2-203-A/B/C	5.74E+19	122.9	5.79E+19	123.0
3-203-A/B/C	5.74E+19	55.1	5.79E+19	55.3
9-203	5.74E+19	72.3	5.79E+19	72.4
PLATES				
D-8906-1	5.74E+19	198.3	5.79E+19	198.5
D-8906-2	5.74E+19	149.7	5.79E+19	149.8
D-8906-3	5.74E+19	179.0	5.79E+19	179.2
D-8907-1	5.74E+19	183.1	5.79E+19	183.3
D-8907-2	5.74E+19	167.0	5.79E+19	167.1
D-8907-3	5.74E+19	128.0	5.79E+19	128.1

Heatup and Cooldown Pressure Temperature Limit Curves and Low Temperature Overpressure Protection – 10 CFR Part 50, Appendix G addresses the limits on pressure and temperature that are placed on heatup and cooldown during normal operation. 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 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.

Upper Shelf Energy - 10 CFR Part 50, Appendix G requires that the upper shelf energy throughout the life of the vessel be no less than 50 ft-lb. Projections were done in accordance with Regulatory Guide 1.99, Revision 2, and were based on the neutron fluence values through the end of the extended license adjusted to represent conditions for power uprate. 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.

Surveillance Capsule Withdrawal Schedule - 10 CFR Part 50, Appendix H defines the RV surveillance program that is to be used by the licensee to monitor the neutron radiation induced changes in fracture toughness of the vessel during the life of the plant. It includes requirements to establish a surveillance capsule withdrawal schedule. The schedule for Calvert Cliffs Units 1 and 2 has been updated based on the fluence projected for the extended license. 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.

IV.6 NUCLEAR FUEL

This section summarizes the evaluations performed to determine the effect of the MUR power uprate on the nuclear fuel. The core design for Calvert Cliffs is performed on a fuel cycle specific basis and varies according to the needs and specifications for each fuel cycle. However, some fuel-related analyses are not cycle specific. The nuclear fuel review for the MUR power uprate evaluated the fuel assembly mechanical performance, the fuel core design, thermal-hydraulic design, and fuel rod performance.

IV.6.1 Fuel Assembly Mechanical Performance

The Calvert Cliffs 14x14 fuel design was evaluated to determine the impact of the MUR power uprate on the fuel assembly design criteria. The evaluation concluded that the Calvert Cliffs fuel design remains acceptable and continues to satisfy the required design criteria under the operating temperature, operating pressure, and flow rates resulting from the MUR power uprate conditions.

The evaluation methodology compared significant operating parameter values, used in the AOR, with the values of those same parameters proposed for the MUR power uprate. The significant parameters evaluated included inlet temperature, system pressure, core average LHRs, maximum fuel rod axial average fluence, minimum coolant flow rate, fuel residence time, and peak fuel rod burnup. These parameters affect such important design criteria issues as the fuel rod stress, strain, fatigue, and clad collapse, as well as the fuel assembly hold-down margin and shoulder gap. The evaluation of the comparison of these significant parameter values showed that the proposed MUR power uprate operating and transient values are the same as or bounded by the existing AOR values except for the core average LHR. Sufficient margins exist, however, to allow for the power uprate increase in that parameter. Since the core plate motions for the seismic and LOCA evaluations are not affected by the uprated conditions, there is no impact on the fuel assembly seismic/LOCA structural evaluation.

Therefore, the fuel mechanical performance design criteria will continue to be satisfied under the proposed MUR power uprate conditions.

IV.6.2 Fuel Core Design

The impact of a bounding 1.7% uprate condition on the fuel core design was evaluated against the data used in the current Calvert Cliffs safety AOR. Since the MUR power uprate is relatively small, the range of parameters used in the current safety AOR are adequate to accommodate the range of parameters expected for future cores that have implemented the MUR power uprate. The core analyses for specific uprate cycles has shown that the implementation of the MUR power uprate does not result in significant changes to the current nuclear design basis for the safety analysis documented in the UFSAR. The impact of the MUR power uprate on peaking factors, rod worths, reactivity coefficients, shutdown margin, and kinetics parameters is either well within normal cycle-to-cycle variation of these values or controlled by the core design and will be addressed on a cycle-specific basis consistent with reload methodology.

The methods and core models used in the MUR power uprate analyses are consistent with those presented in the Calvert Cliffs UFSAR. No changes to the nuclear design philosophy, methods, or models are necessary due to the MUR power uprate. The current range of required cycle specific analysis is sufficient to verify the applicability of these parameters for future cycles.

IV.6.3 Core Thermal-Hydraulic Design

The core thermal-hydraulic design and methodology were evaluated for the MUR power uprate. The thermal hydraulic design is based on the TORC computer code described in Reference IV-2, the ABB-TV and ABB-NV [non-mixing vane] critical heat flux correlations described in Reference IV-3, the simplified TORC modeling methods described in Reference IV-4, and the CETOP-D code described in Reference IV-5. In addition, the DNBR analysis uses the methodology for determining the limiting fuel assembly or assemblies.

The Extended Statistical Combination of Uncertainties presented in Reference IV-6 and approved in Reference IV-7 was applied to validate the design limit of 1.24 on the ABB-TV and ABB-NV minimum DNBR. This DNBR limit includes the following allowances:

1. NRC specified allowances for TORC code uncertainty.
2. Rod bow penalty equivalent to 0.6% on minimum DNBR as discussed in Reference IV-8.

The core thermal-hydraulic design and methodology remain applicable for Calvert Cliffs with the MUR power uprate.

IV.6.4 Fuel Rod Design

As noted in previous sections (e.g., II.3) Calvert Cliffs Unit 1 Cycle 17 was originally targeted as the lead unit for implementation of the MUR power uprate. Subsequent to performance of the Unit 1 Cycle 17 analyses and evaluations, a fuel design change that implements the use of ZrB_2 integral burnable absorbers was submitted to the NRC (Reference IV-17) and approved (Reference IV-18). Unit 2 Cycle 16 was the lead unit for the fuel design change. The analyses performed to support the transition to ZrB_2 have already included the MUR power uprate as an initial condition. No additional analyses to support both ZrB_2 and the MUR power uprate are required. Application of the MUR power uprate analyses and evaluations have been included as part of the normal reload process for all subsequent Calvert Cliffs Units 1 and 2 cores since that time (presently Unit 1 is on Cycle 19 and Unit 2 is on Cycle 17).

Starting with the Calvert Cliffs Unit 1 Cycle 17 core, the thermal performance of Erbia and UO_2 fuel rods with the MUR power uprate were evaluated using the FATES3B version of the fuel EM (References IV-7, IV-8, and IV-9), the Erbia burnable absorber methodology described in Reference IV-10, the maximum pressure methodology described in Reference IV-11, and the ZIRLO™ fuel rod cladding methodology described in Reference IV-12. This evaluation included a power history that enveloped the power and burnup levels expected for the peak fuel rod at each burnup interval, from beginning-of-cycle to end-of-cycle burnups, including a reduction in maximum permitted F_r, consistent with implementation of the power uprate. The maximum predicted fuel rod internal pressure for the uprated core remains below the critical pressure for No-Clad-Lift-Off (Reference IV-14).

The thermal performance of Erbia and UO_2 fuel rods for Calvert Cliffs Unit 1 (subsequent to Cycle 17) and Unit 2 (subsequent to Cycle 15) cores with the MUR power uprate is evaluated using the FATES3B version of the fuel EM (References IV-9, IV-10, and IV-11), the Erbia burnable absorber methodology described in Reference IV-12, the maximum pressure methodology described in Reference IV-13, and the

ZIRLO™ fuel rod cladding methodology described in Reference IV-14. These evaluations include a power history that envelopes the power and burnup levels expected for the peak fuel rod at each burnup interval, from beginning-of-cycle to end-of-cycle burnups. The maximum predicted fuel rod internal pressure for the uprated conditions will be shown to remain below the critical pressure for No-Clad-Lift-Off (Reference IV-13).

The expected fuel rod corrosion performance for Calvert Cliffs Units 1 and 2 cores with MUR power uprate was evaluated and found acceptable. This evaluation was conducted consistent with requirements of the NRC SER on the high burnup topical report for 14x14 CE design fuel of Reference IV-15. This evaluation also considered the impact of recent high duty corrosion observations for OPTIN™ clad fuel that may be resident in Calvert Cliffs Units 1 and 2 (see also Reference IV-16). The fuel rod corrosion performance of OPTIN™ and ZIRLO™ clad fuel specifically for the Calvert Cliffs Unit 1 Cycle 17 core (including MUR power uprate) were evaluated (References IV-14, IV-15, and IV-16) and found to be acceptable. The fuel rod corrosion performance for MUR power uprated Calvert Cliffs Unit 1 (subsequent to Cycle 17) and Unit 2 (subsequent to Cycle 15) cores is evaluated using this same methodology.

IV.7 BALANCE OF PLANT PIPING

The balance of plant (BOP) piping systems impacted by the uprate (main steam, feedwater, extraction steam, moisture separator drains, reheater drains, condensate, and heater drain piping) have been evaluated by comparing the conditions for the proposed power uprate with the current operating conditions. The design temperatures and pressures used in the analyses continue to bound the uprate conditions. The maximum operating temperatures with the MUR power uprate are within 1% of the existing maximum operating temperatures.

The BOP piping systems remain acceptable for operation at the MUR power uprate conditions, and the proposed 1.38% power uprate will not have adverse effects on the BOP piping.

IV.8 CODE OF RECORD

The allowable stress formulae defining the primary stress limits for the core shroud, as specified in the Calvert Cliffs UFSAR, were adopted prior to the establishment, of specific design criteria for core support structures by the ASME Boiler & Pressure Vessel Code. Core support structure-specific design criteria were formally introduced as Subsection NG in the Winter 1973 Addendum to Section III of the ASME Boiler & Pressure Vessel Code. Therefore, the core shroud evaluation described above, used allowable stress values defined in Subsection NG of the Winter 1973 Addendum. Rules for the evaluation of core support structures at elevated temperatures have not yet been approved. Subsection NH in the 1998 Edition of Section III of the ASME Code, which provides rules for the design of Class 1 components in elevated temperature service, was therefore used to adjust the allowable stress values defined in Subsection NG of the Winter 1973 Addendum.

In conclusion, the Calvert Cliffs Units 1 and 2 primary piping and tributary nozzles remain within allowable stress limits in accordance with ASME Section III, 1965 edition, up to and including the Winter 1967 Addendum (and in accordance with ASME Section III, 1986 Edition for components with MNSA). Furthermore, no piping or pipe restraint modifications are required as a result of the increased power level, because conservatively determined LOCA loads due to MCLBs were used to design the pipe restraint systems.

IV.9 COMPONENT INSPECTION, TESTING, AND EROSION/CORROSION PROGRAMS

IV.9.1 Flow Accelerated Corrosion

The MUR power uprate has no immediate impact on the flow accelerated corrosion program scope but does result in a slight increase in long-term scope. This long-term impact includes increased inspection scope for some specific systems and possibly some additional replacement scope prior to the end-of-plant life expectancy. The components included in the increased inspection scope will be determined by analyzing the projected wear rate changes through the use of Chec-Works modeling software. It is expected that the feedwater system would experience the largest increase in wear. However, it should be noted that, even in the feedwater line, the 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.

IV.9.2 Inservice Inspection Program

The inservice inspection (ISI) program defines the scope and method of examination of Class 1, 2, and 3 components, and also supports the procedures and examination schedule of these components at Calvert Cliffs.

The MUR power uprate does not impact the scope, method of examination, schedule and requirements, or criteria of the ISI program. Additionally, the operating condition changes associated with the MUR power uprate are bounded by the design of the ISI components and supports and do not affect the program scope, selection criteria, or acceptance standards. Therefore, the ISI program is not affected by the MUR power uprate.

IV.9.3 Inservice Testing Program

The inservice testing (IST) program at Calvert Cliffs defines the scope of Class 1, 2, and 3 pumps and valves to be tested, the test method, and test schedule.

The MUR power uprate does not impact the scope, test methods, schedule and requirements, or criteria of the IST program. Additionally, the operating condition changes associated with the MUR power uprate are bounded by the design of the IST pumps and valves and do not affect the scope, selection criteria, or acceptance standards. Therefore, the IST program is not affected by the MUR power uprate.

IV.9.4 Alloy 600 Program

Industry experience in PWRs has shown that Alloy 600 (Inconel 600) components and Alloy 82/182 weld filler metals are susceptible to primary water stress corrosion cracking. The program includes all Alloy 600 components and Alloy 82/182 welds that are part of the RCS pressure boundary, integral attachments to the RCS pressure boundary, or can have a direct or indirect effect on the integrity of the RCS pressure boundary. These components include: partial penetration welded nozzles and penetrations in the RCS fabricated from Alloy 600 material, welds made with Alloy 82 or 182 filler metal, full-penetration welds made with Alloy 82 and 182 filler metal, and Alloy 600 piping components, non-pressure boundary Alloy 600 components such as welded internal attachments to vessels, and thermal sleeves. Steam generator tubes and the associated tube-to-tube sheet seal welds, are specifically excluded from this program.

This program has assessed the Alloy 600 components and for each of them has documented the risk of component failure. As part of the program, system reliability is evaluated with respect to potential for equipment degradation. The system reliability is in part based on Calvert Cliffs susceptibility modeling of Alloy 600 components. Primary water stress corrosion cracking has been shown to be predominantly temperature and environment dependent. As such, with an increase in RCS temperature, Alloy 600

susceptibility could potentially be challenged. Therefore, a review was performed on the impact of a temperature increase as a result of the MUR power uprate with regards to Alloy 600 susceptibility.

As part of the MUR power uprate it was determined that the RCS temperature would only increase by 0.8°F on the hot leg. The RCS pressure, flow, and cold leg temperatures would remain the same. Thus, it is anticipated for the worst case scenario that the overall increase experienced by Alloy 600 materials is a 0.8°F increase. The review of this increase on the Alloy 600 components concluded that this increase in temperature affects the Alloy 600 component aging but has an insignificant impact on the components' risk of failure.

IV.9.5 Coatings

Coatings used within the Containment were specified based on their ability to withstand accident conditions. The Containment is designed to withstand an internal pressure of 50 psig at 276°F including all thermal loads resulting from the temperature associated with this pressure (UFSAR Section 14.20.2). The coatings within the Containment are not impacted by the MUR power uprate since the mass and energy values are not changed from previously analyzed conditions.

IV.9.6 Steam Generator Program

The purpose of the SG program is to ensure the structural and leakage integrity of the tubes through the implementation of the following program elements:

- Assessment of existing degradation mechanisms in the reactor coolant pressure boundary within the SG
- Steam generator inspection in accordance with the Electric Power Research Institute PWR SG examination guidelines
- Assessment of tube integrity after each SG inspection to ensure that the performance criteria for the operating period have been met and will continue to be met for the next period
- Maintenance, plugging, and repairs of SG tubes
- Primary-to-secondary leakage monitoring
- Maintenance of SG secondary side integrity
- Primary side and secondary side water chemistry
- Foreign material exclusion
- Self-assessment of the SG program
- Preparation of NRC and industry reports

A review of the SG program elements has concluded that the program elements are symptom based, augmented by regular inspections, maintenance and chemistry activities, and industry experiences. At the MUR power uprate conditions, the SG tubes are exposed to a 0.8°F increase in temperature. This temperature increase slightly increases the chance of stress corrosion cracking in the SG tubes. The existing plugging margin and inspection program elements are sufficient to ensure tube integrity. The SG program elements are independent of the reactor thermal power and therefore, the SG program elements are not affected by the MUR power uprate.

IV.9.7 Containment Leak Rate Program

The containment leak rate testing program performs the Type A, B, and C containment leakage testing to verify the integrity of the Containment and those systems and components which penetrate the containment walls.

The MUR power uprate does not impact the scope, requirements or criteria of the containment leak rate testing program. Additionally, the operating condition changes associated with the MUR power uprate do not affect the Containment or the systems and components which penetrate the containment walls. The containment pressure following a DBA from the MUR power uprate conditions is bounded by the AOR performed at 102% thermal power. Therefore, the containment leak rate testing program is not affected by the MUR power uprate.

IV.9.8 Motor-Operated Valve Program

The proposed MUR power uprate will not impact the Generic Letter 89-10 Motor-Operated Valve (MOV) program. The following systems contain valves within the MOV program:

1. Instrument air system,
2. Safety injection system,
3. Plant drain system,
4. Primary containment heating and ventilation,
5. Containment spray system,
6. Chemical and Volume Control System,
7. Reactor Coolant System, and
8. Feedwater system.

The variables that could affect MOV performance are increased differential pressure across the valve, increased effects of pressure locking/thermal binding, and increased temperature experienced by the actuator motor.

Systems 1 through 5 above are not impacted by the MUR power uprate. The 11 valves in systems 6 through 8 could potentially be impacted by the uprate. The differential pressure calculations for these valves were reviewed and all of them use system design pressures for calculating maximum differential pressure. Since the system design pressures are not changing there is no effect on the calculated differential pressures across the valves.

The valves susceptible to pressure locking or thermal binding are the power-operated relief block valves and the SDC return line valves. These valves possess engineered features that preclude pressure locking or thermal binding.

The maximum design temperature of the room in which the motor is located is used to calculate the torque reduction effect of increased temperature. Since the design temperatures are not changing there will be no effect on the MOV motors.

As identified in Section II.1, there are no changes to the safety analysis (i.e., existing analysis of the MSLB and LOCA remain bounding). Consequently, this results in no impact to the MOV program.

IV.9.9 Air-Operated Valves

The air-operated valve (AOV) program was evaluated for impact due to the proposed MUR power uprate.

A review considered valves in the main steam, feedwater, and other secondary side systems. Valves in primary side systems such as SI and CVCS should not be affected since RCS pressure is not changing. From an AOV program standpoint, the main concern is differential pressure across the valve and flow through the valve. Thrust calculations to determine required outputs from the air-operators conservatively assume worst case differential pressure across a valve. The following valves are addressed in the review:

1. Atmospheric Dump Valves
2. Feedwater Regulating Valves
3. Feedwater Regulating Bypass Valves
4. Main Steam to Auxiliary Feedwater (AFW) Pumps

There is no impact from a thrust standpoint because the calculations assume the highest SG pressure based on pressure limits, MSSV settings, and SG feed pumps running at shutoff head. These are conservatively higher pressures or D/Ps than the MUR power uprate will implement, so there is no impact on actuator capability.

Based on the information that was provided through the heat balance calculation generated using the plant specific model, there is no impact to AOV program valves from an actuator thrust standpoint.

IV.9.10 Non-Program Valves

Since increases in differential pressure are minimal and flow rates will only increase about 2% or less, it was determined that the control valves will be able to handle the increased flow due to the MUR power uprate.

IV.10 FIRE PROTECTION

This evaluation has been conducted in order to evaluate the effects of the MUR power uprate on the plant's fire protection program.

The plant fire protection program is the integrated effort involving systems, structures, components, procedures, and personnel used to carry out all activities of fire protection, fire prevention, and to ensure safe shutdown following a fire event. The fire protection program uses a defense in depth concept to prevent fires from starting, to detect, control, and suppress those fires that do occur; and to ensure that fire will not prevent essential plant safety functions from being performed.

Both units are served by a fire protection system that provides a reliable fire protection water supply delivering fire protection water in quantities sufficient to satisfy the maximum probable demand; and, automatic and manual fire protection systems and equipment that provide fire suppression capabilities.

The fire protection program and fire protection features are described in Calvert Cliffs UFSAR Section 9.9. Fire protection features include the fire water supply; fire pumps and distribution piping; fixed water suppression systems; fixed gaseous suppression systems; manual fire suppression systems; and fire detection and alarm systems. Passive fire protection features include fire barriers and fire rated penetration seals. Fire and emergency response activities are performed by the on site fire brigade.

The changes that will occur as a result of the MUR power uprate which increase the thermal and electrical power of the plant have been evaluated with respect to their impact on plant fire protection. The results of the evaluation are that the MUR power uprate has no affect on the plant's fire protection program.

IV.11 APPENDIX R

The goal of 10 CFR Part 50, Appendix R, is to ensure safe shutdown of the reactor following a fire in any plant area, thereby preventing core damage and protecting the public. Appendix R applies to plants licensed prior to January 1, 1979.

Appendix R compliance can be affected by adding heat to plant areas that could affect Appendix R safe shutdown because the higher temperatures could affect Appendix R equipment and plant operators. However, the overall temperature changes in the primary and secondary systems are very small such that the issue of added heat load to the plant is not a concern.

Appendix R can be affected by additional decay heat due to the higher power levels. This additional decay heat associated with the changes from the MUR power uprate was evaluated and found to be negligible.

IV.12 HIGH ENERGY LINE BREAK

The Calvert Cliffs high energy line break (HELB) analysis was reviewed in support of the MUR power uprate. The activities, elements, and philosophy that currently constitute the HELB analysis are not affected by the MUR power uprate. The slight lowering of the secondary pressure limits the mass flowrate through the break location. Although an extremely slight increase in enthalpy occurs with a decrease in saturated steam pressure and temperature, the lowered choked flowrate more than compensates for this. As a result, the overall impact from the proposed MUR power uprate is bounded by the existing HELB analysis. In accordance with Calvert Cliffs design change process, the design change package for installing the Caldon LEFM CheckPlus system was evaluated against the HELB analysis requirements. No new piping was added, no postulated break locations changed, and no changes were made to the assumed blowdown from any currently-postulated breaks; therefore, there is no impact on the current Calvert Cliffs HELB analysis.

IV.13 REFERENCES

IV-1 CEN-387-P, Revision 1-P-A, "Pressurizer Surge Line Flow Stratification Evaluation," May 1994

IV-2 CENPD-206-P-A, "TORC Code: Verification and Simplified Modeling Methods," June 1981

IV-3 CENPD-387-P-A, "ABB Critical Heat Flux Correlations for PWR Fuel," May 2000

IV-4 CENPD-206-P-A, "TORC Code: Verification and Simplified Modeling Methods," June 1981

IV-5 CEN-191(B)-P, "CETOP-D Code Structure and Modeling Methods for Calvert Cliffs Units 1 and 2," December 1981

IV-6 CEN-348(B)-P-A, Supplement 1-P-A, "Extended Statistical Combination of Uncertainties," January 1997

IV-7 Letter from G.M. Holahan (NRC) to S.A. Toelle (ABB), dated August 31, 1994, "Generic Approval of CEN-348(B)-P-A, 'Extended Statistical Combination of Uncertainties' (TAC No. M90019)"

IV-8 CENPD-225-P-A, "Fuel and Poison Rod Bowing," June 1983

IV-9 CEN-161(B)-P Supplement 1-P-A, "Improvements to Fuel Evaluation Model," January 1992

IV-10 CENPD-139-P-A, "Fuel Evaluation Model," July 1974

IV-11 CEN-161(B)-P-A, "Improvements to Fuel Evaluation Model," August 1989

IV-12 CENPD-382-P-A, "Methodology for Core Designs Containing Erbium Burnable Absorbers," August 1993

IV-13 CEN-372-P-A, "Fuel Rod Maximum Allowable Gas Pressure," May 1990

IV-14 CENPD-404-P-A, Revision 0, "Implementation of ZIRLO™ Cladding Material in CE Nuclear Power Fuel Assembly Designs," November 2001

IV-15 CEN-382(B)-P-A, "Verification of the Acceptability of a 1-Pin Burnup Limit of 60 MWD/kgU for Combustion Engineering 14x14 PWR Fuel," August 1993

IV-16 CENPD-384-P, "Report on the Continued Applicability of 60 MWD/kgU for ABB Combustion Engineering PWR Fuel," September 1995

IV-17 Letter from B.S. Montgomery (CCNPP) to Document Control Desk (NRC), dated July 15, 2004, "License Amendment Request: Incorporate Methodology References for the Implementation of PHOENIX-P, ANC, PARAGON, and Zirconium Diboride into the Technical Specifications"

IV-18 Letter from R.V. Guzman (NRC) to G. Vanderheyden (CCNPP), dated February 24, 2005, "Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2 – Amendment RE: Incorporating Core Operating Limits Analytical Methodology References into Technical Specifications (TAC Nos. MC4019 and MC4020)"

V. ELECTRICAL EQUIPMENT DESIGN

This section summarizes the evaluations performed to determine the effect of the MUR power uprate on the electrical equipment. The electrical equipment included in the evaluation is presented within each subsection.

V.1 EMERGENCY DIESEL GENERATORS/STATION BLACKOUT EQUIPMENT

Calvert Cliffs onsite electrical distribution systems include non-Class 1E plant service transformers and associated busses. The 4.16 kV, 480 Volt, and 120/208 Volt systems include both Class 1E and non-Class 1E equipment. The onsite direct current (DC) distribution system includes both Class 1E and non-Class 1E equipment.

The Class 1E AC distribution system includes two Class 1E 4.16 kV busses per operating unit, each capable of being powered by an associated Class 1E standby emergency diesel generator in the event of a loss of offsite power. A station blackout diesel generator is designed to provide sufficient power to any of the four Class 1E 4.16 kV busses in order to safely shutdown one unit and maintain it in a safe shutdown condition during a station blackout event. Downstream 480 Volt and 120 Volt busses also feed two trains of redundant safety equipment. As referenced in Section II.1, there is no change to the existing accident analyses and they continue to be valid for the MUR power uprate. The electrical motors and supporting equipment are sized for maximum accident load requirements. Thus, the emergency diesel generators remain sufficient to provide all required electrical loads and there is no need to upgrade any other existing Class 1E electrical equipment.

The non-Class 1E AC distribution systems provide power for non-safety-related systems during normal plant conditions. The large non-safety loads powered from these busses include condensate pumps, condensate booster pumps, and heater drain pumps. The MUR power uprate does not result in an increase in mechanical load beyond the design rating of any non-Class 1E equipment. The motors and associated support and protective equipment are sized based on design ratings, thus they are adequately sized for the small load increase resulting from the MUR power uprate.

The onsite DC distribution system will see minor load variations due to the power uprate; however the resulting electrical loads remain within the ratings of the existing distribution system and no changes are required.

V.2 MAIN GENERATOR AND ASSOCIATED SYSTEMS

The Unit 1 Main Generator has a design rating of 1,020 MVA at 25 kV 60 Hz when operated at 0.9 lagging power factor (918 MW) and hydrogen pressure of 60 psig. Unit 2 Main Generator is rated at 1,012 MVA at 22 kV 60 Hz when operated at 0.9 lagging power factor (910.8 MW) and hydrogen pressure of 75 psig. The new operating point corresponding to the MUR power uprate is within the design rating of both machines. The generators are operated to produce power output within the limits of their associated reactive capability curves. If required, the MVAR output of the generator can be adjusted such that the total MVA rating is not exceeded. No modification to auxiliary or support equipment is required.

Applicable calculations were reviewed and determined no changes are required for generator voltage regulator and associated protective relay settings.

Two unit transformers are connected via an isolated phase bus to the output of each main generator and are designed to carry the maximum generator output and transform generator output voltage to transmission system voltage. Each of the paralleled transformers is rated for 810 MVA at 65°C rise. The

maximum MVA rating of the Units 1 and 2 generators remain at 1020 and 1012 MVA which is well within the rating of the paralleled transformers.

The associated isolated phase bus and switchyard equipment are also rated for maximum current flow from the generator, thus no modification to this equipment is required. However, the existing unit limitations for conditions when the unit's isophase bus duct cooling is not available or operating at rated capability, or when one of the unit's generator step-up transformers is not available still remain.

V.3 ENVIRONMENTAL QUALIFICATION OF ELECTRICAL EQUIPMENT

In accordance with 10 CFR 50.49, certain electrical equipment must be qualified to operate when exposed to the postulated harsh accident environments of DBAs (i.e., LOCA, MSLB, HELBs). The qualification includes aging considerations of normal plant operating ambient environments. The effects of the proposed MUR power uprate on the 10 CFR 50.49 EQ program is as follows:

V.3.1 Environmental Qualification Accident (Temperature/Pressure) Environments

As discussed in Section II.5, the current UFSAR Chapter 14 Containment LOCA and MSLB temperature/pressure analyses will not be affected (i.e., remain bounding) considering the MUR power uprate. The current EQ accident (temperature/pressure) environments utilize these UFSAR Chapter 14 analyses. Therefore, the current inside-Containment EQ equipment LOCA/MSLB temperature/pressure qualification is unaffected by the MUR power uprate.

As discussed in Section VII.5, the current UFSAR Chapter 10A outside-Containment HELB temperature/pressure analyses will not be affected (i.e., remain bounding) considering the MUR power uprate. The current EQ accident (temperature/pressure) environments utilize these UFSAR Chapter 10A analyses. Therefore, the current outside Containment EQ Equipment HELB temperature/pressure qualification is unaffected by the MUR power uprate.

V.3.2 Environmental Qualification Accident (Radiation) Environments

As discussed in Section III, the current accident radiation doses, utilized in the EQ program, required re-evaluation as a result of the proposed MUR power uprate. Environmental qualification equipment was re-evaluated against these revised accident radiation doses and confirmed to remain environmentally qualified to these revised accident doses.

V.3.3 Environmental Qualification Normal Plant Operating Ambient (Temperature/Humidity) Environments

As discussed in Section VI.6, the heating, ventilation, and air conditioning (HVAC) systems, control normal plant operating ambient environments in Containment and Auxiliary Building. Environmental qualification equipment is located in both the Containment and the Auxiliary Building. These HVAC systems were reviewed considering the MUR power uprate. The review determined that these existing HVAC systems will continue to maintain the Containment and Auxiliary Buildings within their UFSAR Chapter 9 design ranges. Therefore, the current normal plant operating aging considerations, utilized in the EQ program, are not impacted and are bounded by the current design.

V.3.4 Environmental Qualification Normal Plant Operating Ambient (Radiation) Environments

As discussed in Section III, the current normal operating radiation doses, utilized in the EQ Program, required re-evaluation as a result of the proposed MUR power uprate. Equipment was re-evaluated against these revised normal operating radiation doses and confirmed to remain environmentally qualified to these revised normal operating doses.

V.4 GRID STABILITY

The Pennsylvania, New Jersey, Maryland Interconnection has preliminarily reviewed the power uprate for impact on grid stability. The proposed increase in plant electrical output does not affect the stability of the grid. No switchyard modifications are required as a result of the MUR power uprate.

VI. SYSTEM DESIGN

This section presents the results of the evaluations and analyses performed in the NSSS area to support the revised conditions provided previously in Table IV-1. The systems addressed in this section include fluid systems and control systems. The results and conclusions of each evaluation and analysis are presented within each subsection.

VI.1 NSSS INTERFACE SYSTEMS

VI.1.1 Safety Injection System

The function of the SI system is to remove the stored energy and fission product decay heat from the reactor core following a LOCA. The system is designed such that fuel rod damage is minimized, facilitating the long-term removal of decay heat. The system also provides injection of negative reactivity (boron) in the RCS cooldown events such as a MSLB.

The active part of the SI system consists of high pressure SI pumps, the refueling water tank, low pressure safety injection (LPSI) pumps, and the associated valves, instrumentation, and piping.

The passive portion of the SI system is the safety injection tanks (SITs) that are connected to each of the RCS cold leg pipes. Each SIT contains borated water under nitrogen pressure and automatically injects into the RCS when the RCS pressure drops below the operating pressure of the SITs. The active portion of the SI system (injection pumps) injects borated water from the refueling water tank into the reactor following a break in either the RCS or steam system piping to cool the core and prevent an uncontrolled return to criticality.

Safety Injection system operation is described in two phases; the injection phase and the recirculation phase. The injection phase provides emergency core cooling and additional negative reactivity immediately following a spectrum of accidents including a LOCA by prompt delivery of borated water to the RV. The recirculation phase provides long-term post-accident cooling by recirculating water from the containment sump.

During normal operation the SI system does not operate and has no design function. Thus, during normal operation, there is no impact on the system due to the MUR power uprate. However, the slight increase in RCS stored energy and decay heat resulting from the power uprate are well within the capabilities of the SI system to respond to DBAs. The results of the evaluation of a LOCA are presented in Section II.3. For non-LOCA RCS depressurization events, the SI system is acceptable for the proposed power uprate as demonstrated in Section II.2.

VI.1.2 Chemical and Volume Control System

The CVCS provides for boric acid addition and removal, chemical additions for corrosion control, reactor coolant cleanup and degasification, reactor coolant makeup, and processing of reactor coolant letdown.

During plant operation, reactor coolant letdown is taken from the cold leg on the suction side of the RCP, and is reduced in pressure and temperature prior to it entering the volume control tank. The charging pumps take suction from the volume control tank and return the coolant through the regenerative heat exchanger to the RCS in the cold leg, downstream of the RCP.

The nominal T_{cold} for the power uprate remains unchanged at 548°F. As a result, the temperature of the letdown flow is not changed. Consequently, there is no impact on the thermal performance of the CVCS.

The CVCS provides a source of borated water for post-accident injection. Evaluation of required ECCS water volumes and boric acid concentrations will be performed as part of the normal reload safety evaluation process. The slight increase of N-16 activity at the MUR power uprate conditions has a negligible effect on letdown line decay time requirements. There will be no change to the letdown and makeup requirements as a result of the MUR power uprate.

As previously noted, T_{cold} and the reactor coolant mass flow rate remain unchanged. Increased power is due to a slight increase in T_{hot} and associated increase in T_{ave}. The increase in T_{ave} causes a small increase in the makeup requirements for coolant shrinkage during cooldown. However, this effect is considered negligible, so the system is capable of supporting the MUR power uprate.

VI.1.3 Shutdown Cooling System

The SDC system is designed to remove sensible and decay heat from the core and to reduce the temperature of the RCS during the second phase of plant cooldown.

The SDC system consists of two electrically aligned trains. Each train consists of one heat exchanger, one LPSI pump, associated valves, and instrumentation. Both trains take suction from a common suction line off one reactor coolant hot leg, and then flow is divided through the LPSI pumps, the tube side of the SDC heat exchangers, and back to the RCS cold legs through a four leg header.

The proposed power uprate will affect the SDC system due to an increased heat load from higher decay heat input. Since decay heat is proportional to plant operating power, any increase in RTP will result in an increase in decay heat load.

The SDC system was previously evaluated to be capable of supporting the decay heat that would be present based on 102% of 2700 MWt, which is 2754 MWt including uncertainty. The analytical power level including revised uncertainty with the MUR power uprate remains 2754 MWt. Therefore the system is capable of supporting the MUR power uprate.

VI.1.4 Auxiliary Feedwater System

The purpose of the AFW system is to provide sufficient feedwater to the SGs for the removal of sensible and decay heat, and to cool the primary system to 300°F in case the condensate or the main feedwater systems are inoperable. An evaluation was performed to determine whether the current design of the AFW system will satisfy its safety functions and support an MUR power uprate.

The AFW and condensate storage tank analyses are based on 102% of 2700 MWt (2754 MWt). The analytical power level, including revised uncertainty, with the MUR power uprate remains 2754 MWt. Therefore, the evaluation concluded that the AFW system and condensate storage tank system are capable of supporting the MUR power uprate.

VI.1.5 Main Steam System

The MSS is designed to transfer steam from the SGs to the turbine throttle stop valves, the reheaters, and the turbine-driven pumps. The MSS also controls SG pressure by means of steam bypass, dump, or safety valves (high pressure) and MSIVs (low pressure).

The system is designed to accommodate electrical load changes from 15 to 100% power at a rate of 5% per minute and at greater rates over smaller load change increments, up to a step change of 10%. This is normally accomplished by manual CEA movement and adjustment of RCS soluble boron concentration. The primary impact of the MUR power uprate on the MSS is an increase in main steam flow of about 2%.

There is no change in the MSS operating pressure and temperature. Steam flow to the high pressure turbine will increase by 2.07% for Unit 1 and 2.23% for Unit 2. The MSS and associated components were evaluated for the increased steam flow and are capable of supporting the MUR power uprate.

However, for Unit 2 there is an economic issue concerning the turbine throttle valves. The Unit 2 turbine throttle valves are currently operating at the valve-wide-open position, which is a limiting factor for taking full advantage of the full MUR power uprate. This limitation does not effect the safe operation of the plant and the necessary hardware changes to eliminate this limitation are addressed as an economic concern.

VI.1.5.1 MSSVs

Overpressure protection for the shell side of the SGs and the main steam line piping up to the inlet of the turbine stop valve is provided by 16 spring-loaded ASME Code MSSV which discharge to the atmosphere. Eight of these safety valves are mounted on each of the main steam lines upstream of the steam line isolation valves, but outside the Containment. The MSSVs are designed for full flow relief pressure of 1085 psig, thereby ensuring that the secondary system pressure is limited to within 110% of its design pressure of 1015 psia during the most severe anticipated system operational transient. The opening pressure of the valves is set in accordance with ASME Code allowances, with the minimum set pressure at 935 psig, and the maximum set pressure at 1050 psig. The total relieving capacity for all valves on both of the steam lines in either unit is 12.26×10^6 lbs/hr of saturated steam (6.13×10^6 lbs/hr per SG). This relief capacity is larger than the steam flow at the MUR power uprate conditions. The accident analysis shows there is adequate MSSV capability at 102% power.

Startup and/or power operation is allowable with MSSVs inoperable within the limitations of the Technical Specifications. The number of inoperable MSSVs determines the necessary level of reduction in secondary system steam flow and thermal power required by the reduced reactor trip settings of the Power Level-High channels. The current Technical Specifications were confirmed to remain applicable for supporting the proposed MUR power uprate.

VI.1.5.2 MSIVs

One MSIV assembly is provided on each main steam line header in order to protect the reactor and SG from damage due to a rupture in the main steam header down stream of the valves.

Closure of the MSIV, within a maximum of six seconds after a trip signal is initiated, prevents rapid flashing and blowdown of water stored in the shell side of the SG, thus avoiding a rapid uncontrolled cooldown of the RCS. Also, the isolation valves prevent release of the contents of the secondary side of both SGs to the Containment in the event of the rupture of one main steam line inside the containment structure.

The MSIVs are not impacted by the MUR power uprate because SG and the MSS operational pressure are not increased. Therefore, the ability of the MSIV to close within the Technical Specification limited closure time following a postulated SLB event is not affected. The MSIVs are therefore capable to supporting the proposed MUR power uprate.

VI.1.5.3 Steam Dump and Bypass System

The steam dump and bypass system is used to rapidly remove RCS stored energy and to limit secondary steam pressure following a turbine-reactor trip. The atmospheric steam dump system consists of two automatically actuated ADVs which exhaust to the atmosphere. The turbine bypass system consists of four turbine bypass valves which exhaust to the main condenser. The power-operated steam dump valves

and steam bypass valves obviate opening of the MSSVs following turbine and reactor trips from full power.

The system also provides a means of heat removal during hot standby, startup, and during plant cooldown. The atmospheric steam dump valves are capable of removing reactor decay heat when the condenser is not available.

The total respective capacities of the atmospheric steam dump and turbine bypass valves are nominally 5% and 40% of steam flow with the reactor at full power. This flow is sufficient to control the secondary steam pressure on a turbine trip at the MUR power uprate without necessitating operation of the spring-loaded safety valves. Therefore, the steam dump and bypass system is capable of supporting the proposed MUR power uprate.

VI.1.5.4 Main Turbine-Generator

The turbine-generator is designed to receive steam from the SGs and convert it into electric energy. The condenser transfers unusable heat to the condenser cooling water and deaerates the condensate. The closed regenerative turbine cycle heats the condensate and returns it to the SGs.

Saturated steam is supplied to the turbine throttle from the SGs through four stop valves and four governor control valves. The steam flows through a dual-flow, high-pressure turbine and then through combination moisture separator-reheaters (two in parallel for Unit 1, four in parallel for Unit 2) to three double-flow, low-pressure turbines which exhaust to the main condenser system.

Unit 1 is a General Electric turbine and Unit 2 is a Westinghouse turbine. The two units are similar in construction and type, and have similar performance characteristics and generating capacity.

Each generator has the capability to accept the gross rated output of the turbine at rated steam conditions. The generator shafts are oil-sealed to prevent hydrogen leakage. Each generator has its own shaft-driven excitation equipment.

The main turbine-generator and their associated components were evaluated for the MUR power uprate conditions. The primary impact on the main turbines is the increase in main steam flow. Steam flow to the high pressure turbines is expected to increase by 2.07% for Unit 1 and 2.23% for Unit 2. This increase in steam flow is within the design capabilities of each main turbine. The impact on the main generators was previously evaluated in Section V.2. No changes to the main turbine-generators are necessary as they are capable of supporting the MUR power uprate.

VI.1.6 Steam Generator Blowdown System

Each SG has an upper and lower blowdown line which can be used to control the build-up of soluble and particulate concentrations within the SG. The blowdown system will continue to operate normally with no change at a continuous rate of up to 180 gpm per SG. No changes to SG blowdown are required as the SG blowdown system is capable of supporting the proposed MUR power uprate.

VI.1.7 Feedwater and Condensate Systems

The feedwater and condensate systems are designed to provide a means for transferring the condensate from the condenser hotwell to the SGs (while at the same time raising the temperature and pressure) and providing a means for controlling the quantity of feedwater into the SGs.

The MUR power uprate results in approximately a 2% increase in condensate flow, due to an increase in extraction steam flow through the feedwater heaters and an increase in the condensate temperature. A hydraulic calculation was performed verifying the condensate systems ability to perform its design function of delivering condensate from the condenser hotwells to the feedwater system at the required flow, pressure, temperature, and quality at the MUR power uprate conditions.

The feedwater flow will increase approximately 2% under the MUR power uprate conditions. A hydraulic calculation was performed to evaluate the feedwater systems ability to provide sufficient flow to the SGs at the MUR power uprate conditions. Results of the calculation found the feedwater system capable of providing sufficient flow to the SGs.

However, one level of the feedwater heaters on Unit 1 and two levels of feedwater heaters on Unit 2 have been identified as having possible limitations for full MUR power uprate conditions. The feedwater heaters are not a safety critical component and may be further evaluated with the potential for replacement/modification as an economic concern.

The feedwater and condensate systems are capable of supporting the proposed MUR power uprate.

VI.1.8 Extraction Steam System/Heater Drains System

The extraction steam and heater drain systems provide a means of heating condensate and feedwater, and for returning condensed steam to the condensate system.

The MUR power uprate results in approximately a 2% increase in heater drain flow and a corresponding increase in heater drain temperature, due to the increase in heat load in the feedwater heaters. The system evaluation demonstrated that the equipment can operate at the MUR power uprate conditions, with further action needed to upgrade or evaluate the capability of the Unit 2 heater drain pumps and Unit 2 heater drain tank high level dump valves. These actions are treated as economic issues as they are not safety significant.

The MUR power uprate results in an increase in extraction steam flows and pressures. The temperature and pressure ratings for the Units 1 and 2 bleeder trip valves bound the MUR power uprate service conditions based on the maximum working pressures contained in American National Standards Institute (ANSI) B16.34. The design temperature and pressure ratings for Units 1 and 2 extraction steam drain trip AOVs and MOVs bound the MUR power uprate service conditions based on the maximum working pressure contained in ANSI B16.34 and the pressure rating listed on the MOV drawings.

The extraction steam and heater drain systems are capable of supporting the proposed MUR power uprate.

VI.1.9 Circulating Water System

The condensers on both units have an operational limit of a 12°F temperature rise in the circulating water across the condensers. The MUR power uprate is expected to result in a small increase in the temperature rise across the condenser. Currently, the actual measured temperature rise is approximately 11.6°F which is expected to rise to approximately 11.8°F after the MUR power uprate. Condenser vacuum in-balance is not adversely affected by the MUR power uprate. The circulating water system is capable of supporting the MUR power uprate.

VI.1.10 Condenser

Each unit has one three-shell, single-pass, deaerating-type condenser with divided water boxes. The condenser is capable of condensing the exhaust steam from the main turbine and the SG feed pump turbines under the MUR power uprate plant load. Two of the three condenser shells are connected to discharge lines from the steam dump and bypass system. The condenser is internally equipped to receive the full flow from this system. The condenser is adequately sized and is capable of supporting the proposed MUR power uprate. The condenser will operate with approximately 1.9% higher backpressure under the MUR power uprate during summer conditions (< 3.8 in-HgA). This is still well below the backpressure limit (5.5 in-HgA).

VI.1.11 Heat Balance

A plant specific model was developed for each unit for both summer and winter bay temperature limits at both the current and the MUR power uprate conditions. This detailed model was benchmarked to existing plant operating conditions and used to simulate the estimated impact of the proposed uprate. Output from these simulations (pressure, flows, temperatures) were used as an aid in evaluating the impact the MUR power uprate will have on the plant equipment.

VI.2 CONTAINMENT SYSTEMS

VI.2.1 Reactor Coolant System

The purpose of the RCS is to remove heat from the core and transfer it to the secondary side of the SGs. The RCS consists of the reactor pressure vessel, two hot leg pipes, two SGs, four RCPs, four cold leg pipes, and one pressurizer with attendant interfacing piping, valves and instrumentation.

Evaluations were performed to ensure that the RCS design basis functions could still be met at the revised operating conditions. The principal effects of the MUR power uprate on the RCS are a slight increase in T_{hot} and the increase in decay heat. The normal operating pressure of 2250 psia remains unchanged. The results of the evaluation of uprated conditions on the RCS functions are described below.

a. The increase in T_{hot} will increase the total amount of heat transferred to the MSS. Verification that the major components of the NSSS can support this increase in the normal heat removal function is addressed in this section.

b. The increased thermal power can change the transient response of the RCS to normal and postulated DBEs. The acceptability of the RCS with respect to protection functions is addressed in Section II. The acceptability of the RCS with respect to fatigue evaluations is addressed in Section IV. The setpoints for various control systems will be evaluated for recommended changes prior to plant startup.

c. The cold leg temperature remains unchanged at 548°F. As a result, the RCS mass flow is not affected by the MUR power uprate.

d. Reactor coolant system design temperature and pressure of 650°F and 2500 psia continue to remain applicable for the uprate conditions.

e. The pressurizer design temperature and pressure of 700°F and 2500 psia continue to remain applicable for the uprate conditions.

f. The pressurizer relief requirements increased slightly due to an increase in RCS stored energy and decay heat. However, the change is well within the relieving capacity of the pressurizer safety valves for the design transient condition (Section II.2).

The RCS is capable to supporting the proposed MUR power uprate.

VI.3 SAFETY-RELATED COOLING WATER SYSTEMS

VI.3.1 Service Water System

The Service Water (SRW) system is designed to remove heat from the plant's various auxiliary systems. The Saltwater (SW) system provides the cooling medium for the SRW heat exchangers. System components are rated for maximum duty requirements during normal operation and SDC operation, and are also capable of providing heat removal during a LOCA. The SRW system serves as an intermediate barrier between the various auxiliary systems and the SW system.

The turbine plant components cooled by SRW include:

a. Generator isolated 3 phase bus duct coolers
b. Exciter air coolers
c. Generator hydrogen coolers
d. Stator liquid coolers (Unit 1 only)
e. Circulating Water System priming pump seal water coolers
f. Condenser vacuum pump seal water coolers
g. Feed pump turbine lube oil coolers
h. Condensate booster pump lube oil and seal water coolers
i. Instrument and plant air compressors and aftercoolers
j. Turbine lube oil cooler
k. Electro-hydraulic oil coolers
l. Turbine Building sample cooling system
m. Seal oil system coolers (Unit 2 only)
n. Auxiliary feedwater pump room air cooler

The SRW system does not see significant impact with the MUR power uprate. The increased decay heat and turbine auxiliary cooling loads will cause a small increase in the cooling water temperature. The heat loads increase slightly for the Spent Fuel Pool Cooling (SFPC) in the Auxiliary Building; however, this increase is due to the SDC function, not the MUR power uprate. The impact on the SRW system with the MUR power uprate on the component heat loads has been reviewed. Some system flow adjustments may be necessary to ensure proper cooling to the affected heat loads, the SRW system has adequate margin to perform its design functions within its design parameters. As such, the SRW system is capable to supporting the proposed MUR power uprate.

VI.3.2 Saltwater System

The SW system has three pumps for each unit. The pumps provide the driving head to move SW from the intake structure, through the system, and back to the circulating water discharge conduits. The system is designed such that each pump has sufficient head and capacity to provide cooling water for the SRW and Component Cooling Water (CCW) systems. The system also cools the ECCS pump room air coolers.

The SW system consists of two subsystems in each unit. Each subsystem provides SW to two SRW heat exchangers, a CCW heat exchanger, and the ECCS pump room air cooler in order to transfer heat from those systems to the Chesapeake Bay. Seal water for the circulating water pumps is supplied by both subsystems. A self-cleaning strainer is installed upstream of each SRW heat exchanger.

Operation of the SW system following a LOCA has two phases: before the recirculation actuation signal and after the recirculation actuation signal. Since the LOCA analysis has been performed at 102% of

2700 MWt (2754 MWt) it remains applicable at the MUR power uprate. Therefore, the cooling requirements for both phases are unchanged.

The MUR power uprate results in small increases to the heat loads for the CCW and SRW heat exchangers to be transferred to the SW system, corresponding to a slight increase of the SW discharge piping temperature. These impacts are negligible on the SW system and component operation. The margins in the system remain essentially the same as for current operating power levels.

VI.3.3 Component Cooling Water System

The CCW system is designed to remove heat from the plant's various auxiliary systems. The SW system provides the cooling medium for the CCW heat exchangers. System components are rated for maximum duty requirements during normal and SDC, and are also capable of providing heat removal during a LOCA. The CCW system serves as an intermediate barrier between the various auxiliary systems and the SW system.

The CCW heat exchangers are designed for a CCW supply temperature of 95°F, with a SW cooling supply temperature of 90°F, at normal operating conditions. Component cooling water may reach as high as 120°F during a LOCA, and during plant cooldown and cold shutdowns.

The MUR power uprate results in a change to the CCW system heat loads. The change has a negligible impact on the CCW system. The increased decay heat has a small impact on the cooling water temperature increase. Calvert Cliffs has evaluated the most limiting mode of CCW operation at the analytical power level of 2754 MWt, therefore, the CCW system is capable of supporting the proposed MUR power uprate.

VI.4 SPENT FUEL POOL COOLING SYSTEM

The SFPC system is common to both units. The pool contains water with the proper dissolved concentration of boron and has the capacity to store 1830 fuel assemblies.

The SFPC system is designed to remove the maximum decay heat expected from 1613 fuel assemblies, not including a full core off-load. The maximum pool temperature in this case is 120°F. The system is also capable of being used in conjunction with the SDC system to remove the maximum expected decay heat load from 1830 fuel assemblies, including a full core discharge. The maximum spent fuel pool temperature in this case is 130°F.

The decay heat source-term used in the evaluation of the SFPC system was determined to be conservative for the proposed MUR power uprate conditions. Therefore, the SFPC system is capable of supporting the proposed MUR power uprate.

VI.5 RADIOACTIVE WASTE SYSTEMS

The waste processing systems are designed to provide controlled handling and disposal of radioactive liquid, gaseous, and solid wastes from both units. Design criteria were established to maintain the release of radioactive material from the plant to the environment at levels which are ALARA.

The design of the waste processing systems was based upon processing reactor coolant and miscellaneous waste during operation with 1% failed fuel. The proposed MUR power update is for a small power increase and since the radioactive waste processing system is designed to handle 1% failed fuel, the MUR

power uprate does not represent a significant challenge to the liquid or gaseous radwaste processing system.

All releases meet the Offsite Dose Calculation Manual (ODCM) limits. By meeting the ODCM limits, the guidelines of 10 CFR Part 50, Appendix I are met. This is confirmed by the effluent data and doses reported to the NRC in the Radioactive Effluent Release Reports required by the Technical Specifications and 10 CFR 50.36a.

Therefore, the proposed MUR power uprate has no impact on the radioactive waste system releases.

VI.6 ENGINEERED SAFETY FEATURES HEATING, VENTILATION, AND AIR CONDITIONING SYSTEMS

The plant ventilating systems are designed to provide a suitable environment for equipment and personnel with a maximum amount of safety and operating convenience. Potentially contaminated areas are separated from clean areas. Airflow patterns originate in areas of potentially low contamination and progress toward areas of higher activity. Generally, negative pressures are maintained in potentially contaminated areas and positive pressures in clean areas. The ventilating systems in the Containment, waste processing, and fuel-handling areas are designed for containment of radioactive particles. The path of the discharge from potentially contaminated areas is directed into the respective plant vent where the radioactivity level is monitored. The equipment in most critical systems is redundant.

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 feedwater piping in the Containment, Auxiliary Building (steam tunnel), and Turbine Building were evaluated to account for a < 2°F increase in feedwater process fluid temperature to ensure UFSAR Chapter 9 design basis are not impacted. The remaining BOP piping temperatures do not change appreciably.

VI.6.1 Containment

The Containment is cooled by the containment air coolers. During the summer the air temperature is expected to remain below the 120°F design limit. The total heat load in the Containment during normal operation is calculated to be ~7.44 x 10^6 Btu/hr. The increase of <2.0°F in feedwater temperature could potentially increase the heat load on the cooling system by ~400 Btu/hr, clearly inconsequential given the order of magnitude difference considering the original heat load in the building. This assessment is applicable and valid for both units.

VI.6.2 Main Steam Penetration Rooms

Heat load from the main steam and feedwater piping traversing through these rooms was evaluated previously. The inputs and assumptions used in this calculation are very conservative and the small increase in anticipated feedwater process fluid operating temperature (<2°F) will not have any effect on the calculated or actual overall room temperature. The calculated heat load in the room is already based on a feedwater design temperature of 460°F, in lieu of lower operating feedwater temperature. This assessment is applicable and valid for both units.

VI.6.3 Turbine Building

Heat load from the main steam and feedwater piping in the Turbine Building was evaluated to account for a < 2°F increase in feedwater process fluid temperature. This evaluation indicates that general Turbine Building area air temperatures may increase by less than a fraction of (0.05°F) a degree. This is reasonable since the minimal increase in the feedwater temperature as compared to all of the other large

heat loads in the Turbine Building has minimal effect on the Turbine Building air temperature rise. This assessment is applicable and valid for both units.

VI.6.4 Auxiliary Feedwater Pump Room and 5' Fan Room

Heat load from process piping traversing through these two rooms was previously established. The less than 2°F increase in feedwater fluid temperature has no effect on the results of the calculation. A feedwater design temperature of 460°F was used in the analysis; therefore the calculation predicted room temperatures already bound the room conditions expected as a result of the power uprate. This assessment is applicable and valid for both units.

VI.6.5 Auxiliary Building

There is a minimal amount of piping traversing through the Auxiliary Building to and from the main steam penetration room and the 5' fan room. There is no specific calculation evaluating the heat input from the feedwater piping into this area, given the short run of piping and the minimal increase of feedwater temperature, the effect of air temperature increase in that area is expected to be negligible. This assessment is applicable and valid for both units.

VI.6.6 Control Room Heating, Ventilation, and Air Conditioning System

The Control Room (Elevation 45'0") and the Cable Spreading Room (Elevation 27'0") are incorporated into a single year-round air-conditioning system serving the common Control Room for Units 1 and 2. Therefore, the ambient temperature in the Control Room is expected to be the same as the ambient temperature in the Cable Spreading Room. Air handling and refrigeration equipment are redundant. The Control Room and Cable Spreading Room areas have a third source of cooling, which is not safety-related, in the form of a water chiller supplying a second set of coils in the safety-related air handling systems.

VI.6.7 Auxiliary Building Ventilation System (Auxiliary Building Charcoal Filters)

Key parameters for the Auxiliary Building Ventilation System charcoal filters are total flow rates, and total charcoal weights. The charcoal is Barnebey-Cheney #727 (or equivalent) impregnated with 5 weight% iodine compounds. The flow velocity through the charcoal bed is 40 fpm in all cases and the corresponding residence time is 0.25 seconds. Testing is performed to demonstrate that the installed charcoal absorbers will perform satisfactorily in removing both elemental and organic iodides for design conditions of flow, temperature, and relative humidity. Periodic testing is conducted to ensure filter efficiencies credited in the accident analysis are maintained. These key parameters remain unaffected by the MUR power uprate and, as such, the MUR power uprate has negligible impact on the Auxiliary Building charcoal filters.

Based on the above discussions, none of the design, operational or performance requirements of the various area heat removal systems are significantly affected by the slight increase in feedwater fluid temperature. As such the various HVAC systems are capable to supporting the proposed MUR power uprate.

VII. OTHER

VII.1 OPERATOR ACTIONS

Operator actions that are part of the Abnormal Operating Procedures (AOP) and Emergency Operating Procedures (EOP) have been reviewed, and it was concluded that the proposed MUR power uprate does not adversely impact the available time for operator actions. The small change in decay heat as a result of the proposed MUR power uprate has a negligible impact on operator response times.

VII.2 PROCEDURES, CONTROL ROOM, SIMULATOR, AND TRAINING

VII.2.1 Emergency and Abnormal Operating Procedures

The EOP and AOP procedures have also been reviewed to assess if there are any impacts to these procedures as a result of the proposed MUR power uprate. The proposed MUR power uprate is being implemented under the administrative controls of Calvert Cliffs design change process. The design change process ensures any impacted procedures are revised prior to the implementation of the power uprate.

VII.2.2 Control Room Controls, Displays, and Alarms

Section I.3 describes the physical modifications required to support the implementation of the Caldon LEFM CheckPlus feedwater measurement system. While there are no controls for the LEFM CheckPlus feedwater measurement system located in the Control Room, Control Room Operators have the ability to select the LEFM CheckPlus system output as the source of input data into the Plant Computer calculation of calorimetric calculation via a control room display interface. Additionally, the results of the calorimetric calculation are displayed on the Plant Computer to Control Room Operators. There are no hardwired alarms from the local LEFM CheckPlus System cabinet to the Control Room but system alarms trigger an alarm in the control room annunciation system.

Any additional plant hardware modifications potentially required to support the proposed MUR power uprate have been identified. 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.

VII.2.3 Control Room Plant Reference Simulator

A review of the plant simulator will be conducted, and necessary changes made, prior to implementing the proposed MUR power uprate. 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 are identified during the design change review process.

VII.2.4 Operator Training Program

Prior to actual implementation of the proposed MUR power uprate, training will be conducted to instruct the operations staff on the impact of the uprate on plant operations (e.g., revised scaling for instrumentation, required actions for Caldon LEFM CheckPlus OOS).

VII.3 INTENT TO COMPLETE MODIFICATIONS

All modifications that are required to support the MUR power uprate will be completed prior to Calvert Cliffs implementing the higher reference thermal power level of 2737 MWt. In addition, Calvert Cliffs

will ensure all required Operator training in support of this proposed power uprate is completed prior to implementing the higher reference thermal power level.

VII.4 TEMPORARY OPERATION ABOVE LICENSED POWER LEVEL

Currently Calvert Cliffs uses a rolling eight-hour average of secondary calorimetric power in the surveillance of maximum core power under full-power, steady-state conditions. Currently, the maximum deviation of the indicated power does not exceed 2754 MWt (102% of 2700 MWt). After the proposed MUR power uprate is implemented, the maximum deviation remains at an upper limit of 2754 MWt (100.6% of 2737 MWt). Additional restrictions on secondary calorimetric power may be implemented in accordance with regulatory guidance separate from this project.

VII.5 ENVIRONMENTAL PROTECTION

The Environmental Report, the Final Environmental Statement, and supplements to the Environmental Report were reviewed. The only non-radiological discharge parameter that will be affected by the MUR power uprate is the delta-T across the condenser. The maximum predicted increase in the delta-T across the condensers after the MUR power uprate is described in Section VI.1.9. It is within the 12°F (max) limit in our discharge permit.

The Calvert Cliffs discharge permit contains the following requirement:

> *"All discharges authorized herein shall be consistent with the terms and conditions of this permit. The discharge of any pollutant identified in this permit at a level in excess of that authorized shall constitute a violation of the terms and condition of this permit. Anticipated facility expansions, production increases or decreases, or process modifications, which will result in new, different, or an increased discharge of pollutants, shall be reported by the permittee by submission of a new application or, if such changes will not violate the effluent limitations specified in this permit, by notice to the Department. Following such notice, the permit may be modified by the Department to specify and limit any pollutants not previously limited. "*

The MUR power uprate does constitute a production increase that results in a slight increase in the discharge of pollutants, thus Calvert Cliffs will send a letter to Maryland Department of the Environment describing the change that is made and the impact on the effluents.

The delta-T across the condenser is monitored, consistent with our normal practice, during implementation of the MUR power uprate to verify accuracy of the predicted temperature increase.

VII.6 10 CFR 51.22 DISCUSSION

Title 10 CFR 51.22(c)(9) provides criteria for, and identification of, licensing and regulatory actions eligible for categorical exclusion from performing an environmental assessment. A proposed amendment to an operating license for a facility requires no environmental assessment provided:

a) The amendment involves no significant hazards consideration – This proposed amendment does not involve a significant hazards consideration as previously evaluated in Section 4.3 of Attachment 1.

b) There is no significant change in the types or significant increase in the amounts of any effluents that may be released offsite - The proposed change does not significantly impact installed equipment performance, require significant changes in system operation or significantly increase the release of solid, liquid or gaseous effluents. The specific activity of the primary and secondary coolant is expected to increase by no more than the percentage increase in power level. Therefore,

the gaseous and liquid effluent releases are expected to increase from current values by no more than the percentage of increase in power level. Offsite release concentrations and doses continue to be maintained within the limits of 10 CFR Part 20 and 10 CFR Part 50, Appendix I in accordance with the requirements of the Calvert Cliffs ODCM. The proposed change will not result in changes in the operation or design of the gaseous, liquid or solid waste systems, and will not create any new or different radiological release pathways.

c) There is no significant increase in individual or cumulative occupational radiation exposure - The proposed change does not cause radiological exposure in excess of the dose criteria for restricted and unrestricted access specified in 10 CFR Part 20. General radiation levels in the plant are expected to increase by no more than the percentage increase in power level. Individual worker exposures will continue to be monitored and be maintained ALARA in accordance with Calvert Cliffs Radiation Protection Program.

In summary, the proposed MUR uprate meets the criteria for categorical exclusion from environmental review as identified in 10 CFR 51.22(c)(9) in that the amendment request involves no significant hazards consideration (see Attachment 1), involves no significant change in the types or significant increase in the amount of any effluents that may be released offsite, and involves no significant increase in individual or cumulative occupational radiation exposure.

VIII. CHANGES TO TECHNICAL SPECIFICATIONS, PROTECTION SYSTEM SETTINGS, AND EMERGENCY SYSTEM SETTINGS

INTRODUCTION

This section addresses the impact of the proposed change in RTP on Technical Specifications, Protection System Settings, and Emergency System Settings.

VIII.1 TECHNICAL SPECIFICATIONS

Other than the proposed change to the RTP, there are no other changes required to support the increase in RTP.

VIII.2 REACTOR PROTECTIVE SYSTEM

The RPS at Calvert Cliffs Units 1 and 2 includes three trip functions whose settings could be impacted by the increase in the RTP. The three trip functions, as listed in the Technical Specifications Table 3.3.1-1 are:

- Power Level – High,
- Axial Power Distribution – High, and
- TM/LP.

The setpoints/allowable values for the Power Level – High trip are specified in Technical Specifications Table 3.3.1-1. The setpoints/allowable values for the Axial Power Distribution – High trip and the TM/LP trip are specified in the COLR.

The setpoints/allowable values and coefficients for these three trip functions are calculated and/or verified every cycle using the methodology described in References VIII-1, VIII-2, and VIII-3. No changes are required to the methodology as a result of the increase in the RTP. Therefore, the cycle specific calculation and/or verification of the setpoints/allowable values and coefficients for these trip functions appropriately reflect the increase in the RTP. No changes to the Variable High Power Trip setpoints/allowable values in Technical Specification Table 3.3.1-1 or to the Axial Power Distribution or TM/LP trip settings/allowable values in the COLR have been identified due to the increase in the RTP.

VIII.3 LIMITING CONDITIONS FOR OPERATION

Four of the LCOs in Technical Specification Section 3.2 could be impacted by the increase in the RTP. These Technical Specification LCOs are:

- 3.2.1, Linear Heat Rate (LHR),
- 3.2.2, Total Planar Radial Peaking Factor (F_{xy}^{T}),
- 3.2.3, Total Integrated Radial Peaking Factor (F_{r}^{T}), and
- 3.2.5, Axial Shape Index (ASI).

The limits for these LCOs are specified in the COLR and are calculated and/or verified every cycle using the methodology described in References VIII-1, VIII-3 and VIII-4. No changes are required to the methodology as a result of the increase in the RTP. Therefore, the cycle specific calculation and/or verification of the limits for these LCOs appropriately reflect the increase in the RTP and the COLR is modified as necessary. In addition, coefficients for the Better Axial Shape Selection System, which is used to establish the limits for the Axial Shape Index LCO, are updated as necessary each cycle. The cycle specific updates reflect the increase in the RTP.

VIII.4 EMERGENCY SAFETY FEATURES ACTUATION SYSTEM AND AUXILIARY FEEDWATER ACTUATION SYSTEM

The existing Emergency Safety Features Actuation System and AFAS setpoints and response times were used in the justification of the continued applicability of the safety analysis (see Section II). No changes were required or necessary to support the proposed change in RTP.

VIII.5 REFERENCES

VIII-1 CENPD-199-P, Revision 1-P-A, "C-E Local Power Density and DNB LSSS and LCO Setpoint Methodology for Analog Protection Systems," January 1986

VIII-2 CEN-124(B)-P, Statistical Combination of Uncertainties, Part 1," December 1979

VIII-3 CEN-348(B)-P-A Supplement 1-P-A, "Extended Statistical Combination of Uncertainties," January 1997

VIII-4 CEN-124(B)-P, Statistical Combination of Uncertainties Methodology Part 3, December 1979

ENCLOSURE (1)

CA06945, Revision 0000, Calorimetric Uncertainty Using the Caldon LEFM CheckPlus Flowmeter

Calvert Cliffs Nuclear Power Plant , Inc.
August 29, 2008

CALORIMETRIC UNCERTAINTY USING THE LEFM CHECKPLUS FLOW MEASUREMENT SYSTEM

For Calvert Cliffs Nuclear Power Plant
Units 1 & 2

Calculation No. CCN-IC-08001 Revision 0

Prepared By Hurst Technologies, Corp.

Project: CCNAKC

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ENCLOSURE (1)

TABLE OF CONTENTS

1.0 PURPOSE

The purpose of this evaluation is to estimate the uncertainty in the secondary calorimetric, as computed on the plant computer, using the Caldon LEFM CheckPlus ultrasonic flow measurement system to measure feedwater flow. Uncertainty is evaluated at the proposed Appendix K uprated power of 2737 MW(th). The Appendix K power represents an increase of approximately 1.4% from the current licensed power limit of 2700 MW(th).

2.0 COMPONENT LISTING

2.1. Calorimetric power is calculated using the following instrumentation:

2.1.1. Feedwater Flow

2.1.1.1. Feedwater flow is measured using the Caldon LEFM CheckPlus ultrasonic flow measurement system. Principle components of the LEFM CheckPlus system consist of:

a. Metering Section – The metering section is a spool-piece installed in the feedwater header to each steam generator. The metering section consists of 8 transducer pairs.
Equipment IDs
To be determined

b. Electronic Unit – The electronic unit (one per plant) sequences the operation of the transducers and calculates volumetric flow, temperature, and mass flow. The digital output of the electronic unit provides input to the plant computer via the plant data network.
Equipment IDs (Based on AMAG Cabinet Numbers – To Be Verified Later)
1CPU1C209 - Unit 1 Caldon LEFM CheckPlus Electronic Unit
2CPU2C209 - Unit 2 Caldon LEFM CheckPlus Electronic Unit

2.1.2. Feedwater Pressure

2.1.2.1. Feedwater pressure is measured from pressure transmitters from taps installed in the metering section of the flow measurement system.
Equipment IDs
To be determined

2.1.2.2. The feedwater pressure transmitters provide input to the flow measurement system electronic unit, which transmits the information to the plant computer via the plant data network.

2.1.2.3. Feedwater pressure is used for:

a. Calculation of feedwater mass flow.

b. Calculation of feedwater enthalpy.

2.1.3. Feedwater Temperature

2.1.3.1. Feedwater temperature is measured from RTDs installed in the feedwater header to each steam generator.
Equipment IDs
1TE4516 - 11 SG Feedwater Inlet Temperature
1TE4517 - 12 SG Feedwater Inlet Temperature
2TE4516 - 21 SG Feedwater Inlet Temperature
2TE4517 - 22 SG Feedwater Inlet Temperature

2.1.3.2. The feedwater temperature RTDs provide input to the plant computer via the DAS cabinets.

2.1.3.3. Feedwater temperature is used for the calculation of feedwater enthalpy. The feedwater RTDs are independent of the temperature measurement used to calculate feedwater flow.

2.1.4. Main Steam Pressure

2.1.4.1. Main steam pressure is measured from pressure transmitters installed in the steam headers downstream of each steam generator.
Equipment IDs
1PT3991 - 11 Main Steam Header Pressure
1PT4008 - 12 Main Steam Header Pressure
2PT3991 - 21 Main Steam Header Pressure
2PT4008 - 22 Main Steam Header Pressure

2.1.4.2. Main Steam Pressure is used for the calculation of steam enthalpy.

2.1.5. Steam Generator Blowdown Flow

2.1.5.1. Steam generator blowdown flow is determined from indicated total blowdown tank flow.
Equipment IDs
1FT4089 - Unit 1 BD Tank Effluent Flow
2FT4089 - Unit 2 BD Tank Effluent Flow

2.1.5.2. Blowdown flow for each steam generator is not measured directly, but is manually input to the plant computer in accordance with Reference 6.7. The blowdown flow input represents the total blowdown flow from each steam generator.

2.1.6. Calorimetric Constants

2.1.6.1. Calorimetric constants are assigned to calorimetric inputs not directly measured by plant instruments such as

- Heat addition to the reactor coolant system (RCS) from the pressurizer heaters and reactor coolant pumps.
- Net heat loss from the RCS to letdown flow.
- Net heat loss from the RCS through insulation.
- Steam generator exit steam quality.

3.0 METHOD OF ANALYSIS

3.1. This calculation uses the methodology established in ES-028, "Instrument Loop Uncertainty and Setpoint Methodology" (Reference 6.1).

3.2. Sign Convention:

3.2.1. Uncertainties are applied to actual values, as opposed to calculated or indicated values.

3.2.2. Uncertainty is positive when the indicated or calculated value is greater than the actual value. Uncertainty is negative when the indicated or calculated value is less than the actual value.

3.2.3. If calculated power is less than actual power and the plant is being operated with indicated power near the rated thermal power limit, the Appendix K thermal power limit may be exceeded. Therefore, only the negative component of calorimetric uncertainty needs to be evaluated.

3.2.4. Since only negative calorimetric uncertainty is considered, the methodology established in Reference 6.11, Section 8.1, "Correction for Setpoints with a Single Side of Interest" may be applied.

3.3. The secondary calorimetric computation is performed, using various input process parameters, such as Feedwater Flow, Steam Pressure, Blowdown Flow, etc., which are either measured or estimated. The uncertainty in the measurement or estimate of each of these process parameters is analyzed to determine its individual impact on the secondary calorimetric computation. These impacts are then combined to determine an overall uncertainty in the secondary calorimetric computation.

3.3.1. The impact of the uncertainty for a process parameter is analyzed for a given nominal condition (actual process value). The potential range of nominal conditions is determined from analysis of historical plant readings, as listed and summarized in Reference 6.13. The limiting conditions are based on pooled data from Unit 1 and Unit 2, which are different because of differences in secondary plant design. However, the data is used to provide conservative overall limits to apply to both units. Reference 6.13 extrapolates the measured upper and lower range values for the current 100% power limits to the new, power uprate conditions, based on observations of the change in the parameter with increases in power. The final output of Reference 6.13 is a set of process limits, which are considered "indicated values."

3.3.2. The sensitivity of the secondary calorimetric computation is assessed to determine whether an upper or lower process value is more conservative for use in the secondary calorimetric uncertainty analysis. In some cases, this is difficult to assess without specifically computing the effects, since a given parameter may impact several aspects of the secondary calorimetric computation. (For instance, the process value for feedwater temperature affects the impact of the feedwater flow, feedwater pressure and feedwater temperature uncertainties, which are all used in the secondary calorimetric computation.) In these cases, the overall effects are numerically assessed to determine the most conservative limit for use. The most conservative limit is used in the final uncertainty analysis.

3.3.3. For each process parameter, a directional error of an input process parameter reading will produce a corresponding directional error in the secondary calorimetric computation. For the secondary calorimetric uncertainty analysis, only those uncertainties producing a lower-than-actual power computation will be considered, since a lower-than-actual power computation would cause operation at a higher reactor power level. Therefore, impact of each uncertainty in the input process parameters will be considered to determine the appropriate direction for uncertainty consideration.

3.3.4. The impact of instrument uncertainty for a given process parameter is determined by a few simple steps. First, the contribution of the input parameter to the secondary calorimetric computation is determined from the nominal condition (actual process value). Secondly, the contribution is re-computed, with error applied to the actual process value (indicated value). The difference in these two computations is the impact of the uncertainty of the individual process parameter on the secondary calorimetric computation.

3.3.5. Finally, the uncertainty impacts from each of the input process parameters are combined to determine an overall uncertainty in the secondary calorimetric computation.

3.4. The calculated uncertainty is a bounding uncertainty, applicable to both units at the proposed Appendix K uprated power of 2737 MW(th). The bounding uncertainty is based upon conservative values for feedwater flow uncertainty. The final margin value is expressed as a percent of the new uprated power level of 2737 MW(th).

3.5. The ASME 1967 steam tables are the current basis for the plant computer determination of thermodynamic properties.

3.6. The computations performed were calculated to several significant digits. Hand verification utilizing the rounded values may result in slightly different results due to round off errors.

3.7. No computer codes were utilized in the performance of this calculation.

3.8. Unless otherwise noted, U will be used to designate random uncertainties, and B will be used to designate bias uncertainties.

3.9. Subscripts:

(M) – Maximum Value of Input Parameter
(M1) – Additional Maximum Value of Input Parameter for Analysis Only (Not an Actual Limit; Only Used for Feedwater Temperature in this Analysis)
(m) – Minimum Value of Input Parameter
(SS) – Single Side of Interest
ACT – Actual Value
– Calorimetric Power with No Uncertainties Applied
– Uncertainty Using Actual Value for Parameters
– Actual Value of Parameter
BD – Blowdown
BDT – Total Blowdown
CAL – Calorimetric
CALC – Calculated
FS – Saturated Liquid
FW – Feedwater
GS – Saturated Vapor
h_{FG} – Saturated Vapor Enthalpy – Saturated Liquid Enthalpy
h_{FS} – Saturated Liquid Enthalpy
h_{FW} – Feedwater Enthalpy
h_{GS} – Saturated Vapor Enthalpy
IND – Indicated Values
INPUT – Evaluation of Calorimetric Power or Uncertainty with Uncertainty Applied to a Selected Input
MBD – Blowdown Mass Flow Rate
MBDT – Total Blowdown Mass Flow Rate
MFW – Feedwater Mass Flow Rate
NET – Net Calorimetric Uncertainty
Net Contribution to Uncertainty from each input
OTHER – Contribution to Calorimetric Power from Other Inputs to the Secondary Calorimetric Calculation, not based on Measured Plant Parameters
PC – Plant Computer
PFW – Feedwater Pressure
PSTM – Steam Pressure
SG – Steam Generator
SG1 – First Header Steam Generator
SG2 – Second Header Steam Generator
STM – Steam
TFW – Feedwater Temperature

4.0 DESIGN INPUTS

4.1. EQUATIONS FOR CALORIMETRIC POWER

4.1.1. Per Reference 6.2, the gross thermal output of one steam generator is computed from the expression

$$Q_{SG} = (M_{FW} - M_{BD})[(1-X)h_{FS} + Xh_{GS} - h_{FW}] + M_{BD}(h_{FS} - h_{FW})$$

where,

M_{FW} is feedwater flow,
M_{BD} is blowdown flow,
h_{FS} is the fluid component of steam enthalpy,
h_{GS} is the vapor component of steam enthalpy,
h_{FW} is the feedwater enthalpy (compressed liquid), and
X is the steam quality.

4.1.2. For a steam quality of 1 (no moisture carryover), the above expression simplifies to

$$Q_{SG} = (M_{FW})(h_{GS} - h_{FW}) + (M_{BD})(h_{FS} - h_{GS})$$

$$Q_{SG} = (M_{FW})(h_{GS} - h_{FW}) - (M_{BD})(h_{FG})$$

4.1.3. Calorimetric Constant

4.1.3.1. To determine reactor power from steam generator thermal output, adjustments are made to account for heat transfer to/from the Reactor Coolant System from sources other than the reactor and sinks other than the steam generators, such as the heat added by the pressurizer heaters, and heat losses through pipe insulation.

4.1.3.2. These corrections are input as constants to the calorimetric calculation.

4.1.3.3. For convenience, this calculation represents the net adjustment as a single constant, Q_{OTHER}.

4.1.4. Calorimetric power is the sum of the gross thermal output of both steam generators and the calorimetric constant.

$$Q_{CAL} = Q_{SG1} + Q_{SG2} + Q_{OTHER}$$

4.2. INPUT UNCERTAINTIES:

4.2.1. Input uncertainties for the measured parameters, with exception of feedwater pressure (See Assumptions) are summarized in the table below:

Parameter	Random, U	Positive Bias, B
Feedwater Temperature, T_{FW}, deg. F	+/- 1.88	
Steam Pressure, P_{STM}, PSI	+/- 19.80	+ 3.40
Total Blowdown Flow, M_{BDT}, klbm/hr	+/- 7.9	

4.2.2. References:

4.2.2.1. Main Steam Pressure - Reference 6.3.

4.2.2.2. Feedwater Temperature - Reference 6.5

4.2.3. Blowdown Flow

4.2.3.1. Total blowdown flow uncertainty is evaluated in Reference 6.4 as a function of indicated blowdown flow and power.

4.2.3.2. From a review of Reference 6.4, blowdown flow at 50 gpm indicated flow bounds the uncertainty at higher blowdown flow rates and will be used for this evaluation.

4.2.3.3. Since the sign convention for uncertainties in this reference is not clear, the largest magnitude of uncertainty (positive or negative) for 50 gpm total indicated flow will be used and applied in both the positive and negative directions.

4.2.3.4. The uncertainty of Reference 6.4 is expressed in units of klbm/hr and rounded up to the nearest 0.1 klbm/hr for simplicity and conservatism.

4.2.4. Plant Computer Calculation of Enthalpies: From Reference 6.2, the uncertainty of the plant computer calculation of enthalpy is
U_{PC} = +/- 0.10 BTU/lbm

4.2.5. Steam Quality

4.2.5.1. Per Reference 6.8, steam quality, X, is set at 1. As shown below, this is a conservative input and uncertainties are not considered.

a. If carryover is considered, the gross thermal output of one steam generator is represented by
$$Q_{SG} = (M_{FW})[(1-X)h_{FS} + Xh_{GS} - h_{FW}] + (XM_{BD})(h_{FS} - h_{GS})$$
which reduces to
$$Q_{SG} = (M_{FW})(h_{FS} - h_{FW}) + (X)(M_{FW} - M_{BD})(h_{GS} - h_{FS})$$

b. Since quality only has values of 1 or less, the calculated thermal output, using a quality of 1, will always be equal to or greater than actual thermal output.

c. This results in a positive uncertainty and does not need to be considered per Section 3.2.

4.2.6. Calorimetric Constant Biases:

4.2.6.1. The biases of the calorimetric constants used in Reference 6.6 were reviewed by Reference 6.10. As part of the review, sign conventions of the biases were verified to be consistent with Section 3.2. Corrections to biases were made by Reference 6.6.1.

4.2.6.2. Per Section 3.2, only the negative value of biases needs to be considered in this evaluation.

4.2.6.3. If the values for the biases of the constants are different between Unit 1 and Unit 2, the most conservative value is used to evaluate calorimetric uncertainty.

4.2.6.4. The biases associated with the calorimetric constants are summarized below.

CONSTANT	NEGATIVE BIAS	UNITS
PAK0021, Pressurizer Heater Input	0	MW
PAK0026, Reactor Coolant Pump Heat Addition	-0.61	MW
PAK0022, RCS Heat Loss	-0.35	MW
PAK0024, Letdown Flow Heat Loss	-3.19	MBTU/hr

5.0 ASSUMPTIONS

5.1. Feedwater Flow:

5.1.1. Feedwater flow uncertainty is determined by the vendor, Cameron International Corporation (formally Caldon, Incorporated) based upon hydraulic modeling and testing at a hydraulic laboratory (typically Alden Lab). Although testing has not been completed, typical uncertainties are less than 0.4%.

5.1.2. For this evaluation, a conservative uncertainty of +/- 0.5000% actual flow will be used.

- THIS ASSUMPTION WILL BE VERIFIED UPON RECEIPT OF THE VENDOR UNCERTAINTY FOR FEEDWATER FLOW.

5.2. Feedwater Pressure

5.2.1. The Caldon topical report, Ref. 6.12, and the supplement to the topical report, Ref. 6.12.1, assume a pressure uncertainty of +/- 15.00 PSI, which will be used in this evaluation.

- THIS ASSUMPTION WILL BE VERIFIED UPON SELECTION OF THE PRESSURE INSTRUMENTATION AND EVALUATION OF THE INSTALLATION.

5.2.2. Actual feedwater pressures at the entrance to the steam generators are not known. Calorimetric uncertainty will be evaluated using steam generator pressure. This results in a conservative calculation of feedwater enthalpy since actual pressure at the inlet to the steam generator must be greater than steam generator pressure. See 7.2 for further discussion.

- THIS ASSUMPTION DOES NOT REQUIRE VERIFICATION. THE SELECTED PRESSURE CAN CONTINUE TO BE USED WITHOUT KNOWING ACTUAL FEEDWATER PRESSURE.

5.2.3. The same feedwater pressure instrumentation is used for both feedwater enthalpy and feedwater flow. This activity assumes the principle contribution to feedwater flow is density. For a given temperature, a higher than actual pressure increases density, resulting in a higher feedwater flow measurement. Feedwater enthalpy also increases, but the increase is minimal. From inspection of the steam generator thermal output calculation, a positive error in feedwater flow measurement results in a higher than actual calorimetric power computation. An increase in feedwater enthalpy results in a lower than actual calorimetric power computation. Therefore, the effects are offsetting, and it is more conservative to treat the uncertainties in feedwater flow and enthalpy as independent.

- THIS ASSUMPTION WILL BE VERIFIED UPON RECEIPT OF THE VENDOR UNCERTAINTY FOR FEEDWATER FLOW.

5.3. Blowdown Flow Distribution

5.3.1. The blowdown flow measurement used in the plant calorimetric is total blowdown flow.

5.3.2. From References 6.7.1 and 6.7.2, total indicated blowdown flow through both steam generators is limited to 180 gpm. Maximum indicated blowdown flow through a single steam generator is 150 gpm (Unit 1). The Unit 2 procedure does not provide a single steam generator limit. Therefore the Unit 1 limit from Reference 6.7.1 is used.

5.3.3. This activity evaluates blowdown assuming the following distribution of blowdown flow:

5.3.3.1. Total blowdown flow is set at the maximum permissible flow rate (180 GPM = 125.6 klbm/hr, per Reference 6.7). Blowdown flow for one steam generator is set at the maximum permissible flow rate (150 GPM = 107 klbm/hr, per Reference 6.7).

5.3.3.2. Total blowdown flow is set at the maximum permissible flow rate (180 GPM = 125.6 klbm/hr, per Reference 6.7). Blowdown flow is evenly distributed between the steam generators.

- THIS ASSUMPTION DOES NOT REQUIRE VERIFICATION SINCE IT IS FURTHER EVALUATED FOR CONSERVATISM WITHIN THIS CALCULATION.

5.4. Assumed Plant Parameters at 2737 MW

- THIS ASSUMPTION WILL BE VERIFIED UPON POWER ESCALATION, ALTHOUGH SLIGHT CHANGES IN POWER SHOULD NOT SIGNIFICANTLY AFFECT CALORIMETRIC UNCERTAINTY.

5.4.1. For parameters other than blowdown flow and feedwater pressure, conservative values, based upon trends in plant parameters following a reactor startup after a refueling outage are selected to maximize calorimetric uncertainty. The summary values and the data used in the determination are included within Reference 6.13.

5.4.2. The following table summarizes the bounding conditions for each parameter's indicated value, with exception of blowdown flow, as provided from Reference 6.13. The maximum indicated values have been rounded up to the nearest whole number. The minimum indicated values have been rounded down to the nearest whole number.

Parameter	Minimum (Indicated Value)	Maximum (Indicated Value)
Feedwater Flow, M_{FW}, klbm/hr	5932	6178
Feedwater Temperature, T_{FW}, deg. F	432	443
Feedwater Pressure, P_{FW}, PSIA	854	876
Steam Pressure, P_{STM}, PSIA	819	847

6.0 REFERENCES

6.1. ES-028, Instrument Loop Uncertainty And Setpoint Methodology, Revision 1

6.2. VTM 12138-249, Control Spec, Plant Computer, Revision 25

6.3. DCALC I-93-037, Uncertainty Calculation For The Plant Computer Indication Of Main Steam Pressure, Revision 1

6.4. DCALC CA04564, Uncertainty Calculation for the Blowdown Flow Input to the Secondary Heat Balance, Revision 0.

6.5. DCALC CA00470, Loop Uncertainty For Feedwater RTDS, Revision 0

6.6. DCALC I-93-072, Uncertainties Of The Secondary Calorimetric Constants For Unit 1 & 2, Revision 0

6.6.1. CCN 1-93-072-0001, Revision 0

6.7. Operating Procedures

6.7.1. OI-08A-1, Blowdown System, Revision 39 (Unit 1)

6.7.2. OI-08A-2, Blowdown System, Revision 37 (Unit 2)

6.8. SP 094, System 094 Setpoint File, Revision 9

6.9. ASME Steam Tables, Fifth Edition (1967 ASME Steam Tables)

6.10. ESP ES200400492-000, Review/Revise Calorimetric Constants, Revision 0

6.11. ISA-RP67.04.02-2000, Methodologies for the Determination of Setpoints for Nuclear Safety Related Instrumentation, 1/1/2000

6.12. Caldon, Inc. Engineering Report-80P, Topical Report, Improving Thermal Power Accuracy and Plant Safety While Increasing Operating Power Level Using the LEFM Check System, Revision 5, October 2001

6.12.1. Caldon, Inc. Engineering Report-157P, Supplement to Topical Report ER-80P, Basis for a Power Uprate With the LEFM Check or LEFM CheckPlus System, Revision 0, March, 1997

6.13. Constellation Correspondence DMLS # DE07881, D. A. Dvorak to File, Dated March 31, 2008, "Estimated Parameters for Calorimetric Power for a 1.4% Appendix K Uprate"

7.0 CALCULATION

7.1. EVALUATION OF UNCERTAINTY

7.1.1. The calculation of calorimetric uncertainty has three major components, Q_{SG1}, Q_{SG2}, and Q_{OTHER}. Calorimetric uncertainty may be evaluated by evaluating the uncertainty for each component, then statistically combining the results.

7.1.2. Similarly, the uncertainty of each major component is comprised of individual components. For example, Q_{SG1} is comprised of feedwater flow, blowdown flow, feedwater enthalpy (determined from feedwater pressure and temperature inputs), and steam enthalpies (determined by steam pressure input). The total uncertainty is determined by evaluating the uncertainty for each component, then statistically combining the results.

7.1.3. For the gross thermal output of a steam generator, Q_{SG}, the contribution for each input to uncertainty is determined from

$U_{SG-INPUT} = (Q_{SG})_{CALC-INPUT} - (Q_{SG})_{ACT}$, where

$(Q_{SG})_{CALC-INPUT}$ is the gross thermal output of the steam generator determined by varying the selected input parameter by its uncertainty while using actual values for the other inputs, and

$(Q_{SG})_{ACT}$ is the gross thermal output of the steam generator determined by using actual values for all inputs.

7.1.4. Biases are similarly determined where each input parameter is varied by its associated bias.

7.2. SELECTED FEEDWATER PRESSURE AND TEMPERATURE

7.2.1. Feedwater temperature and pressure are used to calculate feedwater enthalpy. Feedwater pressure is also used in the calculation of feedwater flow, but the effect of feedwater pressure on feedwater flow is included in the uncertainty of the measured flow rate. (See Assumption 5.2.3.)

7.2.2. Enthalpies are taken from Reference 6.9. Feedwater enthalpies for the range of interest are shown summarized below:

h_{FW}, BTU/lbm	P_{FW}, PSIA				
T_{FW}, DEG F	800	850	900	950	1000
420	397.35	397.40	397.45	397.50	397.55
430	408.29	408.33	408.37	408.41	408.46
440	419.31	419.35	419.38	419.42	419.45
450	430.43	430.46	430.49	430.52	430.55
460	441.66	441.68	441.70	441.72	441.74

7.2.2.1. Feedwater Pressure

a. Enthalpy increases as pressure increases. From inspection of the expression for steam generator gross thermal output, a negative calorimetric uncertainty will result if indicated pressure is greater than actual pressure. Therefore, a positive uncertainty value will be applied for use in the secondary calorimetric uncertainty analysis.

b. Maximizing the difference between steam enthalpy and feedwater enthalpy will result in the greatest calorimetric uncertainty contribution from feedwater flow. The difference is maximized by minimizing feedwater enthalpy, using lower values for feedwater pressure.

c. The relative change in feedwater enthalpy with temperature decreases as pressure increases. Therefore, lower values of pressure maximize the contribution of feedwater temperature to uncertainty.

d. For a given temperature, the relative change in enthalpy with pressure change is essentially constant. Therefore, the contribution of feedwater pressure to uncertainty does not vary substantially with changes in initial pressure.

e. In summary, lower values of pressure maximize the contribution of feedwater flow and temperature uncertainties to calorimetric uncertainty. Process pressure values do not significantly impact the contribution of feedwater pressure uncertainties to calorimetric uncertainty.

f. Calorimetric uncertainty is maximized by assuming minimum feedwater pressure with indicated pressure greater than actual pressure ($P_{FW\text{-}IND(m)} > P_{FW\text{-}ACT(m)}$).

7.2.2.2. The actual value of feedwater pressure is determined by subtracting measurement uncertainty from the minimum indicated value of feedwater pressure.

$$P_{FW-ACT(m)} = P_{FW-IND(m)} - U_{PFW}$$

$P_{FW\text{-}IND(m)}$ PSIA	U_{PFW}	$P_{FW\text{-}ACT(m)}$ PSIA	Reference Sections
854	15.00	839.00	5.4.2, 5.2.1

7.2.2.3. Feedwater Temperature

a. Enthalpy increases as temperature increases. From inspection of the expression for steam generator gross thermal output, a negative calorimetric uncertainty will result if indicated temperature is greater than actual temperature. Thus, a negative uncertainty value will be applied for use in the secondary calorimetric uncertainty analysis.

b. Maximizing the difference between steam enthalpy and feed enthalpy will result in the greatest calorimetric uncertainty contribution from feedwater flow. The difference is maximized by minimizing feedwater enthalpy, using lower values for feedwater temperature.

c. The relative change in feedwater enthalpy with pressure decreases as temperature increases. Therefore, lower values of temperature maximize the contribution of feedwater pressure to uncertainty.

d. For a given pressure, the relative change in enthalpy increases as temperature increases. Therefore, higher values of temperature maximize the contribution of feedwater temperature to uncertainty.

e. In summary, lower values of temperature maximize the contribution of feedwater flow and feedwater pressure to calorimetric uncertainty while higher values of feedwater temperature maximize the contribution of feedwater temperature to uncertainty.

f. Calorimetric uncertainty will be evaluated at both minimum and maximum feedwater temperatures, with indicated temperature greater than actual temperature ($T_{FW\text{-}IND(m)} > T_{FW\text{-}ACT(m)}$ and $T_{FW\text{-}IND(M)} > T_{FW\text{-}ACT(M)}$). In addition, analysis is performed at a higher temperature (454 deg. F) than the those specified within Reference 6.13 ($T_{FW\text{-}IND(M1)} > T_{FW\text{-}ACT(M1)}$) to confirm the error trend, and thus be confident of choosing the most conservative process condition for analysis.

$T_{FW\text{-}IND(m)}$ deg. F	U_{TFW}	$TP_{FW\text{-}ACT(m)}$ deg. F	Reference Sections
432	1.88	430.12	5.4.2, 4.2.1

$T_{FW\text{-}ACT(M)}$ deg. F	U_{TFW}	$TP_{FW\text{-}IND(M)}$ deg. F	Reference Sections
443	1.88	441.12	5.4.2, 4.2.1

$T_{FW\text{-}ACT(M1)}$ deg. F	U_{TFW}	$TP_{FW\text{-}IND(M1)}$ deg. F	Reference Sections
454	1.88	452.12	4.2.1

7.3. SELECTED STEAM PRESSURE

7.3.1. Steam pressure is used to calculate steam enthalpy. Enthalpies are taken from Reference 6.9. Steam enthalpies for the range of interest are shown summarized below:

P_{STM}, PSIA	h_{GS}, BTU/lbm	h_{FG}, BTU/lbm
750	1200.7	699.8
760	1200.4	697.7
770	1200.2	695.7
780	1199.9	693.6
790	1199.7	691.6
800	1199.4	689.6
810	1199.1	687.6
820	1198.8	685.5
830	1198.5	683.5
840	1198.2	681.5
850	1198.0	679.5
860	1197.7	677.6
870	1197.3	675.6
880	1197.0	673.6
890	1196.7	671.6

7.3.2. Since the mass flow rate of feedwater is substantially greater than the mass flow rate of blowdown through a single steam generator (the ratio of feedwater flow to blowdown flow is > 65), the principal effect on calorimetric uncertainty is the contribution to calorimetric uncertainty from feedwater flow. Maximizing the difference between steam enthalpy and feed enthalpy will result in the greatest calorimetric uncertainty contribution from feedwater flow. The difference is maximized by maximizing the saturated vapor enthalpy. Therefore, a lower steam pressure maximizes the contribution of feedwater flow to calorimetric uncertainty.

7.3.3. Steam Pressure Contribution to Uncertainty

7.3.3.1. The saturated vapor enthalpy, h_{GS}, decreases as pressure increases. From inspection of the expression for steam generator gross thermal output, a negative calorimetric uncertainty will result from the change in h_{GS} if indicated pressure is greater than actual pressure.

7.3.3.2. The difference between the saturated vapor and saturated liquid enthalpies, h_{FG}, also decreases as pressure increases. From inspection of the expression for steam generator gross thermal output, a negative calorimetric uncertainty will result from the change in h_{FG} if indicated pressure is less than actual pressure.

7.3.3.3. Referring to the information within table of Section 7.3.1 for 750 and 760 PSIA, for a 10 psi increase in indicated pressure above actual pressure, h_{GS} changes by approximately -0.3 BTU/lbm, while h_{FG} changes by approximately -2.1 BTU/lbm (approximately 7 times greater than the change in h_{GS}).

7.3.3.4. For calorimetric uncertainty, the change in h_{GS} is amplified by the feedwater flow rate while the change in h_{FG} is amplified by the blowdown flow rate.

7.3.3.5. Since the mass flow rate of feedwater is substantially greater than the mass flow rate of blowdown through a single steam generator, an increase in indicated pressure is a net negative contribution to calorimetric uncertainty.

7.3.4. The change in h_{GS} and the difference between the saturated vapor and saturated liquid enthalpies, h_{FG}, are essentially constant as steam pressure changes. Therefore, a change in initial pressure has a minimal effect on calorimetric uncertainty.

7.3.5. Calorimetric uncertainty is maximized by assuming minimum steam pressure with indicated pressure greater than actual pressure ($P_{STM\text{-}IND(m)} > P_{STM\text{-}ACT(m)}$).

The actual value of steam pressure is determined by subtracting measurement uncertainty and positive bias from the minimum indicated value of steam pressure.

$$P_{STM\text{-}ACT(m)} = P_{STM\text{-}IND(m)} - U_{PSTM} - B_{PSTM}$$

$P_{STM\text{-}IND(m)}$ PSIA	U_{PSTM}	B_{PSTM+}	$P_{STM\text{-}ACT(m)}$ PSIA	Reference Sections
819	19.80	3.40	795.80	5.4.2, 4.2.1

NOTE – The NIST steam tables show that the change in the saturated vapor enthalpy increases as pressure increases, while the change in the difference between saturated vapor enthalpy and saturated liquid enthalpy decreases, but the change is slight.

This results in opposite contributions to calorimetric uncertainty from steam pressure uncertainty. Since the mass flow rate of feedwater flow is substantially greater than the mass flow rate of blowdown, the contribution to calorimetric uncertainty from steam pressure uncertainty will be greater at higher steam pressures.

However, the overall contribution to calorimetric uncertainty from feedwater flow is also much greater than the contribution from steam pressure. Therefore, maximizing h_{GS}, by using lower values of steam generator pressure, results in a conservative assessment of calorimetric uncertainty.

7.3.6. For this evaluation, calorimetric uncertainty will be evaluated at minimum steam pressure to maximize steam enthalpy.

7.4. ENTHALPIES USED IN EVALUATION

7.4.1. Feedwater Enthalpy

Enthalpies used in this evaluation are summarized in the table below. Feedwater enthalpies are derived from interpolation of values in Reference 6.9.

h_{FW}, BTU/lbm @ T and P		Pressure, PSIA	
		$P_{FW-ACT(m)}$	$P_{FW-IND(m)}$
Temperature, deg. F		839	854
$T_{FW-ACT(m)}$	430.12	408.4534	408.4654
$T_{FW-IND(m)}$	432.00	410.5252	
$T_{FW-ACT(M)}$	441.12	420.5858	420.5967
$T_{FW-IND(M)}$	443.00	422.6749	
$T_{FW-ACT(M1)}$	452.12	432.8325	432.8409
$T_{FW-IND(M1)}$	454.00	434.9423	

7.4.2. Steam Enthalpy

7.4.2.1. Since steam pressure has an applicable bias, enthalpies are not evaluated at indicated pressure. Enthalpies are evaluated by individually applying random and bias components to actual pressure.

7.4.2.2. Steam enthalpies used in this evaluation are summarized in the table below. Enthalpies are derived from interpolation of values in Reference 6.9.

	Pressure, PSIA	h_{GS}, BTU/lbm	h_{FG}, BTU/lbm
$P_{STM-ACT(m)}$	795.80	1199.5260	690.4400
$P_{STM-ACT(m)}+B_{PSTM}$	799.20	1199.4240	689.7600
$P_{STM-ACT(m)}+U_{PSTM}$	815.60	1198.9320	686.4240

7.5. FEEDWATER FLOW UNCERTAINTY CONTRIBUTION

7.5.1. A negative calorimetric uncertainty results from a negative feedwater flow measurement uncertainty. (Indicated flow < Actual flow). Also, the contribution to calorimetric uncertainty from feedwater flow measurement is maximized by maximizing feedwater flow. Therefore, the maximum indicated flow is used in the evaluation of calorimetric uncertainty.

7.5.2. The maximum actual flow is determined from the maximum indicated flow and the associated uncertainty.

$$M_{FW-IND(M)} = M_{FW-ACT(M)}(1-U_{FW})$$

$$M_{FW-ACT(M)} = \frac{M_{FW-IND(M)}}{(1-U_{FW})}$$

$M_{FW-IND(M)}$ klbm/hr	U_{FW}	$M_{FW-ACT(M)}$ klbm/hr	Reference Sections
6178	0.5000%	6209.05	5.1.2, 4.2.1

7.5.3. General Equations for Uncertainty:

7.5.3.1. For each steam generator,

$$U_{SG-MFW} = \left\{ \begin{matrix} [(M_{FW})(h_{GS}-h_{FW})-(M_{BD})(h_{FG})]_{CALC-MFW} \\ -[(M_{FW})(h_{GS}-h_{FW})-(M_{BD})(h_{FG})]_{ACT} \end{matrix} \right\}$$

7.5.3.2. For an error in feedwater flow measurement, the expression reduces to

$$U_{SG-MFW} = (M_{FW-IND})(h_{GS} - h_{FW}) - (M_{FW-ACT})(h_{GS} - h_{FW}) \text{ or}$$

$$U_{SG-MFW} = (M_{FW-IND} - M_{FW-ACT})(h_{GS} - h_{FW}).$$

7.5.3.3. The error is the same for each steam generator,

$$U_{SG-MFW} = U_{SG1-MFW} = U_{SG2-MFW}$$

7.5.4. Contribution to Calorimetric Uncertainty

7.5.4.1. Using the expression above, the contribution of feedwater flow uncertainty to the gross thermal output of one steam generator is:

a. Minimum Indicated Feedwater Temperature

$M_{FW-IND(M)}$ klbm/hr	$M_{FW-ACT(M)}$ klbm/hr	h_{GS}, BTU/lbm @$P_{STM-ACT(m)}$	h_{FW}, BTU/lbm @$T_{FW-ACT(m)}$, $P_{FW-ACT(m)}$	U_{SG-MFW} MBTU/hr	Reference Sections
6178.00	6209.05	1199.5260	408.4534	-24.5590	7.5.2, 7.4.2, 7.4.1
-31.05		791.0726			

b. Maximum Indicated Feedwater Temperature

$M_{FW-IND(M)}$ klbm/hr	$M_{FW-ACT(M)}$ klbm/hr	h_{GS}, BTU/lbm @$P_{STM-ACT(m)}$	h_{FW}, BTU/lbm @$T_{FW-ACT(M)}$, $P_{FW-ACT(m)}$	U_{SG-MFW} MBTU/hr	Reference Sections
6178.00	6209.05	1199.5260	420.5858	-24.1824	7.5.2, 7.4.2, 7.4.1
-31.05		778.9402			

c. Additional Maximum Feedwater Temperature (Beyond Upper Limit)

$M_{FW-IND(M)}$ klbm/hr	$M_{FW-ACT(M)}$ klbm/hr	h_{GS}, BTU/lbm @$P_{STM-ACT(m)}$	h_{FW}, BTU/lbm @$T_{FW-ACT(M1)}$, $P_{FW-ACT(m)}$	U_{SG-MFW} MBTU/hr	Reference Sections
6178.00	6209.05	1199.5260	432.8325	-23.8022	7.5.2, 7.4.2, 7.4.1
-31.05		766.6935			

d. The tables above demonstrate that the contribution of feedwater flow to calorimetric uncertainty is maximized using lower values of feedwater temperature.

7.5.4.2. The net contribution to calorimetric uncertainty is

$$U_{CAL-MFW} = \sqrt{2} \times U_{SG-MFW}$$

FW Temperature	$U_{CAL-MFW}$, MBTU/hr
@$T_{FW-ACT(m)}$	-34.7317
@$T_{FW-ACT(M)}$	-34.1990
@$T_{FW-ACT(M1)}$	-33.6614

7.6. BLOWDOWN FLOW UNCERTAINTY CONTRIBUTION

7.6.1. General Equations for Uncertainty:

Since the same bounding conditions are established for feedwater flow, main steam pressure, feedwater temperature and feedwater pressure, the net thermal output of both steam generators, Q_{SG1} + Q_{SG2}, can be re-written as

$$Q_{SG1} + Q_{SG2} = (2M_{FW})(h_{GS} - h_{FW}) + (M_{BDT})(h_{FS} - h_{GS})$$

where M_{BDT} is the total blowdown flow.

The contribution to calorimetric uncertainty from total blowdown flow measurement is

$$U_{CAL-MBDT} = (Q_{SG1} + Q_{SG2} + Q_{OTHER})_{CALC-MBDT} - (Q_{SG1} + Q_{SG2} + Q_{OTHER})_{ACT-MBDT}$$

$$U_{CAL-MBDT} = \left\{ \begin{array}{l} [(2M_{FW})(h_{GS} - h_{FW}) - (M_{BDT-CALC})(h_{FG}) + Q_{OTHER}] \\ -[(2M_{FW})(h_{GS} - h_{FW}) - (M_{BDT-ACT})(h_{FG}) + Q_{OTHER}] \end{array} \right\}$$

$$U_{CAL-MBDT} = -U_{MBDT}(h_{FG})$$

7.6.2. From inspection of the above equations, a negative calorimetric uncertainty results from a positive total blowdown flow measurement uncertainty since h_{FG} is positive. (Indicated flow > Actual flow)

7.6.3. The net contribution to calorimetric uncertainty from blowdown is

U_{MBDT}, klbm/hr	h_{FG}, BTU/lbm @$P_{STM-ACT(m)}$	$U_{CAL-MBDT}$, MBTU/hr	Reference Sections
7.9	690.4400	-5.4545	4.2.1

7.7. FEEDWATER ENTHALPY UNCERTAINTY

7.7.1. Feedwater Temperature

7.7.1.1. General Equations for Uncertainty:

a. For each steam generator,

$$U_{SG-TFW} = \left\{ \begin{array}{l} [(M_{FW})(h_{GS} - h_{FW}) - (M_{BD})(h_{FG})]_{CALC-TFW} \\ -[(M_{FW})(h_{GS} - h_{FW}) - (M_{BD})(h_{FG})]_{ACT} \end{array} \right\}$$

b. For an error in feedwater temperature measurement, the expression reduces to

$$U_{SG-TFW} = (M_{FW})(h_{GS} - h_{FW-CALC}) - (M_{FW})(h_{GS} - h_{FW-ACT}) \text{ or}$$

$$U_{SG-TFW} = (M_{FW})(h_{FW-ACT} - h_{FW-CALC})$$

where h_{FW-ACT} is evaluated at actual feedwater temperature and pressure while h_{FW-IND} is evaluated at indicated feedwater temperature and actual pressure.

c. The error is the same for each steam generator,

$$U_{SG-TFW} = U_{SG1-TFW} = U_{SG2-TFW}$$

7.7.1.2. Contribution to Calorimetric Uncertainty

a. Using the expression above, the contribution of feedwater temperature measurement uncertainty to the gross thermal output of one steam generator is

(1) Minimum Indicated Feedwater Temperature

$M_{FW\text{-}ACT(M)}$, klbm/hr	$h_{FW\text{-}ACT}$ BTU/lbm @$T_{FW\text{-}ACT(m)}$, $P_{FW\text{-}ACT(m)}$	$h_{FW\text{-}CALC}$ BTU/lbm @$T_{FW\text{-}IND(m)}$, $P_{FW\text{-}ACT(m)}$	$U_{SG\text{-}TFW}$, MBTU/hr	Reference Sections
6209.05	408.4534	410.5252	-12.8637	7.5.2, 7.4.1
6209.05	-2.0718			

(2) Maximum Indicated Feedwater Temperature

$M_{FW\text{-}ACT(M)}$, klbm/hr	$h_{FW\text{-}ACT}$ BTU/lbm @$T_{FW\text{-}ACT(M)}$, $P_{FW\text{-}ACT(m)}$	$h_{FW\text{-}CALC}$ BTU/lbm @$T_{FW\text{-}IND(M)}$, $P_{FW\text{-}ACT(m)}$	$U_{SG\text{-}TFW}$, MBTU/hr	Reference Sections
6209.05	420.5858	422.6749	-12.9713	7.5.2, 7.4.1
6209.05	-2.0891			

(3) Additional Maximum Feedwater Temperature (Beyond Upper Limit)

$M_{FW\text{-}ACT(M)}$, klbm/hr	$h_{FW\text{-}ACT}$ BTU/lbm @$T_{FW\text{-}ACT(M1)}$, $P_{FW\text{-}ACT(m)}$	$h_{FW\text{-}CALC}$ BTU/lbm @$T_{FW\text{-}IND(M1)}$, $P_{FW\text{-}ACT(m)}$	$U_{SG\text{-}TFW}$, MBTU/hr	Reference Sections
6209.05	432.8325	434.9423	-13.0997	7.5.2, 7.4.1
6209.05	-2.1098			

(4) The tables above demonstrate that the contribution of feedwater temperature measurement to calorimetric uncertainty is maximized using higher values of feedwater temperature.

b. The net contribution to calorimetric uncertainty is

$$U_{CAL-TFW} = \sqrt{2} \times U_{SG-TFW}$$

FW Temperature	$U_{CAL\text{-}TFW}$, MBTU/hr
@$T_{FW\text{-}ACT(m)}$	-18.1920
@$T_{FW\text{-}ACT(M)}$	-18.3442
@$T_{FW\text{-}ACT(M1)}$	-18.5257

7.7.2. Feedwater Pressure

7.7.2.1. General Equations for Uncertainty:

a. For each steam generator,

$$U_{SG-PFW} = \left\{ \begin{array}{l} [(M_{FW})(h_{GS} - h_{FW}) - (M_{BD})(h_{FG})]_{CALC-PFW} \\ -[(M_{FW})(h_{GS} - h_{FW}) - (M_{BD})(h_{FG})]_{ACT} \end{array} \right\}$$

b. For an error in feedwater pressure measurement, the expression reduces to
$U_{SG-PFW} = (M_{FW})(h_{GS} - h_{FW-CALC}) - (M_{FW})(h_{GS} - h_{FW-ACT})$ or
$U_{SG-PFW} = (M_{FW})(h_{FW-ACT} - h_{FW-CALC})$
where h_{FW-ACT} is evaluated at actual feedwater temperature and pressure while h_{FW-IND} is evaluated at actual feedwater temperature and indicated pressure.

c. The error is the same for each steam generator,
$U_{SG-PFW} = U_{SG1-PFW} = U_{SG2-PFW}$

7.7.2.2. Contribution to Calorimetric Uncertainty

a. Using the expression above, the contribution of feedwater pressure measurement uncertainty to the gross thermal output of one steam generator is

(1) Minimum Indicated Feedwater Temperature

$M_{FW-ACT(M)}$ klbm/hr	h_{FW-ACT} BTU/lbm @$T_{FW-ACT(m)}$, $P_{FW-ACT(m)}$	$h_{FW-CALC}$ BTU/lbm @$T_{FW-ACT(m)}$, $P_{FW-IND(m)}$	U_{SG-PFW} MBTU/hr	Reference Sections
6209.05	408.4534	408.4654	-0.0744	7.5.2, 7.4.1
6209.05	-0.0120			

(2) Maximum Indicated Feedwater Temperature

$M_{FW-ACT(M)}$ klbm/hr	h_{FW-ACT} BTU/lbm @$T_{FW-ACT(M)}$, $P_{FW-ACT(m)}$	$h_{FW-CALC}$ BTU/lbm @$T_{FW-ACT(M)}$, $P_{FW-IND(m)}$	U_{SG-PFW} MBTU/hr	Reference Sections
6209.05	420.5858	420.5967	-0.0680	7.5.2, 7.4.1
6209.05	-0.0110			

(3) Additional Maximum Feedwater Temperature

$M_{FW-ACT(M)}$ klbm/hr	h_{FW-ACT} BTU/lbm @$T_{FW-ACT(M1)}$, $P_{FW-ACT(m)}$	$h_{FW-CALC}$ BTU/lbm @$T_{FW-ACT(M1)}$, $P_{FW-IND(m)}$	U_{SG-PFW} MBTU/hr	Reference Sections
6209.05	432.8325	432.8409	-0.0519	7.5.2, 7.4.1
6209.05	-0.0084			

(4) The tables above demonstrate that the contribution of feedwater pressure measurement to calorimetric uncertainty is maximized using lower values of feedwater temperature.

b. The net contribution to calorimetric uncertainty is
$U_{CAL-PFW} = \sqrt{2} \times U_{SG-PFW}$

FW Temperature	$U_{CAL-PFW}$, MBTU/hr
@$T_{FW-ACT(m)}$	-0.1053
@$T_{FW-ACT(M)}$	-0.0962
@$T_{FW-ACT(M1)}$	-0.0734

7.8. MAIN STEAM ENTHALPY UNCERTAINTY FROM PRESSURE MEASUREMENT

7.8.1. Random Component of Pressure Measurement:

7.8.1.1. General Equations for Uncertainty:

a. For each steam generator,

$$U_{SG-PFW} = \left\{ \begin{array}{l} [(M_{FW})(h_{GS} - h_{FW}) - (M_{BD})(h_{FG})]_{CALC-PSTM} \\ -[(M_{FW})(h_{GS} - h_{FW}) - (M_{BD})(h_{FG})]_{ACT} \end{array} \right\}$$

b. For an error in steam pressure measurement, the expression reduces to

$$U_{SG-PSTM} = \left\{ \begin{array}{l} [(M_{FW})(h_{GS-CALC}) - (M_{BD})(h_{FG-CALC})] \\ -[(M_{FW})(h_{GS-ACT}) - (M_{BD})(h_{FG-ACT})] \end{array} \right\} \text{ or}$$

$$U_{SG-PSTM} = [(M_{FW})(h_{GS\ CALC} - h_{GS\ ACT})] - [(M_{BD})(h_{FG\ CALC} - h_{FG\ ACT})]$$

where h_{GS-ACT} and h_{FG-ACT} are evaluated at actual steam pressure, while $h_{GS-CALC}$ and $h_{FG-CALC}$ are evaluated by applying the random component of steam pressure measurement uncertainty to the actual pressure.

c. Since blowdown flow can vary between steam generators, the error may not be the same for each generator.

7.8.1.2. Contribution to Calorimetric Uncertainty

a. Using the expression above, the contribution of steam pressure measurement uncertainty to the gross thermal output of one steam generator is

(1) Case 1 – Maximum Flow Through One Steam Generator, Maximum Total Flow

$M_{FW-ACT(M)}$ klbm/hr	$h_{GS-CALC}$ BTU/lbm @($P_{STM-ACT(m)}$ + U_{PSTM})	h_{GS-ACT} BTU/lbm @$P_{STM-ACT(m)}$	M_{BD-ACT} klbm/hr	$h_{FG-CALC}$ BTU/lbm @($P_{STM-ACT(m)}$ + U_{PSTM})	h_{FG-ACT} BTU/lbm @$P_{STM-ACT(m)}$	$U_{SG-PSTM}$ MBTU/hr
6209.05	1198.9320	1199.5260	107.00	686.4240	690.4400	-3.2585
6209.05	-0.5940		107.00	-4.0160		
-3688.17			-429.71			
6209.05	1198.9320	1199.5260	18.60	686.4240	690.4400	-3.6135
6209.05	-0.5940		18.60	-4.0160		
-3688.17			-74.70			

Note 1: Values Obtained from Sections 7.4.2, 7.5.2 and 5.3.3.
Note 2: The table above is split into two sets of three rows each. The second row in each set provides a difference in the enthalpies as computed in row 1, and the third row provides the flow multiplied by the difference in enthalpies.

(2) Case 2 – Evenly Distributed Blowdown Flows, Maximum Total Flow

$M_{FW-ACT(M)}$ klbm/hr	$h_{GS-CALC}$ BTU/lbm @($P_{STM-ACT(m)}$ + U_{PSTM})	h_{GS-ACT} BTU/lbm @$P_{STM-ACT(m)}$	M_{BD-ACT} klbm/hr	$h_{FG-CALC}$ BTU/lbm @($P_{STM-ACT(m)}$ + U_{PSTM})	h_{FG-ACT} BTU/lbm @$P_{STM-ACT(m)}$	$U_{SG-PSTM}$ MBTU/hr
6209.05	1198.9320	1199.5260	62.80	686.4240	690.4400	-3.4360
6209.05	-0.5940		62.80	-4.0160		
-3688.17			-252.20			
6209.05	1198.9320	1199.5260	62.80	686.4240	690.4400	-3.4360
6209.05	-0.5940		62.80	-4.0160		
-3688.17			-252.20			

Note: Values Obtained from Sections 7.4.2, 7.5.2 and 5.3.3.
Note 2: The table above is split into two sets of three rows each. The second row in each set provides a difference in the enthalpies as computed in row 1, and the third row provides the flow multiplied by the difference in enthalpies.

b. The net contribution to calorimetric uncertainty is

$$U_{CAL-PSTM} = \pm\sqrt{(U_{SG1-PSTM})^2 + (U_{SG2-PSTM})^2}$$

CASE	$U_{SG1-PSTM}$ MBTU/hr	$U_{SG2-PSTM}$ MBTU/hr	$U_{CAL-PSTM}$ MBTU/hr
1	-3.2585	-3.6135	-4.8657
2	-3.4360	-3.4360	-4.8592

This table shows that the contribution of steam pressure measurement to calorimetric uncertainty is maximized by assuming maximum flow through one steam generator and maximum total flow (Case 1), but the effect is slight.

7.8.2. Bias component of pressure measurement:

7.8.2.1. General Equations for Uncertainty:

a. For each steam generator,

$$B_{SG-PFW} = \left\{ \begin{matrix} [(M_{FW})(h_{GS} - h_{FW}) - (M_{BD})(h_{FG})]_{CALC-PSTM} \\ -[(M_{FW})(h_{GS} - h_{FW}) - (M_{BD})(h_{FG})]_{ACT} \end{matrix} \right\}$$

b. For an error in steam pressure measurement, the expression reduces to

$$B_{SG-PSTM} = \left\{ \begin{matrix} [(M_{FW})(h_{GS-CALC}) - (M_{BD})(h_{FG-CALC})] \\ -[(M_{FW})(h_{GS-ACT}) - (M_{BD})(h_{FG-ACT})] \end{matrix} \right\} \text{ or}$$

$$B_{SG-PSTM} = [(M_{FW})(h_{GS-CALC} - h_{GS-ACT})] - [(M_{BD})(h_{FG-CALC} - h_{FG-ACT})]$$

where h_{GS-ACT} and h_{FG-ACT} are evaluated at actual steam pressure, while $h_{GS-CALC}$ and $h_{FG-CALC}$ are evaluated by applying the bias component of steam pressure measurement uncertainty to the actual pressure.

c. As demonstrated previously, the contribution of steam pressure measurement to calorimetric uncertainty is maximized by assuming maximum flow through one steam generator and is the only case evaluated when evaluating bias.

7.8.2.2. Contribution to Calorimetric Bias

a. Using the expression above, the contribution of steam pressure measurement bias to the gross thermal output of one steam generator is

$M_{FW-ACT(M)}$ klbm/hr	$h_{GS-CALC}$ BTU/lbm @($B_{STM-ACT(m)}$ + U_{PSTM})	h_{GS-ACT} BTU/lbm @$P_{STM-ACT(m)}$	$M_{BD-ACT(m)}$ klbm/hr	$h_{FG-CALC}$ BTU/lbm @($B_{STM-ACT(m)}$ + U_{PSTM})	h_{FG-ACT} BTU/lbm @$P_{STM-ACT(m)}$	$B_{SG-PSTM}$ MBTU/hr
6209.05	1199.4240	1199.5260	107.00	689.7600	690.4400	-0.5606
6209.05	-0.1020		107.00	-0.6800		
-633.32			-72.76			
6209.05	1199.4240	1199.5260	18.60	689.7600	690.4400	-0.6207
6209.05	-0.1020		18.60	-0.6800		
-633.32			-12.65			

Note: Values Obtained from Sections 7.4.2, 7.5.2 and 5.3.3.
Note 2: The table above is split into two sets of three rows each. The second row in each set provides a difference in the enthalpies as computed in row 1, and the third row provides the flow multiplied by the difference in enthalpies.

b. The net contribution to calorimetric bias is
$B_{CAL-PSTM} = B_{SG1-PSTM} + B_{SG2-PSTM}$

$B_{SG1-PSTM}$ MBTU/hr	$B_{SG2-PSTM}$ MBTU/hr	$B_{CAL-PSTM}$ MBTU/hr
-0.5606	-0.6207	-1.1812

7.8.3. The net contribution of steam pressure measurement error to uncertainty is the sum of the random contribution to uncertainty and the bias contribution to uncertainty

$U_{CAL-PSTM}$ MBTU/hr	$B_{CAL-PSTM}$ MBTU/hr	$U_{CAL-PSTM(NET)}$ MBTU/hr
-4.8657	-1.1812	-6.0469

7.9. PLANT COMPUTER UNCERTAINTY CONTRIBUTION

7.9.1. Feedwater Enthalpy

7.9.1.1. General Equations for Uncertainty:

a. For each steam generator,

$$U_{SG-PC(hFW)} = \left\{ \begin{array}{l} [(M_{FW})(h_{GS} - h_{FW}) - (M_{BD})(h_{FG})]_{CALC-PC} \\ -[(M_{FW})(h_{GS} - h_{FW}) - (M_{BD})(h_{FG})]_{ACT} \end{array} \right\}$$

b. An error in the computation of feedwater enthalpy is determined from
$U_{SG-PC(hFW)} = (M_{FW})(h_{FW-ACT} - h_{FW-CALC})$ or
$U_{SG-PC(hFW)} = (M_{FW})(U_{PC})$

c. The error is the same for each steam generator,
$U_{SG-PC(hFW)} = U_{SG1-PC(hFW)} = U_{SG2-PC(hFW)}$
and the net contribution to calorimetric uncertainty is
$U_{CAL-PC(hFW)} = \sqrt{2} \times U_{SG-PFW}$

7.9.1.2. Using the expression above, the contribution of feedwater enthalpy uncertainty to the calorimetric uncertainty from the plant computer computation of enthalpy is

$M_{FW-ACT(m)}$ klbm/hr	U_{PC} BTU/lbm	$U_{SG-PC(hFW)}$ MBTU/hr	$U_{CAL-PC(hFW)}$ MBTU/hr	Reference Sections
6209.05	-0.10	-0.6209	-0.8781	7.5.2, 4.2.4

7.9.2. Main Steam Enthalpy, Saturated Vapor

7.9.2.1. The plant computer does not calculated h_{FG} directly but takes the difference between the calculated saturated vapor enthalpy and the saturated liquid enthalpy.

7.9.2.2. General Equations for Uncertainty:

a. For each steam generator,

$$U_{SG-PC(hGS)} = \begin{Bmatrix} [(M_{FW})(h_{GS} - h_{FW}) - (M_{BD})(h_{GS} - h_{FS})]_{CALC-PC} \\ -[(M_{FW})(h_{GS} - h_{FW}) - (M_{BD})(h_{GS} - h_{FS})]_{ACT} \end{Bmatrix}$$

b. An error in the computation of main steam saturated vapor enthalpy is determined from

$$U_{SG-PC(hGS)} = \begin{Bmatrix} [(M_{FW})(h_{GS-CALC}) - (M_{BD})(h_{GS-CALC})] \\ -[(M_{FW})(h_{GS-ACT}) - (M_{BD})(h_{GS-ACT})] \end{Bmatrix} \text{ or}$$

$$U_{SG-PC(hGS)} = (M_{FW} - M_{BD})(h_{GS-CALC} - h_{GS-ACT}) \text{ or}$$

$$U_{SG-PC(hGS)} = (M_{FW} - M_{BD})(U_{PC})$$

c. Since blowdown flow can vary between steam generators, the error may not be the same for each generator. As demonstrated previously, assuming maximum blowdown flow through one steam generator is the more conservative approach. The net contribution to calorimetric uncertainty is

$$U_{CAL-PC(hGS)} = \sqrt{(U_{SG1-PC(hGS)})^2 + (U_{SG2-PC(hGS)})^2}$$

7.9.2.3. Using the expression above, the contribution of main steam saturated vapor enthalpy uncertainty to the calorimetric uncertainty from the plant computer computation of enthalpy is

$M_{FW-ACT(m)}$ klbm/hr	M_{BD} klbm/hr	U_{PC} BTU/lbm	$U_{SG-PC(hGS)}$ MBTU/hr	$U_{CAL-PC(hGS)}$ MBTU/hr	Reference Sections
6209.05	107.0	-0.10	-0.6102	-0.8692	7.5.2, 5.3, 4.2.4
6102.0		-0.10			
6209.05	18.6	-0.10	-0.6190		
6190.4		-0.10			

7.9.3. Main Steam Enthalpy, Saturated Liquid

7.9.3.1. The plant computer does not calculate h_{FG} directly, but takes the difference between the calculated saturated vapor enthalpy and the saturated liquid enthalpy.

7.9.3.2. General Equations for Uncertainty:

a. For each steam generator,

$$U_{SG-PC(hFS)} = \begin{Bmatrix} [(M_{FW})(h_{GS} - h_{FW}) - (M_{BD})(h_{GS} - h_{FS})]_{CALC-PC} \\ - [(M_{FW})(h_{GS} - h_{FW}) - (M_{BD})(h_{GS} - h_{FS})]_{ACT} \end{Bmatrix}$$

b. An error in the computation of main steam saturated liquid enthalpy is determined from

$$U_{SG-PC(hFS)} = (M_{BD})(h_{FS-ACT} - h_{FS-CALC}) \text{ or}$$

$$U_{SG-PC(hFS)} = (M_{BD})(U_{PC})$$

c. Since blowdown flow can vary between steam generators, the error may not be the same for each generator. As demonstrated previously, assuming maximum blowdown flow through one steam generator is the more conservative approach.

$$U_{CAL-PC(hFS)} = \sqrt{(U_{SG1-PC(hFS)})^2 + (U_{SG2-PC(hFS)})^2}$$

7.9.3.3. Using the expression above, the contribution of main steam saturated liquid enthalpy uncertainty to the calorimetric uncertainty from the plant computer computation of enthalpy is:

M_{BD} klbm/hr	U_{PC} BTU/lbm	$U_{SG-PC(hFS)}$ MBTU/hr	$U_{CAL-PC(hFS)}$ MBTU/hr	Reference Sections
107.0	-0.10	-0.0107	-0.0109	5.3, 4.2.4
18.6	-0.10	-0.0019		

7.9.4. The combined uncertainty for the plant computer calculation of enthalpies is given by

$$U_{CAL-PC} = \sqrt{(U_{CAL-PC(hFW)})^2 + (U_{CAL-PC(hGS)})^2 + (U_{CAL-PC(hFS)})^2}$$

$U_{CAL-PC(hFW)}$ BTU/hr	$U_{CAL-PC(hGS)}$ BTU/hr	$U_{CAL-PC(hFS)}$ BTU/hr	U_{CAL-PC} MBTU/hr
-0.8781	-0.8692	-0.0109	-1.2356

7.10. CALORIMETRIC CONSTANT BIAS

7.10.1. Since the inputs to Q_{OTHER} only consist of bias terms, the contribution of Q_{OTHER} to net calorimetric uncertainty is the sum of all biases associated with the inputs to Q_{OTHER}.

7.10.2. A conversion of 3.412141 MBTU/hr/MW is used when summing the biases. The total bias, $B_{CAL-OTHER}$, is expressed in units of MBTU/hr, rounded up to the nearest 0.0001 MBTU/hr.

CONSTANT	NEGATIVE BIAS	UNITS	REFERENCE SECTIONS
PAK0021, Pressurizer Heater Input	0	MW	4.2.6.4
PAK0026, Reactor Coolant Pump Heat Addition	-0.61	MW	4.2.6.4
PAK0022, RCS Heat Loss	-0.35	MW	4.2.6.4
PAK0024, Letdown Flow Heat Loss	-3.19	MBTU/hr	4.2.6.4
$B_{CAL-OTHER}$	-6.4657	MBTU/hr	

7.11. NET CALORIMETRIC RANDOM UNCERTAINTY

7.11.1. The contribution of all random terms to net calorimetric uncertainty is determined using the most limiting uncertainties for each input.

7.11.1.1. Feedwater Temperature

a. Feedwater temperature was evaluated at minimum and maximum actual temperatures, as well as a temperature higher than the actual temperature range, to confirm the trend in the uncertainty data. Feedwater temperatures impact feedwater flow measurement uncertainty, feedwater temperature measurement uncertainty and feedwater pressure measurement uncertainty. The final temperature to use in the assessment of calorimetric uncertainty is determined by using the most conservative limit:

FW Temperature	$U_{CAL-MFW}$, MBTU/hr	$U_{CAL-TFW}$, MBTU/hr	$U_{CAL-PFW}$, MBTU/hr	SRSS
$@T_{FW-ACT(m)}$	-34.7317	-18.1920	-0.1053	39.2078
$@T_{FW-ACT(M)}$	-34.1990	-18.3442	-0.0962	38.8084
$@T_{FW-ACT(MI)}$	-33.6614	-18.5257	-0.0734	38.4226

b. It is observed that the overall calorimetric uncertainty increases with decreasing feedwater temperature in a near linear fashion over the temperature range of interest. Therefore, the minimum value of feedwater temperature results in the most limiting assessment of calorimetric uncertainty.

7.11.1.2. Blowdown Flow – As demonstrated previously, calorimetric uncertainty is maximized by maximizing blowdown flow through one steam generator.

7.11.2. General Equation:

$$U_{CAL} = \sqrt{(U_{CAL-MFW})^2 + (U_{CAL-MBDT})^2 + (U_{CAL-TFW})^2 + (U_{CAL-PFW})^2 + (U_{CAL-PSTM})^2 + (U_{CAL-PC})^2}$$

$U_{CAL-MFW}$ MBTU/hr	$U_{CAL-MBDT}$ MBTU/hr	$U_{CAL-TFW}$ MBTU/hr	$U_{CAL-PFW}$ MBTU/hr	$U_{CAL-PSTM}$ MBTU/hr	U_{CAL-PC} MBTU/hr	U_{CAL} MBTU/hr
-34.7317	-5.4545	-18.1920	-0.1053	-4.8657	-1.2356	-39.9024

7.11.3. Single Side of Interest Correction

7.11.3.1. Since only negative calorimetric uncertainty is considered, the methodology established in Reference 6.11, Section 8.1, "Correction for Setpoints with a Single Side of Interest" may be applied.

7.11.3.2. The calorimetric uncertainty calculated previously is based upon a 95% confidence level (1.96 standard deviations). The random component of calorimetric uncertainty may be reduced by a correction of (1.645/1.96).

$$U_{CAL(SS)} = \frac{1.645}{1.96}(U_{CAL}) = -33.4895 \text{ MBTU/hr}$$

7.12. CALORIMETRIC BIAS

7.12.1. Calorimetric bias is the sum of all bias components.

7.12.2. Contributions to calorimetric bias are limited to the calorimetric constants and main steam pressure.

$B_{CAL-OTHER}$ MBTU/hr	$B_{CAL-PSTM}$ MBTU/hr	B_{CAL} MBTU/hr
-6.4657	-1.1812	-7.6469

7.13. NET CALORIMETRIC UNCERTAINTY

7.13.1. Net calorimetric uncertainty is

$$U_{CAL-NET} = U_{CAL} + B_{CAL}$$

7.13.2. A conversion of 3.412141 MBTU/hr/MW is used to express uncertainty in MW.

7.13.3. Uncertainty is expressed in %RTP by dividing the uncertainty, in MW, by 2737 MW.

7.13.4. Net Uncertainty

U_{CAL} MBTU/hr	B_{CAL} MBTU/hr	$U_{CAL-NET}$ MBTU/hr	$U_{CAL-NET}$ MW	$U_{CAL-NET}$ %RTP
-39.9024	-7.6469	-47.5493	-13.9353	-0.5091%

7.13.5. Net Uncertainty, Single Side of Interest

$U_{CAL(SS)}$ MBTU/hr	B_{CAL} MBTU/hr	$U_{CAL-NET}$ MBTU/hr	$U_{CAL-NET(SS)}$ MW	$U_{CAL-NET(SS)}$ %RTP
-33.4895	-7.6469	-41.1365	-12.0559	-0.4405%

7.14. MARGIN

7.14.1. Available margin is obtained by adding the net uncertainty to the Appendix K power limit of 2754 MW and subtracting the rated thermal power of 2737 MW.

7.14.2. Margin is expressed in %RTP by dividing the uncertainty, in MW, by 2737 MW.

7.14.3. Available Margin

Appendix K Limit	Rated Thermal Power	$U_{CAL-NET}$ MW	Margin MW	Margin %RTP
2754	2737	-13.9353	3.0647	0.1120%

7.14.4. Available Margin, Single Side of Interest

Appendix K Limit	Rated Thermal Power	$U_{CAL-NET(SS)}$ MW	Margin MW	Margin %RTP
2754	2737	-12.0559	4.9441	0.1806%

7.15. RELATIVE CONTRIBUTIONS OF EACH INPUT TO OVERALL UNCERTAINTY

Providing a relative contribution permits a method of combining random and bias components of uncertainty for each input to evaluate the effect of each input on the overall calorimetric uncertainty.

7.15.1. The relative contribution of each random input to calorimetric uncertainty is given by

$$U'_{INPUT} = \left(\frac{U_{INPUT}}{U_{CAL}}\right) U_{INPUT}$$

7.15.2. This expression can be combined with the associated bias input to find the net contribution of that input to calorimetric uncertainty

$$U'_{INPUT-NET} = U'_{INPUT} + B_{INPUT}$$

7.15.3. The net contribution of each input to net calorimetric uncertainty can then be found by

$$U_{INPUT-NET\%} = \frac{U'_{INPUT-NET}}{U_{CAL-NET}}$$

and is summarized in the following table:

INPUT	U_{INPUT} MBTU/hr	(U'_{INPUT}) MBTU/hr	B_{INPUT} MBTU/hr	$(U'_{INPUT-NET})$ MBTU/hr	$U_{INPUT-NET\%}$
Feedwater Flow	-34.7317	-30.2310		-30.2310	63.578%
Blowdown Flow	-5.4545	-0.7456		-0.7456	1.568%
Feedwater Temperature	-18.1920	-8.2939		-8.2939	17.443%
Feedwater Pressure	-0.1053	-0.0003		-0.0003	0.001%
Steam Pressure	-4.8657	-0.5933	-1.1812	-1.7746	3.732%
Plant Computer Computation of Enthalpies	-1.2356	-0.0383		-0.0383	0.080%
Other Inputs			-6.4657	-6.4657	13.598%
Totals		-39.9024	-7.6469	-47.5493	100.000%

7.15.4. Single Side of Interest:

7.15.4.1. A similar approach is used to find the relative contribution of each input to calorimetric uncertainty for the single side of interest.

7.15.4.2. For single side of interest, the relative contribution of each input to calorimetric uncertainty is given by

$$U'_{INPUT(SS)} = \frac{1.645}{1.96}\left(\frac{U_{INPUT}}{U_{CAL}}\right)U_{INPUT}$$

$$U'_{INPUT-NET(SS)} = U'_{INPUT(SS)} + B_{INPUT}, \text{ and}$$

$$U_{INPUT-NET(SS)\%} = \frac{U'_{INPUT-NET(SS)}}{U_{CAL-NET(SS)}}$$

7.15.4.3. The relative contributions to calorimetric uncertainty using single side of interest is summarized in the table below:

INPUT	U_{INPUT} MBTU/hr	$(U'_{INPUT(SS)})$ MBTU/hr	B_{INPUT} MBTU/hr	$(U'_{INPUT-NET(SS)})$ MBTU/hr	$U_{INPUT-NET(SS)\%}$
Feedwater Flow	-29.1498	-25.3725		-25.3725	61.679%
Blowdown Flow	-4.5779	-0.6258		-0.6258	1.521%
Feedwater Temperature	-15.2682	-6.9610		-6.9610	16.922%
Feedwater Pressure	-0.0884	-0.0002		-0.0002	0.001%
Steam Pressure	-4.0837	-0.4980	-1.1812	-1.6792	4.082%
Plant Computer Computation of Enthalpies	-1.0370	-0.0321		-0.0321	0.078%
Other Inputs			-6.4657	-6.4657	15.718%
Totals		-33.4895	-7.6469	-41.1365	100.000%

8.0 CONCLUSION

This calculation determines the calorimetric uncertainty using the Caldon LEFM CheckPlus ultrasonic flow measurement system to measure feedwater flow. Uncertainty is evaluated at the proposed Appendix K uprated power of 2737 MW(th). The Appendix K power represents an increase of approximately 1.4% from the current licensed power limit of 2700 MW(th).

This calculation contains various unverified assumptions, which will be verified as design progresses for the UFMs. Specifically, assumptions 5.1.2, 5.2.1 and 5.2.3 will be verified as a part of the design process. Additionally, the conservatism of the assumptions regarding maximum and minimum uprated plant process parameters to be used for the computation (5.4.1 and 5.4.2) will be verified upon startup after the power uprate. However, the process for which these limits were chosen was very conservative, and the analyzed limits are not anticipated to change.

Results Using Single-Side-of-Interest Approach:

This calculation determines total secondary calorimetric calculation uncertainties using standard methodology. The calculation is performed in general accordance with Reference 6.1, considering the single-side-of-interest approach, as defined in Section 8.1 of Reference 6.11. The following table presents the results of this analysis, where $U_{CAL\text{-}NET(SS)}$ is the total secondary calorimetric uncertainty.

Appendix K Limit	Rated Thermal Power	$U_{CAL\text{-}NET(SS)}$ MW	Margin MW	Margin %RTP
2754	2737	-12.0559	4.9441	0.1806%

Since adequate margin exists between the proposed Rated Thermal Power and the Appendix K limit to account for instrument uncertainty and additional margin, the proposed Rated Thermal Power limit is deemed acceptable, considering instrument uncertainty.

The relative contributions of each uncertainty term to the total secondary calorimetric uncertainty are provided in the table below.

INPUT	U_{INPUT} MBTU/hr	$(U'_{INPUT(SS)})$ MBTU/hr	B_{INPUT} MBTU/hr	$(U'_{INPUT\text{-}NET(SS)})$ MBTU/hr	$U_{INPUT\text{-}NET(SS)\%}$
Feedwater Flow	-29.1498	-25.3725		-25.3725	61.679%
Blowdown Flow	-4.5779	-0.6258		-0.6258	1.521%
Feedwater Temperature	-15.2682	-6.9610		-6.9610	16.922%
Feedwater Pressure	-0.0884	-0.0002		-0.0002	0.001%
Steam Pressure	-4.0837	-0.4980	-1.1812	-1.6792	4.082%
Plant Computer Computation of Enthalpies	-1.0370	-0.0321		-0.0321	0.078%
Other Inputs			-6.4657	-6.4657	15.718%
Totals		-33.4895	-7.6469	-41.1365	100.000%

Results if Single-Side-of-Interest Approach is Not Credited:

This calculation also presents the same set of results as above for the case where the single-side-of-interest approach, as defined in Section 8.1 of Reference 6.11, is not credited. Under this case, the following table presents the results of this analysis; where $U_{CAL-NET}$ is the total secondary calorimetric uncertainty.

Appendix K Limit	Rated Thermal Power	$U_{CAL-NET}$ MW	Margin MW	Margin %RTP
2754	2737	-13.9353	3.0647	0.1120%

Since adequate margin exists between the proposed Rated Thermal Power and the Appendix K limit to account for instrument uncertainty and additional margin, the proposed Rated Thermal Power limit is deemed acceptable, considering instrument uncertainty.

The relative contributions of each uncertainty term to the total secondary calorimetric uncertainty are provided in the table below.

INPUT	U_{INPUT} MBTU/hr	(U'_{INPUT}) MBTU/hr	B_{INPUT} MBTU/hr	$(U'_{INPUT-NET})$ MBTU/hr	$U_{INPUT-NET\%}$
Feedwater Flow	-34.7317	-30.2310		-30.2310	63.578%
Blowdown Flow	-5.4545	-0.7456		-0.7456	1.568%
Feedwater Temperature	-18.1920	-8.2939		-8.2939	17.443%
Feedwater Pressure	-0.1053	-0.0003		-0.0003	0.001%
Steam Pressure	-4.8657	-0.5933	-1.1812	-1.7746	3.732%
Plant Computer Computation of Enthalpies	-1.2356	-0.0383		-0.0383	0.080%
Other Inputs			-6.4657	-6.4657	13.598%
Totals		-39.9024	-7.6469	-47.5493	100.000%

ATTACHMENT (3)

MARKED UP TECHNICAL SPECIFICATION PAGES

Renewed Operating License Page 3 (Unit 1)

Renewed Operating License Page 3 (Unit 2)

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 ~~2700~~ 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. 286, 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 ~~2700~~ 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. 263 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 ~~2700~~ 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