February 27, 2008

Mr. Tom E. Tynan
Vice President - Vogtle
Vogtle Electric Generating Plant
7821 River Road
Waynesboro, GA 30830

SUBJECT: VOGTLE ELECTRIC GENERATING PLANT, UNITS 1 AND 2, ISSUANCE OF AMENDMENTS REGARDING MEASUREMENT UNCERTAINTY RECAPTURE POWER UPRATE (TAC NOS. MD6625 AND MD6626)

Dear Mr.Tynan:

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 149 to Facility Operating License NPF-68 and Amendment No. 129 to Facility Operating License NPF-81 for the Vogtle Electric Generating Plant, Units 1 and 2. The amendments consist of changes to the Technical Specifications (TSs) in response to your application dated August 28, 2007, as supplemented by letters dated October 9, 2007, December 21, 2007, January 18, 2008, and January 30, 2008.

The amendments increase the licensed core power level by 1.7% to 3625.6 megawatts thermal. This increase will be achieved by the installation of the Caldon leading edge flow measurement (LEFM) CheckPlus ultrasonic feedwater flow element, which allows for more accurate measurement of feedwater flow.

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

Sincerely,

/RA/

Siva P. Lingam, Project Manager
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-424 and 50-425

Enclosures:
1. Amendment No. 149 to NPF-68
2. Amendment No. 129 to NPF-81
3. Safety Evaluation

cc w/encls: See next page

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NAME	AKlein	AMendiola	GCranston	DHarrison	
DATE	1/25/08*	1/14/08*	1/14/08*	2/20/08	
OFFICE	LPL2-2/PM	LPL2-2/LA	OGC	LPL2-2/BC (A)	DORL/D
NAME	SLingam	MO'Brien LOlshan for	E.Williamson** DRoth for	MWong LOlshan for	CHaney
DATE	2/11/08	2/26/08	2/25/08	2/26/08	2/26/08

OFFICIAL RECORD

Letter to Tom E. Tynan from Siva P. Lingam dated February 27, 2008

SUBJECT: VOGTLE ELECTRIC GENERATING PLANT, UNITS 1 AND 2 - ISSUANCE OF AMENDMENT REGARDING MEASUREMENT UNCERTAINTY RECAPTURE POWER UPRATE (TAC NOS. MD6625 AND MD6626)

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RidsNrrDraAadb(JParillo)

SOUTHERN NUCLEAR OPERATING COMPANY, INC.

GEORGIA POWER COMPANY

OGLETHORPE POWER CORPORATION

MUNICIPAL ELECTRIC AUTHORITY OF GEORGIA

CITY OF DALTON, GEORGIA

VOGTLE ELECTRIC GENERATING PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 149
License No. NPF-68

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

   A. The application for amendment to the Vogtle Electric Generating Plant, Unit 1 (the facility) Facility Operating License No. NPF-68 filed by the Southern Nuclear Operating Company, Inc. (the licensee), acting for itself, Georgia Power Company, Oglethorpe Power Corporation, Municipal Electric Authority of Georgia, and City of Dalton, Georgia (the owners), dated August 28, 2007, as supplemented by letters dated October 9, 2007, December 21, 2007, January 18, 2008, and January, 30, 2008, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations as set forth in 10 CFR Chapter I;

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

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I;

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

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

2. Accordingly, the license is hereby amended by page changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. NPF-68 is hereby amended to read as follows:

   Technical Specifications and Environmental Protection Plan

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 149, and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated into this license. Southern Nuclear shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. Further, Facility Operating License No. NPF-68 is amended by changes to License Condition 2.C.(1) which reads:

   Maximum Power Level

   Southern Nuclear is authorized to operate the facility at reactor core power levels not in excess of 3625.6 megawatts thermal (100 percent power) in accordance with the conditions specified herein.

4. This license amendment is effective as of its date of issuance and shall be implemented at the completion of Unit 1 spring 2008 refueling outage.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Catherine Haney, Director
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to License No. NPF-68
and the Technical Specifications

Date of Issuance: February 27, 2008

SOUTHERN NUCLEAR OPERATING COMPANY, INC.

GEORGIA POWER COMPANY

OGLETHORPE POWER CORPORATION

MUNICIPAL ELECTRIC AUTHORITY OF GEORGIA

CITY OF DALTON, GEORGIA

VOGTLE ELECTRIC GENERATING PLANT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 129
License No. NPF-81

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

   A. The application for amendment to the Vogtle Electric Generating Plant, Unit 2 (the facility) Facility Operating License No. NPF-81 filed by the Southern Nuclear Operating Company, Inc. (the licensee), acting for itself, Georgia Power Company Oglethorpe Power Corporation, Municipal Electric Authority of Georgia, and City of Dalton, Georgia (the owners), dated August 28, 2007, as supplemented by letters dated October 9, 2007, December 21, 2007, January 18, 2008, and January 30, 2008, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations as set forth in 10 CFR Chapter I;

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

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I;

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

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

2. Accordingly, the license is hereby amended by page changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. NPF-81 is hereby amended to read as follows:

   Technical Specifications and Environmental Protection Plan

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 129, and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated into this license. Southern Nuclear shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. Further, Facility Operating License No. NPF-81 is amended by changes to License Condition 2.C.(1) which reads:

   Maximum Power Level

   Southern Nuclear is authorized to operate the facility at reactor core power levels not in excess of 3625.6 megawatts thermal (100 percent power) in accordance with the conditions specified herein.

4. This license amendment is effective as of its date of issuance and shall be implemented at the completion of Unit 2 fall 2008 refueling outage.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Catherine Haney, Director
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to License No. NPF-81
and the Technical Specifications

Date of Issuance: February 27, 2008

ATTACHMENT

TO LICENSE AMENDMENT NO. 149

FACILITY OPERATING LICENSE NO. NPF-68

DOCKET NO. 50-424

AND

TO LICENSE AMENDMENT NO. 129

FACILITY OPERATING LICENSE NO. NPF-81

DOCKET NO. 50-425

Replace the following pages of the Licenses and the Appendix A Technical Specifications (TSs) with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove Pages	Insert Pages
License	License
License No. NPF-68, page 3	License No. NPF-68, page 3
License No. NPF-68, page 4	License No. NPF-68, page 4
License No. NPF-81, page 4	License No. NPF-81, page 4
TSs	TSs
1.1-2	1.1-2
1.1-5	1.1-5
3.3.1-18	3.3.1-18

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO

AMENDMENT NO. 149 TO FACILITY OPERATING LICENSE NPF-68

AND

AMENDMENT NO. 129 TO FACILITY OPERATING LICENSE NPF-81

SOUTHERN NUCLEAR OPERATING COMPANY, INC.

VOGTLE ELECTRIC GENERATING PLANT, UNITS 1 AND 2

DOCKET NOS. 50-424 AND 50-425

1.0 INTRODUCTION

By application dated August 28, 2007 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML072470688), as supplemented by letters dated October 9, 2007 (ADAMS Accession No. ML072850108), December 21, 2007 (ADAMS Accession No. ML073580035), January 18, 2008 (ADAMS Accession No. ML080240086), and January 30, 2008 (ADAMS Accession No. ML080300576), Southern Nuclear Operating Company, Inc. (the licensee), requested changes to the Technical Specifications (TSs) to increase the licensed thermal power level for the Vogtle Electric Generating Plant, Units 1 and 2 (Vogtle 1 and 2).

The amendment would increase the licensed core power level for Vogtle 1 and 2 by 1.7% to 3625.6 megawatts thermal (MWt). This increase will be achieved by the installation of the Caldon leading edge flow measurement (LEFM) CheckPlus (or √+) ultrasonic flow measurement (UFM) system, which allows for more accurate measurement of feedwater (FW) flow. The supplements dated October 9, 2007, December 21, 2007, January 18, 2008, and January 30, 2008, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the Nuclear Regulatory Commission (NRC or the Commission) staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on November 20, 2007 (72 FR 65372).

2.0 BACKGROUND

Nuclear power plants are licensed to operate at a specified core thermal power. Title 10 of the *Code of Federal Regulations* (10 CFR), Part 50, Appendix K, requires licensees to assume that the reactor has been operating continuously at a power level at least 1.02 times the licensed power level when performing loss-of-coolant accident (LOCA) and emergency core cooling

system (ECCS) analyses. This requirement is included to ensure that instrumentation uncertainties are adequately accounted for in the analyses. Appendix K to 10 CFR, Part 50 allows licensees to assume a power level less than 1.02 times the licensed power level (but not less than the licensed power level), provided the licensee has demonstrated that the proposed value adequately accounts for instrumentation uncertainties. The licensee has proposed to use a power measurement uncertainty of 0.3%. To achieve this level of accuracy, the licensee will install a Caldon LEFM CheckPlus UFM system for measuring the main FW flow at Vogtle 1 and 2. The Caldon system provides a more accurate measurement of FW flow than the FW flow measurement accuracy assumed during the development of the original 10 CFR Part 50, Appendix K requirements and that of the current method of FW flow measurement used to calculate reactor thermal output. The Caldon system will measure FW mass flow to within plus or minus (±) 0.25% for Vogtle 1 and 2. This bounding FW mass flow uncertainty would be used to calculate a total power measurement uncertainty of 0.3%. On the basis of this, the licensee proposed to reduce the power measurement uncertainty required by 10 CFR Part 50, Appendix K to 0.3%. The improved power measurement uncertainty would obviate the need for the 2% power margin originally required by 10 CFR Part 50, Appendix K, thereby allowing an increase in the reactor power available for electrical generation. This accuracy is supported by Caldon Topical Report ER-80P, "Improving Thermal Power Accuracy and Plant Safety While Increasing Operating Power Level Using the LEFM Check™ System," Caldon Engineering Report ER-160P, "Supplement to Topical Report ER-80P: Basis for a Power Uprate with the LEFM Check™ System" and Caldon Engineering Report ER-157P, "Supplement to Topical Report ER-80P: Basis for a Power Uprate with the LEFM Check™ or LEFM CheckPlus™ System." The NRC staff approved Topical Report ER-80 and its supplements Topical Report ER-160 and Topical Report ER-157 in Safety Evaluations (SEs) dated March 8, 1999, January 19, 2001, and December 20, 2001, respectively, for use in justifying MUR power uprates up to 1.7%.

3.0 EVALUATION

3.1 Instrumentation and Controls

3.1.1 Regulatory Evaluation

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

Because the maximum power level of a nuclear plant is a licensed limit, a proposal to raise the licensed power level must be reviewed and approved under the license amendment process. The license amendment request should include a justification for the reduced power measurement uncertainty to support the proposed power uprate. Caldon Topical Report ER-80P and its

Supplement, ER-157P, describe the LEFM √+ System for the measurement of feedwater flow and provide a basis for the proposed 1.7% uprate of the licensed reactor power. The NRC staff also considered the guidance of NRC Regulatory Issue Summary (RIS) 2002-03, "Guidance on the Content of Measurement Uncertainty Recapture Power Uprate Applications" in its review of the licensee's submittals for the proposed power uprate request.

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

3.1.2 Technical Evaluation

The licensee's request is based on a reduced measurement uncertainty of core thermal power due to the installation of a Caldon LEFM CheckPlus system to measure FW flow at Vogtle 1 and 2. The licensee's submittal referenced Caldon Topical Report ER-80P, "Improving Thermal Power Accuracy and Plant Safety While Increasing Operating Power Level Using the LEFM Check™ System," and its Supplement ER-157P, "Supplement to Topical Report ER-80P: Basis for a Power Uprate with the LEFM Check™ or LEFM CheckPlus™ System." These two reports together provide a generic basis for the proposed 1.7% power uprate. The Caldon Topical Report ER-80P and its Supplement ER-157P were respectively approved by the NRC staff in March 1999 (Reference 1) and December 2001 (Reference 2).

The plant-specific basis for the proposed uprate is provided in Cameron Engineering Report ER-477, Revision 5, "Bounding Uncertainty Analysis for Thermal Power Determination at Vogtle Electric Generating Plant [VEGP] Using LEFM √+ System," dated May 2007 (Reference 14).

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

The instrumentation for measuring feedwater flow rate typically is a venturi. This device generates a differential pressure proportional to the feedwater velocity in the pipe. Due to the high cost of calibration of the venturi and the need to improve flow instrumentation measurement uncertainty, the nuclear industry assessed other flow measurement techniques and found the LEFM √ and the LEFM √+ system UFMs to be a viable alternative. Both these systems use the transit time methodology to measure fluid velocity. The basis of the transit time methodology to measure fluid velocity and temperature is that ultrasonic pulses transmitted into a fluid stream travel faster in the direction of the fluid flow than opposite the flow. The difference in the upstream and downstream traversing times of the ultrasonic pulses is proportional to the fluid velocity in the pipe and the temperature is determined using a pre-established correlation between the mean propagation velocity of the ultrasound pulses in the fluid and the fluid pressure. The mean fluid density may be obtained using the measured pressure and the derived mean fluid temperature as an input to a table of thermodynamic properties of water.

Both UFMs use multiple diagonal acoustic paths, instead of a single diagonal path, so that velocities measured along each path can be numerically integrated over the pipe cross section to determine the average fluid velocity in the pipe. This fluid velocity is multiplied by a velocity profile correction factor, the pipe cross section area, and the fluid density to determine the feedwater mass flow rate in the piping. The velocity profile correction factor is derived from calibration testing of the LEFM in a plant-specific piping model at a calibration laboratory.

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

Currently, the instrumentation used for measuring FW flow rate at Vogtle 1 and 2 is a venturi flow element. The licensee will install the Caldon LEMF √+ ultrasonic FW flow element into Vogtle 1 and 2 to reduce the uncertainty in the FW flow measurement. The licensee stated that this reduced uncertainty, in combination with other uncertainties, will result in an overall power level measurement uncertainty of 0.3% reactor thermal power (RTP). The remaining margin of 1.7% RTP forms the basis for the proposed MUR power uprate of 1.7% RTP.

The NRC staff's review in the area of instrumentation and controls covers the proposed plant-specific implementation of the feedwater flow measurement technique and the power increase gained as a result of implementing this technique in accordance with the guidelines (A through H) provided in Section I of Attachment 1 to RIS 2002-03. The NRC staff's review was conducted to confirm that the licensee's implementation of the proposed feedwater flow measurement device was consistent with the NRC staff-approved Caldon Topical Report ER-80P and ER-157P and adequately addressed the four additional requirements listed in the NRC staff SE. The NRC staff also reviewed the power uncertainty calculations to ensure that (1) the conservatively proposed uncertainty value of 0.3% correctly accounted for all uncertainties due to power level instrumentation errors, and (2) the calculations met the relevant requirements of Appendix K to 10 CFR Part 50 as described in Section 3.1.1 of this SE.

Items A through C of Section I of Attachment 1 to RIS 2002-03

The licensee's submittals provided the following information regarding the LEFM √+ system feedwater flow measurement technique and its implementation in Vogtle 1 and 2.

The FW flow measurement system to be installed in each Vogtle unit is a Caldon LEFM √+ ultrasonic multi-path transit time flow meter as described in Caldon Topical Report ER- 157P, "Supplement to Topical Report ER-80P: Basis for a Power Uprate with the LEFM √™ or LEFM Check-Plus™ System," dated October 2001 and approved by the NRC on December 20, 2001.

Each LEFM √+ System at Vogtle 1 and 2 consists of one spool piece measurement section integrally welded into each unit's common feedwater header upstream of the four feedwater loops. The header length and elbow configuration is the same for each of the Vogtle units. A Caldon LEFM √+ electronic unit will also be installed in each turbine building and will contain an integral air conditioning unit to maintain an acceptable internal cabinet temperature.

As stated in the license amendment request (LAR), the Caldon LEFM √+ Systems will be permanently installed at Vogtle 1 and 2 in accordance with the requirements of Topical Reports ER-80P and ER-157P and approved VEGP procedures. The systems will determine FW parameters to be used for continuous thermal power calorimetric measurement and communicate these parameters to the Vogtle 1 and 2 Integrated Plant Computer (IPC) system for incorporation into the secondary calorimetric algorithm. Each system will incorporate self-verification features to ensure that the system continually operates within design basis uncertainty analysis. The Caldon LEFM √+ System will communicate with the IPC via an Ethernet digital communications interface, and FW data will be transmitted to the IPC via fiber optic cables and data converters. Dual data outputs from the LEFM √+ cabinet will provide redundancy and fault tolerance for IPC communications.

Based on the NRC staff's review of the licensee's submittals as reflected in the above discussion, the NRC staff finds that the licensee has sufficiently addressed the plant-specific implementation of the LEFM √+ UFM system topical report guidelines (References 7, 8 and 9), and that the licensee's description of the FW flow measurement technique and the MUR power uprate due to implementing this technique adequately addresses the guidance in Items A through C of Section I of Attachment 1 to RIS 2002-03, "Guidance on the Content of Measurement Uncertainty Recapture Power Uprate Applications."

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

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

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

The licensee's response stated that implementation of the MUR power uprate license amendment will include developing the necessary procedures and documents required for operation, maintenance, calibration, testing, and training with the new Caldon LEFM √+ UFM system. These procedures will incorporate Caldon's maintenance and calibration requirements for the LEFM √+ UFM system.

The licensee also stated that selected instrumentation and control personnel will be trained and qualified, per the licensee's instrumentation and controls training program, on the LEFM √+ System prior to performing maintenance or calibration on the system and training will be completed prior to Caldon LEFM √+ System commissioning.

Routine preventative maintenance (PM) activities for the Caldon LEFM √+ system will include, but not be limited to, physical inspections of system components, power supply checks, internal oscillator frequency verification, and calibration of instrumentation which provides input to the LEFM √+ electronics unit. Ultrasonic signal verification and alignment is performed automatically by the LEFM √+ System. Signal verification will be determined by review of signal quality measurements performed and displayed by the LEFM √+ System.

Plant instrumentation that affects the power calorimetric, including the Caldon LEFM √+ inputs, will be monitored by the licensee's System Engineering personnel. These same instruments are included in the licensee's PM program and/or the TS surveillance program for periodic calibration.

The LEFM √+ UFM system is assumed to be inoperable if one or more paths are lost. The proposed allowed outage time (AOT) in the Technical Requirements Manual (TRM) for operation at any power level in excess of the current licensed core power level (3565 MWt) with the LEFM √+ UFM system out of service is 48 hours provided steady-state conditions persist (i.e., no power changes in excess of 10%) throughout the 48-hour period.

For the LEFM √+ UFM system out-of-service condition or loss of plant computer communication, the 48-hour AOT in the TRM will start at the time of the failure and this failure will be annunciated in the control room. The licensee stated that the plant operating procedures will be revised to state that if the inoperable LEFM √+ UFM system is not restored to an operable status or the plant experiences a power change of greater than 10% during the 48-hour period, then the permitted maximum power level will be reduced to the current licensed core thermal power level of 3565 MWt.

Additionally, the licensee stated that a back-up calorimetric algorithm which receives input from alternate plant instruments for the calculation of FW mass flow rate (FW venturis and resistance temperature detectors (RTDs)) will be available if the LEFM √+ UFM system is out of service . Specifically, the total FW flow from the four venturis will be normalized to the Caldon LEFM √+ FW mass flow rate so that the alternate calorimetric closely matches the primary LEFM-based calorimetric.

In its response to the Request for Additional Information (RAI) dated December 21, 2007, the licensee further indicated that the venturis transmitter drift data showed the worst case drift for any one transmitter (eight transmitters installed per Unit with two installed on each of the four loops) to be on the order of 0.5% over the 18-month calibration interval. Even conservatively assuming all transmitters drifted by this magnitude in the same direction, the impact on the thermal power measurement over the 48-hour AOT has been calculated to be less than 0.1 MWt. Also, based on the calculated average drift value for all feedwater transmitters over the 18-month interval, the impact on the thermal power measurement over the AOT would be undetectable.

A main plant computer system failure will be treated as a loss of both the Caldon LEFM √+ UFM system and the ability to obtain a corrected calorimetric power using alternate plant instrumentation. Thus, operation at the MUR core power level of 3625.6 MWt may continue until the next required nuclear instrumentation heat balance adjustment which could be up to 24 hours. The IPC failure will result in reducing core thermal power to less than or equal to 3565 MWt as needed to support a manual calorimetric power calculation.

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

The licensee's response stated that Vogtle 1 and 2 currently has flow measurement venturis on the FW system. The FW system flow venturis instrumentation will serve as backup inputs to the secondary calorimetric to be used when the LEFM √+ UFM system is not available. The new LEFM √+ UFM system will be independent of the FW system venturis. Thus, operational and maintenance history associated with the venturis devices is not applicable to the new LEFM √+ UFM system.

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

The licensee's response stated that the total power calorimetric uncertainty using the Caldon LEFM √+ System is determined by evaluating the reactor thermal power sensitivity to deviations in the process parameters used to calculate reactor thermal power. Channel statistical allowances (CSA) calculations have been performed for the plant instrumentation which provide input to the calorimetric calculation. Uncertainties for the parameters that are not statistically independent are arithmetically summed to produce groups that are independent of each other, which can be statistically combined. Then, all independent parameters/groups that contribute to the power measurement uncertainty are combined using a square root of sum of squares (SRSS) approach to determine the overall power measurement uncertainty. The licensee further clarified that the basis for the Vogtle 1 and 2 reactor trip system (RTS) and engineered safety features actuation system (ESFAS) setpoint methodology is WCAP-11269, Revision 1, "Westinghouse Setpoint Methodology for Protection Systems - Vogtle Station," November 1986. This method was used for the setpoints for the initial licensing of Vogtle 1 and 2 and is referenced in TS Bases B3.3.1 and B3.3.2.

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

The licensee's response stated that Criterion 4 does not apply to Vogtle 1 and 2. The calibration meter factors for the Caldon measurement sections were established by weigh tank tests performed at Alden Research Laboratory during October 30 through November 3, 2006. These tests were performed using a full scale piping model of the Vogtle 1 and 2 feedwater header

hydraulic geometry and tests in a straight pipe. The same Alden test model was used for testing of Vogtle 1 and 2 measurement sections. An Alden data report for these tests and Caldon uncertainty reports evaluating the test data were provided to the licensee. The calibration meter factor and the uncertainty in the calibration factor used for the Caldon LEFM √+ system at Vogtle 1 and 2 are based on these reports. The site-specific uncertainty analysis documents these analyses, and this documentation will be maintained as part of the technical basis for the Vogtle 1 and 2 MUR per Quality Assurance (QA) record retention requirements.

The licensee also stated that final acceptance of the Vogtle 1 and 2 specific uncertainty analyses will occur after the completion of the commissioning process. The commissioning process verifies that in-situ test data is bounded by the calibration test data (See Appendix F of ER-80P). This step provides final positive confirmation that actual performance in the field meets the uncertainty bounds established for the instrumentation. Final commissioning of the Caldon LEFM √+ Systems is expected to be completed following the Unit 1 refueling outage in spring 2008 and the Unit 2 refueling outage in fall 2008.

Based on the above listed responses and commitment provided by the licensee to the four criteria, the NRC staff finds that the licensee has fully addressed the four criteria specified in the NRC staff's SE of Topical Reports ER-80P and ER-157P and, therefore, has adequately addressed the guidance in Items D, G, and H of Section I of Attachment 1 to RIS 2002-03.

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

To address Item E of RIS 2002-03, the licensee provided a summary of the Vogtle 1 and 2 core thermal power measurement uncertainty in a table format listing uncertainty values from Cameron Engineering Report ER-477 Rev 5, and ER-586 Rev 1, which provide a detailed calculation of the uncertainties for Unit 1 and Unit 2, respectively. The licensee stated that the values in the uncertainty column of the table and the total power uncertainty determination are bounding values. After auditing of ER-477 and ER-586, the NRC staff found that the calculations determined individual measurement uncertainties of all parameters contributing to the core thermal power measurement uncertainty and those uncertainties were then combined using SRSS methodology, as described in Regulatory Guide (RG) 1.105 and Instrument Society of America S67.04.

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

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

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

i) Maintaining Calibration

Calibration and maintenance for the Caldon LEFM √+ System hardware and instrumentation will

be performed using procedures based on the appropriate Caldon LEFM √+ technical manuals. The other calorimetric process instrumentation and computer points are maintained and periodically calibrated in accordance with approved procedures. Preventative maintenance tasks are periodically performed on the plant computer system and support systems to ensure continued reliability. All work will be planned and executed in accordance with established Vogtle 1 and 2 work control processes and procedures.

ii) Controlling Hardware and Software Configuration

The LEFM √+ System is designed and manufactured in accordance with Caldon's 10 CFR, Part 50, Appendix B, Quality Assurance Program and its Verification and Validation (V&V) program. Caldon's V&V program fulfills the requirements of American National Standards Institute (ANSI)/Institute of Electrical and Electronics Engineers (IEEE)-American Nuclear Society (ANS) Std. 7-4.3.2, 1993, "IEEE Standard Criteria for Digital Computers in Safety Systems of Nuclear Power Generating Stations," Annex E, and American Society of Mechanical Engineers (ASME) Nuclear Quality Assurance (NQA) NQA-2a- 1990, "Quality Assurance Requirements for Nuclear Facility Applications." In addition, the program is consistent with guidance for software V&V in Electric Power Research Institute (EPRI) TR-103291s, "Handbook for Verification and Validation of Digital Systems," December 1994. After installation, the LEFM CheckPlus System software configuration will be maintained using existing procedures and processes. The plant computer software configuration is maintained in accordance with the Vogtle 1 and 2 change control process which includes verification and validation of changes to software configuration. Configuration of the hardware associated with the LEFM √+ System and the calorimetric process instrumentation will be maintained in accordance with Vogtle 1 and 2 configuration control processes.

iii) Performing Corrective Actions

The LEFM √+ System performance will be monitored and documented per the requirements of the Vogtle 1 and 2 system health monitoring program. The system diagnostic information will be trended for identification of conditions that are adverse to quality. Such conditions will be documented in the Vogtle 1 and 2 corrective action program and needed actions will be controlled by the Vogtle 1 and 2 work control process.

iv) Reporting Deficiencies to the Manufacturer

Conditions found to be adverse to quality will be documented per the Vogtle 1 and 2 corrective action program and reported to the vendor as needed to support corrective action.

v) Receiving and Addressing Manufacturer Deficiency Reports

Manufacturer's deficiency reports will be documented in the Vogtle 1 and 2 corrective action program and actions will be controlled by the Vogtle 1 and 2 work control process.

Based on the information provided by the licensee, the NRC staff finds that the licensee has addressed the calibration and maintenance aspects of the LEFM √+ UFM system and all other instruments affecting power calorimetric and, thus, complied with the guidance in item F of

Section I of Attachment 1 to RIS 2002-03.

Change of TS Table 3.3.1-1, P-9 Nominal Setpoint and the Allowable Value

Vogtle 1 and 2 are designed for 50% load rejection capability without a reactor trip, and P-9 interlock, at the current nominal setpoint of 50% RTP, prevents direct reactor trip below 50% RTP. Also before power operation, it was demonstrated that P-9 permissive will not result in the opening of the pressurizer power operated relief valves (PORVs). However, because of the slightly lower steam dump system capacity at the MUR power uprate conditions, the P-9 setpoint analyses were revised. Based on this analysis, the licensee proposed to revise the P-9 nominal setpoint in TS Table 3.3.1-1 from 50% to 40% RTP, and the Allowable Value from 52.3% to 40.6% RTP. The licensee stated that for normal operation with all normal control systems assumed operational, the pressurizer PORVs were not challenged with P-9 at the current nominal setpoint of 50% RTP. However, for certain single failures in the steam dump system and pressurizer spray flow, the pressurizer PORVs were challenged. With the reduced P-9 nominal setpoint of 40% RTP, the PORVs will not be challenged.

In response to NRC staff's request for additional information, the licensee stated that the P-9 setpoint and allowable value do not protect any safety limit. They do, however, enable reactor trip on turbine trip when power is above the P-9 setpoint to minimize the transient on the reactor and prevent challenges to the PORVs. Therefore, P-9 falls into the category of a limiting safety system setpoint (LSSS) that does not protect a safety limit but does perform a significant safety function. The Allowable Value was determined consistent with the Allowable Values for overtemperature differential temperature (OTDT) and overpower differential temperature (OPDT) approved by the NRC in Vogtle 1 and 2 Amendment Nos. 128 and 106, respectively, issued June 4, 2003.

Based on the licensee's response, the NRC staff finds that the licensee followed NRC staff guidelines in RIS-2006-17, "NRC Staff Position on the Requirements of 10 CFR 50.36, "Technical Specifications," Regarding Limiting Safety System Settings During Periodic Testing and Calibration of Instrument Channels," for the methodology used for establishing the limiting setpoint and limiting acceptable values for P-9 and, therefore, the proposed change to P-9 setpoint and setpoint Allowable Value is acceptable.

3.1.3 Summary

The NRC staff reviewed the licensee's proposed plant-specific implementation of the FW flow measurement device and the power uncertainty calculations and determined that the licensee's proposed amendment is consistent with the NRC staff's approved Topical Reports ER-80P, and its supplement ER-157P. The NRC staff has also determined that the licensee adequately accounted for all instrumentation uncertainties in the reactor thermal power measurement uncertainty calculations and demonstrated that the calculations meet the relevant requirements of 10 CFR Part 50, Appendix K as described in Section 3.1.1 of this SE. Therefore, the NRC staff finds the instrumentation and controls aspect of the proposed MUR power uprate acceptable.

Therefore, there is reasonable assurance that when the licensee implements FW flow measurement to determine plant thermal power with the LEFM √+ system ultrasonic flow meter in accordance with the NRC staff's safety evaluation basis for ER-80P and ER-157P, the plant

thermal power measurement uncertainty of Vogtle 1 and 2 will be limited to ±0.3% of the reactor thermal power and, therefore, can support the proposed 1.7% thermal power uprate proposed by the licensee for Vogtle 1 and 2.

3.2 Reactor Systems

3.2.1 Regulatory Evaluation

Appendix K and Section 50.46 of 10 CFR Part 50 require licensees to base their LOCA analysis on an assumed power level of at least 102% of the licensed thermal power level to account for power measurement uncertainty. On June 1, 2000, the NRC published a final rule (65 FR 34913) that allows licensees to justify a smaller margin for power measurement uncertainty. Licensees may apply the reduced margin to operate the plant at a level higher than the previously licensed power. The licensee proposed to use a Caldon LEFM CheckPlus system to decrease the uncertainty in the measurement of feedwater flow, thereby, decreasing the power level measurement uncertainty from 2.0% to 0.3%.

The licensee stated the LAR is consistent with the guidelines in NRC RIS 2002-03, “Guidance on the Content of Measurement Uncertainty Recapture Power Uprate Applications” (Reference 5).

In the licensee’s original application dated August 28, 2007 (Reference 6), the licensee identified the NRC staff’s original evaluation of the Caldon Ultrasonic Flow Meter (UFM) in References 7, 8, and 9. Generic hydraulic issues were identified by the licensees when installing and using UFMs following the NRC staff’s evaluation.

Consequently, the NRC staff re-evaluated the hydraulic aspects of the Check and CheckPlus UFMs. This re-evaluation is reported in Reference 10, which provided the following information:

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

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

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

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[1] This does not apply if the CheckPlus is located too close to a flow perturbation such as an elbow. The condition exists in the Vogtle application where the entrance to the CheckPlus spool piece is approximately one pipe diameter downstream of the exit from an elbow.

[2] This conclusion does not apply if the flow profile consists of multiple individual flow paths such as may exist immediately downstream of a tubular flow straightener.

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

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

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

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

The NRC letter (Reference 10) applicability addressing the evaluation of the hydraulic aspects of the Caldon flowmeters to the licensee's application request, dated August 28, 2007 (Reference 6), is covered in Section 3.2.2.A, below.

3.2.2 Technical Evaluation

A. FEEDWATER FLOW MEASUREMENT DEVICES-CALDON LEFM CHECKPLUS SYSTEM

A.1 Installation

RIS 2002-03 (Reference 5) describes the CheckPlus UFMs to be installed in accordance with the requirements of Topical Reports ER-80P and ER-157P (References 7 through 9). Each CheckPlus is to be installed in a 31-foot 10-inch vertical section of 36-inch diameter feedwater pipe with the entrance to the CheckPlus spool piece located 3.5 feet downstream of the exit from a 90-degree elbow. Feedwater from feedwater heater A flows through a 24- to 36-inch expansion and feedwater from feedwater heater B enters the 36-inch pipe via a 36 X 36 X 24-inch Tee that is 17 feet 6 inches upstream of the 90-degree elbow. Complete mixing and removal of swirl may not occur between the feedwater junctions and the CheckPlus. Thus, the CheckPlus system may operate with swirl and some thermal stratification if there is a difference in temperature exiting the feedwater heaters, and it will operate with a poorly-developed flow profile. These aspects are addressed in Section 3.2.2.A.4, below.

A.2 CheckPlus Inoperability

To operate above the presently licensed power of 3565 MWt, the licensee proposes that the CheckPlus cannot have been out-of-service for more than 48 hours and there cannot have been any power changes that exceeded 10% during the 48 hours. Without an operational CheckPlus system, power will be monitored using the existing four venturis that have been recalibrated to agree with CheckPlus before the failure. The licensee justifies this operation on the basis that

[3] See Footnote 1.

there is not a significant uncertainty associated with using venturis for 48 hours as long as the plant is essentially operated in a steady state condition and, if a power change in excess of 10% should occur during the 48 hours, then the plant thermal power will be reduced to the presently licensed 3565 MWt and the flow rate will be determined using the present, pre-uprate venturi calibration. Stated differently, after 48 hours without an operable CheckPlus, or if core thermal power changes by more than 10% while the CheckPlus is inoperable, the plant will be operated as though the CheckPlus was never installed and the power uprate was not in effect. These actions are covered by the proposed technical requirement (TR) 13.3.7 in the licensee's letter dated January 30, 2008 (Reference 3). It is necessary for the licensee to derate to the current license power level (3565 MWt), if the Caldon LEFM CheckPlus flowmeter is out of service beyond the AOT.

A.3 Transducer Replacement

The Reference 7 qualification to establish the effect of transducer replacement on the Check and CheckPlus system uncertainties has been addressed in References 11 and 12. A number of tests were conducted in which the transducers were removed and replaced for each test. Each of the tests consisted of a statistically meaningful number of individual determinations of the calibration factor. The calibration factors, and uncertainty associated with each calibration factor, were provided. Calibration factor variation was shown to be bounded by changes in the fourth significant figure. The bounding uncertainty due to transducer installation variability was incorporated into the overall CheckPlus uncertainty calculation in Reference 13. The NRC staff finds that transducer installation variability has been acceptably addressed.

A.4 CheckPlus Calibration

CheckPlus calibration was accomplished at ARL. Reference 13 covers the ARL test configuration. The NRC staff compared the test configuration to drawings of the Vogtle 1 and 2 system and noted the following:

- The general test configuration is rotated 90 degrees from the plant configuration so that the test CheckPlus is in a horizontal pipe segment in contrast to the vertical plant orientation. The effect of gravity on the flow characteristics is judged negligible for the flow rates used in the tests.
- Axial pipe dimensions of the test configuration upstream of the tee that is upstream of the CheckPlus are substantially shorter than in the plant. This will increase flow profile distortion and swirl in contrast to the plant application.
- The axial dimension of the test pipe downstream of the CheckPlus before encountering an elbow is shorter than in the plant. This is judged to have a negligible effect on CheckPlus test results.
- The test configuration from the upstream Tee to downstream of the CheckPlus is an accurate representation of the plant, including provision of detail such as a small vent line located upstream of the upstream elbow.

Calibration tests were conducted with the Vogtle 1 and 2 Checkplus UFMs with flow rate splits ranging from all flow through the side-entering tee to all flow entering the straight section of the tee. Additional tests were conducted with a flow straightener upstream of the side-entering tee. Each

test consisted of a statistically meaningful number of individual tests that were parametric in flow rate with multiple CheckPlus flow rate indications per individual test. Flatness ratios, swirl, and flow rate behavior were generally consistent with the Seabrook calibrations discussed in Reference 10 in that there was little variation of calibration factor with changes limited to changes in the fourth significant figure. Swirl was minimal for the tests using straight pipe and with equal flow rates from each tee connection. Unequal flow rates are of limited concern since they will generally correspond to a reduced thermal power and CheckPlus accuracy will have little impact on the safety aspects of interest here. As was the case for Seabrook, maximum flow rates were less than maximum flow rate in the plants and test temperature was room temperature in contrast to a plant feedwater temperature of 445 ^{o}F.

A correlation of flatness ratio with calibration factor was not used to extrapolate the test results to plant operating conditions as was used for some previous CheckPlus applications and was discussed in Reference 10 because the close distance downstream of the elbow produced a distorted flow profile that could not be reasonably modeled via flatness ratio. Consequently, a straight line fit to a function of Reynolds number was used to extrapolate the test data. This increased the test correction factor by about 0.06%. The resulting plant correction factor was assigned a 0.08% uncertainty due to the Reynolds number extrapolation.

Operation with unequal heating from the feedwater heaters could introduce thermal stratification in feedwater passing through the CheckPlus. The NRC letter regarding the evaluation of the hydraulic aspects of the Caldon flowmeters (Reference 10) addressed this situation by noting that, “where feedwater enters a common header upstream of the UFM, there is a possibility that temperature will not be uniform at the UFM location. In some cases with off-normal feedwater heater operation, a temperature difference of as much as 30 °F or 40 °F may occur.” The CheckPlus, by measuring transit time in both directions in 8 paths, will provide the average velocity of sound along those eight paths. The sound velocity will, in turn, provide temperatures. This capability led to the Reference 10 statement “that CheckPlus would continue to measure bulk average feedwater flowrate within its design basis accuracy because the sound velocity is integrated over the pipe cross section.” The use of CheckPlus to determine temperature and the associated temperature uncertainty is addressed in Appendix A.2 of Cameron Engineering Report ER-477 (Reference 14).

A.5 Caldon Topical Reports ER-80P and ER-157P Safety Evaluation Criterion

The NRC staff reviewed and approved Caldon Topical Reports ER-80P and ER-157P related to the LEFM Check or LEFM CheckPlus Systems. In approving the Caldon Topical Reports, the NRC staff established four criteria to be satisfied by each licensee as follows:

Criterion 1

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

Licensee Response to Criterion 1

The licensee will develop the necessary procedures and documents required for operation, maintenance, calibration, testing, and training at the MUR power level with the new Caldon LEFM CheckPlus system in accordance with Caldon's requirements prior to declaring the system operational and raising the core power level above 3565 MWt. The incorporation of the Caldon maintenance calibration requirements into Vogtle 1 and 2 procedures and continued adherence to these requirements will assure that the Caldon LEFM CheckPlus System is properly maintained and calibrated.

Plant procedures for calibration and maintenance will be performed using site-specific procedures developed using Caldon technical manuals. Selected Instrumentation and Controls personnel in the Maintenance Department will be trained and qualified per the Vogtle 1 and 2 Institute for Nuclear Power Operations (INPO) accredited training program before maintenance or calibration is performed and prior to increasing the power level above 3565 MWt. Initially, Caldon will provide formal training to Vogtle 1 and 2 personnel. Plant instrumentation that affects the power calorimetric, including Caldon LEFM Checkplus inputs, will be monitored by the Vogtle 1 and 2 System Engineering personnel. These instruments are included in the Vogtle 1 and 2 preventive maintenance program and/or TS surveillance program for periodic calibration.

The MUR power uprate is based on the use of the Caldon LEFM CheckPlus flowmeter. It is necessary for the licensee to derate to the current license power level (3565 MWt), if the Caldon LEFM CheckPlus flowmeter is out of service beyond the AOT. After the AOT has expired and without an operable CheckPlus, or if core thermal power changes by more than 10% while the CheckPlus is inoperable, the plant will be operated as though the CheckPlus was never installed and the power uprate was not in effect.

Criterion 2

"For plants that currently have LEFMs installed, provide an evaluation of the operational and maintenance history of the installed instrumentation and confirmation that the installed instrumentation is representative of the LEFM system and bounds the analysis and assumptions set forth in Topical Report ER-80P."

Licensee Response to Criterion 2

This criterion is not applicable to Vogtle 1 and 2. The licensee currently uses venturis to measure feedwater flow to support the secondary calorimetric power measurements.

Criterion 3

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

Licensee Response to Criterion 3

The total power calorimetric accuracy using the Caldon LEFM CheckPlus System is determined by evaluating the reactor thermal power sensitivity to deviations in the process parameters used to calculate reactor thermal power. Channel Statistical Allowances (CSA) calculations have been performed for the plant instrumentation which provides input to the calorimetric calculation. Uncertainties for the parameters that are not statistically independent are arithmetically summed to produce groups that are independent of each other, which can be statistically combined, Then, all independent parameters/groups that contribute to the power measurement uncertainty are combined using an SRSS approach to determine the overall power measurement uncertainty.

Criterion 4

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

Licensee Response to Criterion 4

The calibration meter factors for the Caldon measurement sections were established by weigh tanks tests performed at ARL during October 30 through November 2, 2006, in accordance with the National Institute of Standards and Technology Standards. These tests were performed using a full-scale model of the Vogtle 1 and 2 feedwater header hydraulic geometry and tests in a straight pipe. The same ARL test model was used for testing both the Vogtle 1 and 2 measurement sections. An ARL data report for these tests and Caldon uncertainty reports evaluating the test data were provided to the licensee. The calibration meter factor and the uncertainty in the calibration factor used for the Caldon LEFM CheckPlus system at Vogtle 1 and 2 are based on these reports. The site-specific uncertainty analysis documents these analyses, and this documentation will be maintained as part of the technical basis for the Vogtle 1 and 2 MUR per quality assurance (QA) record retention requirements. Final acceptance of the Vogtle 1 and 2 specific uncertainty analysis will occur after the completion of the commission process. The commissioning process verifies that in-situ test data is bounded by the calibration test data. This step provides final positive confirmation that actual performance in the field meets the uncertainty bounds established for the instrumentation. Final commissioning of the Caldon LEFM CheckPlus system is expected to be completed following the Vogtle 1 refueling outage in the spring 2007 and the Vogtle 2 refueling outage in fall 2008.

Based on its review of the licensee's responses, the NRC staff determined that the licensee has addressed the four criteria specified in the NRC staff's evaluation of Topical Reports ER-80P and ER-175P, and it is consistent with the guidelines of RIS 2002-03 (Reference 5).

B. NSSS PARAMETERS

The NSSS design parameters provide the reactor coolant system (RCS) and secondary system conditions (pressures, temperatures, and flow) that are used as the basis for the design transients and for systems, components, accidents and transient analyses and evaluations. The parameters are established using conservative assumptions to provide bounding conditions to be used in the NSSS analyses and evaluations. Table 6.6.2-1 of Enclosure 10 (WCAP-16736-NP) of Reference 3 lists the NSSS design operating parameters in detail for various cases analyzed by the licensee. The major input parameters and assumptions from Table 6.6.2-1 used in the analyses are as follows:

Parameter	Case A	Case B	Case C	Case D
Total RCS Power (MWt)	3,653	3,653	3,653	3,653
Core Thermal Power (MWt)	3,636	3,636	3,636	3,636
Reactor Coolant Pressure (psia)	2,250	2,250	2,250	2,250
SG A/B Tube Plugging (%)	0	10	0	10
Core outlet/vessel outlet (ºF)	609.2/603.8	609.2/603.8	625.7/620.6	625.7/620.6
Core Bypass, percent	8.4	8.4	8.4	8.4
Vessel/core Inlet (ºF)	537.6	537.6	556.2	556.2
Vessel Average (ºF)	570.7	570.7	588.4	588.4
RCS Flow (E6 lb/hr)	143.1	143.1	139.5	139.5
Thermal Design Loop Flow (gpm)	93,600	93,600	93,600	93,600
Steam Outlet Temperature (ºF)	520.6	518.1	539.8	537.3
Steam Outlet Flow (E6 lb/hr)	16.22	16.22	16.32	16.31
Steam Outlet Pressure (psia)	817	799	961	941
Feedwater Temperature (ºF)	448.7	448.7	448.7	448.7

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

In the submittal, the licensee evaluated the current Updated Final Safety Analysis Report (UFSAR) Chapter 15 LOCA and non-LOCA transients and accidents in support of the Vogtle 1 and 2, 1.7% power uprate. The licensee used NRC-approved computer codes and methodologies for each accident and transient analysis.

The results of the NRC staff's review are summarized in the following table.

Accident/Transient	Analyzed Core Power Level	Analysis of Record Bounds MUR Uprate	NRC Staff Conclusion/ Discussion
Large and Small Break LOCA	102%	Yes	Acceptable
Post-LOCA Long Term Cooling	102%	Yes	Acceptable
Hot Leg Switchover	102%	Yes	Acceptable

Accident/Transient	Analyzed Core Power Level	Analysis of Record Bounds MUR Uprate	NRC Staff Conclusion/ Discussion
Steamline Break at Hot Full Power	102%	Yes	Acceptable See Section 3.2.2.C.2
Steamline Break at Hot Zero Power	Hot zero power	Yes	Acceptable
Loss of Non-Emergency AC Power to Plant Auxiliaries	102%	Yes	Acceptable
Anticipated Transients Without Scram	96%	Yes	Acceptable See Section 3.2.2.C.1
Uncontrolled Rod Control Cluster Assembly (RCCA) Withdrawal Bank at Power	100%	Yes	Acceptable See Section 3.2.2.C.2
RCCA Misalignment	100%	Yes	Acceptable See Section 3.2.2.C.2
Partial and Complete Loss of Forced Reactor Coolant Flow	100%	Yes	Acceptable See 3.2.2.C.2
Startup of an Inactive Reactor Coolant Pump	72%	Yes	Acceptable See 3.2.2.C.2
Loss of Normal Feedwater Flow	102%	Yes	Acceptable
Feedwater System Pipe Break	102%	Yes	Acceptable
Steam Generator Tube Rupture	102%	Yes	Acceptable
Reactor Coolant Pump Shaft Seizure	102%	Yes	Acceptable
Uncontrolled RCCA Bank Withdrawal from Subcritical	Hot zero power	Yes	Acceptable
RCCA Ejection Accident	102%	Yes	Acceptable
RCCA Ejection Accident	Hot zero power	Yes	Acceptable
Inadvertent Operation of ECCS During Power Operation	102%	Yes	Acceptable
Steam System Piping Failure, and Turbine Trip, and Steam Generator Tube Rupture	102%	Yes	Acceptable See Section 3.2.2.C.2
Inadvertent Opening of a Pressurizer Safety or Relief Valve	100%	Yes	Acceptable See Section 3.3.2
Feedwater System Malfunctions that Result in an Increase in Feedwater Flow	100%	Yes	Acceptable See Section 3.2.2.C.2

Accident/Transient	Analyzed Core Power Level	Analysis of Record Bounds MUR Uprate	NRC Staff Conclusion/ Discussion
Excessive Increase in Secondary Steam Flow	100%	Yes	Acceptable See Section 3.2.2.C.2
Chemical and Volume Control System Malfunction that Increases Reactor Coolant Inventory	0/100	Yes	Acceptable See Section 3.2.2.C.3
Station Blackout	102%	Yes	Acceptable See Section 3.2.2.C.4

C.1 Anticipated Transients Without Scram

The licensee states that Vogtle 1 and 2 comply with the requirements set forth in 10 CFR 50.62(b), that all Westinghouse designed PWRs must install anticipated transients without Scram (ATWS) mitigation system actuation circuitry (AMSAC). In letter NS-TMA-2182, dated December 30, 1979, from T.M. Anderson (Westinghouse) to Dr. Hanauer (NRC), “ATWS Submittal,” Westinghouse provided a generic ATWS analyses based on guidelines in NUREG-0460, “Anticipated Transients Without Scram for Light Water Reactors.” These analyses were performed for various Condition II events considering various Wesinghouse PWR configurations at that time. For Vogtle 1 and 2, the applicable Westinghouse ATWS generic analyses were for a 4-loop PWR with Model F steam generators and a core power of 3427 MWt. The licensee states that Vogtle 1 and 2 continue to operate with Model F steam generators and utilize the Westinghouse AMSAC Logic 2 for AMSAC actuation on low main feedwater flow. The licensee performed an evaluation to determine the effect of the power uprate on the ATWS analysis. The uprate evaluation assumed a 6.3% higher reactor power and a 27% high auxiliary feedwater flow rate than that assumed in the generic limiting loss of load ATWS event applicable to Vogtle 1 and 2. The higher reactor power and auxiliary feedwater flow result in a combined overall peak RCS pressure penalty (increase) of 107 psia relative to the peak pressure of 2,974 psia from the generic analysis. The penalty added to the generic analysis applicable to Vogtle 1 and 2 results in a peak pressure of 3,081 psia. This is below the ATWS peak RCS pressure limit of 3,200 psia.

Based on the acceptability of the original Westinghouse ATWS analyses and the margin determined by the licensee, the NRC staff finds the plant response to an ATWS at Vogtle 1 and 2 acceptable for the proposed MUR uprate.

C.2 Departure from Nucleate Boiling (DNB)-Related Non-LOCA Events

The statistical departure from nucleate boiling methods are based on WCAP-11397-P-A, “Revised Thermal Design Procedure,” (RTDP) which describes the procedure for predicting the departure from nucleate boiling ratio (DNBR) design limit in Westinghouse PWRs for use in analyzing DNB-related non-LOCA events. With the RTDP methodology, uncertainties in plant operating parameters, nuclear and thermal parameters, fuel fabrication parameters, computer codes and the DNB correlation predictions are combined statistically (rather than directly or

deterministically) to obtain the overall DNB uncertainty factor. The factor is used to define the design limit DNBR that satisfies the 95/95 DNB design criterion. The MUR power uprate can potentially impact a number of areas in the non-LOCA safety analysis. These areas include reload-related inputs as used in the NRC-approved Westinghouse reload methodology, the protection system setpoints and the initial condition uncertainties. The licensee's evaluation assumed that the reload-related inputs will not be impacted. In an RAI response dated December 21, 2007 (Reference 4), the licensee stated that the assumption was based on an evaluation of representative uprated Vogtle 1 and 2 cores for impacts on the reload-related inputs. This assumption will be verified as part of the standard Westinghouse reload methodology that is performed for each reload.

The licensee proposed not to reanalyze the non-LOCA events for the MUR power uprate provided that the DNBR margin was quantified and allocated to offset the effect of the 1.7% power increase. DNBR calculations were performed using the VIPRE-W code to address the increase in the nominal heat flux associated with the MUR power uprate and to determine the amount of margin that needed to be allocated. The licensee states that DNB margin is allocated specifically to address the 1.7% power uprate to ensure that the core thermal DNB limits do not change and are not violated. In Tables 7.10.4-1 and 7.10.4-2 of Enclosure 10 of Reference 3 (WCAP-16736-NP), the licensee shows that for events protected by the overtemperature-delta temperature trip, the total DNBR penalty will increase from 8.1 and 7.9 for the typical cell and thimble cell, respectively, to 11.6 and 11.5 for the uprated conditions. In the RAI response dated December 21, 2007, the licensee states a maximum of 3.6% (thimble cell) of the available DNBR margin has been allocated for the 1.7% power increase, while maintaining the same plant operating limits and trip setpoints. Currently, a higher DNBR limit is used for the safety analysis than is required to meet the 95/95 DNB design criterion. As such, the licensee states that the current DNB design basis continues to be met and the UFSAR conclusions remain valid. The DNB margin that is allocated to the uprate may not be credited for any other purpose. Any future license amendment request must specify the DNB margin allocated for the uprate. The following events utilized the DNB margin evaluation:

Feedwater System Malfunctions that Result in an Increase in Feedwater Flow

The licensee states that the limiting licensing basis case is the automatic rod control case, which has 34% margin to the DNBR limit. The calculations confirmed that the DNB design basis continues to be met and the minimum DNBR remains valid.

Excessive Increase in Secondary Steam Flow

The licensee states that the limiting licensing basis case is the minimum feedback/automatic rod control case, which has more than 24% margin to the DNBR limit. The calculations confirmed that the DNB design basis continues to be met and the minimum DNBR remains valid.

Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power

The licensee states that the limiting licensing basis case is based on an initial core power level equal to 60% of the rated thermal power. The licensing basis analysis shows approximately 14% margin to the DNBR limit. The calculations confirmed that the DNB design basis continues to be

met and the minimum DNBR remains valid.

Inadvertent Opening of a Pressurizer Safety or Relief Valve

The licensing basis analysis shows there is more than 5% margin to the DNBR limit. The calculations confirmed that the DNB design basis continues to be met and the minimum DNBR remains valid.

Partial Loss of Forced Reactor Coolant Flow and Complete Loss of Forced Reactor Coolant Flow

A detailed DNB analysis using the VIPRE-W code was performed using the current licensing basis analysis statepoints and an increased nominal heat flux associated with the 1.7% power uprate. The calculations confirmed that the DNB design basis continues to be met and the minimum DNBR remains valid.

Rod Cluster Control Assembly Misalignment

An evaluation of the DNB design basis was performed and confirmed that the acceptance criteria continued to be met and the conclusion in the UFSAR remain valid. Additionally, the licensee cited a similar four-loop Westinghouse plant that performed sensitivity studies for a 5% increase (over the current Vogtle 1 and 2 3565 MWt) for the dropped rod events and found that the generic statepoints remain applicable for use at the uprated conditions.

Steam System Piping Failure, and Turbine Trip, and Steam Generator Tube Rupture

These events are currently analyzed at 102% power. However, the nominal heat pump has increased from 14 MWt to 17 MWt. The increased pump heat of 3 MWt results in an initial NSSS power of 3654 MWt, which is greater than the current analysis at 3651 MWt. The licensee states the NSSS power is not a significant contributor to the margin. Additionally, in an RAI response dated December 21, 2007, the licensee states that since DNBR margin was allocated to offset the 1.7% power uprate, the conclusion in the submittal remains valid.

Startup of an Inactive Reactor Coolant Pump (RCP) at an Incorrect Temperature

The current licensing basis for the startup of an inactive RCP at an incorrect temperature event is performed at part-power conditions. Vogtle 1 and 2 cannot operate with one reactor coolant pump (RCP) out of service during power operation because the TSs do not permit N-1 operation. However, the event has been historically included as part of the licensee's analyses, as such, the licensee included it as part of the MUR. The licensing basis analysis shows there is more than 6% margin to the DNBR limit. The calculations confirmed that the DNB design basis continues to be met and the minimum DNBR remains valid.

C.3 Chemical and Volume Control System Malfunction that Increases Reactor Coolant inventory

The chemical and volume control system (CVCS) can be used to add unborated water to the reactor coolant system (RCS). This addition may happen inadvertently because of operator error or a system malfunction, and cause an unwanted increase in reactivity and a decrease in

shutdown margin. The operator must stop this unplanned dilution before the shutdown margin is eliminated. The current licensing basis for an uncontrolled boron dilution is performed at all plant operating modes with offsite power available and with manual rod operation. The current licensing basis must demonstrate that at least 15 minutes in any operating mode (except in Mode 6, which allows 30 minutes) to terminate the RCS dilution before a complete loss of shutdown margin occurs. The analysis does not explicitly model an initial power level. Therefore, operator reaction time is unaffected. In addition to response time, the minimum DNBR must remain above the limit value and the peak RCS system pressure must remain below 110% of their design values. The licensee evaluated the inadvertent operation of the emergency core cooling system and found that the DNBR limit and RCS pressure limits were maintained. The evaluation confirmed that the licensing basis continues to be met for this event.

C.4 Station Blackout (SBO)

For an SBO, licensees are required by 10 CFR 50.63 to be capable of maintaining core cooling and containment integrity for a minimum of 4 hours. The licensee evaluated the impact of the power uprate on SBO capability. The design bases include capability to provide core cooling, ability to maintain adequate RCS inventory, ability to maintain appropriate containment integrity, effects of loss of ventilation, equipment environmental evaluation, access to plant area requirements, emergency lighting requirements, required operator actions, procedure interface requirements, and emergency diesel generator reliability program requirements. The condensate storage tanks volume for dedicated safety grade water that is required for an SBO is based on a reactor trip from 102% power. The licensee's evaluation of the SBO design criteria revealed that the design bases parameters are not adversely affected by the MUR power uprate and are bounded by the existing design bases and analyses.

C.5 SETPOINT CHANGE FOR REACTOR TRIP SYSTEM PERMISSIVE P-9

The P-9 permissive prevents a reactor trip following a turbine trip, provided that the turbine trip occurs when the plant is operating at a relatively low power level (i.e., at a power level below the P-9 setpoint). No credit is taken in the accident analyses for any reactor trips derived directly from any of the turbine trip signals.

The P-9 permissive enables a reactor trip, on turbine trip, when reactor power level rises above the P-9 interlock setpoint in at least 2 out of 4 neutron power range flux channels. The requirements that pertain to the P-9 interlock are included in TS 3.3.1, "Reactor Trip System Instrumentation." Section 50.36(c)(2)(ii)(c), Criterion 3, applies to the Reactor Trip System Instrumentation, since the Reactor Trip System Instrumentation is, "a structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier."

The current P-9 setpoint analysis was performed with a 2% uncertainty, which bounds the MUR power uprate. The Condition I limiting transient is a 50% load rejection. For MUR power uprate conditions, the steam dump capacity has been reduced from 33% of nominal steam flow at the current conditions to 31.8% of nominal steam flow. No other transients actuate the steam dump control system. The licensee performed a sensitivity study considering the reduced steam dump

control system capability that was consistent with the system failures of the current analysis. The licensee performed the analysis using the Vogtle 1 and 2 model of the NRC-approved LOFTRAN code. For normal plant operating conditions and for end-of-cycle T_{avg} coastdown maneuver conditions, the licensee analyzed a nominal case, in which all control systems are available and function as designed at both the low and high RCS full-power T_{avg} conditions. Additionally, eight sensitivity scenarios were performed at both the low and high RCS full-power T_{avg} conditions.

The licensee's acceptance criteria for the analysis are, with all NSSS control systems assumed operational and in the automatic mode of control, a turbine trip without a direct reactor trip at or below the P-9 setpoint should not challenge the pressurizer power-operated relief valve (PORV). This acceptance criterion will also address the NRC's position with respect to a stuck-open PORV, as specified in Item II.K.3.10 of NUREG-0737, "Clarification of TMI Action Plan Requirements." Item II.K.3.10 concerns the effect of certain anticipatory trip modifications upon the probability of a small-break LOCA resulting from a stuck-open PORV.

The licensee's analysis demonstrated that for normal plant operating conditions, the peak pressure was 2,317 psia (low-T_{avg}) and 2,331 psia (high-T_{avg}). The lowest setpoint for a pressurizier PORV is 2,350 psia. Therefore, no challenge would be made to the pressurizer PORVs. However, three sensitivity cases (three steam dump valves from bank 2 failing to actuate, three valves from bank 1 failing to actuate, and the failure to receive any pressurizer spray flow) challenged the pressurizer PORVs. To address this, the licensee proposed a reduced P-9 nominal setpoint of 40%, which will not challenge the pressurizer PORVs.

For the end-of-cycle T_{avg} coastdown maneuver conditions, the licensee's analysis demonstrated that the peak pressurizer pressure was 2,312 psia. Therefore, no challenge would be made to the pressurizer PORVs. However, for two sensitivity cases regarding the failure to receive any pressurizer spray flow and three steam dump valves from bank 1 failing to actuate with the rods in manual control challenged the pressurizer PORVs. To address this, the licensee proposed a reduced P-9 nominal setpoint of 40%, which will not challenge the pressurizer PORVs.

The NRC staff concludes that the licensee has demonstrated that their proposal, lowering the P-9 setpoint from 50% RTP to 40% RTP, would continue to satisfy the analysis acceptance criteria for the affected licensing basis events (e.g., the load rejection events), and Item II.K.3.10 of NUREG-0737, "Clarification of TMI Action Plan Requirements."

C.6 CODE CHANGE FROM THINC IV to VIPRE-W

The VIPRE-W code, which was used for the Vogtle 1 and 2 measurement uncertainty recapture power uprate (MUR-PU), is a configured QA version of the Westinghouse VIPRE-01 code that was approved in WCAP-14565-P-A (Reference 15). The NRC staff stated in the SE for WCAP-14565-P-A that VIPRE-01, as described in WCAP-14565-P-A, is acceptable for licensing calculations and may be used to replace THINC-IV and FACTRAN computer codes in Westinghouse refueling methodology provided the following conditions are met:

1. Selection of the appropriate critical heat flux (CHF) correlation, DNBR limit, engineered hot channel factors for enthalpy rise and other fuel-dependent parameters for a specific plant application should be justified with each submittal.

2. Reactor core boundary conditions determined using other computer codes are generally input into VIPRE for reactor transient analyses. These inputs include core inlet coolant flow and enthalpy, core average power, power shape and nuclear peaking factors. These inputs should be justified as conservative for each use of VIPRE.

3. The NRC staff's generic review of VIPRE (Reference 16) set requirements for use of new CHF correlations with VIPRE. Westinghouse has met these requirements for using the WRB-1, WRB-2 and WRB-2M correlations. The DNBR limit for WRB-1 and WRB-2 is 1.17. The WRB-2M correlation has a DNBR limit of 1.14. Use of other CHF correlations not currently included in VIPRE will require additional justification.

4. Westinghouse proposes to use the VIPRE code to evaluate fuel performance following postulated design-basis accidents, including beyond-CHF heat transfer conditions. These evaluations are necessary to evaluate the extent of core damage and to ensure that the core maintains a coolable geometry in the evaluation of certain accident scenarios. The NRC staff's generic review of VIPRE (Reference 16) did not extend to post CHF calculations. VIPRE does not model the time-dependent physical changes that may occur within the fuel rods at elevated temperatures. Westinghouse proposes to use conservative input in order to account for these effects. The NRC staff requires that appropriate justification be submitted with each usage of VIPRE in the post-CHF region to ensure that conservative results are obtained.

As required by the SE for WACP-14565-P-A, the licensee documented how these conditions were met in WCAP-16736-P and RAI response dated December 21, 2007, therefore the NRC staff finds that this use of VIPRE as a replacement for THINC-IV is consistent with the approved WACP-14565-P-A.

3.2.3 Summary

The proposed amendment is based on the use of the Caldon LEFM CheckPlus UFM system that would decrease the uncertainty in the feedwater flow rate, thereby decreasing the power level measurement uncertainty from 2.0% to 0.3%. The NRC staff reviewed the reactor systems and thermal-hydraulic aspects of the proposed license amendment request in support of implementation of a measurement uncertainty recapture.

The NRC staff finds that the hydraulic aspects of the Caldon LEFM CheckPlus UFM system have been accurately described in applicable documentation and that there is a firm theoretical and operational understanding of behavior. The NRC staff further finds that the calibration accomplished at ARL is appropriate for installation at Vogtle 1 and 2 and is acceptable. Therefore, the NRC staff has concluded, based on the considerations discussed above, that the proposed changes are acceptable with respect to the hydraulic aspects of the CheckPlus UFM when installed at Vogtle 1 and 2. During implementation, the licensee shall complete the activities associated with installation, certification, and long-term operation consistent with the descriptions of such activities provided to the NRC.

Based on the above evaluation, the NRC staff determined that the results of licensee's analyses related to these areas continue to meet the applicable acceptance criteria following

implementation of the power uprate. Most of the current analyses of record are based on 102% power that includes 2.0% measurement uncertainty. In these cases, the proposed MUR rated thermal power is bounded by the current analyses of record. The NRC staff reviewed those instances where the licensee's analyses of record were performed at the current licensed thermal power and determined that those analyses either (1) were bound by the analysis at 100% original licensed thermal power, or (2) were bound by a more limiting analysis performed. In those cases, the appropriate disposition has been noted. The DNB margin that is allocated to the uprate may not be credited for any other purpose. Any future license amendment request must specify the DNB margin allocated for the uprate.

The NRC staff reviewed the proposal to change from THINC-IV to VIPRE-W and finds that the licensee met all conditions associated with the generic NRC approval of WCAP-14565-P-A.

The MUR power uprate is based on the use of the Caldon LEFM CheckPlus flowmeter. It is necessary for the licensee to derate to the current license power level (3565 MWt), if the Caldon LEFM CheckPlus flowmeter is out of service beyond the AOT. After the AOT has expired and without an operable CheckPlus, or if core thermal power changes by more than 10% while the CheckPlus is inoperable, the plant will be operated as though the CheckPlus was never installed and the power uprate was not in effect.

3.3 Electrical Systems

3.3.1 Regulatory Evaluation

The licensee developed the LAR consistent with the guidelines in NRC RIS 2002-03, "Guidance on the Content of Measurement Uncertainty Recapture Uprate Applications."

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

General Design Criterion (GDC) 17, "Electric power systems," of Title 10 to the *Code of Federal Regulations* (10 CFR) Part 50, Appendix A requires that an onsite power system and an offsite electrical power system be provided with sufficient capacity and capability to permit functioning of structures, systems, and components important to safety.

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

Section 50.49, "Environmental Qualification of Electric Equipment important to Safety for Nuclear Power Plants," requires licensees to establish programs to qualify electric equipment important to safety.

3.3.2 Technical Evaluation

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

- AC Distribution System
- Power Block Equipment (Generator, Exciter, Transformers, Isolated-phase bus duct, Generator circuit breaker)

- Direct Current (DC) System
- Emergency Diesel Generators (EDGs)
- Switchyard
- Grid Stability
- SBO
- Equipment Qualification Program

3.3.2.1 AC Distribution System

The AC Distribution System is the source of power to nonsafety-related buses, and to safety-related emergency buses supplying the redundant engineered safety feature loads.

The AC Distribution system consists of the 13.8 kilovolt (kV), 4.16 kV, 480 volt (V), 277 V, 240 V, 208 V, and 120 V systems. As the current Vogtle 1 and 2 heater drain system pumping capability is marginal, the licensee has proposed replacing the associated pumps and motors to improve system performance. The existing heater drain pumps have 1500 horsepower (hp) motors whereas the new ones will have 1800 hp motors. Each motor will represent a load increase of 0.3 megavolt ampere (MVA), and there are two motors per unit. The heater drain pumps are powered from different non-Class 1E 4.16 kV buses. A discussion of the loading effects on the transformers is provided in Section 3.3.2.2.

Additionally, the licensee stated that the new Caldon electronics cabinet power supply will be connected to a non-Class 1E 120 V AC inverter with a rating of 3 kilovolt ampere (kVA). The load being added is 5 ampere (A) or correspondingly, 0.6 kVA, for a total loading of 0.8 kVA on the inverter, which remains within the current inverter rating of 3 kVA.

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

3.3.2.2 Power Block Equipment (Generator, Exciter, Transformers, Iso-Phase Bus Duct, Generator Circuit Breaker)

As a result of the power uprate, Vogtle 1 and 2 rated thermal power will increase to 3625.6 MWt from the previously analyzed core power level of 3565 MWt. Both generators are rated at 1350 MVA with a 1.0 power factor (pf). Currently, the generator outputs 1225.3 MWe at 0.91 pf. At uprated conditions, the pf is estimated to be 0.93, corresponding to an output of 1249.8 MWe. The generator electrical power output for the MUR power uprate falls within the range of operation on the Generator Reactive Capability Curve. Additionally, the increase in electrical output remains bounded by the design ratings of the generators. Although there will be an increase in cooling water temperature and flow rate in the generator stator water cooling system, the increase in temperature is bounded by the system design basis. In regards to the generator hydrogen system, there is no change in the hydrogen pressure or flow rate as a result of the power uprate. Furthermore, the turbine-generator carbon dioxide system purging requirements are unaffected by the power uprate. As a result, the generator is capable of operation at uprated conditions.

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

The Vogtle 1 and 2 reserve auxiliary transformers (RATs) feed Class 1E loads and non-Class 1E loads under certain operating conditions. As the heater drain pumps are changing from a 1500 hp motor to a 1800 hp motor, there will be additional load on the transformers. The heater drain pumps are normally fed from the unit auxiliary transformers (UATs) but if they are not available, they are fed from the RATs. The added load to each UAT is approximately 0.3 MVA, and the total loading (12.69 MVA) is less than the UAT rating of 20 MVA. The additional load added to each RAT is approximately 0.3 MVA, but the total loading (19.07 MVA) on each RAT remains less than the 25 MVA rating. Thus, the NRC staff finds that no changes to the RATs or UATs are needed.

The main generator voltage is stepped up to 230 kV by the Vogtle 1 main power transformer (MPT) and 500 kV by the Vogtle 2 MPT. Each MPT has three single phase units, each unit rated at 452 MVA. Each MPT has a three-phase rating of 1356 MVA. At uprated conditions, the maximum generator output is 1343 MVA (rated for 1350 MVA), which is below the rating of each MPT. Therefore, the NRC staff finds that each MPT is capable of operation at uprated conditions.

The generators can be separated from the system by disconnecting the high voltage switchyard breakers (230 kV for Vogtle 1 and 500 kV breaker for Vogtle 2). The licensee stated that these breakers are sized with ample margin, based on future and present transmission system conditions and will continue to have adequate margin after the MUR power uprate. The licensee evaluated the impact of the increase in load due to the new heater drain pumps and concluded that the short circuit fault level remains within existing limits. In addition, the steady-state voltages remain within acceptable limits. Based on this information, the NRC staff finds that no changes are needed to the breakers that disconnect the generators from the system. The NRC staff finds that the small increase in generator output (25 MWe) will not cause overloading of the generator circuit breaker, the iso-phase bus ducts, or the transformers. Therefore, the ratings of the Vogtle 1 and 2 UATs and RATs would not be impacted by MUR power uprate conditions.

3.3.2.3 DC System

The station 125 VDC system is comprised of batteries, battery chargers, and distribution equipment that supply 125 VDC power to station loads. The nuclear safety-related (Class 1E) portion of the DC System consists of 125 VDC batteries, battery chargers, inverters, and 125 VDC distribution panels. It provides the source of power for control, instrumentation, and DC motors for Class 1E and selected non-Class 1E electrical equipment.

The NRC staff reviewed the LAR and Vogtle 1 and 2 UFSAR and confirmed that there are no significant changes in DC system loads. Therefore, the NRC staff finds that the analyses for the DC system reasonably bound the MUR power uprate conditions.

3.3.2.4 Emergency Diesel Generators (EDGs)

The EDG system provides a safety-related source of AC power to sequentially energize and

restart loads necessary to shutdown the reactor safely and to maintain the reactor in a safe shutdown condition. The EDG system is capable of performing this function during a loss of offsite power, with or without a coincident LOCA. There are two EDG sets of identical design, each dedicated to one of the safety-related, redundant engineered safety features electrical trains.

According to the licensee, there are no changes to the existing Class 1E LOCA loading requirements for the EDG and no changes to the sequencing of these loads. Hence, the EDG system has adequate capacity and capability to power the safety-related loads at MUR power uprate conditions.

Based on the above, the NRC staff finds that there are no significant changes in EDG system loads. Therefore, the NRC staff finds that the analyses for the EDG system bound the MUR power uprate conditions.

3.3.2.5 Switchyard

The switchyard equipment and associated components are classified as nonsafety-related. The primary function of the Vogtle 1 and 2, 230 kV and 500 kV switchyards and distribution system is to distribute the generated power to the transmission grid. The switchyards supply two independent and immediately available offsite sources of power to the RATs, which are the preferred power source for the Class 1E safety-related AC distribution system, and also, provide startup power to the plant auxiliaries. The licensee contends that the two offsite sources are not affected by the MUR power uprate.

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

3.3.2.6 Grid Stability

The grid stability impact of the power uprate is discussed in Section 5 of the LAR. The licensee stated that the impact of an uprate of 33 MWe for each unit was evaluated. The results of the evaluation showed that the critical breaker failure clearing time would be reduced and that the proposed electrical output uprate for the VEGP units will not cause any grid stability problems.

The NRC staff requested additional information on the grid stability study, specifically asking the licensee for the assumptions, methodology, and cases studied to support the conclusion that the uprate does not impact grid stability. In its December 21, 2007, letter, the licensee stated that the grid stability study involved two dynamic ready cases, one with a valley case load level (17 gigawatts (GW)) and the other with a 50% load level (23.3 GW). Seven faults with normal clearing were studied, four at the 230 kV level and three at the 500 kV level. The licensee stated that the simulation results indicated that for the normal clearing faults the critical clearing times were reduced for the power uprate but the actual clearing time is less than the critical clearing time.

In addition, breaker failures were studied at both the 230 kV and 500 kV level. The licensee stated that the two worst-case scenarios involved breaker failures that resulted in the loss of power

delivery components (i.e., transmission lines or transformers). In both cases, the fault was assumed to occur on the bus-work for the autotransformers as the transformer differential relaying is slightly slower than the line relaying, producing the most severe contingencies from a transient stability point of view. The licensee has shown that the actual clearing times for these faults is less than the critical clearing time.

The reactive power output capability of Vogtle 1 and 2 is discussed in the LAR, and the licensee stated that the grid stability review indicated that the units continue to meet the system reactive policy after the MUR power uprate. Presently, the megavolt ampere reactive (MVAR) output is 575 MVAR for both Vogtle 1 and 2. The grid stability study found that the most demanding MVAR output for Vogtle 1 and 2 (after the uprate) is 440 MVAR for Vogtle 2 when Vogtle 1 is offline and the Vogtle 1 and 2-Scherer 500 kV line is out of service. Following the uprate, the MVAR output will be 567 MVAR for Vogtle 1 and comparable for Vogtle 2. Hence, both Vogtle 1 and 2 will be able to provide the necessary MVAR support following the uprate. Additionally, there are two 90 MVAR capacitor banks that can be switched onto the 230 kV bus by transmission system operators. Furthermore, in all of the cases considered, the minimum required 230 kV switchyard voltage is maintained within the acceptable range.

The NRC staff reviewed the grid stability study, and finds that the Vogtle 1 and 2 MUR power uprate allows for continued stable and reliable grid operation.

3.3.2.7 Station Blackout

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

The Vogtle 1 and 2 SBO coping duration is 4 hours. This is based on the licensee's evaluation of the offsite power design characteristics, emergency AC power system configuration, and EDG reliability, in accordance with the evaluation procedure outlined in NUMARC 87-00 and Regulatory Guide 1.155. The offsite power design characteristics include the expected frequency of grid-related loss of offsite power, the estimated frequency of loss of offsite power from severe and extremely severe weather, and the independence of offsite power.

The design bases for an SBO include capability to provide core cooling, ability to maintain adequate RCS inventory, ability to maintain appropriate containment integrity, effects of loss of ventilation, equipment environmental evaluation, access to plant area requirements, emergency lighting requirements, and EDG reliability program requirements. The licensee stated that the proposed MUR has no affect on Vogtle 1 and 2 station battery capacity as the MUR does not increase loads. The licensee stated that the areas containing equipment required to cope with an SBO will not see an increase in temperature due to the power uprate, and additionally, there is no effect of loss of ventilation during an SBO. Furthermore, as all containment isolation valves are designed to fail in their safe position, the MUR power uprate has no impact on the requirements of containment isolation. The licensee's evaluation of the SBO design criteria indicated that the design bases parameters are not adversely affected by the MUR power uprate and continue to remain bounded by the existing design bases and analyses. Based on this information, the NRC staff finds that the MUR power uprate will have no impact on the Vogtle 1 and 2 SBO coping duration. Therefore, the NRC staff finds that Vogtle 1 and 2 will continue to meet the requirements

of 10 CFR 50.63.

3.3.2.8 Equipment Qualification (EQ) Program

In its LAR, the licensee stated that the environmental qualification of electrical equipment was performed at a core power level of 102% of 3565 MWt, which bounds the MUR operating conditions. The MUR power uprate causes the gamma and beta dose inside containment to increase by less than 10% and the gamma doses outside containment to increase by 5%. The NRC staff finds that these increases are within the accuracy of the analyses and radiological doses used in equipment qualification evaluations. Based on this information, the NRC staff finds that the current EQ parameters remain bounding for the MUR power uprate. Therefore, the NRC staff finds that the MUR power uprate will have no impact on the Vogtle 1 and 2 EQ Program.

Based on the technical evaluation provided above, the NRC staff finds that Vogtle 1 and 2 will continue to meet GDC 17, 10 CFR 50.63, and 10 CFR 50.49. Therefore, the NRC staff finds the MUR power uprate acceptable.

3.4 Mechanical and Civil Engineering

3.4.1 Regulatory Evaluation

The NRC staff's review in the area of mechanical engineering covers the structural and pressure boundary integrity of NSSS and balance-of-plant (BOP) systems and components. This review focuses on the impact of the proposed MUR power uprate on (1) NSSS piping, components, and supports; (2) BOP piping, components, and supports; (3) the reactor vessel (RV) and its supports; (4) control rod drive mechanisms (CRDMs); (5) the U-tube steam generators (SGs) and their supports; (6) reactor coolant pumps (RCPs) and their supports; (7) the pressurizer and its supports; (8) reactor internals and core supports; and (9) safety-related valves. Technical areas covered by this review include stresses, cumulative usage factors (CUFs), flow-induced vibration, high-energy line break locations, jet impingement and thrust forces, and safety-related valve programs.

The above affected piping systems, components and their supports, including core support structures, are designed in accordance with the rules of the American Society of Mechanical Engineers, *Boiler and Pressure Vessel Code* (ASME Code), Section III, and the American National Standards Institute (ANSI) Power Piping code B31.1. The NRC staff's evaluation considered General Design Criteria (GDC) 1, 2, 4, 10, 14, and 15. The NRC staff review focused on verifying that the licensee has provided reasonable assurance of the structural and functional integrity of piping systems, components, component internals, and their supports under normal and vibratory loadings, including those due to fluid flow, postulated accidents, and natural phenomena such as earthquakes.

The acceptance criteria are based on the requirements of the following regulations:
(1) 10 CFR 50.55a, and GDC 1 as they relate to structures and components being designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety function to be performed; (2) GDC 2 as it relates to structures and components important to safety being designed to withstand the effects of earthquakes combined

with the effects of normal or accident conditions; (3) GDC 4 as it relates to structures and components important to safety being designed to accommodate the effects of, and to be compatible with, the environmental conditions of normal and accident conditions; (4) GDC 10 as it relates to reactor internals being designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences; (5) GDC 14 as it relates to the reactor coolant pressure boundary being designed, fabricated, erected, and tested to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture; and (6) GDC 15 as it relates to the RCS being designed with sufficient margin to ensure that the design conditions are not exceeded.

The specific review areas are contained in the NRC Standard Review Plan (SRP), NUREG-0800, Chapter 3, and Section 3.9. The review also includes the plant-specific provisions of Generic Letter (GL) 89-10, "Safety-Related Motor-Operated Valve Testing and Surveillance," and GL 96-05, as related to plant-specific program for motor-operated valves, GL 95-07, "Pressure Locking and Thermal Binding of Safety-Related Power-Operated Gate Valves," as related to the pressure locking and thermal binding for safety-related gate valves, and the plant-specific evaluation of the GL 96-06 program, "Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions," regarding the over-pressurization of isolated piping segments.

3.4.2 Technical Evaluation

The NRC staff reviewed the Vogtle 1 and 2 MUR power uprate amendment request dated August 28, 2007. The review focused on the effects of power uprate on the structural and pressure boundary integrity of piping systems and components, their supports, and reactor vessel and internal components, the control rod drive mechanisms, and the BOP piping systems.

The proposed 1.7 percent power uprate will increase the RTP level from 3565 MWt to 3625.6 MWt. The power uprate will be achieved by an increase in steam flow and FW flow, and an increase in the temperature difference across the core. The reactor coolant system (RCS) pressure and RCS average temperature will remain the same.

Table 4-2 of Enclosure 9 and Enclosure 10 in the licensee's August 28, 2007, submittal shows the pertinent temperatures and pressures for the current conditions. Table 2-1 of Enclosure 9 and Enclosure 10 in the licensee's August 28, 2007, submittal shows the pertinent temperatures and pressures for the uprated conditions. Section VI of Enclosure 5 in the licensee's August 28, 2007, submittal describes the pertinent flow rates for the current and uprated conditions. At full power, the hot-leg temperature increases from 620.0 to 620.6 degrees Fahrenheit while the cold-leg temperature decreases from 556.8 to 556.2 degrees Fahrenheit. The SG pressures decrease from 950.0 to 941.0 psia and the steam flow increases from 15.92 to 16.32 million pounds per hour (Mlbm/hr). The FW temperature increases from 446.0 to 448.7 degrees Fahrenheit and the FW flow increases from 16.0 to 16.4 Mlbm/hr. The proposed uprate does not change heatup or cooldown rates or the number of cycles assumed in the design analyses. In addition, the limiting analyses for design transients are still bounding.

The design parameters for the SGs and RCS are found in Tables 5.4.2-1 and 5.4.3-1, respectively,

of the Vogtle 1 and 2 UFSAR. The RCS components, including the reactor vessel, core support structures, and SGs, were designed to operate at a core power level of 3636 MWt. The RCS components are designed to 650 degrees Fahrenheit (except the pressurizer, which is designed to 680 degrees Fahrenheit) and 2,485 psig. The FW system design temperature is 460 degrees Fahrenheit (Licensee Submittal Enclosure 5 Section VI Page 4).

3.4.2.1 Reactor Pressure Vessel and Internals

The code of record for the reactor vessel, nozzles, and supports is ASME Code, Section III, 1971 edition with addenda through the summer 1972. The licensee compared the expected temperatures and pressures for the proposed power uprate condition against the analyses of record. The licensee confirmed in its submittal that there is no change in RCS design or operating pressure, and the effects of operating temperature changes for cold and hot legs are within design limits. The MUR power uprate conditions are bounded by the design conditions. In addition, the operating transients continue to bound the uprate conditions and no additional transients have been proposed. The existing loads, stresses, and fatigue CUF values for reactor vessel and internals remain valid for the proposed power uprate. The NRC staff concurs with the licensee's assessment that the RPV and internals are acceptable for operation at the uprated power level.

3.4.2.2 Control Rod Drive Mechanisms (CRDMs)

The code of record for the pressure retaining components of the CRDMs is the ASME Code, Section III,1974 edition with addenda through summer 1975. The licensee confirmed in its submittal that the temperatures and pressures used in the CRDM design analyses continue to bound the conditions at the proposed uprated power level. In addition, the operating transients continue to bound the uprate conditions and no additional transients have been proposed. The existing loads, stresses, and fatigue CUF values for CRDM components remain valid for the proposed power uprate. The NRC staff concurs with the licensee's assessment that the CRDMs are acceptable for operation at the uprated power level.

3.4.2.3 Reactor Coolant Piping and Components

The RCS piping was designed to ASME Code, Section III, 1977 edition with addenda through summer 1979. The licensee reviewed the revised design conditions for impact on the existing design-basis analyses for the reactor coolant piping and supports. It was stated that there is no change in RCS design or operating pressure, and the effects of operating temperature changes for cold and hot legs are within design limits. The MUR power uprate conditions are bounded by the design conditions. In addition, the operating transients continue to bound the uprate conditions and no additional transients have been proposed. The existing loads, stresses, and fatigue CUF values for RCS piping and supports remain valid for the proposed power uprate.

The SGs were designed to the ASME Code, Section III, 1971 edition with addenda through summer 1972. The licensee reviewed the revised design conditions for impact on the existing design-basis analyses for the SGs including the SGs tubes, secondary side internal support structures, shell and nozzles. There is a negligible change in RCS mass flow rate and the RCS temperatures and pressures used in the design continue to bound the uprate conditions. There is an increase in the steam flow and FW flow. The steam and FW pressures and flow rates used in

the design of the SGs continue to bound the expected uprate conditions. In addition, the operating transients continue to bound the uprate conditions and no additional transients have been proposed. Also, since the design of the SGs included modeling of flow-induced vibration and the steam and FW flow rates remain bounded by the design flow rates, the licensee concluded that the MUR uprate will have no effect on flow-induced vibration. The licensee stated that the existing tube loads due to LOCA, main steam line break (MSLB), and feedwater line break (FWLB) will not change as a result of the power uprate. The existing loads, stresses, and fatigue CUF values for the SGs remain valid for the proposed MUR uprate.

The pressure retaining parts of the RCPs were designed in accordance with the ASME Code, Section III, 1971 edition with addenda through winter 1972. The licensee reviewed the revised design conditions for impact on the existing design-basis analyses for the RCPs. It was stated that the temperature changes due to the MUR uprate are bounded by those used in the existing analyses. In addition, the operating transients continue to bound the uprate conditions and no additional transients have been proposed. The existing loads, stresses, and fatigue CUF values for RCPs remain valid for the proposed power uprate.

The code of record for the pressurizer, including the nozzles, is the ASME Code, Section III, 1971 edition with addenda through summer 1972. The licensee reviewed the revised design conditions for impact on the existing design-basis analyses for the pressurizer. The licensee stated that the temperature changes due to the MUR power uprate are bounded by those used in the existing analyses. In addition, the operating transients continue to bound the uprate conditions and no additional transients have been proposed. The existing loads, stresses, and fatigue CUF values for the pressurizer remain valid for the proposed power uprate.

Based on the above, the NRC staff agrees with the licensee's conclusion that the design of the reactor coolant piping and components, including the SGs, RCPs, and pressurizer, and their supports, is adequate to maintain the structural and pressure boundary integrity of the reactor coolant loop since the analyses of record parameters are bounding for the proposed 1.7 percent power uprate condition.

3.4.2.4 High Energy Line Break Locations (HELB)

The licensee stated that an engineering evaluation was performed to determine the impact of power uprate on HELB systems. The HELB evaluations were performed at 3,625.6 MWt (102% of LTP) to bound the expected range of operation resulting from the MUR uprate. There are no new line breaks postulated for current HELB systems. It was stated that the impact of the MUR uprate on HELB systems remains bounded by the values in existing analyses. Also, there are no new systems that qualify as HELB systems as a result of the uprate. The NRC staff agrees with the licensee's conclusion regarding HELB.

3.4.2.5 Balance of Plant (BOP) Piping (NSSS Interface Systems, Safety-Related Cooling Water Systems, and Containment Systems) and Safety-Related Valves

The licensee evaluated the BOP piping systems by comparing the conditions for the proposed power uprate with the analyses of record conditions and the current operating conditions. The licensee stated that the revised design conditions were reviewed for impact on the existing

design-basis analyses for the BOP piping and concluded that the MUR power uprate conditions are bounded by the design conditions. Since the operating transients continue to bound the uprate conditions and no additional transients have been proposed, the licensee concluded that the existing loads, stresses, and fatigue CUF values remain valid.

The revised conditions were reviewed for impact on the existing design-basis analyses for the main steam and main feedwater piping and supports. No significant changes in SG design or operating pressure were made as a part of the power uprate. The changes in the operating temperatures and flow rates due to the MUR power uprate have been evaluated and were determined by the licensee to have a negligible effect on the existing design-basis analyses. The existing loads and stresses remain valid.

The licensee stated that the revised design conditions were reviewed for impact on the existing design-basis analyses for the safety-related valves and showed that the temperature changes due to the MUR uprate are bounded by those used in the existing analyses. Safety analyses confirmed that the installed capacities and lift set points of the RCS and main steam relief valves continue to be valid for the MUR conditions. None of the safety-related valves required a change to their design or operation as a result of the MUR. The existing loads, stresses, and fatigue CUF values remain valid. The licensee did not identify any changes to the plant-specific provisions of GL 89-10 and GL 96-05, related to motor-operated valves, GL 95-07, related to pressure locking and thermal binding of safety-related gate valves, or GL 96-06, related to over-pressurization of isolated piping segments. The NRC staff does not anticipate any changes to the analysis of over-pressurization of isolated piping segments because the analyses of record for containment temperature and pressure was performed at 102% of current RTP and remains bounding for the uprate conditions. Therefore, the NRC staff does not expect any changes to the plant-specific provisions of GL 89-10, GL 96-05, GL 95-07, or GL 96-06.

The licensee concluded that the Vogtle 1 and 2 BOP piping systems remain acceptable for operation at the uprated conditions. Based on the above, the NRC staff agrees with the licensee's conclusion that the proposed 1.7% power uprate will not have adverse effects on BOP systems including safety-related valves.

3.4.3 Summary

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

3.5 Component Performance and Testing

3.5.1 Effect of Power Uprate on Major Components

3.5.1.1 Control Rod Drive Mechanism (CRDM)

The licensee reviewed the impact on the existing CRDM design-basis analysis for the measurement uncertainty recapture (MUR) power uprate conditions. No changes in the RCS design or operating pressure were made as part of the MUR power uprate. The effect of operating temperature changes (T_{hot}/T_{cold}) are within the design limits. The design conditions in the existing analyses are based on the RCS functional specification. The licensee further stated that the MUR power uprate conditions are bounded by the design conditions. Since the operating transients will not change as a result of the MUR power uprate and no additional transients have been developed, the existing loads, stresses, and fatigue values remain valid. Thus, the NRC staff concludes that the existing stresses for the CRDMs remain applicable for the MUR power uprate conditions, and that the existing CRDM design-basis analyses support the MUR power uprate.

3.5.1.2 Safety-Related Valves

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

The licensee reviewed the impact of the proposed MUR power uprate conditions on the existing design-basis analyses for the safety-related valves. The evaluation showed that the temperature changes due to MUR power uprate are insignificant and bounded by those used in the existing analyses, and analyses also confirmed the installed capacities and lift setpoints of the RCS and main steam safety valves remain valid for the MUR power uprate conditions. Due to the insignificant changes in temperature and operating pressure, none of the safety-related valves required a change to their design or operation as a result of the MUR power uprate. The licensee stated that the plant uprate accident analysis required flows are not changing, the specific nuclear class valve response times are not changing, and the nuclear grade valve components are not physically changing to support the MUR power uprate, and, therefore, the inservice testing program for safety-related valves will not be affected. The licensee also stated that systems which have valves maintained within the air-operated valve program, the GL 89-10 motor-operated valve program, and the GL 95-07 pressure locking/thermal binding program were reviewed. The review concluded that the MUR power uprate does not impact program valves since the operating temperature and pressure ranges are bounded by the original design parameters and the MUR power uprated accident analysis required flows are not changing. Therefore, the NRC staff finds the performance of existing safety-related valves acceptable with respect to the MUR power uprate.

3.5.1.3 Safety-Related Pumps

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

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

3.6 Steam Generator Tube Integrity and Chemical Engineering

3.6.1 Chemical and Volume Control System (CVCS)

3.6.1.1 Regulatory Evaluation

The CVCS provides a means for (1) maintaining water inventory and quality in the reactor coolant system (RCS), (2) supplying seal-water flow to the reactor coolant pumps and pressurizer auxiliary spray, (3) controlling the boron neutron absorber concentration in the reactor coolant, (4) controlling the primary-water chemistry and reducing coolant radioactivity level, and (5) supplying recycled coolant for demineralized water makeup for normal operation and high-pressure injection flow to the emergency core cooling system in the event of postulated accidents. The NRC staff has reviewed the safety-related functional performance characteristics of CVCS components. The acceptance criteria are based on (1) General Design Criteria (GDC) 14, "Reactor Coolant Pressure Boundary (RCPB)," as it requires that the RCPB be designed to have an extremely low probability of abnormal leakage, of rapidly propagating fracture, and of gross rupture; and (2) GDC 29, "Protection Against Anticipated Operational Occurrences," as it requires that the reactivity control systems be designed to ensure an extremely high probability of accomplishing their functions in the event of condenser in-leakage or primary-to-secondary leakage. Specific review criteria are contained in SRP, NUREG-0800, Chapter 9, Section 9.3.4, "Chemical and Volume Control System (PWR)."

3.6.1.2 Technical Evaluation

The primary functions of the CVCS are to adjust boron concentration, maintain coolant inventory, and maintain the chemistry of the RCS. In addition, the CVCS provides seal water flow to the reactor coolant pumps (RCPs), emergency boric acid addition, purification of reactor coolant, and high head injection as part of the emergency core cooling system.

During plant operation, reactor coolant letdown is taken from the cold-leg on the suction side of the RCP, through the shell side of the regenerative heat exchanger, through letdown orifices, a letdown heat exchanger, and letdown pressure regulating valves. The regenerative heat

exchanger reduces the temperature of the reactor coolant and the letdown orifices limit the flow rate through the CVCS system. The letdown heat exchanger further reduces the reactor coolant temperature and the letdown pressure regulating valves maintain pressure on the coolant to prevent it from flashing to steam. Flow continues through the purification ion exchangers, where ionic impurities are removed, thus keeping the reactor coolant activity within design limits. The reactor coolant then passes through the letdown filter and enters the volume control tank (VCT). The charging pumps take suction from the VCT and return the coolant through the tube side of the regenerative heat exchanger to the RCS in the cold-leg, downstream of the RCP. Under power uprate conditions, the licensee indicated that the hot-leg and cold-leg temperatures will increase and decrease by 0.6 degrees Fahrenheit, respectively. This will result in a slightly lower temperature for the letdown line since the CVCS system takes suction from the cold-leg. The licensee concluded that the slightly lower temperature of the letdown line does not affect the performance of the letdown coolers because they remain bounded by current operation. In addition, the licensee reported that no changes to the makeup requirement are required under power uprate conditions. A very slight increase of N-16 activity will have only a negligible effect on the radioactivity of fluids in the excess lines.

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

3.6.2 Steam Generator Blowdown System (SGBS)

3.6.2.1 Regulatory Evaluation

Control of secondary-side water chemistry is important for preventing degradation of steam generator (SG) tubes. The SGBS provides a means for removing SG secondary-side impurities, and thus assists in maintaining acceptable secondary-side water chemistry in the SGs. The design basis of the SGBS includes consideration of expected design flows for all modes of operation. The staff reviewed the ability of the SGBS to remove particulate and dissolved impurities from the SG secondary-side during normal operation, including condenser in-leakage and primary-to-secondary leakage. The NRC's acceptance criterion for the SGBS is based on GDC 14, "Reactor Coolant Pressure Boundary," as it requires that the RCPB be designed to have an extremely low probability of abnormal leakage, of rapidly propagating fracture, and of gross rupture. Specific review criteria are contained in SRP, NUREG-0800, Chapter 10, Section 10.4.8, "Steam Generator Blowdown System (PWR)."

3.6.2.2 Technical Evaluation

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

of SG secondary side chemistry reduces the probability of secondary-side-initiated SG tube degradation.

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

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

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

3.6.3 Steam Generator Program

3.6.3.1 Regulatory Evaluation

Steam generator tubes constitute a large part of the RCPB. The staff reviewed the effects of changes in operating parameters (e.g., pressure, temperature, and flow velocities) resulting from the proposed power uprate on the design and operation of the SGs. Specifically, the staff evaluated whether changes to these parameters continue to be bounded by those considered in the plant design and licensing basis (i.e., the technical specification plugging limits).

3.6.3.2 Technical Evaluation

Vogtle 1 and 2 each have four Westinghouse Model F SGs. Each SG contains 5,626 stress-relieved, thermally treated, Alloy 600 tubes. Each tube has a nominal outside diameter of 0.688 inches and a nominal wall thickness of 0.040 inches. The tubes were hydraulically expanded for the full-length of the tubesheet. A number of stainless steel tube support plates (TSPs) support the tubes.

Feedwater flow, operating temperature, and differential pressure across the tubes will change under power uprate conditions. The licensee expects a marginal increase in the stress corrosion cracking susceptibility due to a SG inlet temperature increase of 0.6° F. The licensee stated that the primary-to-secondary differential pressure across the tubes would experience a slight increase under uprated conditions, but that the increase was bounded by the original specification limit. As for the temperature, the expected increase of 0.6° F remains bounded by the original design specification temperature.

The licensee stated that the impact of the MUR-PU on steam generator tube wear was evaluated

based on the current design-basis analysis. The effect of the MUR-PU changes on the fluid-elastic stability ratio, the amplitude of tube vibrations due to turbulence, and the potential for future tube wear were addressed. The licensee stated that the results showed the 1.7% power uprate did not affect the straight-leg stability ratios and that the vortex shedding mechanism did not contribute significantly to tube vibration. For the most-limiting tube support condition, it was determined that turbulence-induced displacement could increase slightly, but that the displacements were not of sufficient magnitude to produce tube-to-tube contact.

Based on the increased turbulence-induced displacement, the original, design-basis analysis calculations were updated to account for wear in both the straight and U-bend portions of the steam generator tubes. The calculations indicated that there might be increased wear at the MUR-PU conditions, but that the increased level is not significant.

Portions of two tubes were removed from SG 2 at Vogtle 2 during the spring 2004 refueling outage, thus leaving some cut tube remnants in this SG. These cut tube remnants (which had been evaluated previously) were re-evaluated at MUR-PU conditions to see if they would experience significant flow-induced vibration (FIV) resulting from the fluid-elastic and/or turbulent excitation. The analysis considered the effects of secondary-side flow on the remnants for various boundary conditions that could credibly occur at the remnant location, as well as the faulted steamline break event. While the stability ratios of the remnants increased slightly, the ratios are still well below the allowable value of 1.0.

The licensee stated that the current plug and tube stabilizer design parameters bound the power uprate conditions. Loose parts that were previously evaluated for Vogtle 1 and 2 were re-evaluated using updated operating conditions associated with the MUR-PU. The limiting loose parts from the previous evaluations performed for Vogtle 1 and 2 were re-evaluated. It was concluded that these results would envelop all other objects.

The licensee also stated that the Regulatory Guide (RG) 1.121, “Bases for Plugging Degraded PWR Steam Generator Tubes,” analysis for Vogtle 1 and 2 was not impacted by the MUR-PU. The analysis considered both the effects of current operation and chemical cleaning, and the input assumptions were either unchanged by the MUR-PU or remained bounded by the original analysis; therefore, the structural limits are still acceptable.

The licensee discusses in Section 6.6.11 of Enclosures 9 and 10 to the August 28, 2007, letter, loose part, tube-end damage that occurred in the hot-leg primary chamber in SG 4 of Vogtle 1 in 1996. Post-damage examination of the tube-to-tubesheet welds in 1996 showed that the weld material present was equal to or greater than that modeled in the original design analysis, which the licensee stated demonstrated compliance with the ASME Code allowable values for normal, emergency, and faulted conditions. As a “defense-in-depth” type of evaluation, the licensee also performed a SG analysis in 1996 that assumed the tube-to-tubesheet welds were not present, and it was concluded that subsequent SG operation would be safe from tube bundle structural integrity and leak tightness perspectives, at a power level of 3,579 MWt. The licensee stated that they used the same evaluation process that was used in 1996 in the current MUR-PU report to support a 3,653 MWt power uprate condition, again as a “defense-in-depth” type of evaluation. The NRC staff verified with the licensee that the current design-basis analysis was still bounding for the MUR-PU conditions, and that the “defense-in-depth” analysis was not required to meet the

MUR-PU analyzed conditions. The NRC staff has been performing a separate, concurrent review of an analysis provided by Westinghouse, called H*/B*, which is similar to the "defense-in-depth" analysis that was performed for the licensee in 1996. The NRC staff has concluded that the technical basis of the finite element analysis has not been conducted with sufficient technical rigor to provide adequate evidence for the NRC staff to accept the reduced inspection scope proposed in the separate requested H*/B* amendment (H*: The structural integrity of the primary-to-secondary pressure boundary is unaffected by tube degradation of any magnitude below a certain depth, designated as H*, from the top of the 21 inch thick steam generator tubesheet. The H* depth varies depending on the radial location of the tube in the tubesheet; B*: The accident condition leak rate integrity can be bounded by a multiple of the normal operating leak rate, due to degradation at or below a calculated distance, designated as B*, from the top of the 21 inch thick steam generator tubesheet, including degradation of the tube end welds). Therefore, the reduced inspection depths related to H*/B*, and referenced in the MUR-PU amendment request, are beyond the scope of this review and were not reviewed as part of this license amendment request. However, the NRC staff notes that the critical primary-side components with the highest stress and fatigue usage in the MUR-PU (divider plate, tubesheet and shell junctions, tube-to-tubesheet weld, and tubes), are bounded by the analysis done for current operation, as stated in enclosure 9 of the licensee's original application dated August 28, 2007 (Reference 6).

The licensee also stated that the Vogtle 1 and 2 Model F SGs are not regarded as susceptible to the TSP packing phenomena, referenced in NRC Bulletin 88-02 (Rapidly Propagating Fatigue Cracks In Steam Generator Tubes), since the stainless steel Quatrefoil© TSPs are resistant to the corrosion that affected the carbon steel TSPs at North Anna Unit 1.

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

3.6.4 Flow Accelerated Corrosion (FAC) Program

3.6.4.1 Regulatory Evaluation

FAC is a corrosion mechanism occurring in carbon steel components exposed to single-phase or two-phase water flow. Components made from stainless steel are immune to FAC, and FAC is significantly reduced in components containing small amounts of chromium or molybdenum. Material loss rates due to FAC depend on flow velocity, fluid temperature, steam quality, oxygen content, and pH. During plant operation, control of these parameters is limited and often the conditions for minimizing FAC cannot be achieved; therefore, loss of material by FAC is likely to occur. The staff reviewed the proposed MUR power uprate for potential effects on FAC and the adequacy of the licensee's FAC program to predict loss rates so that repair or replacement of damaged components could be made before reaching critical minimum thickness.

3.6.4.2 Technical Evaluation

FAC is a corrosion mechanism occurring in carbon steel components exposed to flowing single or two-phase water. FAC results in wall thinning and possible failure of high-energy carbon steel pipes in the power conversion system. The rates of material loss by FAC depend on velocity of flow, temperature, steam quality, oxygen content, and pH. During plant operation, control of these parameters is limited and the optimum conditions for minimizing FAC effects, in most cases, cannot be achieved. Loss of material by FAC is, therefore, likely to occur. Since undesirable challenges to the plant's safety systems may result from piping system component failure, licensees maintain a FAC-related failure prediction, inspection, and component repair/replacement program.

The licensee stated that to support the MUR-PU, a new high-pressure turbine will be installed to accept the increased main steam flow. As a result, most of the secondary plant fluid systems will experience flow increases of less than 10 percent, with insignificant changes in fluid temperatures. The largest increases are in the high-pressure extraction steam lines. These lines were reviewed, and only minimal increases in wear rates were predicted. Other system pressures and temperatures are not changing significantly; thus, significant corrosion increases are not expected. As part of the MUR-PU implementation, the fluid conditions will be input into the FAC predictive models to establish and validate component erosion/wear rates. Major changes to the scope of FAC inspections or inspection frequencies are not expected. The FAC Susceptibility Analysis is a continuous effort and is reviewed and revised as necessary. No changes or additions to currently scheduled component replacements are required.

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

3.6.5 Coatings

3.6.5.1 Regulatory Evaluation

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

3.6.5.2 Technical Evaluation

The licensee indicated that the design-basis accident temperature and pressure profiles are a function of RCS average temperature and that the RCS average temperature is not changing due to power uprate conditions. In response to the staff's request for additional information, the licensee confirmed that the original coating temperature and pressure qualification profile remain bounding for MUR-PU pressure and temperature conditions.

Based on its review, the staff concludes that the coatings will not be adversely impacted by the MUR-PU temperature and pressure conditions, as they will continue to be bounded by the conditions to which the coatings were qualified. Therefore, the NRC staff finds the proposed MUR power uprate acceptable with respect to the coatings program.

3.6.6 Summary

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

3.7 Operator Licensing and Human Performance

3.7.1 Regulatory Evaluation

The NRC's acceptance criteria for human factors are based on 10 CFR 50.54, "Conditions of licenses," Sections (i) and (m), 10 CFR 50.92, "Issuance of amendment," 10 CFR 50.120, "Training and qualification of nuclear power plant personnel," and GDC 19. The NRC staff is also responsible for reviewing the licensee's human factors evaluation to determine if it conforms to the NRC staff's guidance in Section VII of RIS 2002-03, "Guidance on the Content of Measurement Uncertainty Recapture Power Uprate Applications." RIS 2002-03 provides guidance to the licensee in evaluating the need for changes to the areas of operator manual actions, procedures, human-system interfaces, and operator training related to the MUR power uprate. The NRC staff's human factors evaluation was conducted to confirm that operator performance would not be adversely affected as a result of changes made to implement the proposed MUR power uprate.

3.7.2 Technical Evaluation

The NRC staff has requested the licensee on November, 20, 2007 (ADAMS Accession No. ML073200227) for additional information for human factors reviews of MURs based on RIS 2002-03, Attachment 1, Section VII. The following sections evaluate the licensee's response to these questions in the LAR and additional clarifications in its RAI response, dated December 21, 2007 (Reference 4).

3.7.2.1 Operator Manual Actions

The licensee evaluated five design bases events in the Vogtle 1 and 2 UFSAR at 102% power for possible changes to operator manual actions and response action times. The licensee's analysis concluded that no new operator manual actions or changes to existing operator action times were needed for the power uprate conditions. The licensee performed additional evaluations of overall operator manual actions and response action times in plant procedures, emergency operating procedures (EOPs), and abnormal operating procedures (AOPs) and found that existing operator manual actions and response action times will remain unchanged by the MUR power uprate.

The NRC staff has reviewed the licensee's statements in the original submittal and responses to the RAI relating to any impacts of the MUR power uprate to existing or new operator manual actions. The NRC staff concludes that the proposed MUR power uprate will not have any impact on the overall existing operator manual actions and their response times. The proposed MUR power uprate will not adversely impact safety in the area of human performance as required by 10 CFR 50.92. The NRC staff also finds that the statements provided by the licensee follow the guidance of Section VII.1 of Attachment 1 to RIS 2002-03.

3.7.2.2 Emergency and Abnormal Operating Procedures

The licensee reviewed Vogtle 1 and 2 EOPs and AOPs for potential changes related to the proposed MUR power uprate. The EOPs and AOPs include setpoints that will be revised to reflect the increased power level and increased decay heat at shutdown conditions during MUR power uprate conditions. The licensee's review also concluded that no additional operator actions or changes to the response action times will be made in the EOPs and AOPs. The changes being made to reflect the revised setpoints will be incorporated in the normal operator training cycles prior to the implementation of the MUR power uprate.

The NRC staff reviewed the licensee's evaluation of the effects of the MUR power uprate on Vogtle 1 and 2 EOPs and AOPs. The NRC staff concludes that the proposed MUR power uprate does not present any adverse impacts on the EOPs and AOPs. This conclusion is based upon the licensee making revisions to the EOPs and AOPs that will reflect the new power level and increased decay heat due to the MUR power uprate. The setpoint changes being made in the EOPs and AOPs will be reflected in the operator training program prior to MUR implementation. The proposed changes to the EOPs and AOPs are being made in accordance with 10 CFR 50.92 and 10 CFR 50.120, in which safety will not be adversely affected and the revisions being reflected in the operator training program. The NRC staff also finds that the statements provided by the licensee follow the guidance of Sections VII.2.A and VII.4 of Attachment 1 to RIS 2002-03.

3.7.2.3 Control Room Controls, Displays, and Alarms

In its submittal, the licensee described changes to control room controls, displays (including the Safety Parameter Display System), and alarms related to the proposed MUR power uprate. Notable proposed changes to controls, displays, and alarms will include:

- The Caldon Leading Edge Flow Meter (LEFM) system will be installed for each unit with a main control board annunciator alarm, computer display and diagnostic functions to be displayed on the plant computer control room monitors. The LEFM in both Units will not share components, displays, or annunciators, which will allow the operators to operate on both Units without adverse impacts. The main control room annunciator alarm for the respective LEFM system will alert operators when function errors have been diagnosed. An annunicator response procedure (ARP) will be created to provide operators instruction to diagnose the alarms using the control room display and or local cabinet display.

The licensee will also make modifications to the high pressure turbine and heater drain pump as part of the physical changes to the plant to support the MUR power uprate. These modifications will not require any changes to the control room alarms, displays, or controls. The licensee also

stated in the LAR that no changes will be made to the Safety Parameter Display System (SPDS) monitoring panel. Operator training on the control room changes involving the new Caldon LEFM system will be developed and completed prior to implementation of the MUR power uprate.

The NRC staff has reviewed the licensee's evaluation and proposed changes to the control room in Vogtle 1 and 2. The NRC staff concludes that the proposed changes do not present any adverse effects to the operators' functions in the control room. The licensee stated that the modifications to the control room related to the new Caldon LEFM systems and the incorporation of these changes into the operator training program will be made prior to MUR power uprate implementation. The proposed control room changes will not alter existing requirements for the control room as stated in GDC 19. The NRC staff also finds that the statements provided by the licensee follow the guidance of Sections VII.2.B and VII.3 of Attachment 1 to RIS 2002-03.

3.7.2.4 Control Room Plant Reference Simulator and Operator Training

The licensee has provided the following items on the plant reference simulator to be modified based on the physical plant modifications related to the MUR power uprate:

- Core model update
- Heater drain pump flow model
- Turbine model
- New calorimetric and displays for Caldon LEFM interface

The licensee's Operations Training department will be involved to incorporate the actual plant modifications and plant reference simulator modifications into the design change process to support the MUR power uprate. The Operations Training program will be revised to address the plant modifications, technical specification changes, control room modifications, and EOPs, AOPs, and plant procedure revisions prior to the implementation of the MUR power uprate. The Operations Training department will also develop training on the operation of the new Caldon LEFM system and calorimetric impacts.

The NRC staff has reviewed the licensee's proposed changes to the operator training and plant simulator related to the MUR power uprate. The NRC staff concludes that the changes do not present any adverse effects on the plant simulator or the operator training program. The licensee stated that the modifications to the plant simulator and incorporating these changes into the operator training program will take place prior to MUR power uprate implementation. The NRC staff finds that these changes are being made in accordance with 10 CFR 50.120, where the operator training program is expected to be revised as necessary to reflect the amendment changes. The NRC staff also finds that the statements provided by the licensee follow the guidance of Sections VII.2.C, VII.2.D, and VII.3 of Attachment 1 to RIS 2002-03.

3.7.3 Summary

The NRC staff has completed its human factors review of the licensee's proposed changes for Vogtle 1 and 2 and concludes that the licensee has adequately considered the impact of the proposed MUR power uprate on operator manual actions, EOPs and AOPs, control room components, the plant simulator, and operator training programs.

3.8 Vessels and Internals Integrity

3.8.1 Reactor Vessel Material Surveillance Program

3.8.1.1 Regulatory Evaluation

The reactor pressure vessel (RPV) material surveillance program provides a means for determining and monitoring the fracture toughness of the RPV beltline materials to support analyses for ensuring the structural integrity of the ferritic components of the RPV. Appendix H of 10 CFR Part 50 provides the NRC staff's requirements for the design and implementation of the RPV material surveillance program. The NRC staff's review primarily focused on the effects of the proposed MUR-PU report on the licensee's RPV surveillance capsule withdrawal schedule. The NRC staff's acceptance criteria are based on: (1) General Design Criterion (GDC) 14, which requires that the reactor coolant pressure boundary (RCPB) be designed, fabricated, erected, and tested so as to have an extremely low probability of rapidly propagating failure; (2) GDC 31, which requires that the RCPB be designed with margin sufficient to assure that, under specified conditions, it will behave in a non-brittle manner and the probability of a rapidly propagating fracture is minimized; (3) Appendix H of 10 CFR Part 50, which provides for monitoring changes in the fracture toughness properties of materials in the RPV beltline region; and (4) Section 50.60, which requires compliance with the requirements of 10 CFR Part 50, Appendix H. Specific review criteria are contained in SRP, Chapter 5, Section 5.3.1.

3.8.1.2 Technical Evaluation

The NRC's regulatory requirements related to the establishment and implementation of a facility's RPV materials surveillance program and surveillance capsule withdrawal schedule are given in 10 CFR Part 50, Appendix H. Appendix H of 10 CFR Part 50 invokes, by reference, the guidance in American Society for Testing and Materials (ASTM) Standard Practice E185, "Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels," through the 1982 edition (ASTM E185-82). ASTM Standard Practice E185 provides guidelines for designing and implementing the RPV materials surveillance programs for operating light-water reactors, including guidelines for determining RPV surveillance capsule withdrawal schedules based on the RPV material predicted transition temperature shifts (ΔRT_{NDT}). The surveillance capsule withdrawal schedule is prepared in terms of effective full power years (EFPY) of plant operation with a projected end of current operating license (EOL) value of 32 EFPY. The licensee is applying ASTM E185-82 as its basis for implementing the Vogtle 1 and 2 materials surveillance program.

The licensee discussed the impact of the 1.7 percent MUR-PU on the RPV material surveillance program in Section 6.1.2.3 of the analysis report and stated that the revised MUR-PU fluence projections have been used in the MUR-PU assessment of the current withdrawal schedule for Vogtle 1 and 2. This calculation determined that the maximum ΔRT_{NDT} using the MUR-PU fluences corresponding to 3625.6 MWt for Vogtle 1 and 2 at 36 EFPY is less than 100 °F and does not change the required number of capsules to be withdrawn from the reactors in the current RPV materials surveillance program withdrawal schedule. ASTM E185-82 requires a minimum of three surveillance capsules to be withdrawn from each unit for which the maximum ΔRT_{NDT} is less than 100 °F.

The Vogtle 1 and 2 RPV materials surveillance program withdrawal schedule in the licensee's RAI response dated December 21, 2007, shows that Vogtle 1 and 2 have withdrawal schedules of six capsules from each unit's RPV. To date, four capsules have been withdrawn and tested from each unit (Capsules U, Y, V, and X for VEGP, Unit 1 and Capsules U, Y, X, and W for Vogtle 2). The four capsules from each unit were withdrawn at neutron fluence levels which meet the criteria specified in ASTM E185-82 for surveillance capsule withdrawals, including consideration of the effect of the MUR-PU. Hence, the NRC staff finds that the Vogtle 1 and 2 RPV surveillance program continues to support operation of the units through the period of their current operating license.

3.8.1.3 Summary

The NRC staff has reviewed the licensee's evaluation of the effects of the proposed MUR-PU on the RPV surveillance capsule withdrawal schedule and concludes that the licensee has adequately addressed changes in neutron fluence on the Vogtle 1 and 2 material surveillance program withdrawal schedules. The NRC staff further concludes that the RPV capsule withdrawal schedule is appropriate to ensure that the material surveillance program will continue to meet the requirements of 10 CFR Part 50, Appendix H and 10 CFR 50.60, and will provide the licensee with information to ensure intended compliance with GDC 14 and GDC 31 following implementation of the proposed MUR-PU. Therefore, the NRC staff finds the proposed MUR-PU is acceptable with respect to the RPV material surveillance program.

3.8.2 Neutron Fluence, Upper-Shelf Energy, Pressure-Temperature Limits, and Fracture Integrity Evaluation

3.8.2.1 Regulatory Evaluation

Appendix G of 10 CFR Part 50, provides fracture toughness requirements for ferritic materials (low alloy steel or carbon steel) in the RCPB, including requirements on the upper shelf energy (USE) values used for assessing the remaining safety margins of the RPV materials against ductile tearing and requirements for calculating pressure-temperature (P-T) limits for the plant. Appendix G of 10 CFR Part 50, requires that RCPB materials satisfy the criteria in Appendix G of Section XI of the ASME Code to ensure the structural integrity of the ferritic components of the RCPB is maintained during any condition of normal operation, including anticipated operational occurrences and hydrostatic tests.

The NRC's acceptance criteria are based on (1) GDC 14, which requires that the RCPB be designed, fabricated, erected, and tested so as to have an extremely low probability of rapidly propagating fracture; (2) GDC 31, which requires that the RCPB be designed with margin sufficient to assure that, under specified conditions, it will behave in a nonbrittle manner and the probability of a rapidly propagating fracture is minimized; (3) 10 CFR Part 50, Appendix G, which specifies fracture toughness requirements for ferritic components of the RCPB; and (4) 10 CFR 50.60, which requires compliance with the requirements of 10 CFR Part 50, Appendix G. Specific review criteria are contained in SRP, Chapter 5, Section 5.3.2.

3.8.2.2 Technical Evaluation

A. Neutron Fluence

The methodology used to determine the fast (E > 1.0 MeV) neutron fluence is based on the guidance in Regulatory Guide (RG) 1.190, “Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence,” which was issued in March 2001. Elements of the methodology are also described in WCAP-14040-NP-A, Revision 4, “Methodology Used to Develop Cold Overpressure Mitigation System Setpoints and RCS Heatup and Cooldown Limit Curves.”

The calculations were carried out using the DORT Code, a discrete ordinates transport code in the forward mode using the BUGLE-96 cross section library. BUGLE-96 is based on the Sixth Brookhaven Evaluated Nuclear Data File (ENDF/B-VI) cross sections which are recommended in RG 1.190. All forward calculations are based on a P_5 Legendre polynomial expansion and an S_{16} order of angular quadrature. Both the P_5 Legendre polynomial expansion and the S_{16} order of angular quadrature exceed the minimum guidelines specified in RG 1.190. The method derives three-dimensional fluence distributions by synthesis of one- and two-dimensional solutions. This is also in accordance with the guidance in RG 1.190.

The neutron sources were derived from cycle-specific calculations which accounted for the power uprate from 3635 MWt to 3636 MWt on the onset cycle 12 at both units. Finally, the licensee made plant-specific dosimetry comparisons (using this methodology) of measured and calculated values from the plant surveillance capsules. The results documented in Tables 8.1-1 and 8.1-2 of WCAP-16736-NP are satisfactory and within the limits prescribed in RG 1.190. The NRC staff finds the fluence values acceptable because the methodology used for the estimation of the vessel fast neutron fluence follows the guidance in RG 1.190 and because it is supported by plant-specific measurements.

B. USE Value Calculations:

Appendix G of 10 CFR Part 50, provides the NRC staff’s criteria for maintaining acceptable levels of USE for the RPV beltline materials of operating reactors throughout the licensed lives of the facilities. The rule requires RPV beltline materials to have a minimum USE value of 75 ft-lb in the unirradiated condition and to maintain a minimum USE value above 50 ft-lb throughout the life of the facility, unless it can be demonstrated through analysis that lower values of USE would provide acceptable margins of safety against fracture equivalent to those required by Appendix G of Section XI of the ASME Code. The rule also mandates that the methods used to calculate USE values must account for the effects of neutron radiation on the USE values for the materials and must incorporate any relevant RPV surveillance capsule data that are reported through implementation of a plant’s 10 CFR Part 50, Appendix H RPV material surveillance program. Regulatory Guide 1.99, Revision 2, “Radiation Embrittlement of Reactor Vessel Materials,” provides an expanded discussion regarding the calculation of Charpy USE values and describes two methods for determining Charpy USE values for RPV beltline materials, depending on whether or not a given RPV beltline material is represented in the plant’s RPV material surveillance program. If surveillance data is not available, the Charpy USE is determined in accordance with position 1.2 in RG 1.99, Revision 2. If surveillance data are available, the Charpy

USE should be determined in accordance with position 2.2 in RG 1.99, Revision 2. These methods refer to Figure 2 in RG 1.99, Revision 2, which indicates that the percentage drop in Charpy USE depends upon the amount of copper in the material and the neutron fluence. Since the analyses performed in accordance with Appendix G to 10 CFR Part 50, are based on a flaw with a depth equal to one-quarter of the vessel wall thickness (1/4T), the neutron fluence used in the Charpy USE analysis is the neutron fluence at the 1/4T depth location.

In its letter dated December 21, 2007, the licensee noted that:

> In the evaluation of the pressure vessel exposure applicable to the VEGP, MUR-PU, the fluence levels accrued through the end of Cycle 11 (ϕeoc_{11}) from References 1[4] and 2[5] were used as the baseline fluence values representing actual past operation, and the incremental fluence accrued after the beginning of Cycle 12 was increased by the ratio of [3636]/[3565] providing an overall pressure vessel fluence based on operation at a core power level of 3565 MWt for fuel Cycles 1 through 11, followed by operation at a core power level of 3636 MWt for fuel Cycles 12 and beyond. This approach was applied to both Vogtle 1 and 2.

In the licensee's MUR-PU analysis report, it was noted that the 1/4T neutron fluence values for the MUR-PU will exceed the fluence projections used to develop the projected EOL USE values by less than 2 percent. The neutron fluence values for MUR-PU conditions were calculated using a methodology consistent with the guidance in RG 1.190.

The licensee discussed the impact of the 1.7% MUR-PU on the USE values for the RPV beltline materials in Section 6.1.2.4 of the MUR-PU analysis report and its letter dated December 21, 2007. Tables in Enclosure 2 of the licensee's letter dated December 21, 2007, provide the predicted USE values for all Vogtle 1 and 2 beltline materials at EOL. The USE value for the Vogtle 1 limiting material, Intermediate Shell Plate B8805-2, was determined based on data from Vogtle 1 Surveillance Capsule X. The USE value for the Vogtle 2 limiting material, Lower Shell Plate R8-1, was determined based on data from Vogtle 2 Surveillance Capsule W.

The applicant's analysis demonstrated that all Vogtle 1 and 2 RPV beltline materials will have a USE value greater than 50 ft-lb through EOL as required by Appendix G to 10 CFR Part 50. Hence, the units continue to comply with the requirements in 10 CFR Part 50, Appendix G for operating reactors. Therefore, the NRC staff concludes that the Vogtle 1 and 2 RPV beltline materials will have acceptable USE values under the MUR-PU conditions for duration of the Vogtle 1 and 2 operating licenses.

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4. WCAP-16278-NP, Revision 0, "Analysis of Capsule X from the Southern Nuclear Operating Company, Vogtle Unit 1 Reactor Vessel Radiation Surveillance Program," K. G. Knight, et al., July 2004. WCAP-16278-NP, Revision 0 is not included in this safety evaluation (SE).

5. WCAP-16382-NP, Revision 0, "Analysis of Capsule W from the Southern Nuclear Operating Company, Vogtle, Unit 2, Reactor Vessel Radiation Surveillance Program," T. J. Laubham, et al., January 2005. WCAP-16382-NP, Revision 0 is not included in this SE.

C. P-T Limit Calculations:

Section IV.A.2 of 10 CFR Part 50, Appendix G requires that the P-T limits for operating reactors be at least as conservative as those that would be generated if the methods of calculation in the ASME Code, Section XI, Appendix G were used to calculate the P-T limits. The rule also requires that the P-T limit calculations account for the effects of neutron radiation on the P-T limit values for the RPV beltline materials and to incorporate any relevant RPV surveillance capsule data that are required to be reported as part of the licensee's implementation of its materials surveillance program.

In its letter dated December 21, 2007, the licensee stated:

> The MUR impacts the current fluence projections of VEPG, Units 1 and 2, i.e., the MUR fluence projections exceed the current fluence projections. The Pressurized Thermal Shock reference temperature (RT_{PTS}) values are dependent on the fluence values; therefore RT_{PTS} calculations were performed using the MUR fluence projections for the reactor vessel beltline materials at EOL (36 Effective Full Power Years (EFPY)). The PTS calculations were performed using the latest requirements specified in the PTS Rule (10 CFR Part 50.61) for both 36 and 57 EFPY. Tables 6.1.2-4, 6.1.2-5, 6.1.2-6, and 6.1.2-7 in WCAP-16736 provide a summary of the limiting beltline material RT_{PTS} values. All RT_{PTS} values remain below the 10 CFR Part 50.61 screening criteria values using the projected MUR-PU fluence values for 36 and 57 EFPY for VEGP, [Units 1 and 2].
>
> Tables 6.1.2-1 and 6.1.2-2 of the MUR-PU analysis report are provided to calculate the shift in RT_{NDT}, which is used to justify the continued applicability of the surveillance capsule withdrawal schedule. Table 6.1.2-3 [of the MUR-PU analysis report] is provided to show that VEGP, Units 1 and 2 will remain in the same category for the Emergency Response Guideline [ERG] Limits as discussed on page 6-7 of WCAP-16736.

The NRC staff reviewed the licensee's tables contained in Section 6.1 Reactor Vessel in the MUR-PU evaluation and found that the current peak inside surface RT_{NDT} values for the limiting materials (Intermediate Shell Plate B8805-2 for Vogtle 1 and Lower Shell Plate R8-1 for Vogtle 2) at EOL were calculated to be 117.4 °F for Vogtle 1 and 125.7 °F for Vogtle 2. When comparing these values with the limits contained in Table 6.1.2-3 of the MUR-PU analysis report, this would currently put Vogtle 1 and 2 in ERG Category 1. The ERG category is determined by the magnitude of the ART value.

The licensee noted that revised fluence projections after the MUR-PU were used to recalculate the peak inside surface RT_{NDT} values at EOL. The revised RT_{NDT} (or RT_{PTS}) values at EOL were calculated to be 117.6 °F (Vogtle 1) and 125.9 °F (Vogtle 2). Therefore, the NRC staff determined that Vogtle 1 and 2 will remain in the same ERG categories as shown in Tables 6.1.2-4 and 6.1.2-6 of the MUR-PU analysis report for the revised peak RT_{NDT} (or RT_{PTS}) values for 36 EFPY for Vogtle 1 and 2, respectively.

D. Fracture Integrity Evaluation of Ferritic Class 1 Components:

Appendix G of Section III of the ASME Code provides requirements for obtaining the allowable loadings for ferritic pressure-retaining components. These requirements are based on the principles of linear elastic fracture mechanics. For section thicknesses of 4 inches to 12 inches, a maximum postulated 1/4T flaw depth is evaluated. For section thicknesses less than 4 inches, a postulated 1-inch deep flaw is evaluated. Appendix G of Section III of the ASME Code indicates that smaller defect sizes may be utilized on an individual case basis if a smaller size of a maximum postulated defect can be ensured.

D.1 Technical Evaluation

The licensee discussed the effects of the MUR-PU on the fracture integrity of ferritic Class 1 components, specifically the RPV in Section 6.1 of the MUR-PU analysis report. The stresses in the RPV were evaluated under the conditions specified in Section 6.1.1 of the MUR-PU analysis report. The licensee used a range of RPV outlet temperatures from 603.8 °F to 620.0 °F and a range of RPV inlet temperatures from 538.3 °F to 556.2 °F to determine if the RPV would be within the bounds of the current RPV structural analysis. The licensee found that the current transients used in the analysis remain applicable to all nuclear steam supply system (NSSS) components other than the steam generators (SG). The licensee also found that the MUR-PU program has not affected the current RPV system seismic analysis. Therefore, based on the licensee's findings, the NRC staff determined that the current RPV and NSSS components (except for the SG) analysis remains bounding for the faulted condition blowdown LOCA and safe shutdown earthquake seismic loads.

E. NSSS Components Steam Generator:

In Section 6.6.2 of the MUR-PU analysis report, Structural Integrity Evaluation Introduction, it is stated that:

> The structural evaluation for the MUR-PU focused on the critical steam generator components discussed below. The critical components are those with the highest stress and fatigue usage. The critical components are affected by changes in the pressure and temperature in the primary and secondary side of the steam generator. The following critical primary side components were evaluated: divider plate, tubesheet and shell junction, tube-to-tubesheet weld, and tubes. Additionally, the following critical secondary side components were evaluated: feedwater nozzle, auxiliary feedwater nozzle, secondary manway bolts, and steam nozzle.
>
> Key Input Parameters and Assumptions:
>
> 1. The primary side component stresses are proportional to the primary-to-secondary ΔP. Since the NSSS design transient responses and the secondary side pressure are unchanged, the evaluation done for current operation bounds the MUR-PU.

2. The increase in feedwater temperature due to the MUR-PU only affects the feedwater nozzle and auxiliary feedwater nozzle. This effect may be considered by scaling the appropriate transient stresses by $\Delta T_{fw}(MUR) / \Delta T_{fw}(Ref)$.

3. The evaluation performed for current operation for the other secondary side components remains valid, since the bounding parameters discussed in Section 2 [of the MUR-PU analysis report] and the NSSS design transient responses discussed in Section 4 [of the MUR-PU analysis report] have not changed.

The plant operating conditions provide for both a low T_{avg} temperature operating condition case and a high T_{avg} temperature operating condition. The low T_{avg} case results in a lower steam pressure (and, therefore, a greater primary-to-secondary side ΔP) and as such, will envelop the high T_{avg} case. On this basis, the bounding scale factors based on the low T_{avg} were used to perform a bounding evaluation of the critical components of both the primary and secondary side components discussed above. The low T_{avg} steam pressure used for the structural evaluation was 825 psia [pounds per squire inch absolute] (which corresponds to a temperature of 521.8 °F).

E.1 Primary Side Components:

The NSSS design transient responses for the MUR-PU Program are the same as the current NSSS design transients, except for the revised feedwater temperature curves. A comparison between the MUR-PU parameters in Table 2-1 [contained in Section 2 of the MUR-PU submittal] with the current operating parameters identified that the minimum steam pressure of 825 psi [pounds per square inch] and the NSSS transient responses are the same for the MUR-PU, except for feedwater temperature. As such, the analysis performed for current operation is bounding.

E.2 Secondary Side Components:

The effect of the higher feedwater temperature (+ 2.7°F) on the feedwater nozzle fatigue evaluation was determined. For the other components, the evaluation for current operation remains valid for the MUR-PU.

The effect of the increase in the feedwater temperature due to the MUR-PU was accounted for in the fatigue evaluation of the feedwater nozzle and the auxiliary feedwater nozzle. Since the affected alternating stresses were less than the endurance limit in [1971 Edition through 1977 Addenda of the ASME Code, Section III,] it has a negligible effect on the fatigue usage.

E.3 Acceptance Criteria

The acceptance criteria for each component are consistent with the criteria used in the design basis analysis for that component. The maximum range of primary-plus-secondary stresses was compared with the corresponding allowable stress intensity $3S_m$ limits of ASME Code, Section III. For situations in which these limits were exceeded, a plastic analysis or simplified elastic-plastic analysis had been performed in the original analysis to

meet the limits in [ASME Code, Section III.] The results of these original analyses were updated for current operation. A cumulative fatigue usage factor less-than-or-equal-to unity demonstrates the adequacy of the steam generators for a 40-year design life.

E.4 Technical Evaluation

Table 6.6.2-1, Evaluation Summary - Primary and Secondary Side Components, and Section 6.5.2, Structural Integrity Evaluation, of the licensee's MUR-PU submittal indicate all steam generator ferritic high stress and fatigue components will meet the requirements of Appendix G of Section III of the ASME Code. The licensee's structural evaluation was performed for the bounding condition for the low T_{avg} case, where P_{stm} = 825 psia. In addition, scale factors were calculated based on the revised steam pressure of 825 psia at the MUR-PU conditions. The licensee evaluated the critical components with the highest stresses and fatigue loading.

The NRC staff determined that for the primary side of the steam generator high stress components (divider plate, tubesheet and shell junctions, tube-to-tubesheet welds, and tubes) and secondary side components (feedwater nozzle, auxiliary feedwater nozzle, steam nozzle, and secondary manway bolts), the stress ranges and fatigue usages of the licensee's evaluations are acceptable for MUR-PU conditions.

The NRC staff determined that based on the licensee's evaluation of the critical components with the highest stresses and fatigue loading that they were acceptable under MUR-PU conditions, it can be concluded that components with lower stresses and fatigue usages (such as steam generator secondary side internal support structures or steam generator shell) are acceptable at the MUR-PU conditions.

3.8.2.3 Summary

The NRC staff concludes that the licensee has demonstrated that the neutron fluence, upper-shelf energy, pressure-temperature limits, and fracture integrity evaluation of carbon steel components will continue to be acceptable and will continue to meet the requirements of the ASME Code, Section III, 10 CFR Part 50, Appendix G, and guidance in RG 1.190. Therefore, the NRC staff finds, the proposed MUR-PU acceptable with respect to the neutron fluence, upper-shelf energy, pressure-temperature limits, and fracture integrity evaluations.

3.8.3 Pressurized Thermal Shock

3.8.3.1 Regulatory Evaluation

The pressurized thermal shock (PTS) evaluation provides a means for assessing the susceptibility of the RPV beltline materials to PTS events to assure that adequate fracture toughness is provided during reactor operation. The staff's requirements, methods of evaluation, and safety criteria for PTS assessments are given in 10 CFR 50.61. The NRC staff's review covered the PTS methodology and the calculations for the reference temperature, RT_{PTS}, considering neutron embrittlement effects. The NRC's acceptance criteria for PTS are based on (1) GDC 14, which requires that the RCPB be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture; (2)

GDC 31, which requires that the RCPB be designed with margin sufficient to assure that, under specified conditions, it will behave in a nonbrittle manner and the probability of a rapidly propagating fracture is minimized; and (3) Section 50.61, which sets fracture toughness criteria for protection against PTS events. Specific review criteria are contained in SRP Section 5.3.2.

3.8.3.2 Technical Evaluation

Section 50.61 of 10 CFR Part 50 provides the fracture toughness requirements protecting the RPVs of pressurized water reactors against the consequences of PTS. Licensees are required to perform an assessment of the RPV's projected values of the PTS reference temperature, RT_{PTS}, through the end of their units' operating license. The rule requires each licensee to calculate the RT_{PTS} value for each material located within the beltline of the RPV. The RT_{PTS} value for each beltline material is the sum of the unirradiated nil-ductility reference temperature (RT_{NDT}) value, a shift in the RT_{NDT} value caused by exposure to high energy neutron radiation (i.e., ΔRT_{NDT} value), and an additional margin value to account for uncertainties (i.e., M value). Section 50.61 of 10 CFR Part 50 also provides screening criteria against which the calculated values are to be evaluated. RPV beltline base-metal materials (forging or plate materials) and longitudinal (axial) weld materials are considered to provide adequate protection against PTS events if the calculated RT_{PTS} values are less than or equal to 270 °F. Reactor vessel beltline circumferential weld materials are considered to provide adequate protection against PTS events if the calculated RT_{PTS} values are less than or equal to 300 °F. RG 1.99, Revision 2 provides an expanded discussion regarding the calculations of the ΔRT_{NDT} and the margin value. In this RG, the ΔRT_{NDT} value is the product of a chemistry factor (CF) and a fluence factor. The fluence factor is dependent upon the neutron fluence and the CF may be determined from surveillance material or from the tables in the RG. If the RPV beltline material is not represented by surveillance material, its CF and the ΔRT_{NDT} is using the methodology documented in position 1.1 and the tables in this RG. The CF determined from the tables in the RG depends upon the amount of copper and nickel in the material. If the RPV beltline material is represented by surveillance material, its CF may be determined from the surveillance data using the methodology documented in position 2.1 of RG 1.99, Revision 2. This RG indicates that if the surveillance data gives a higher value of adjusted reference temperature (the adjusted reference temperature at EOL neutron fluence is equivalent to the RT_{PTS} value) than that given by position 1.1, the surveillance data should be used. If the surveillance data gives a lower value of adjusted reference temperature than that given by position 1.1, either may be used. Section 50.61 of 10 CFR Part 50, contains methods of determining adjusted RT_{NDT} values equivalent to RG 1.99, Revision 2.

In the licensee's MUR-PU evaluation for Pressurized Thermal Shock (PTS) the licensee stated:

> The PTS calculations were performed using the latest procedures specified by the NRC in the PTS Rule for both 36 and 57 EFPY. The calculated neutron fluence values for the updated MUR-PU condition for VEGP at 36 and 57 EFPY have exceeded the current fluences. Therefore, to evaluate the effects of the MUR-PU, the PTS values for the beltline materials from each unit were recalculated using the MUR-PU fluences. A summary of the limiting beltline material RT_{PTS} values is presented in Tables 6.1.2-4 through 6.1.2-7 [of the MUR-PU evaluation]. All RT_{PTS} values remain below the NRC screening criteria values using the projected uprated

> fluence values for 36 and 57 EFPY for VEGP (to include the extended beltline materials). The evaluation done at 57 EFPY addresses the extension to 60 years.

The licensee discussed the impact of the MUR-PU on the PTS assessment in Section 6.1.2 of the MUR-PU analysis report. Tables 6.1.2-4 through 6.1.2-7 identify the RT_{PTS} values for all RPV beltline materials for 36 and 57 EFPY of operation. The limiting material in the Vogtle 1 beltline was Intermediate Shell Plate 8805-2. The limiting RT_{PTS} value at 36 EFPY for Vogtle 1 was 117.6 °F based on using the tables in the RG. For Vogtle 2, the limiting material was Lower Shell Plate R8-1. The limiting RT_{PTS} value at 36 EFPY for Vogtle 2 was 125.9 °F based on using the tables in the RG. Both of these values comply with the limits established in 10 CFR 50.61.

3.8.3.3 Summary

The NRC staff has reviewed the licensee's evaluation of the effects of the proposed MUR-PU on the PTS evaluation for the Vogtle 1 and 2 RPVs, and concludes that the licensee has adequately addressed changes in neutron fluence and their effects. The staff further concludes that the licensee has demonstrated that the plants will continue to meet the requirements of GDC 14, GDC 31, and 10 CFR 50.61 following implementation of the proposed MUR-PU. Therefore, the NRC staff finds the proposed MUR-PU acceptable with respect to the issue of PTS.

3.8.4 Reactor Internal and Core Support Materials

3.8.4.1 Regulatory Evaluation

The RPV internals and core supports include structures, systems, and components (SSCs) that perform safety functions or whose failure could affect safety functions performed by other SSCs. These safety functions include reactivity monitoring and control, core cooling, core support, and fission product confinement (within both the fuel cladding and the RCS). The NRC staff's review covered the materials' specifications and mechanical properties, welds, weld controls, nondestructive examination procedures, corrosion resistance, and susceptibility to degradation. The NRC's acceptance criteria for RPV internal and core support materials are based on GDC 1 and 10 CFR 50.55a for material specifications, controls on welding, and inspection of RPV internals and core supports. Specific review criteria are contained in SRP, Chapter 4, Section 4.5.2.

In Section 6.2.1.2 of the MUR-PU analysis report, Reactor Internals Core Support Structures, the licensee stated that:

> Evaluations were performed to demonstrate that the structural integrity of the reactor internal components is not adversely affected by the MUR-PU conditions. The presence of heat generated in the reactor internal components, along with the various fluid temperatures, results in thermal gradients within and between the components. These thermal gradients result in thermal stresses and thermal growth that must be accounted for in the design and analysis of various components.

> The reactor internals and core support structure components subjected to heat generation effects (either directly or indirectly) are the upper core plate, the lower core plate, the core baffle plates, the former plates, the core barrel, the baffle-former bolts, and the barrel-former bolts. For these reactor internal components, the stresses and cumulative fatigue usage factors were unaffected by the MUR-PU uprate conditions. Therefore, the current analyses of record bound the MUR-PU conditions.

In its RAI response dated December 21, 2007, the licensee also stated that:

> The structural integrity of the VEGP Units 1 and 2 reactor internals design has been ensured by analyses performed on both a generic and plant-specific basis. These analyses were used as the basis for evaluating Vogtle, Units 1 and 2 reactor internal components for the MUR-PU.
>
> The criteria established by the 1974 Edition of the ASME Boiler and Pressure Vessel Code, Section III, Sub-section NG applies to these reactor internals analyses. However, it should be noted that Sub-section NG of the 1974 Edition of the ASME Code, [Section III] was not approved by the NRC for use at the time the VEGP Units 1 and 2 reactor internals were procured.
>
> Since there were no changes in the NSSS design transients and fuel assembly design, and the impact due to seismic and LOCA loads is not changing from the current analysis of record due to the MUR-PU, the reactor internals components remain bounded by the current analysis of record. Therefore, the MUR-PU does not affect the [RPV] internals for the loadings evaluated due to structure deadweight, temperature differences, flow loads, fuel assembly pre-load, control rod assembly dynamic loads, vibratory loads, and earthquake accelerations.
>
> Based on these conclusions, the current analysis of record for the reactor internals components remains applicable for the Vogtle 1 and 2 MUR-PU.

3.8.4.2 Technical Evaluation

The licensee discussed the impact of the MUR-PU on the structural integrity of the Vogtle 1 and 2 RPV internal components in Section 6.2.1.2 of the MUR-PU analysis report. The licensee discussed the effect of changes due to the MUR-PU on their evaluation of RPV internals based on loading due to structure deadweight, temperature differences, flow loads, fuel assembly pre-load, control rod assembly dynamic loads, vibratory loads, and earthquake accelerations. The evaluations were performed in accordance with the 1974 edition of Section III of the ASME Code (i.e., the construction code for Vogtle 1 and 2).

The NRC staff determined that the licensee's evaluation of the effect of the MUR-PU of the RPV internals is acceptable and that the MUR-PU will not affect the structural integrity of the internal components.

In addition, the RPV internals of PWR-designed light-water reactors may be susceptible to the following aging effects:

- cracking induced by thermal cycling (fatigue-induced cracking), stress corrosion cracking (SCC), or irradiation-assisted stress corrosion cracking (IASCC)

- loss of fracture toughness properties induced by radiation exposure for all stainless steel grades, or the synergistic effects of radiation exposure and thermal aging for cast austenitic stainless steel (CASS) grades;

- stress relaxation in bolted, fastened, keyed or pinned RPV internal components induced by radiation exposure and/or exposure to elevated temperatures void swelling (induced by radiation exposure).

In its letter dated December 21, 2007, the licensee committed to participate in the industry program for investigating and managing aging effects on RPV internals. The licensee will also evaluate and implement the results of the industry programs, such as the Electric Power Research Institute (EPRI) Material Reliability Program (MRP), as applicable to the Vogtle 1 and 2 RPV internals.

3.8.4.3 Summary

The NRC staff concludes that the licensee has demonstrated that the RPV internal and core support materials will continue to be acceptable and will continue to meet the requirements of GDC 1 and 10 CFR 50.55a following implementation of the proposed MUR-PU. Therefore, the NRC staff finds, with inclusion of the commitment noted above, the proposed MUR-PU is acceptable with respect to the maintenance of the integrity of RPV internal and core support materials.

3.8.5 Overall Summary

The NRC staff has reviewed the licensee's proposed license amendment to increase the rated core thermal power for Vogtle 1 and 2 by 1.7% (to 3625.6 MWt) and has evaluated the impact that the MUR-PU conditions will have on the structural integrity assessments for the RPV and RPV internals. The NRC staff has determined that the proposed license amendment will not significantly impact the remaining safety margins required for the following RPV and RPV internals structural integrity assessments: (1) RPV Surveillance Program for Vogtle 1 and 2 (2) USE assessment for the RPVs, (3) P-T limits for the Vogtle 1 and 2 RPVs, (4) facture evaluation of carbon steel components including SGs, (5) PTS assessment for the Vogtle 1 and 2 RPV beltline materials, and (6) structural integrity assessment of the Vogtle 1 and 2 RPV internal components.

3.9 Fire Protection

3.9.1 Regulatory Evaluation

The purpose of the fire protection program is to provide assurance, through a defense-in-depth design, that a fire will not prevent the performance of necessary plant safe-shutdown functions nor

will it significantly increase the risk of radioactive releases to the environment. The NRC staff's review focused on the effects of the increased decay heat on the plant's safe-shutdown analysis to ensure that structures, systems, and components (SSCs) required for the safe-shutdown of the plant are protected from the effects of the fire and will continue to be able to achieve and maintain safe-shutdown following a fire. The NRC's acceptance criteria for the fire protection program are based on (1) 10 CFR 50.48, "Fire protection," and associated Appendix R to 10 CFR Part 50, insofar as they require the development of a fire protection program to ensure, among other things, the capability to safely shutdown the plant; (2) General Design Criterion 3 (GDC) of Appendix A to 10 CFR Part 50, insofar as it requires that (a) SSCs important to safety be designed and located to minimize the probability and effect of fires, (b) noncombustible and heat resistant materials be used, and (c) fire detection and suppression systems be provided and designed to minimize the adverse effects of fires on SSCs important to safety; and (3) GDC 5 of Appendix A to 10 CFR Part 50, insofar as it requires that SSCs important to safety not be shared among nuclear power units unless it can be shown that sharing will not significantly impair their ability to perform their safety functions.

A revision to 10 CFR Part 50, Appendix K, effective July 31, 2000, allowed licensees to use a power uncertainty of less than 2% in design-basis loss-of-coolant accident analyses, based on the use of state-of-the-art FW flow measurement devices that provide for a more accurate calculation of reactor power. Appendix K to 10 CFR Part 50 did not originally require the reactor power measurement uncertainty be determined, but instead required a 2% margin. The revision allows licensees to justify a smaller margin for power measurement uncertainty based on power level instrumentation error. This type of change is also commonly referred to as a MUR power uprate.

Regulatory Information Summary (RIS) 2002-03, "Guidance on the Content of Measurement Uncertainty Recapture Power Uprate Applications," January 31, 2002, Attachment 1, Sections II and III (ADAMS Accession No. ML013530183), recommends that, to improve efficiency of the NRC staff's review, licensees requesting an MUR power uprate should identify current accidents and transients analyses of record which bound plant operation at the proposed uprated power level. For any design-basis accident for which the existing analyses of record do not bound the proposed uprated power level, the licensee should provide a detailed discussion of the reanalysis.

3.9.2 Technical Evaluation

The licensee developed the LAR consistent with the guidelines in RIS 2002-03. In the LAR, the licensee re-evaluated the applicable SSCs and safety analyses at the proposed MUR core power level of 3625.6 MWt against the previously analyzed core power level of 3565 MWt. The NRC staff's review of the licensee's August 28, 2007, LAR, Enclosures 1 and 5 and Table II-1, identified areas in which additional information was necessary to complete the review of the proposed MUR LAR. The licensee responded to the NRC staff's RAI as discussed below.

In Question 1, the NRC staff stated that LAR, Section II, "Accidents and Transients for Which the Existing Analyses of Record Bound Plant Operation at the Proposed Uprated Power Level," mentions safe-shutdown fire analysis. The results of the Appendix R evaluation for MUR power uprate are provided in Table II-1. However this section does not discuss the time necessary for the repair of systems required to achieve and maintain cold shutdown or the increase in decay heat generation following plant trips. The NRC staff requested the licensee to verify that, with the

increased reactor power level of 3625.6 MWt, the safe-shutdown equipment for Vogtle 1 and 2, would remain in compliance with 10 CFR Part 50, Appendix R. By letter dated December 21, 2007, the licensee provided the following response.

> Per the Vogtle supplemental safety evaluation 4, there are only two cold shutdown repairs that may be required. The first is a repair to the Unit 1 diesel fuel oil transfer pump. A control room fire may damage circuits for this pump which would prevent the emergency fuel oil day tank from being refilled. A repair is needed within 1.4 hours to provide a power source to this pump. The second repair involves installing portable ventilation equipment in the Train B Engineered Safety Feature Chiller Room. A control room fire could damage the installed ventilation in the room. At least 48 hours are available to accomplish this repair. There are no changes in the Control Building heat load; therefore, the room heat-up rate is not affected. These repairs are part of the original fire protection licensing bases. The aspects of these bases remain valid. The power uprate will not have any effect on these cold shutdown repairs.
>
> The increase in decay heat generation due to the increased power level does not impact safe-shutdown equipment. The safe-shutdown analysis assumes safe-shutdown is accomplished by only one train of equipment. Decay heat removal is still accomplished by only one train of auxiliary feed water and residual heat removal, for example. The systems necessary to achieve and maintain safe-shutdown remain unchanged. Therefore, compliance is still maintained with the existing safe shutdown analysis. No new components that have not been previously analyzed are required for safe-shutdown in any fire area; therefore, there are no impacts.

The licensee's response satisfactorily addresses the staff's concerns, and this RAI issue is considered resolved based on the following: The licensee evaluated the effect on safe-shutdown by comparing the conditions for the proposed power uprate with conditions from the analyses of record and current operating conditions. The licensee stated that systems necessary to achieve and maintain safe-shutdown remain unchanged and increase in decay heat generation due to the increase power level does not impact safe-shutdown equipment. The licensee indicated that there are two cold shutdown repairs required in the event of a fire accident scenario. The first cold shutdown repair is required to repair to the Vogtle 1 diesel fuel oil transfer pump and the second repair involves installing portable ventilation equipment in the Train B Engineered Safety Feature Chiller Room. The licensee indicated that at least 48 hours are available to accomplish these repairs. With consideration for increased decay heat, plant shutdown and cold shutdown can still be accomplished within the time requirements.

In Question 2, the NRC staff stated that LAR, Section II, "Accidents and Transients for Which the Existing Analyses of Record Bound Plant Operation at the Proposed Uprated Power Level," mentions safe-shutdown fire analysis. This section states that …"the MUR power uprate will increase the thermal and electrical power of the plant, therefore, adding heat to the plant areas. Overall temperature changes in the primary and secondary systems are very small [such that any] added heat load to the plant environment is not significant…" The NRC staff requested the licensee to provide the temperature changes in the primary and secondary systems. Further, the

staff requested the licensee to verify that additional heat in the plant environment from the MUR power uprate will not prevent required manual actions from being performed at their designated time. By letter dated December 21, 2007, the licensee provided the following response.

> Potential operator manual actions due to fires are outlined in procedure 17103A-C for fires in zones outside the Control Room and procedures 18038-1 and 18038-2 for fires inside the Control Room. The MUR power uprate does not create any adverse environmental conditions which would impact performance of these actions. Based on the engineering evaluation, there are no significant environmental changes in plant areas due to the power uprate. There are no increases in heat load in various Control Building rooms (Control Room, Cable Spreading Room, Aux Relay rooms, computer rooms, heat ventilation and air-conditioning area). There are no increases in heat load in the Auxiliary Building or Fuel Handling Building. There are no actions required to be performed inside Containment. There are potential manual operator actions performed in the main steam isolation valve and feed water penetration areas. Temperatures in these areas do not increase significantly. Main steam temperature is reduced slightly (from 543.8 °F to 543.0 °F). Feed water temperature increases slightly (from 446.3 °F to 448.2 °F). Main steam and feed water piping is insulated, therefore, any room temperature changes are minimal. Operator access to these areas is unaffected. Radiation (gamma) levels may increase in the Auxiliary Building by 5%. This increase does not affect access into any areas that may be required for performance of safe-shutdown manual actions. There are no environmental conditions that would affect the timing of the performance of these actions.

The licensee's response satisfactorily addresses the staff's concerns, and this RAI issue is considered resolved based on the following: The licensee indicated that the proposed power uprate does not impact the required operator manual action times, since the MUR power uprate does not create any adverse environmental condition which would impact performance of operator manual actions. The licensee indicated that the feedwater temperature increases slightly from 446.3 °F to 448.2 °F and main steam and feed water piping is insulated; therefore, room temperature changes are minimal. The response times are not impacted for the post-MUR power uprate conditions nor is operator access to these areas affected.

In Question 3, the NRC staff stated that LAR, Section III, "Accidents and Transients for Which the Existing Analyses of Record do not Bound Plant Operation at the Proposed Uprated Power Level," does not include any discussion regarding changes to the fire protection program or other operating conditions that may adversely impact the post-fire safe shutdown capability in accordance with 10 CFR Part 50, Appendix R. The NRC staff requested the licensee to clarify whether this LAR involves changes to the fire protection program or other operating conditions that may adversely impact the post-fire safe-shutdown capability in accordance with 10 CFR Part 50, Appendix R. The NRC staff also requested that the licensee provide the technical justification for whether and, if so, why existing analyses do not bound any impact on accidents or transients resulting from any changes. By letter dated December 21, 2007, the licensee provided the following response.

This does not involve changes to the fire protection program. Changes to operating conditions as a result of the MUR power uprate have been evaluated, and it was determined that those changes do not adversely impact the post-fire safe-shutdown capability of the plant. The following fire-event safe-shutdown plant systems were reviewed to determine if the MUR power uprate has any impact on post-fire safe-shutdown operator actions:

- Reactor Coolant System (RCS)
- Chemical and Volume Control System (CVCS)
- Main Steam System (MSS)
- Auxiliary Feedwater (AFW) System
- Residual Heat Removal (RHR) System
- Nuclear Service Cooling Water (NSCW) System
- Diesel Generator Fuel Oil Transfer System (Unit 1 only)
- Essential Safety Features Room Coolers
- Electrical Distribution System
- Containment Spray System (CSS)

The impact of fire-induced hot-shorts, open-circuits, and shorts-to-ground in the control room electrical circuitry on the ability to safely shut down the plant from outside the control room, both with and without offsite power available, was determined by a Control Room Fire Alternate Shutdown Evaluation (CRFASE). This evaluation defines fire-induced situations requiring special operating procedural requirements and identifies time constraints due to potential spurious control actions/inactions. The results of the CRFASE do not change as a result of the MUR power uprate since the power uprate does not create any new adverse conditions or significant changes to operating conditions that would impact the fire protection program or the performance of post-fire operator actions.

Potential operator manual actions due to fires are outlined in plant procedure 17103A-C for fires in zones outside the Control Room and procedures 18038-1 and 18038-2 for fires inside the Control Room. Operator access to the remote shutdown panel and other areas outside the control room remain accessible to facilitate post-fire safe shutdown operator manual actions. There are no changes in operating conditions that would affect the timing of the performance of these actions, change the safe-shutdown operator actions, or require any changes to the fire protection program.

The basis for determining any MUR power uprate impact on post-fire safe-shutdown operator actions is provided below. For the RCS, there are no changes in pressure and no changes in configuration due to the MUR power uprate; therefore, time constraints and compensatory measures related to potential RCS depressurization and loss of inventory, as a result of spurious opening of various pressurizer valves, are not affected by the MUR power uprate. For the CVCS, there are no changes in configuration, setpoints, pressure, or flow rate as a Result of the MUR power uprate. Hence, CVCS time constraints and compensatory measures related to potential pump failures, RCS inventory loss,

and pressurizer overfill as a result of spurious closure, opening, and fire-induced failure of various valves and instrumentation circuits are not affected by the MUR power uprate.

For the MSS, pressure and temperature will slightly decrease at MUR power uprate conditions. The MSS configuration is not changed for the MUR power uprate. MSS time constraints related to potential steam generator overfill and steam generator time to boil dry may change with the propensity for more rapid overcooling due to increased decay heat as a result of the MUR power uprate. MSS time constraints and compensatory measures related to atmospheric dump valves control are not changed by the MUR power uprate. The MSS compensatory measures to minimize the chances of a significant overcooling transient, as a result of steam generator overfill or boil dry, are to isolate (close) the MSIVs and MSIV bypass valves, the MFIVs and MFIV bypass valves, and the steam generator blowdown isolation valves. If, upon arrival at the shutdown panel, there is indication of uncontrolled flow into or out of any steam generator, compensatory measures consist of tripping various breakers to ensure closure of the MSIVs and MSIV bypass valves, the MFIV and MFIV bypass valves, and the steam generator blowdown isolation valves. Although the MUR power uprate may change the times related to SG overfill or boil dry, the corresponding compensatory measures are not changed and, therefore, there are no changes to operator actions.

The licensee's response satisfactorily addresses the staff's concerns, and this RAI issue is considered resolved based on the following: As stated in the above response the MUR power uprate does not involve changes to the fire protection program. Further, the licensee evaluated changes to operating conditions as a result of the MUR power uprate and determined that the changes do not adversely impact the post-fire safe-shutdown capability of the plant

The information provided in the LAR, as supplemented by the response to staff RAIs, satisfactorily demonstrates that compliance with the fire protection and safe-shutdown program will not be affected because the MUR power uprate evaluation did not identify changes to design or operating conditions that will adversely impact the post-fire safe-shutdown capability. MUR uprate does not change the credited equipment necessary for post-fire safe-shutdown nor does it require reroute of essential cables or relocation of essential components/equipment credited for post-fire safe-shutdown. The licensee has made no changes to the plant configuration or combustible loading as a result of modifications necessary to implement the MUR power uprate that affect the Vogtle 1 and 2 fire protection program. The NRC staff concludes that the proposed 1.7% power uprate will not have adverse effects on post-fire safe-shutdown capability of the plant.

3.9.3 Summary

The NRC staff has reviewed the licensee's fire-related safe-shutdown assessment and concludes that the licensee has adequately accounted for the 1.7% effects of the increased decay heat on the ability of the required systems to achieve and maintain safe-shutdown conditions. The staff finds this aspect of the capability of the associated SSCs to perform their design-basis functions at increased core power level of 3625.6 MWt acceptable with respect to fire protection.

3.10 Accident Dose and Meteorological Analyses

3.10.1 Regulatory Evaluation

According to 10 CFR 50.34, “Contents of construction permit and operating license applications; technical information," each applicant for a construction permit or operating license before January 10, 1997 (Operating Reactors), such as Vogtle shall comply with paragraph (a)(1)(i) of this section. Section 50.34 of 10 CFR Part 50, paragraph (a)(1)(i) requires that the acceptability and safety assessment of the site be evaluated in accordance with factors identified in 10 CFR 100, “Reactor Site Criteria.” Section 100.11, “Determination of exclusion area boundary (EAB), low population zone (LPZ), and population center distance,” provides the accident dose limitations that were considered by an Operating Reactor licensee.

Paragraph (a)(1) of 10 CFR 100.11, requires an exclusion area of such size that an individual located at any point on its boundary for two hours immediately following onset of the postulated fission product release would not receive a total radiation dose to the whole body in excess of 25 rem or a total radiation dose in excess of 300 rem to the thyroid from iodine exposure.

Paragraph (a)(2) of 10 CFR 100.11, requires an LPZ of such size that an individual located at any point on its outer boundary who is exposed to the radioactive cloud resulting from the postulated fission product release (during the entire period of its passage) would not receive a total radiation dose to the whole body in excess of 25 rem or a total radiation dose in excess of 300 rem to the thyroid from iodine exposure.

Appendix A, “General Design Criteria for Nuclear Power Plants,” to 10 CFR Part 50 establishes minimum requirements for the design criteria for water-cooled nuclear power plants. General Design Criteria (GDC) 19, “Control Room,” states, “Adequate radiation protection shall be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident.” A holder of an operating license issued prior to January 10, 1997, would be expected to be in compliance with GDC 19 or the equivalent draft GDC for control room (CR) habitability.

For operating reactors, the characteristics of the fission product release from the reactor core into the containment were set forth in Regulatory Guides 1.3 and 1.4 and were derived from Technical Information Document (TID) 14844, U.S. Atomic Energy Commission (AEC), 1962, “Calculation of Distance Factors for Power and Test Reactor Sites.” This source term has been used in the design-basis applications for light-water-cooled nuclear power plants. The analyses and evaluations required by 10 CFR 50.34 for an operating license are documented in the facility’s UFSAR including the source term.

The regulatory requirements for which the NRC staff based its safety review for the licensee’s accident consequence re-analysis as part of their proposed MUR amendment include the accident dose criteria in 10 CFR 100.11 and 10 CFR Part 50, Appendix A, GDC-19. The NRC staff also utilized the regulatory guidance provided in the following documents, except where the licensee proposed an acceptable alternative.

Regulatory Guide (RG) 1.195, “Methods and Assumptions For Evaluating Radiological Consequences Of Design Basis Accidents At Light-Water Nuclear Power Reactors”

RG 1.109, “Calculation of Annual Doses to Man From Routine Releases of Reactor Effluents For the Purpose Of Evaluating Compliance With 10 CFR Part 50, Appendix I”

Relevant Sections of NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants”

In NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants” (SRP), Section 6.4, for Control Room Habitability, provides guidelines defining the dose equivalency of five rem whole body as 30 rem for both the thyroid and skin dose. For licensees adopting the guidance from RG 1.196, “Control Room Habitability at Light Water Nuclear Power Reactors, Section C.4.5 of RG 1.195 states that in lieu of the dose equivalency guidelines from Section 6.4 of NUREG-0800, the 10 CFR 20.1201 annual organ dose limit of 50 rem can be used for both the thyroid and skin dose equivalent of five rem whole body.

In its determination of adequate protection, the NRC staff reviewed the licensee’s accident re-analysis for compliance with regulations, adherence to NRC acceptable accident consequence analysis assumptions and methods, and accepted precedents. The NRC staff also performed confirmatory accident dose calculations where appropriate. Finally, the NRC staff considered the current licensee’s design basis, including the Vogtle 1 and 2 UFSAR and the Vogtle 1 and 2 TSs.

3.10.2 Technical Evaluation

3.10.2.1 Background

The current licensed core power level for Vogtle 1 and 2 is 3,565 MWt. The licensee has proposed an MUR power uprate to 3,625.6 MWt. The current licensing basis (CLB) TID-14844 source term used for DBA dose consequence analysis is based on the current licensed power level. Therefore, the licensee performed a re-analysis of all the original licensed dose consequence events utilizing a TID-14844 source term calculated at 2% power increase above the current licensed limit, bounding the proposed MUR power uprate. In addition to the increase power affect on the reactor core and primary coolant source terms, the licensee considered in the calculation a fuel cycle length of 18 months and ongoing fuel cycle design variations. As part of the re-analysis for the proposed MUR, the licensee also revised the calculation methods and assumptions for the iodine appearance rates, revised decontamination factors (DCF’s), and other accident specific changes as described in the licensee’s submittal.

The current Vogtle 1 and 2 accident and transients analyzed for the radiological consequences in Section 15, “Accident Analyses” of the Vogtle 1 and 2 UFSAR include the following events:

- Steam System Piping Failure (MSLB)
- Locked Rotor Accident (LRA)
- Spectrum of RCCA Ejection Accidents (REA)
- Break in Instrumentation Line (BIL)
- Steam Generator Tube Failure (SGTF)

- Loss-of-Coolant-Accident (LOCA)
- Fuel-Handling Accident (FHA)
- Radioactive Waste Gas Decay Tank Failure
- Radioactive Liquid Waste System Leak or Failure
- Accidental Release of Liquid Effluents in Ground and Surface Water
- Loss of Non-Emergency AC

The NRC staff reviewed the technical analyses related to the radiological consequences of design basis accidents (DBAs) performed by the licensee in support of this proposed MUR license amendment. Specifically, the NRC staff focused on the licensee's revised radiation source term and dose consequence analyses for the LOCA and FHAs. In addition, the NRC staff reviewed the revised meteorological assumptions used for the revised dose consequence analysis.

3.10.2.2 DBA Radiation Source Terms

The Vogtle 1 and 2 reactor core source terms used for the MUR DBA analyses were calculated using an updated version of the ORIGEN code and the calculation incorporated the 2% power uprate (3626 MWt), an 18-month fuel cycle and extended fuel burn-up which were not reflected in the licensee's current licensing basis. Also the nuclide inventories were increased by a nominal amount to provide margin for ongoing fuel cycle design variations. The increases were 15 percent for Kr-85, 5 percent for Xe-133 and Xe-131m, and 2% for the other noble gases and for the iodines.

The primary coolant source term used for the DBA analyses was also re-calculated to include the 2% power uprate (3626 MWt), an 18-month fuel cycle and extended fuel burn-up. For dose consequence analyses in which primary coolant activity is released, the source term for iodines is based on the licensee's TS limit of 1.0 µCi/g of Dose Equivalent I-131 and reflects a fuel clad defect level of approximately 0.25% while the source term for the noble gases is based on 1.0% fuel clad defects.

The licensee proposed conservative changes to its iodine appearance rates from those in the Vogtle 1 and 2 current licensing basis (CLB). These include a letdown flow rate increase from 75 gallons per minute (gpm) to 140 gpm and an iodine removal increase assumption in the letdown line from 90% to 100%. Also the primary coolant leakage losses were revised from a leakage of 0 gpm to 12 gpm and a primary coolant water mass increase as identified in Table 7.8-3 of WCAP-16736-NP.

The reactor cleanup rate in the Vogtle 1 and 2 CLB value is 67.5 gpm, while the proposed revised licensing basis cleanup rate is 152 gpm. This is more than twice the iodine removal. With the increased cleanup, higher iodine appearance rates can be accommodated without exceeding the TS limit on iodine activity concentration. Therefore, based on the assumption that the plant is operating at the TS limit of 1.0 µCi/g of Dose Equivalent I-131, it is expected that appearance rates for the longer-lived iodines (I-131 and I-133) would increase by a factor of approximately two. The impact would be much less for the other isotopes which have short half-lives because their removal is dominated by radioactive decay. The change in water mass has only a slight impact on the calculated iodine appearance rates.

The licensee revised its radiation source terms to reflect the increase in reactor power and to reflect the higher fuel burn-up and increased fuel cycle length and change in iodine appearance rates; therefore, the NRC staff finds this acceptable for use in calculating revised radiological consequence DBA source terms for the proposed Vogtle 1 and 2 MUR amendment.

3.10.2.3 Loss-of-Coolant-Accident (LOCA)

A LOCA is the result of a pipe rupture of the RCS pressure boundary. In the Vogtle 1 nad 2 UFSAR, a major pipe break (large break) is defined as a rupture with a total cross-sectional area equal to or greater than 1.0 ft^2. This event is considered a limiting fault, an American Nuclear Society (ANS) Condition IV event, in that it is not expected to occur during the lifetime of the plant but is postulated as a conservative design basis. The limiting Vogtle 1 and 2 large-break LOCA was found to be the double-ended cold-leg guillotine (DECLG) break.

The LOCA, as with all DBAs, is a conservative surrogate accident that is intended to challenge selective aspects of the facility design. Analyses are performed using a spectrum of break sizes to evaluate reactor fuel and emergency core cooling system (ECCS) performance in accordance with 10 CFR 50.46, “Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors.”

To demonstrate that the operation of a nuclear power plant does not represent an undue radiological hazard to the public, a large-break LOCA is assumed as the design-basis case for evaluating the performance of release mitigation systems and the reactor containment and for evaluating the proposed siting of a facility in accordance with 10 CFR Part 100.

The radiological consequence DBA LOCA analysis is a deterministic evaluation based on the assumption of a major rupture of the primary RCS piping. The accident scenario assumes the deterministic failure of the ECCS to provide adequate core cooling which results in a significant amount of core damage and significant release of fission products to the containment as specified in 10 CFR 100.11 and RG 1.195.

In accordance with the TID 14844 source term it is assumed that 100% of the noble gases and 50% of the iodine equilibrium core fission product inventory are released to the containment atmosphere. The iodine and the noble gas activity are assumed to be immediately available for leakage from the containment.

Once the gaseous fission product activity is released to the containment atmosphere, it is subject to various mechanisms of removal which operate simultaneously to reduce the amount of activity in the containment atmosphere. The removal mechanisms include radioactive decay, containment sprays, deposition, and containment leakage. For the noble gas fission products, the only removal processes considered in the containment are radioactive decay and containment leakage. Credit for radioactive decay of fission products located within the containment is assumed throughout the course of the accident. Once the activity is released to the environment, no credit is taken for radioactive decay or deposition. The containment leakage to the environment is assumed to be direct and unfiltered.

In its radiological consequence DBA LOCA re-evaluation for the proposed MUR license

amendment, the licensee maintained a majority of its previously licensed calculation assumptions with the exception of the proposed 2% bounding power level increase to 3,636 MWt, a revised source term based on the increased power level, and the use of updated dose conversion factors.

In addition, the licensee made the following adjustments to its LOCA calculation model:

- Control room atmospheric dispersion factor (χ/Q) meteorological values were recalculated;
- Control room unfiltered inleakage increased from 755 cfm and 5 cfm to 835 cfm and 130 cfm (including 10 cfm ingress/egress) for an unpressurized and pressurized control room; respectively;
- Containment spray and plate out removal rates were recalculated; and
- The chemical form of iodine for elemental, organic, and particulate was changed to 91%, 4%, and 5%, respectively, in accordance with Regulatory Guide 1.195.

A. LOCA Source Term

The LOCA analysis assumes that iodine will be removed from the containment atmosphere by both containment sprays and natural diffusion to the containment walls. As a result of these removal mechanisms, a large fraction of the released activity will be deposited in the containment sump. The sump water will retain soluble gaseous and soluble fission products such as iodines and cesium, but not noble gases. The maximum decontamination factor (DF) for elemental iodine is based on the maximum iodine activity in the primary containment atmosphere when the sprays actuate, divided by the activity of iodine in the containment atmosphere remaining in equilibrium with the dissolved iodine in the containment water. This equilibrium is determined by the effective iodine partition coefficient.

The license recalculated the containment spray and plate out removal rates in conformance with NUREG-0800, Section 6.5.2, Revision 2, as described in Regulatory Guide 1.195. As stated in RG 1.195, since the wall deposition by containment sprays is modeled as a time-dependent process, such as in Revision 2 of SRP, Chapter 6, Section 6.5.2, 50% of the equilibrium radioactive iodine inventory is assumed released into the containment atmosphere and is available to be deposited on the walls on the containment. Elemental plate out using the time-dependent models in Revision 2 of SRP, Chapter 6, Section 6.5.2 may be assumed.

In conformance with RG 1.195, the licensee revised the LOCA assumptions for the chemical form of iodine. RG 1.195 requires that the radioiodine released from the RCS to the containment in a postulated accident should be 91% elemental iodine, 5% particulate iodine and 4% organic iodide. This includes releases from the gap and the fuel pellets.

B. Containment Mixing, Natural Deposition and Leak Rate

The licensee did not describe any changes to its containment spray functional licensing basis as part of the dose consequence reanalyses for the LOCA DBA. The Vogtle 1 and 2 UFSAR

describes the containment spray system model as a two-region spray model which is used to calculate the integrated activity released to the environment. The model consists of sprayed and unsprayed regions in containment and with a constant mixing rate between them. As it is assumed that there are no additional sources after the initial release of the fission products, the remaining processes are removal and transfer so that the multivolume containment is described by a system of coupled first-order differential equations.

The containment building atmosphere may be considered a single, well mixed volume if the spray covers at least 90% of the volume and if adequate mixing of unsprayed compartments can be shown. In addition, SRP, Chapter 6, Section 6.5.2, Paragraph III.1.C states, “The containment building atmosphere may be considered a single, well-mixed space if the spray covers regions comprising of at least 90% of the containment building space and if a ventilation system is available for adequate mixing of any unsprayed compartments.”
The leak rates from containment were not changed from the Vogtle 1 and 2 CLB values which are 0.2% containment volume leakage per day for the first 24 hours and 0.1% containment volume leakage per day prescribed by RG 1.195.

C. Assumptions on Engineered Safety Feature (ESF) System Leakage

The ESF System is assumed to leak during their intended period of operation. This release source includes leakage through valve packing glands, pump shaft seals, flanged connections, and other similar components. This release source may also include leakage through valves isolating interfacing systems. The radiological consequences from the postulated leakage should be analyzed and combined with consequences postulated for other fission product release paths to determine the total calculated radiological consequences from the LOCA.

All calculation parameters other than the source term values were consistent with the licensee’s CLB. The licensee assumes leakage of 2 gpm measured at 70 °F. The actual ECCS ESF leakage would not begin until after the recirculation phase of the accident begins. The licensee assumed that ESF leakage will start 30 minutes after the LOCA occurred and continue for the 30-day duration of the accident evaluation.

D. Control Room Habitability

The control room heating ventilation and air conditioning (HVAC) system maintains the control room envelope at a minimum of 0.125-inch water gauge positive pressure relative to the surrounding area, following postulated accidents with the exception of hazardous chemical/smoke releases. Control room doses are calculated for the limiting accidents grouped by the control room emergency filtration system (CREFS) initiation signal. The CREFS is initiated by a safety injection (SI) signal, for all accidents inside containment except the fuel-handling accident (FHA) or by the ventilation intake radiation monitor for accidents outside containment, which do not generate a safety injection signal. Therefore, the control room dose analysis considers the most limiting combination of release magnitude and location from accidents inside the containment and those outside the containment.

For radiological consequence DBAs inside the containment where the containment is isolated and CREFS is initiated by an SI signal, the largest release is for the DBA LOCA. The licensee

determined that the LOCA is bounding for all radiological consequence accidents for the MUR accident re-analysis. Since the licensee did not change significantly from its CLB assumptions, this is consistent with previously approved CLB results.

For accidents outside the containment, the licensee determined that the most-limiting release (largest activity release and closest to the control room) is the FHA in the fuel-handling building. The FHA in the fuel-handling building bounds the dose consequence of the FHA in the containment without isolation. Since the FHA offsite dose consequences bounded all other non-LOCA accident dose consequences in the Vogtle CLB, the licensee re-calculated control room accident doses for the LOCA and the bounding FHA only. The licensee maintained these assumptions from its CLB.

The control room HVAC system is activated on an ESF signal and/or high radiation in the outside air. The system is designed to introduce makeup air equivalent to the expected exfiltration air during plant emergency conditions.

Additionally, during postulated accident conditions, on detection of high radiation in the outside air or on an SI signal, outside makeup air for the control room envelope is automatically routed through makeup air units and cleanup units containing charcoal filters.

The licensee revised the unfiltered inleakage for the MUR DBA analysis to conform with RG 1.78 and to bound its design requirement described in UFSAR 6.4.2.4 G. For the Vogtle 1 and 2 LOCA control room habitability evaluation, the licensee assumed ½ the pressurization flow rate plus 10 cubic feet per minute (cfm) for ingress/egress. This value provides margin to the CLB control room dose analysis for the limiting accident (LOCA) which uses only 5 cfm for ingress/egress and has no margin for pressurized unfiltered inleakage. To bring the analysis into conformance with the industry practice, ingress/egress inleakage was increased to 10 cfm and the pressurized unfiltered inleakage was derived to meet GDC 19 dose requirements. The resulting value used in the updated MUR DBA licensee radiological consequence analysis (130 cfm total) for the proposed amendment bounds the licensee GL 2003-01 as-tested inleakage and provides margin for future design changes, or boundary leakage changes.

E. Summary

The licensee re-evaluated the radiological consequences resulting from the postulated radiological consequence DBA LOCA for the proposed MUR and concluded that the radiological consequences at the EAB, LPZ, and control room are within the accident dose requirements provided in 10 CFR 100.11, RG 1.195 and accident dose criteria specified in SRP Section 6.4, for Control Room Habitability.

The NRC staff's review found that the licensee used analysis assumptions and inputs consistent with applicable regulatory guidance identified in Section 3.10.1 of this Safety Evaluation and has maintained a significant portion of its CLB assumptions in their analyses. The assumptions found acceptable to the NRC staff are presented in Table 15.6.5-9 of the December 21, 2007, licensing amendment supplement and the licensee's calculated dose results are given in Table III-4 of the August 28, 2007, licensing amendment submittal as well as listed below in Table 1. The NRC staff finds that the EAB, LPZ, and control room doses revised by the licensee for the LOCA meet the

applicable accident dose criteria, and are, therefore, acceptable for the proposed MUR amendment.

3.10.2.4 Fuel Handling Accident (FHA)

An FHA involves the drop of a fuel assembly on top of other fuel assemblies during refueling operations. The Vogtle 1 and 2 CLB analyzed postulated FHAs for three cases: an FHA outside the containment in the fuel-handling building; an FHA inside the reactor containment building with the containment airlocks and equipment hatch closed; and an FHA inside the containment with the personnel airlock doors and/or the equipment hatch open. The Vogtle 1 and 2 accident is defined as the dropping of a spent fuel assembly onto another fuel assembly in the fuel storage area or refueling pool, resulting in the rupture of the cladding of all the fuel rods in the dropped assembly plus additional rods in the struck assembly.

According to the Vogtle 1 nad 2 CLB the first fuel transfer operation shall not begin until at least 100 hours after reactor shutdown. The FHA is assumed to occur after a fuel assembly has been removed from the core but before it has been placed in its designated location in the spent fuel storage racks.

The licensee had determined as part of their CLB that the most severe radiological consequences from a dose consequence DBA other than the DBA LOCA would result from the FHA in containment or in the fuel building with no isolation assumed. Further, the licensee determined that the FHA in the fuel-handling building for Vogtle 1 to the Vogtle 1 control room is bounding because the fuel-handling building χ/Q bounds those for releases from the containment and from Vogtle 2.

The barriers between the released activity and the environment for an FHA at Vogtle 1 and 2 are the containment building or the fuel building. A release of radioactivity for a postulated FHA in the fuel building would normally be via the fuel building emergency filtration system. An FHA in the containment building with the airlocks and equipment hatch closed would result in a minimal amount of radioactivity release.

For a postulated accident in the containment building with the airlock and/or equipment hatch open, the limiting pathway for the release of activity is via the equipment building ventilation fan as described in the Vogtle 1 and 2 UFSAR Section 15.7.4.

The equipment hatch and the emergency air lock are farther away from the control room air intake than the personnel airlock. Therefore, the release path from the personnel airlock remains bounding for control room dose as described in the Vogtle 1 and 2 CLB. The updated control room dose calculation for the FHA as in the CLB was done for the case where there is no isolation of containment. Offsite dose is not affected by the relative locations of the personnel and emergency airlocks, the containment purge supply and exhaust ventilation, or the equipment hatch.

A. Source Term

The dropped fuel assembly is assumed to be the assembly containing the peak fission product

inventory. All the fuel rods contained in the dropped assembly are assumed to be damaged. In addition, for the analyses of the accident in the containment building the dropped assembly is assumed to damage 50 rods of an additional assembly.

The fission product inventory that constitutes the source term for this event is the gap activity in the fuel rods assumed to be damaged as a result of the postulated design-basis FHA. Volatile constituents of the core fission product inventory migrate from the fuel pellets to the gap between the pellets and the fuel rod cladding during normal power operations. The fission product inventory in the fuel rod gap of the damaged fuel rods is assumed to be instantaneously released to the surrounding water as a result of the accident. The licensee used the fraction of fission product gases contained in the gap region of the fuel assembly as described in RG 1.195 for the MUR radiological consequence DBA FHA analysis.

The licensee stated in response to a request for additional information dated December 21, 2007, "A comparison between the assumptions used and RG 1.195 indicates the dose consequence re-analysis calculations conform to RG 1.195 guidance for the fuel handling accident." The licensee maintained their CLB assumptions including the use of the overall decontamination factor of 200 and revised the core source terms after 100 hours of decay consistent with the requirements in RG 1.195.

B. CR habitability for the FHA

As stated previously the licensee assumed from its CLB that the FHA release path from the personnel airlock remains bounding for the CR dose. The re-calculated dose consequence for the MUR radiological consequence DBA FHA bounding event remains less than the DBA LOCA control room dose consequence, which is consistent with the Vogtle CLB.

C. Summary

The licensee evaluated the radiological consequences resulting from their defined limiting postulated FHA and concluded that the radiological consequences at the EAB, LPZ, and CR are within the dose guidelines provided in 10 CFR 100.11 and RG 1.195 and accident dose criteria specified in SRP, Chapter 15, Section 15.7.4, Radiological Consequences of Fuel Handling Accidents.

The NRC staff's review has found that the licensee used analysis assumptions and inputs consistent with applicable regulatory guidance identified in Section 3.10.1 of this Safety Evaluation. The assumptions found acceptable to the NRC staff are presented in Table 15.7.4-1 of the December 21, 2007 (Reference 4), licensing amendment supplement and the licensee's calculated dose results are given in Table 7.8-8 of WCAP-16736-NP (Enclosure 10 to Reference 6) and in Table 1. The NRC staff finds that the EAB, LPZ, and CR doses estimated by the licensee for the FHA meet the applicable accident dose criteria, and are, therefore, acceptable.

3.10.2.5 Other Vogtle 1 and 2 DBA Accidents

The licensee also performed radiological consequence analyses for the following accidents for the proposed MUR power level increase. These accidents were re-analyzed to determine the revised

doses associated with the increased source terms. Control room habitability was not calculated for the following accidents, which is consistent with the Vogtle 1 and 2 CLB. The licensee considered the LOCA and FHA as the limiting radiological consequence accident combination of release magnitude and location from accidents inside the containment and those outside the containment.

- Steam System Piping Failure (MSLB)
- Locked Rotor Accident (LRA)
- Spectrum of RCCA Ejection Accidents (REA)
- Break in Instrumentation Line (BIL)
- Steam Generator Tube Failure (SGTF)
- Radioactive Waste Gas Decay Tank Failure
- Radioactive Liquid Waste System Leak or Failure
- Accidental Release of Liquid Effluents in Ground and Surface Water
- Loss of Non-Emergency AC

In response to request for additional information on December 21, 2007, the licensee described the extent at which the assumptions used for the dose consequence analysis met the requirements of RG 1.195. In most cases the accident analysis assumptions were in conformance with RG 1.195 or were assumptions approved as part of the licensee original design basis. The minor differences with RG 1.195 are not significant to the dose calculation results. The dose consequence analysis results for these accidents are bounded by the DBA LOCA for accidents inside containment and are bounded by the FHA for accidents outside containment.

A. Summary

The NRC staffs review has found that the licensee used analysis assumptions and inputs consistent with applicable regulatory guidance identified in Section 3.10.1 of this Safety Evaluation. The licensee-calculated dose consequence results for these accidents are provided in Table 7.8-7 of WCAP- 16736-NP and Table III-4 of the licensee's original application dated August 28, 2007 (Reference 6). The NRC staff did not perform independent confirmatory dose evaluations due to the limited changes associated with this accident dose re-analysis for the Vogtle 1 and 2 MUR. The NRC staff finds that the EAB, and LPZ doses estimated by the licensee for these accidents meet the applicable accident dose criteria, and are, therefore, acceptable.

3.10.3 Meteorological Analyses

3.10.3.1 Meteorological Data used for Radiological Consequences Analyses

Vogtle 1 and 2 is located 26 miles southeast of Augusta, Georgia, in NRC's Region II. For the purpose of performing radiological consequence analysis for the proposed MUR at Vogtle 1 and 2, atmospheric dispersion factors (i.e., χ/Q values) were used as input. However, new χ/Q values were derived for use in only the onsite (i.e., control room or CR) dose analysis. The licensee elected to use the currently licensed χ/Q values to reanalyze the offsite (i.e., at the EAB and outer boundary of the LPZ) dose values for both the LOCA and FHA at VEGP. These values are found in Chapter XV of the Vogtle 1 and 2 UFSAR for Vogtle 1 and 2. Although no offsite χ/Q values were revised in the radiological consequence analysis, the licensee used 3 years of onsite

meteorological data in the reevaluation of onsite χ/Q values.

The licensee used onsite hourly meteorological data collected during calendar years 1998, 1999, and 2000. The data were used as input to the ARCON96 onsite atmospheric dispersion computer code (NUREG/CR-6331, Revision 1, "Atmospheric Relative Concentrations in Building Wakes") in hourly data format to generate control room χ/Q values. Wind speed and wind direction data used in the atmospheric dispersion analyses were measured on the Vogtle 1 and 2 onsite primary meteorological tower at heights of 10.0 meters (~ 33 feet) and 46.0 meters (~ 151 feet) above ground level (AGL). Temperature sensors provided atmospheric stability data (via temperature difference) as well. The combined data recovery (i.e., validity of the collected data) of the wind speed, wind direction, and atmospheric stability data were in the upper 90th percentile during each year of the full data set for measurement levels of 10.0 meters and 46.0 meters. The NRC staff determined there was an overall data recovery rate of 98.9%.

The NRC staff performed confirmatory and quality assurance evaluations of the meteorological data presented using the methodology described in NUREG-0917, "Nuclear Regulatory Commission Staff Computer Programs for Use with Meteorological Data." Further review was performed using computer spreadsheets and the Meteorological Quality Assurance (METQA) Codes computer program. Similar results were found for each individual year of data regarding the wind speed, wind direction, and atmospheric stability data. There was an average wind speed value of 2.3 meters per second (m/s) and 4.4 m/s at the 10.0 meter and 46.0 meter heights AGL, respectively, for the three consecutive years of meteorological data presented. During each year provided, winds predominantly blew from the southwest direction at both measurement levels. Similar trends are shown for nearby Augusta, Georgia, located about 26 miles NW of the Vogtle 1 and 2 site, in the NOAA National Climatic Data Center climatological database for the period 1930 through 1996. Trends show an annual average wind speed of 6.0 mph (2.7 m/s) and prevailing winds blowing from the west-southwest direction. This is generally consistent with the data presented for the Vogtle 1 and 2 site for years 1998, 1999, and 2000.

Regarding atmospheric stability, measured as the temperature difference between the 46.0 meter and 10.0 meter heights, the time of occurrence and duration of reported stability conditions were generally consistent with expected meteorological conditions (e.g., neutral and slightly stable conditions predominated during the year with stable and neutral conditions occurring at night and unstable and neutral conditions occurring during the day).

The meteorological data presented for calendar years 1998, 1999, and 2000 were found acceptable by staff evaluation and are considered adequate for use in making atmospheric dispersion estimates used in the LOCA and FHA dose assessments performed in support of the proposed MUR power uprate at the Vogtle 1 and 2 site.

3.10.3.2 Control Room Atmospheric Dispersion Factors

The licensee generated new control room χ/Q values for postulated ground level releases from Vogtle Units 1 and 2 for the LOCA and FHA using guidance provided in RG 1.194, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants." These new χ/Q estimates were calculated using the ARCON96 onsite atmospheric dispersion computer code. RG 1.194 states that ARCON96 is an acceptable

methodology for assessing onsite χ/Q values for use in DBA radiological analyses. The NRC staff evaluated the applicability of the ARCON96 model and determined that there are no unusual siting, building arrangements, release characterization, source-receptor configuration, meteorological regimes, or terrain conditions that preclude use of this model in support of the proposed MUR power uprate for operation at Vogtle 1 and 2.

The wind speed, wind direction, and atmospheric stability measured at the 10-meter and 46-meter heights AGL served as input for the control room χ/Q calculations. Other inputs included the release/source height, the control room receptor heights, and the straight-line or taut string distance between the source and intake/receptor, all in meters, the direction between intake to source, in degrees, and the default values of 0.2 meters for surface roughness, 0.5 m/s for minimum wind speed, and sector averaging constant of 4.3 (found in Table A-2 of RG 1.194). Radioactive releases from the LOCA and FHA were assumed to discharge to the environment via three different source locations for each Vogtle 1 and 2: Vogtle 1 and 2 containment/reactor, Vogtle 1 and Vogtle 2 hatch door, and Vogtle 1 and Vogtle 2 fuel-handling building (FHB). All releases were assumed to occur at ground level for the purpose of atmospheric dispersion analyses. The primary onsite receptors used for the Vogtle 1 and 2 atmospheric dispersion evaluations were the Vogtle 1 control room intake and the Vogtle 2 control room intake. The licensee notes that all DBAs are bounded by the LOCA and thus, presents onsite χ/Q values for this event only, as noted in Table 2.

The NRC staff performed confirmatory χ/Q calculations of the Vogtle 1 FHB to Vogtle 1 control room and Vogtle 2 containment to Vogtle 2 control room source/receptor pairs. We found that the χ/Q estimates for the Vogtle 1 FHB release to the Vogtle 1 control room bounds the χ/Q estimates for the Vogtle 2 containment release to the Vogtle 2 control room. This finding was addressed by the licensee in their responses to the NRC questions submitted by letter dated December 21, 2007. The licensee states that for accidents outside the containment, the most-limiting release is the FHA in the fuel-handling building. However, the LOCA provides the largest release for accidents inside the containment (except for the FHA), where the containment is isolated and the CREFS initiates a signal. Consequently, considering these χ/Q values, the automatic control room intake radiation monitor isolation times, and the other input parameters (as described in Vogtle 1 and 2 UFSAR Tables 15.6.5.-9, 15A-1, and 15.7.4-1), the control room dose estimates for the LOCA bound the FHA as reanalyzed for this proposed MUR power uprate. Thus, assumptions made in the source term attribute to the LOCA bounding all other accidents in the dose analyses, despite the determination of higher χ/Q values for the Vogtle 1 FHB releases to the Vogtle 1 control room.

3.10.3.3 Summary

The NRC staff qualitatively reviewed the inputs to the ARCON96 calculations and found them generally consistent with the air intake locations and release points drawing and staff practice (Reference Figure 1 in Enclosure 2 to supplemental letter dated October 9, 2007). Additionally, the NRC staff performed selective confirmatory calculations of the licensee's assessments of control room post-accident dispersion conditions generated using the calendar years 1998, 1999, and 2000 meteorological data. The NRC staff has concluded that the resulting onsite χ/Q values generated by the licensee are acceptable for use in the LOCA dose assessments for the proposed increased maximum power level of 3625.6 MWt at the Vogtle 1 and 2 site.

3.10.4 Radiological Consequence Analysis Summary

The NRC staff has evaluated the licensee's revised radiological consequence DBA analysis performed in support of the proposed Vogtle 1 and 2 MUR amendment and concludes that the licensee has adequately accounted for the effects of the power uprate on the plants' CLB applicable DBA accidents. The NRC staff further concludes that the plant site and the dose-mitigating ESFs remain acceptable with respect to the radiological consequences of postulated DBAs. Since, as outlined above, the calculated thyroid and whole-body doses at the EAB, at the LPZ outer boundary, and in the control room meet the exposure guideline values specified in 10 CFR 50.100 and GDC 19, as well as applicable acceptance criteria denoted in the specified SRP Sections. Therefore, the NRC staff finds the licensee's proposed MUR amendment acceptable with respect to the radiological consequences of DBAs. The pertinent dose results for the limiting analyzed accidents are summarized in Table 1 of this Safety Evaluation.

TABLE 1
Radiological Consequences (REM)
For Vogtle MUR
Postulated Limiting Design-Basis Accidents

Accident Dose	EAB	(limit)	LPZ	(limit)	CR	(limit)
LOCA						
Thyroid Dose	84.6	(300)	124	(300)	29.7	(50)
Whole-Body Dose	2	(25)	1.5	(25)	1.0	(5)
Beta Skin Dose	4.2	(NA)	3.5	(NA)	16.7	(50)
Limiting Fuel-Handling Accident						
Thyroid Dose	49	(300)	20	(300)	29.7	(50)
Whole-Body Dose	0.23	(25)	0.1	(25)	0.3	(5)
Beta Skin Dose	NA	(NA)	NA	(NA)	14	(50)

Table 2
(From Table III-2 of Enclosure 5 to Reference 6)
Vogtle Meteorological Data

Control Room χ/Q Values (sec/m^3) for Vogtle 1 and 2

Time Interval	Revised for MUR Power Uprate*
0 - 2 Hours	1.04 E-03
2 - 8 Hours	7.10 E-04
8 - 24 Hours	3.08 E-04
24 - 96 Hours	2.69 E-04
96 - 720 Hours	2.08 E-04

* χ/Q values are for the Vogtle 2 containment to Vogtle 2 CR source/receptor pair for the limiting accident relative to the dose analyses. Licensee defines the limiting accident as the loss-of-coolant accident (LOCA). Note that these χ/Q values are not limiting for all design-basis accidents.

3.11 Plant Systems

In Section 6 of Enclosure 5 of the licensee's application dated August 28, 2007 (Reference 6), the licensee evaluated the effect of the MUR-PU on NSSS interface systems, containment systems, safety-related cooling water systems, spent fuel pool storage and cooling systems, radioactive waste systems and engineered safety features heating, ventilation, and air conditioning systems. For the above systems, the licensee stated that the changes in various parameters such as flows, flow capacities, setpoints, required closure times, pressures and temperatures due to MUR-PU are bounded by the system design bases and analyses, and do not adversely affect system components or functions. All these systems remain bounded by the existing design bases and are capable of performing their design functions at MUR-PU conditions. The licensee also stated that all the above systems will continue to perform their design function at MUR-PU conditions.

In summary, the licensee reviewed the design and operation of the above plant systems and determined that the proposed MUR-PU does not adversely impact any of the systems. Based on the guidance in SRP, Chapters 3, 6, 9, 10, and 11, and RIS 2002-03, Attachment 1, Sections II, III, and VI, the NRC staff concurs with the licensee's conclusion and finds that the above plant systems will be acceptable to the MUR-PU.

In Enclosure 1 of the licensee's letter dated December 21, 2007 (Reference 4), the licensee made it clear that Caldon Check-Plus LEFM, high pressure turbine, heater drain pumps and motors, steam generator steam pressure transmitters and performance improvement of the steam dump valves for Vogtle 1 and 2 will be or were installed/made under 10 CFR 50.59 process. The review of these changes is not in NRC scope.

4.0 STATE CONSULTATION

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

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to the installation or use of facility components located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendments involve no adverse increase in the amounts and no adverse change in the types of any effluents that may be released offsite and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding (72 FR 65372). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

6.0 CONCLUSION

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

7.0 REFERENCES

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2. Stuart A. Richards (NRC) letter to Michael A. Krupa (Entergy), “Review of Caldon, Inc. Engineering Report ER-157P,” dated December 20, 2001 (ADAMS Accession No. ML013540256).

3. L. M. Stinson (Southern Nuclear Operating Company, Inc. (SNC)) to USNRC, “Vogtle Electric Generating Plant, Request to Change Licensed Maximum Power Level,” Measurement Uncertainty Recapture Power Uprate, dated August 28, 2007 (ADAMS Accession No. ML072470688). L. M. Stinson (SNC) to USNRC, “Vogtle Electric Generating Plant, Supplemental Information Supporting Review of the Measurement Uncertainty Recapture Power Uprate, Amendment Request, dated January 30, 2008 (ADAMS Accession No. ML080300576).

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8. "Supplement to Topical Report ER-80P: Basis for a Power Uprate with the LEFM Check System," Caldon Engineering Report ER-160P, May 2000, Rev. 0 (ADAMS Accession No. ML003723785).

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11. "Caldon Ultrasonics, Engineering Report: ER-551P Rev.1, LEFM✔ + Transducer Installation Sensitivity," ML071500360 (proprietary, non-proprietary is ML072740228), March, 2007.

12. "Flow Measurement Uncertainty Due To Transducer (Re)placement in Caldon® LEFM Check and CheckPlus Systems," Caldon Ultrasonics, PR-612P, Rev. 0, ML070870441 (Proprietary, non-proprietary is ML070870435), March 15, 2007.

13. "LEFM✔ + Meter Factor Calculation and Accuracy Assessment for Vogtle Electric Generating Plant Unit 1 and Unit 2 (Alden Reports No. 2005-235/C0730)," Caldon Ultrasonics Engineering Report ER-565 Rev. 3, (proprietary), May, 2007 (ADAMS Accession No. ML072470701).

14. Cameron Engineering Report ER-477, "Bounding Uncertainty Analysis for Thermal Power Determination at Vogtle Electric Generating Plant Unit 1 Using the LEFM✔+ System," Caldon Ultrasonics, Revision 5, (proprietary), May 2007 (ADAMS Accession No. ML072470699).

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