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SOUTHERN COMPANY
Energy to Serve Your World℠

August 28, 2007

Docket Nos.: 50-424
50-425

NL-07-1020

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D. C. 20555-0001

Vogtle Electric Generating Plant
Request to Change Licensed Maximum Power Level

Ladies and Gentlemen:

In accordance with the requirements of 10 CFR 50.90, Southern Nuclear Operating Company (SNC) proposes to revise the "Maximum Power Level" in paragraph 2.C(1) of the Vogtle Electric Generating Plant (VEGP) Facility Operating Licenses NPF-68 and NPF-81 for Unit 1 and Unit 2, respectively. In addition, SNC proposes to revise the definition of "Rated Thermal Power" in Technical Specification 1.1 for both units to reflect the change to the Maximum Power Level. The Maximum Power Level and Rated Thermal Power (RTP) represent the same reactor core power level.

SNC proposes to increase the RTP of VEGP Units 1 and 2 from 3565 MWt to 3625.6 MWt. The proposed change is an increase in RTP of 1.7% from the current reactor output. The proposed increase in reactor output is the result of utilizing margin in the analyses for the Emergency Core Cooling System (ECCS) Evaluation Model as described in 10 CFR 50, Appendix K. In accordance with Appendix K, the proposed change justifies assuming an alternative power level other than 102% of RTP in the ECCS analyses based on reduced uncertainty in the power level measurement. This increase in reactor core power level is referred to as a Measurement Uncertainty Recapture (MUR) power uprate. The proposed change also includes a change to the definition of Dose Equivalent Iodine (DEI) in Technical Specification 1.1 as well as a change to the Power Range Neutron Flux P-9 permissive nominal setpoint and allowable value in Technical Specification 3.3.1. Changes to Technical Specification Bases affected by the MUR power uprate are also included.

SNC will install the Caldon Check-Plus ultrasonic feedwater flow element into both units to reduce the uncertainty in the feedwater flow measurement. This reduced uncertainty, in combination with other uncertainties, results in an overall power level measurement uncertainty of 0.3% RTP. The remaining margin of 1.7% RTP forms the basis for the proposed MUR power uprating of 1.7% RTP. In addition to this modification to the plant, the high-pressure turbines and heater drain pumps will be replaced in both units to accommodate the additional flow requirements as a result of the MUR power uprate.

A001

NRR

Steam generator pressure transmitters will be replaced to be consistent with the power level measurement uncertainty analyses.

MUR power uprates have been approved by the NRC for several plants, including SNC Hatch Nuclear Plant Units 1 and 2 on September 23, 2003. Most recently, the NRC approved the request for a 1.7% MUR power uprate for Seabrook Station Unit 1 on May 22, 2006 (Amendment 110 to Facility Operating License NPF-86). Seabrook is a four-loop Westinghouse PWR similar to VEGP Units 1 and 2. The Caldon Check-Plus ultrasonic feedwater flow element, as proposed for VEGP, has been installed at Seabrook.

Enclosure 1 provides a description and justification for the proposed change. Enclosure 2 contains the 10 CFR 50.92 significant hazards evaluation and justification for the categorical exclusion from performing an environmental assessment. Enclosure 3 provides the marked-up Operating License, Technical Specification, and Bases pages. Enclosure 4 provides the clean typed Operating License, Technical Specification, and Bases pages. Enclosure 5 contains the summary of the engineering reviews performed to support the MUR. Enclosures 6 and 7 contain the Caldon power measurement uncertainty analysis reports for Unit 1 and Unit 2, respectively. Enclosure 8 contains the Caldon ultrasonic feedwater flow element meter factor test report for both units. Enclosure 9 contains Westinghouse WCAP-16736-P (Revision 1) with the results of the Westinghouse evaluations. Enclosure 10 contains WCAP-16736-NP (Revision 1) which is a non-proprietary version of Enclosure 9.

This license amendment request does not affect, nor is affected by, any other request currently under review by the Staff. However, the MUR power uprate will result in changes to some Emergency Action Level Setpoints. SNC will submit, under separate cover, proposed changes to the VEGP Emergency Plan for NRC approval. Approval will be required prior to implementation of the MUR power uprate.

SNC requests approval of this application by March 1, 2008. The proposed implementation will be during the Unit 1 refueling outage in Spring 2008 and the Unit 2 refueling outage in Fall 2008. The Spring 2008 refueling outage is currently scheduled to begin on March 16, 2008.

In accordance with 10 CFR 50.91, a copy of this letter and all applicable enclosures will be sent to the designated official of the Environmental Protection Division of the Georgia Department of Natural Resources.

Enclosures 6, 7, and 8 contain proprietary information. These enclosures also contain applications for withholding by Cameron International Corporation, a Delaware Corporation (herein called "Cameron") on behalf of its operating unit, Caldon Ultrasonics Technology Center, signed by Cameron, the owner of the information. The affidavits set forth the basis upon which the information may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in paragraph (b)(4) of 10 CFR 2.390 of the Commission's regulations. Accordingly, it is respectfully requested that the information, which is proprietary to Cameron, be withheld from public disclosure in accordance with 10 CFR 2.390 of the Commission's regulations. The applications for withholding proprietary information, along with affidavits CAW-07-08, CAW-07-10, and CAW-07-09, are contained in Enclosures 6, 7, and 8, respectively.

Enclosure 9 contains proprietary information. This enclosure also contains an application for withholding by Westinghouse Electric Company LLC, a Pennsylvania Corporation (herein called "Westinghouse"), signed by Westinghouse, the owner of the information. The affidavit sets forth the basis upon which the information may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in paragraph (b)(4) of 10 CFR 2.390 of the Commission's regulations. Accordingly, it is respectfully requested that the information, which is proprietary to Westinghouse, be withheld from public disclosure in accordance with 10 CFR 2.390 of the Commission's regulations. The application for withholding proprietary information, along with affidavit CAW-07-2294, is contained in Enclosure 9.

This letter contains no regulatory commitments. If you have any questions, please advise.

Mr. L. M. Stinson states he is a Vice President of Southern Nuclear Operating Company, is authorized to execute this oath on behalf of Southern Nuclear Operating Company and to the best of his knowledge and belief, the facts set forth in this letter are true.

Respectfully submitted,

SOUTHERN NUCLEAR OPERATING COMPANY

L. M. Stinson

L. M. Stinson
Vice-President - Fleet Operations Support

Sworn to and subscribed before me this 28th *day of* August , *2007.*

Gail A. Hicks
Notary Public

My commission expires: July 5, 2010

GAIL A. HICKS NOTARY PUBLIC ALABAMA STATE AT LARGE

LMS/RJF/daj

Enclosures:
1. Description and Justification for Change
2. 10 CFR 50.92 Significant Hazards Evaluation and Environmental Assessment
3. Marked-Up License, Technical Specifications, and Bases Pages
4. Clean Typed License, Technical Specifications, and Bases Pages
5. Summary of Engineering Reviews
6. Caldon Uncertainty Analysis Report for Unit 1 with Affadavit (Proprietary)
7. Caldon Uncertainty Analysis Report for Unit 2 with Affadavit (Proprietary)
8. Caldon Meter Factor Report with Affadavit (Proprietary)
9. Westinghouse Licensing Report WCAP-16736-P with Affadavit (Proprietary)
10. Westinghouse Licensing Report WCAP-16736-NP (Nonproprietary)

cc: Southern Nuclear Operating Company
Mr. J. T. Gasser, Executive Vice President
Mr. T. E. Tynan, Vice President - Vogtle
Mr. D. H. Jones, Vice President – Engineering
RType: CVC7000

U. S. Nuclear Regulatory Commission
Dr. W. D. Travers, Regional Administrator
Mr. S. P. Lingam, NRR Project Manager – Vogtle
Mr. G. J. McCoy, Senior Resident Inspector – Vogtle

State of Georgia
Mr. N. Holcomb, Commissioner – Department of Natural Resources

Enclosure 1
Vogtle Electric Generating Plant
Request to Change Licensed Maximum Power Level

Description and Justification for Change

Description and Justification for Change

DESCRIPTION OF PROPOSED CHANGE

In accordance with the requirements of 10 CFR 50.90, Southern Nuclear Operating Company (SNC) proposes to revise the "Maximum Power Level" in paragraph 2.C(1) of the Vogtle Electric Generating Plant (VEGP) Facility Operating Licenses NPF-68 and NPF-81 for Unit 1 and Unit 2, respectively. In addition, SNC proposes to revise the definition of "Rated Thermal Power" in Technical Specification 1.1 for both units to reflect the change to the Maximum Power Level. The Maximum Power Level and Rated Thermal Power (RTP) represent the same reactor core power level.

SNC proposes to increase the RTP of VEGP Units 1 and 2 from 3565 MWt to 3625.6 MWt. The proposed change is an increase in RTP of 1.7% from the current reactor output. The proposed increase in reactor output is the result of utilizing margin in the analyses for the Emergency Core Cooling System (ECCS) Evaluation Model as described in 10 CFR 50, Appendix K. In accordance with Appendix K, the proposed change justifies assuming an alternative power level other than 102% of RTP in the ECCS analyses based on reduced uncertainty in the power level measurement. This increase in reactor core power level is referred to as a Measurement Uncertainty Recapture (MUR) power uprate. The proposed change also includes a change to the definition of Dose Equivalent Iodine (DEI) in Technical Specification 1.1 as well as a change to the Power Range Neutron Flux P-9 permissive nominal setpoint and allowable value in Technical Specification 3.3.1. Changes to Technical Specification Bases affected by the MUR power uprate are also included.

SNC will install the Caldon Check-Plus ultrasonic feedwater flow element into both units to reduce the uncertainty in the feedwater flow measurement. This reduced uncertainty, in combination with other uncertainties, results in an overall power level measurement uncertainty of 0.3% RTP. The remaining margin of 1.7% RTP forms the basis for the proposed MUR power uprating of 1.7% RTP. In addition to this modification to the plant, the high-pressure turbines and heater drain pumps will be replaced in both units to accommodate the additional flow requirements as a result of the MUR power uprate. Steam generator pressure transmitters will be replaced to be consistent with the power level measurement uncertainty analyses.

JUSTIFICATION FOR PROPOSED CHANGE

Current Licensing Basis Power Level

The current licensing basis reactor power level is 3565 MWt. The originally licensed reactor power level was 3411 MWt. License Amendments 60 (Unit 1) and 39 (Unit 2) were approved on March 22, 1993, to increase the reactor power level to 3565 MWt. This uprating was a "stretch" power uprate.

Description and Justification for Change

Summary of Operating License, Technical Specification, and Bases Changes

Operating License - Maximum Power Level

Paragraph 2.C(1), "Maximum Power Level," of the Unit 1 and Unit 2 Operating Licenses (license numbers NFP-68 and NFP-81, respectively) authorizes the facility to operate at a reactor core power level not in excess of 3565 MWt. The proposed change is to increase the Maximum Power Level from its current value of 3565 MWt to 3625.6 MWt.

Detailed evaluations and analyses were performed that demonstrated that operation of VEGP at a reactor power level of 3625.6 MWt is acceptable. The detailed evaluations and analyses considered the effects of operation at this power level on: power level measurement uncertainty; accidents and transients; mechanical, structural, and material components integrity and design; electrical equipment design; system design; operator actions, emergency and abnormal operating procedures, control room, plant reference simulator, and operator training; environment; individual and occupational exposures; and, Technical Specifications, protection system settings, and emergency system settings.

The evaluations and analyses were performed using the current licensing basis acceptance criteria and Technical Specifications with the exception of the change to the P-9 nominal setpoint which is discussed below. This assures the same level of protection for public health and safety at the uprated conditions as at the currently licensed power level. These evaluations and analyses are described in Enclosure 5.

Enclosure 3 shows the mark-up to the Operating Licenses and Enclosure 4 shows the clean typed changes.

Technical Specification 1.1 - Definition of Dose Equivalent Iodine (DEI)

The current definition of Dose Equivalent Iodine (DEI) states that the thyroid dose conversion factors for calculating DEI "... shall be those listed in Table E-7 of Regulatory Guide 1.109, Rev. 1, NRC, 1997." For the MUR power uprate analyses, Environmental Protection Agency (EPA) Federal Guidance Report 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," EPA-520/1-88-020, September 1998, was used to determine the thyroid dose conversion factors. The definition of DEI is being revised to replace the reference to the Regulatory Guide with the reference to the EPA report. The change in licensed power level requires that the radiological consequences of the accidents analyzed in Chapter 15 of the VEGP Updated Final Safety Analysis Report (UFSAR) be re-analyzed due to the increase in source terms. As part of this re-analysis, the dose conversion factors (DCFs) were reviewed. The current Technical Specification DEI definition uses Regulatory Guide 1.109 Revision 1; however, VEGP UFSAR has used thyroid DCFs from a variety of acceptable sources, i.e., TID-14844, ICRP-2 and ICRP-30, Regulatory Guide 1.109, and Federal Guidance Report No. 11. All of these DCFs have been considered acceptable at various times as described in Regulatory Issue Summary (RIS) 2001-19. In order to provide consistency among all the dose analyses, the DCFs from Federal Guidance Report No. 11 have been used for the MUR power uprate dose re-analyses. Consequently, the Technical Specification definition for DEI is being revised to also use Federal Guidance Report No. 11, as suggested by RIS 2001-19.

Enclosure 3 contains the mark-ups to the Technical Specifications and Enclosure 4 contains the clean typed changes.

<u>Technical Specification 1.1 - Definition of Rated Thermal Power</u>

The definition of Rated Thermal Power (RTP) in Technical Specification also limits the reactor core power level to 3565 MWt. The MUR power uprate is equivalent to a 1.7% increase in the current RTP. The definition of RTP is being revised to change the value from 3565 MWt to 3625.6 MWt to be consistent with the change to the Maximum Power Level in Paragraph 2.C(1) of the Facility Operating Licenses.

Refer to the discussion above for the maximum Power Level.

Enclosure 3 shows the mark-up to the Technical Specifications and Enclosure 4 shows the clean typed changes.

<u>Technical Specification 3.3.1, Table 3.3.1-1, Function 16 - P-9 Setpoint</u>

The VEGP units are designed with 50% load rejection capability. The Westinghouse design criterion is that load rejections up to 50% should not require a reactor trip if all other control systems function properly. Therefore, Westinghouse has implemented an interlock that would eliminate direct reactor trips below 50% RTP, thereby decreasing unnecessary challenges to the reactor protection system. The interlock is the permissive P-9. The current nominal setpoint is 50% RTP.

The NRC expressed concerns regarding the potential increase in the probability of a stuck-open pressurizer power operated relief valve (PORV) following the implementation of a turbine trip without a reactor trip below 50% power. The NRC position is addressed in NUREG-0737, Item II.K.3.10. In response to this concern, during the initial licensing of the VEGP units, it was demonstrated that for normal operation with all normal control systems assumed operational, the implementation of the P-9 permissive will not result in the opening of the pressurizer PORVs. Further, by assuming a single failure in the control system, it was demonstrated that the implementation of the P-9 permissive will not result in opening the pressurizer PORVs. This is discussed in Section 6.3 of the

Safety Evaluation Report (NUREG-1137, June 1985) for the initial licensing of the VEGP units. These conclusions were confirmed for the VEGP stretch power uprate previously approved on March 22, 1993.

Because of the slightly lower steam dump system capacity at the MUR power uprate conditions, the P-9 setpoint analyses were revised. It was demonstrated 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 a reduced P-9 nominal setpoint of 40% RTP, the PORVs will not be challenged.

The P-9 nominal setpoint in Technical Specification Table 3.3.1-1 is being revised to 40% RTP and the Allowable Value is being revised to 40.6% RTP.

Section 7.11 of Enclosures 9 and 10 contains a detailed discussion of the analyses for the P-9 setpoint under MUR power uprate conditions.

On August 24, 2006, the NRC issued Regulatory Issue Summary (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." This RIS discusses issues that could occur during testing of limiting safety system settings (LSSSs) and which, therefore, may have an adverse effect on equipment operability. This RIS also presents an approach, found acceptable to the NRC staff, for addressing these issues for use in licensing actions that require prior NRC staff approval. The P-9 setpoint and allowable value does not protect either a core safety limit or the reactor coolant system pressure safety limit. The Vogtle TS specify the reactor core safety limits in TS 2.1.1, and the associated Bases state that automatic enforcement of the reactor core safety limits is provided by the following functions: high pressurizer pressure trip; low pressurizer pressure trip; overtemperature delta-T trip; overpower delta-T trip; power range neutron flux trip; reactor coolant flow trips; and the main steam safety valves. The reactor coolant system pressure safety limit is specified in TS 2.1.2, and the associated Bases state that the reactor coolant system pressurizer safety valves, the main steam safety valves, and the reactor high pressure trip ensure that the reactor coolant system safety limit will not be exceeded. Further, the Bases state that the safety analyses for both the high pressure trip and the pressurizer safety valves do not take credit for the operation of PORVs or the steam dumps, among other things. Therefore, 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 an LSSS that does not protect a safety limit but does perform a significant safety function. In this case, following surveillance testing, resetting the trip setpoint to within the setting tolerance of the nominal trip setpoint (specified in the Vogtle Units 1 and 2 TS Nominal Trip Setpoint column of LCO 3.3.1, Table 3.3.1-1) will ensure that P-9 performs its specified safety function. In fact, the Vogtle TS LCO 3.3.1, Table 3.3.1-1 contains a note applied to the Nominal Trip Setpoint column of the table that states, in part:

> "A channel is OPERABLE with an actual Trip Setpoint value outside its calibration tolerance band provided the Trip Setpoint value is conservative with respect to its associated Allowable Value and the channel is readjusted to within the established calibration tolerance band of the Nominal Trip Setpoint."

The P-9 allowable value was determined by the Rack Calibration Accuracy (RCA) for the process racks in accordance with Westinghouse Technical Bulletin ESBU-TB-97-02. The RCA for the process racks is + or - 0.5% of span (0-120 %) or 0.6 %. The allowable value for P-9 is therefore the upper limit of the calibration tolerance which is 40.6 %. Therefore, when the trip setpoint is set to within the calibration tolerance band around the nominal value, as required by TS Table 3.3.1-1, it will be conservative with respect to the allowable value. Channels that experience excessive drift (i.e., outside the calibration tolerance band) are entered into the Vogtle Corrective Action Program, evaluated, and appropriate corrective action(s) are determined. Therefore, the requirements of 10 CFR 50.36(c)(1)(ii)(A) are met, the intent of the RIS is met, and no further action is necessary.

Enclosure 3 contains the mark-ups to the Technical Specifications and associated Bases and Enclosure 4 contains the clean typed changes.

<u>Bases B3.3.1 - RTS Instrumentation</u>

Surveillance Requirement (SR) 3.3.1.2 required the comparison of the calorimetric heat balance calculation to the power range channel output and an adjustment of the channel output if the heat balance results exceed the channel output by more than +2% RTP. As discussed in the Bases B3.3.1, this SR does not preclude adjusting the channel in the downward direction if the heat balance result is less than the channel output. At part-power, this may be non-conservative due to the increased uncertainty in the calorimetric heat balance calculation. This, in turn, is due to the increased uncertainty in the feedwater flow measurement based on the venturi feedwater flow elements. To compensate for this, the Power Range Neutron Flux - High Bistable setpoint must be reduced. With the installation of the ultrasonic feedwater flow elements, the uncertainty in the feedwater flow measurement is not significantly affected at part-power.

However, while the part-power calorimetric uncertainty based on a feedwater flow measurement from the leading-edge flow meter (LEFM) is less than that based on the feedwater venturi, it is prudent to continue to apply the same adjustments to the setpoint. The Bases for SR 3.3.1.2 are being revised to reflect this.

In addition, the list of references in the Bases B3.3.1 is being revised to include a reference to the amendments associated with the revised P-9 setpoint and allowable value.

Enclosure 3 contains the mark-ups to the Bases and Enclosure 4 contains the clean typed changes.

<u>Bases B3.7.1 - Main Steam Safety Valves (MSSVs)</u>

The MSSV relieving capacity is 117% of the current rated steam flow at 110% of the steam generator design pressure. With the increase in rated steam flow, the actual relieving capacity will change from 117% to 114% of rated steam flow due to the MUR power uprate. This continues to meet the Westinghouse design criterion for the relieving capacity of the MSSVs. The Bases B3.7.1 are being revised to reflect this change.

Section 5.2 of Enclosure 8 contains a detailed discussion of the evaluation of MSSV capacity under MUR power uprate conditions.

Enclosure 3 contains the mark-ups to the Bases and Enclosure 4 contains the clean typed changes.

Regulatory Basis for Proposed Changes

The requirements for the ECCS Evaluation Models are set forth in Code of Federal Regulations (CFR) 10 CFR 50, Appendix K. The NRC approved a change to these requirements (Federal Register Notice 65 FR 34913, June 1, 2000) that provides licensees with the option of maintaining the 2% power measurement uncertainty in the ECCS analyses or using a lower value provided the proposed alternative value has been demonstrated to account for the uncertainties due to power level instrumentation error.

SNC has chosen to exercise this option by installing the Caldon Check-Plus ultrasonic feedwater flow element to reduce the uncertainty in the feedwater flow measurement. This reduced uncertainty, in combination with other uncertainties, resulted in an overall power level measurement uncertainty of 0.3% RTP. The remaining margin of 1.7% RTP forms the basis for the proposed MUR power uprating of 1.7% RTP.

Regulatory Precedence

MUR power uprates have been approved by the NRC for several plants, including SNC Hatch Nuclear Plant Units 1 and 2 on September 23, 2003. Most recently, the NRC approved the request for a 1.7% MUR power uprate for Seabrook Station Unit 1 on May 22, 2006 (Amendment 110 to Facility Operating License NPF-86). Seabrook is a four-loop Westinghouse PWR similar to VEGP Units 1 and 2 and has installed the Caldon Check-Plus ultrasonic feedwater flow element, as proposed for VEGP.

Summary of Engineering Review

SNC performed a systematic review of the current licensing basis safety analyses and design bases to demonstrate the ability of the plant to operate at the MUR power uprate conditions. The review followed the guidance in NRC Regulatory Information Summary (RIS) 2002-03, "Guidance on the Content of Measurement Uncertainty Recapture Power Uprate Applications."

In order to implement the MUR power uprate, two major plant design changes are required:

1. Installation of the Caldon Check-Plus LEFM, and
2. Replacement of the high pressure turbine.

The installation of the Caldon Check-Plus LEFM provides a measured feedwater flow with reduced uncertainty in comparison to the uncertainty in the venturi-based feedwater flow measurement. This, in turn, results in a reduced uncertainty in the power calorimetric heat balance calculation. The resulting uncertainty in the power calorimetric heat balance calculation is 0.3% RTP. The uncertainty analyses are discussed in Section I of Enclosure 5 and in Enclosures 6 and 7.

With the present turbine, the volumetric flow limit of the turbine has been reached, resulting in operation with all four turbine control valves fully open. Under the MUR power uprate conditions, the steam mass flow rate will increase and the steam pressure will decrease. This results in an increased volumetric flow rate. The replacement high pressure turbines have been designed with increased volumetric flow capacity.

The current heater drain system pumping capability is marginal. Though the pumps and motors can perform their function at the MUR power uprate conditions, SNC plans to replace the pumps and motors to improve system performance at these conditions. In addition, SNC will complete an ongoing program to replace the steam generator pressure transmitters. The replacement transmitters are assumed in the revised power calorimetric uncertainty analyses for the MUR power uprate. SNC will also complete an ongoing program to improve steam dump valve performance to ensure that the valve stroke and trip-open time are consistent with that assumed in the applicable analyses.

The above-described plant design changes apply to both units.

The detailed engineering review followed the guidance in NRC RIS 2002-03. The major areas of review included:

1. Feedwater flow measurement technique and power measurement uncertainty
2. Accidents and transients bounded by the existing analyses
3. Accidents and transients not bounded by the existing analyses
4. Mechanical, structural, and material components integrity and design
5. Electrical equipment design
6. System design
7. Reviews including operator actions, emergency operating procedures, etc.
8. Changes to Technical Specifications, protection system settings, and emergency system settings

The outcome of the engineering review was that VEGP Units 1 and 2 are capable of operating safely at the MUR power uprate conditions.

The details of the engineering review are included in Enclosure 5.

Regulatory Analysis

The MUR power uprate engineering reviews discussed in Enclosure 5 considered the applicable requirements of 10 CFR 50.36(c)(2)(ii), 50.46, 50.48, 50.49, 50.61, 50.62, 50.63, 50.71(e), Appendix A of 10 CFR 50, and Appendix K of 10 CFR 50. The reviews concluded that the applicable requirements continue to be met.

NON-RADIOLOGICAL ENVIRONMENTAL EVALUATION

Background

VEGP is proposing a MUR power uprate on both Unit 1 and Unit 2. Section 3.1 of the VEGP Environmental Protection Plan (EPP), Appendix B to Facility Operating Licenses NPF-68 and NPF-81, states that:

> *"The licensee may make changes in plant design or operation or perform tests or experiments affecting the environment provided such activities do not involve an unreviewed environmental question and do not involve a change in the EPP."*

Section 3.1 requires that an environmental evaluation be prepared and recorded prior to engaging in any activity which may significantly affect the environment. Section 3.1 further states that:

> *"A proposed change, test or experiment shall be deemed to involve an unreviewed environmental question if it concerns: (1) a matter which may result in a significant increase in any adverse environmental impact previously evaluated in the FES-OL, environmental impact appraisals, or in any decisions of the Atomic Safety and Licensing Board; or (2) a significant change in effluents or power level; or (3) a matter, not previously reviewed and evaluated in the documents specified in (1) of this Subsection, which may have a significant adverse environmental impact."*

In accordance with the above requirements, an environmental evaluation assessing the environmental impact of the proposed MUR power uprate has been performed. This evaluation confirms that the proposed MUR power uprate will not result in a significant adverse environmental impact.

Evaluation

The VEGP Final Environmental Statement (FES) (NUREG-1087, March 1985) evaluates the non-radiological impact of operation of the two units at VEGP. In support of the VEGP EPP, the parameters evaluated in the Environmental Report – Operating License Stage, the subsequent FES, and current operating parameters were reviewed relative to the proposed MUR power uprate to determine if the MUR power uprate is significant relative to adverse environmental impact.

The proposed MUR power uprate will produce slight changes in operational parameters, particularly those parameters associated with the circulating water system. There will be no new non-radiological effluent types or changes in previously evaluated non-radiological effluent types.

Due to the slightly increased heat load on the circulating water system, additional cooling tower makeup will be required to replace evaporated cooling water. A slight increase in cooling tower blowdown will also be required to maintain the proper cycles of concentration (COC) with the higher evaporative loss. The additional circulating water system cooling tower makeup needed has been projected to be 533 gpm per unit (1,066 gpm total). This is the only item resulting in an increased demand on surface water withdrawal. The FES evaluated an average river withdrawal rate of 40,000 gpm to be used for cooling tower makeup (at 4 COC). The projected average cooling tower makeup for the uprated conditions is 38,400 gpm, which is bounded by the FES. The increased surface water withdrawal also remains below the permitted limits established in the surface water withdrawal permit issued to VEGP by the State of Georgia.

Groundwater withdrawal will only be slightly impacted by the proposed MUR power uprate. Makeup to the Nuclear Service Cooling Water (NSCW) cooling towers will increase by 6 gpm per tower (24 gpm total). This de minimus increase is well within the permitted limits established by the VEGP groundwater withdrawal permit issued by the State of Georgia.

Most of the liquid effluent streams at VEGP will be unaffected by the proposed MUR power uprate. No change is anticipated in the flow or non-radiological characteristics of the following effluents: sewage treatment plant effluent, waste water retention basin (WWRB) effluent, water treatment plant production water and reject water, and radwaste dilution flow. Additionally, no change in cooling tower chemical treatment or noise levels is anticipated due to the proposed power uprate. No new non-radiological effluents will be created as a result of the MUR power uprate.

The effluents most impacted by the proposed power uprate are circulating water cooling tower blowdown and consumptive loss. NSCW cooling towers are minimally impacted as discussed above. Cooling tower blowdown flow is projected to increase by 133 gpm per unit, and the blowdown temperature is projected to increase by 0.1 °F. The increased blowdown flow is bounded by the FES, and the 0.1 °F temperature increase is bounded by engineering evaluations performed during VEGP licensing. The changes in cooling tower blowdown are not significant compared to the previously evaluated environmental impacts of VEGP. The final plant effluent to the Savannah River will continue to be discharged in accordance with National Pollutant Discharge Elimination System (NPDES) permit limitations and established water quality criteria.

Consumptive loss from the circulating water cooling towers is the sum of the water discharged to the atmosphere via evaporation and drift. Drift loss from the cooling towers remains unchanged at 0.015% of the cooling tower flow. The evaporative loss has been projected to increase by up to 400 gpm per unit due to the increased heat load on the circulating water system. This projected increase will result in a total consumptive use of 14,481 gpm per unit, which is bounded by the FES average value of 15,000 gpm per unit. Therefore, the change in consumptive loss from the circulating water cooling towers is not significant when compared to previously evaluated environmental impacts of VEGP.

Conclusion

Based on the above information, the plant operating parameters impacted by the proposed Unit 1 and Unit 2 power uprate do not result in significant adverse environmental impact. The proposed MUR power uprate does not involve an unreviewed environmental question. The FES concluded that no significant environmental impact would result from the operation of VEGP. This conclusion remains valid for operation at the proposed MUR power uprate conditions. The proposed MUR power uprate will not result in a change in the types of non-radioactive effluents, nor will it result in a significant increase in the amount of non-radioactive effluents released off-site.

REGULATORY COMMITMENTS

There are no regulatory commitments in this License Amendment Request.

Enclosure 2

Vogtle Electric Generating Plant
Request to Change Licensed Maximum Power Level
10 CFR 50.92 Significant Hazards Evaluation and Environmental Assessment

Enclosure 2

Vogtle Electric Generating Plant
Request to Change Licensed Maximum Power Level
10 CFR 50.92 Significant Hazards Evaluation and Environmental Assessment

Proposed Changes

In accordance with the requirements of 10 CFR 50.90, Southern Nuclear Operating Company (SNC) proposes to revise the "Maximum Power Level" in paragraph 2.C(1) of the Vogtle Electric Generating Plant (VEGP) Facility Operating Licenses NPF-68 and NPF-81 for Unit 1 and Unit 2, respectively. In addition, SNC proposes to revise the definition of "Rated Thermal Power" in Technical Specification 1.1 for both units to reflect the change to the Maximum Power Level. The Maximum Power Level and Rated Thermal Power (RTP) represent the same reactor core power level.

SNC proposes to increase the RTP of VEGP Units 1 and 2 from 3565 MWt to 3625.6 MWt. The proposed change is an increase in RTP of 1.7% from the current reactor output. The proposed increase in reactor output is the result of utilizing margin in the analyses for the Emergency Core Cooling System (ECCS) Evaluation Model as described in 10 CFR 50, Appendix K. In accordance with Appendix K, the proposed change justifies assuming an alternative power level other than 102% of RTP in the ECCS analyses based on reduced uncertainty in the power level measurement. This increase in reactor core power level is referred to as a Measurement Uncertainty Recapture (MUR) power uprate. The proposed change also includes a change to the definition of Dose Equivalent Iodine (DEI) in Technical Specification 1.1 as well as a change to the Power Range Neutron Flux P-9 permissive nominal setpoint and allowable value in Technical Specification 3.3.1. Changes to Technical Specification Bases affected by the MUR power uprate are also included.

SNC will install the Caldon Check-Plus ultrasonic feedwater flow element into both units to reduce the uncertainty in the feedwater flow measurement. This reduced uncertainty, in combination with other uncertainties, results in an overall power level measurement uncertainty of 0.3% RTP. The remaining margin of 1.7% RTP forms the basis for the proposed MUR power uprating of 1.7% RTP. In addition to this modification to the plant, the high-pressure turbines and heater drain pumps will be replaced in both units to accommodate the additional flow requirements as a result of the MUR power uprate. Steam generator pressure transmitters will be replaced to be consistent with the power level measurement uncertainty analyses.

MUR power uprates have been approved by the NRC for several plants, including SNC Hatch Nuclear Plant Units 1 and 2 on September 23, 2003. Most recently, the NRC approved the request for a 1.7% MUR power uprate for Seabrook Station Unit 1 on May 22, 2006 (Amendment 110 to Facility Operating License NPF-86). Seabrook is a four-loop Westinghouse PWR similar to VEGP Units 1 and 2. The Caldon Check-Plus ultrasonic feedwater flow element, as proposed for VEGP, has been installed at Seabrook.

10 CFR 50.92 Evaluation

In 10 CFR 50.92(c), the Nuclear Regulatory Commission (NRC) provides the following standards to be used in determining the existence of a significant hazards consideration:

"...a proposed amendment to an operating license for a facility licensed under 50.21(b) or 50.2, or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not: (1) Involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) Create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) Involve a significant reduction in a margin of safety."

Southern Nuclear Operating Company (SNC) has reviewed the proposed amendment request and determined that its adoption does not involve a significant hazards consideration based upon the following discussion:

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

   Operating License - Maximum Power Level and Technical Specification 1.1 - Definition of Rated Thermal Power

   The increase in Maximum Power Level and Rated Thermal Power (RTP) does not involve a significant increase in the probability or consequences of an accident previously evaluated, because operation at the higher power level will not cause any design or analysis acceptance criteria to be exceeded. As a result, structural and functional integrity of the plant systems is maintained. Power level is an input assumption to the equipment design and accident analyses, but it is not itself an initiator for any transient. Therefore, the probability of occurrence of an accident previously evaluated is not affected.

   The radiological consequences of operation at the Measurement Uncertainty Recapture (MUR) power uprate conditions have been assessed. It was concluded that offsite dose predictions remain within the acceptance criteria for each of the accidents affected. Therefore, the consequences of an accident previously evaluated are not increased.

Technical Specification 1.1 - Definition of Dose Equivalent Iodine

The proposed change to the definition of dose equivalent iodine (DEI) impacts the reactor coolant activity surveillance and calculations of accident consequences and makes these activities consistent with each other. Neither of these functions affects the probability of any accident previously evaluated.

In order to support the MUR power uprate, the accidents previously evaluated in the Updated Final Safety Analysis Report (UFSAR) were re-analyzed. As part of this re-analysis, the dose conversion factors (DCFs) were reviewed, and a consistent set of DCFs was used for all re-analyses based on Federal Guidance Report No. 11, as suggested by RIS 2001-19. The results of these re-analyses continue to meet the acceptance limits as currently described in the UFSAR.

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

Technical Specification 3.3.1, Table 3.3.1-1, Function 16 - P-9 Setpoint

The revised Power Range Neutron Flux P-9 permissive nominal setpoint and allowable value do not involve a significant increase in the probability or consequences of an accident previously evaluated, because operation with these revised values will not cause any design or analysis acceptance criteria to be exceeded. The structural and functional integrity of any plant system is unaffected. The P-9 permissive function is part of the transient mitigation response and is not itself an initiator for any transient. Therefore, the probability of occurrence of an accident previously evaluated is not affected.

The changes to the P-9 nominal setpoint and allowable value do not affect the integrity of the fission product barriers utilized for the mitigation of radiological dose consequences as a result of an accident. The change continues to ensure that the pressurizer power operated relief valves (PORVs) are not challenged following a turbine trip without a reactor trip which, in turn, minimizes the potential for a release. There are no offsite dose predictions for this transient. Since it has been determined that the transient results are unaffected by the change to the P-9 nominal setpoint and allowable value, it is concluded that the consequences of an accident previously evaluated are not increased.

2. Does the proposed change create the possibility of a new or different kind of accident from any previously evaluated?

<u>Operating License - Maximum Power Level and Technical Specification 1.1 - Definition of Rated Thermal Power</u>

The increase in Maximum Power Level and RTP does not create the possibility of a new or different kind of accident from any previously evaluated, because no new operating configuration is being imposed that will create a new failure scenario, and no new failure modes are being created for any plant equipment. System and component design bases have been reviewed. The proposed change does not have an adverse effect on safety-related systems or components and does not challenge the integrity of any safety-related system. Therefore, the types of accidents defined in the UFSAR continue to represent the credible spectrum of events to determine safe plant operation.

<u>Technical Specification 1.1 - Definition of Dose Equivalent Iodine</u>

The proposed change to the definition of Dose Equivalent Iodine (DEI) ensures the reactor coolant activity surveillances are consistent with the assumptions for initial conditions used in the accident analyses. The proposed change does not involve the addition or modification of any plant equipment. Neither does it alter the design, configuration or method of operation of the plant.

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

<u>Technical Specification 3.3.1, Table 3.3.1-1, Function 16 - P-9 Setpoint</u>

The revised Power Range Neutron Flux P-9 permissive nominal setpoint and allowable value do not create the possibility of a new or different kind of accident from any previously evaluated, because these changes do not affect accident initiation sequences. No new operating configuration is being imposed by the P-9 nominal setpoint and allowable value changes that will create a new failure scenario. In addition, no new failure modes are being created for any plant equipment. Therefore, the types of accidents defined in the UFSAR continue to represent the credible spectrum of events to determine safe plant operation.

3. Does the proposed change involve a significant decrease in a margin of safety?

Operating License - Maximum Power Level and Technical Specification 1.1 - Definition of Rated Thermal Power

The increase in Maximum Power Level and RTP does not involve a significant reduction in a margin of safety, because power level is one of the inherent assumptions that determine the safe operating range defined by the accident analyses, which are in turn protected by the Technical Specifications. The acceptance criteria for the accident analyses are conservative with respect to the operating conditions defined by the Technical Specifications. The engineering reviews performed for the MUR power uprate confirmed that the accident analyses criteria are met at the revised value of MPL and RTP. Therefore, the adequacy of the revised Facility Operating Licenses and Technical Specifications to maintain the plant in a safe operating range is also confirmed, and the increase in MPL and RTP do not involve a significant decrease in a margin of safety.

Technical Specification 1.1 - Definition of Dose Equivalent Iodine

The proposed change to the definition of dose equivalent iodine (DEI) has the potential to affect the dose consequences offsite and in the control room. However, the results of the re-analyses of the accidents previously evaluated demonstrate the dose consequences at all locations remain within the regulatory acceptance limits, and the margin of safety as defined by 10 CFR 100 and GDC 19 has not been significantly reduced.

Technical Specification 3.3.1, Table 3.3.1-1, Function 16 - P-9 Setpoint

The change to the P-9 nominal setpoint and allowable value does not involve a significant reduction in a margin of safety because the margin of safety associated with the P-9 setpoint, as verified by the results of the applicable transient analyses, is within acceptable limits. The adequacy of the revised Technical Specification values to maintain the plant in a safe operating range has been confirmed. Therefore, the change to the P-9 nominal setpoint and allowable value does not involve a significant decrease in a margin of safety.

Vogtle Electric Generating Plant
Request to Change Licensed Maximum Power Level
10 CFR 50.92 Significant Hazards Evaluation and Environmental Assessment

Environmental Review

SNC has performed an environmental evaluation in accordance with Section 3.1 of the non-radiological Environmental Protection Plan (EPP) for both Unit 1 and Unit 2. The EPP is contained in Appendix B of both Facility Operating Licenses. The EPP provides the standards for determining the existence of an unreviewed environmental question (UEQ). The environmental evaluation concluded that there is no UEQ resulting from the proposed MUR power uprate.

10 CFR 51.22(c)(9) provides criteria for the categorical exclusion from performing an environmental assessment. A proposed amendment to an operating license for a facility requires no environmental assessment if operation of the facility in accordance with the proposed license amendment will not:

1. Involve a significant hazards consideration;
2. Result in a significant change in the types, or a significant increase in the amounts of any effluents that may be released off-site;
3. Result in a significant increase in individual or cumulative occupational radiation exposure.

Southern Nuclear Operating Company has evaluated the proposed changes and determined the changes do not involve (1) a significant hazards consideration, (2) a significant change in the types or significant increase in the amounts of any effluents that may be released offsite, or (3) a significant increase in the individual or cumulative occupational exposure. Accordingly, the proposed changes meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9), and an environmental assessment of the proposed changes is not required.

Enclosure 3

Vogtle Electric Generating Plant
Request to Change Licensed Maximum Power Level
Marked-Up Operating Licenses, Technical Specifications, and Bases Pages

B. Subject to the conditions and requirements incorporated herein, the Commission hereby licenses:

(1) Southern Nuclear, pursuant to Section 103 of the Act and 10 CFR Part 50, to possess, manage, use, maintain, and operate the facility at the designated location in Burke County, Georgia, in accordance with the procedures and limitations set forth in this license;

(2) Georgia Power Company, Oglethorpe Power Corporation, Municipal Electric Authority of Georgia, and City of Dalton, Georgia, pursuant to the Act and 10 CFR Part 50, to possess, but not operate the facility at the designated location in Burke County, Georgia, in accordance with the procedures and limitations set forth in this license;

(3) Southern Nuclear, pursuant to the Act and 10 CFR Part 70, to receive, possess, and use at any time special nuclear material as reactor fuel, in accordance with the limitations for storage and amounts required for reactor operation, as described in the Final Safety Analysis Report, as supplemented and amended;

(4) Southern Nuclear, pursuant to the Act and 10 CFR Parts 30, 40, and 70 to receive, possess, and use at any time any byproduct, source and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

(5) Southern Nuclear, pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess, and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components;

(6) Southern Nuclear, pursuant to the Act and 10 CFR Parts 30, 40, and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility authorized herein.

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

(1) <u>Maximum Power Level</u>

3625.6

Southern Nuclear is authorized to operate the facility at reactor core power levels not in excess of ~~3565~~ megawatts thermal (100 percent power) in accordance with the conditions specified herein.

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. ~~148~~, 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) Southern Nuclear Operating Company shall be capable of establishing containment hydrogen monitoring within 90 minutes of initiating safety injection following a loss of coolant accident.

(4) DELETED

(5) DELETED

(6) DELETED

(7) DELETED

(8) DELETED

(9) DELETED

(10) Additional Conditions

The Additional Conditions contained in Appendix D, as revised through Amendment No. 102, are hereby incorporated into this license. Southern Nuclear shall operate the facility in accordance with the Additional Conditions.

D. The facility requires exemptions from certain requirements of 10 CFR Part 50 and 10 CFR Part 70. These include (a) an exemption from the requirements of 10 CFR 70.24 for two criticality monitors around the fuel storage area, and (b) an exemption from the requirements of Paragraph III.D.2(b)(ii) of Appendix J of 10 CFR 50, the testing of containment air locks at times when containment integrity is not required. The special circumstances regarding exemption b are identified in Section 6.2.6 of SSER 5.

An exemption was previously granted pursuant to 10 CFR 70.24. The exemption was granted with NRC materials license No. SNM-1967, issued August 21, 1986, and relieved GPC from the requirement of having a criticality alarm system. GPC and Southern Nuclear are hereby exempted from the criticality alarm system provision of 10 CFR 70.24 so far as this section applies to the storage of fuel assemblies held under this license.

These exemptions are authorized by law, will not present an undue risk to the public health and safety, and are consistent with the common defense and security. The exemptions in items b and c above are granted pursuant to 10 CFR 50.12. With

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

(1) <u>Maximum Power Level</u>

3625.6

Southern Nuclear is authorized to operate the facility at reactor core power levels not in excess of ~~3565~~ megawatts thermal (100 percent power) in accordance with the conditions specified herein.

(2) <u>Technical Specifications and Environmental Protection Plan</u>

The Technical Specifications contained in Appendix A, as revised through Amendment No. ~~128~~, 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.

The Surveillance Requirements (SRs) contained in the Appendix A Technical Specifications and listed below are not required to be performed immediately upon implementation of Amendment No. 74. The SRs listed below shall be successfully demonstrated prior to the time and condition specified below for each:

a) DELETED

b) DELETED

c) SR 3.8.1.20 shall be successfully demonstrated at the first regularly scheduled performance after implementation of this license amendment.

(3) Southern Nuclear Operating Company shall be capable of establishing containment hydrogen monitoring within 90 minutes of initiating safety injection following a loss of coolant accident.

(4) <u>Additional Conditions</u>

The Additional Conditions contained in Appendix D, as revised through Amendment No. 80, are hereby incorporated into this license. Southern Nuclear shall operate the facility in accordance with the Additional Conditions.

D. The facility requires exemptions from certain requirements of 10 CFR Part 50 and 10 CFR Part 70. These include (a) an exemption from the requirements of 10 CFR 70.24 for two criticality monitors around the fuel storage area, and (b) an exemption from the requirements of Paragraph III.D.2(b)(ii) of Appendix J of 10 CFR 50, the testing of containment air locks at times when containment integrity is not required. The special circumstances regarding exemption b are identified in Section 6.2.6 of SSER 8.

1.1 Definitions (continued)

CHANNEL CHECK	A CHANNEL CHECK shall be the qualitative assessment, by observation, of channel behavior during operation. This determination shall include, where possible, comparison of the channel indication and status to other indications or status derived from independent instrument channels measuring the same parameter.
CHANNEL OPERATIONAL TEST (COT)	A COT shall be the injection of a simulated or actual signal into the channel as close to the sensor as practicable to verify the OPERABILITY of required alarm, interlock, and trip functions. The COT shall include adjustments, as necessary, of the required alarm, interlock, and trip setpoints so that the setpoints are within the required range and accuracy.
CORE ALTERATION	CORE ALTERATION shall be the movement of any fuel, sources, or other reactivity control components within the reactor vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe position.
CORE OPERATING LIMITS REPORT (COLR)	The COLR is the unit specific document that provides cycle specific parameter limits for the current reload cycle. These cycle specific parameter limits shall be determined for each reload cycle in accordance with Specification 5.6.5. Unit operation within these limits is addressed in individual Specifications.
DOSE EQUIVALENT I-131	DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries/gram) that alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in ~~Table E-7 of Regulatory Guide 1.109, Rev. 1, NRC, 1977.~~

Insert A

(continued)

INSERT "A"

CHANGE TO TECHNICAL SPECIFICATION 1.1

EPA Federal Guidance Report No. 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," EPA-520/1-88-020, September 1988.

1.1 Definitions

PHYSICS TESTS (continued)

a. Described in Chapter 14 of the FSAR;

b. Authorized under the provisions of 10 CFR 50.59; or

c. Otherwise approved by the Nuclear Regulatory Commission.

PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

The PTLR is the unit specific document that provides the reactor vessel pressure and temperature limits, including heatup and cooldown rates, Cold Overpressure Protection System (COPS) arming temperature and the nominal PORV setpoints for the COPS, for the current reactor vessel fluence period. These pressure and temperature limits shall be determined for each fluence period in accordance with Specification 5.6.6. Unit operation within these operating limits is addressed in individual specifications.

QUADRANT POWER TILT RATIO (QPTR)

QPTR shall be the ratio of the maximum upper excore detector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of the maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs, whichever is greater.

RATED THERMAL POWER (RTP)

RTP shall be a total reactor core heat transfer rate to the reactor coolant of ~~3565~~ 3625.6 MWt.

REACTOR TRIP SYSTEM (RTS) RESPONSE TIME

The RTS RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its RTS trip setpoint at the channel sensor until loss of stationary gripper coil voltage. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and the methodology for verification have been previously reviewed and approved by the NRC.

(continued)

Table 3.3.1-1 (page 5 of 9)
Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	NOMINAL TRIP SETPOINT(n)
14. Turbine Trip						
a. Low Fluid Oil Pressure	1(j)	3	O	SR 3.3.1.10 SR 3.3.1.16	≥ 500 psig	580 psig
b. Turbine Stop Valve Closure	1(j)	4	P	SR 3.3.1.10 SR 3.3.1.14	≥ 90% open	96.7% open
15. Safety Injection (SI) Input from Engineered Safety Feature Actuation System (ESFAS)	1,2	2 trains	Q	SR 3.3.1.13	NA	NA
16. Reactor Trip System Interlocks						
a. Intermediate Range Neutron Flux, P-6	2(d)	2	R	SR 3.3.1.11 SR 3.3.1.12	≥ 1.2E-5% RTP	2.0E-5% RTP
b. Low Power Reactor Trips Block, P-7	1	1 per train	S	SR 3.3.1.5	NA	NA
c. Power Range Neutron Flux, P-8	1	4	S	SR 3.3.1.11 SR 3.3.1.12	≤ 50.3% RTP	48% RTP
d. Power Range Neutron Flux, P-9	1	4	S	SR 3.3.1.11 SR 3.3.1.12	≤ ~~52.3~~% RTP 40.6	~~50~~% RTP 40
e. Power Range Neutron Flux, P-10 and input to P-7	1,2	4	R	SR 3.3.1.11 SR 3.3.1.12	(l,m)	(l,m)
f. Turbine Impulse Pressure, P-13	1	2	S	SR 3.3.1.10 SR 3.3.1.12	≤ 12.3% Impulse Pressure Equivalent turbine	10% Impulse Pressure Equivalent turbine

(continued)

(d) Below the P-6 (Intermediate Range Neutron Flux) interlocks.
(j) Above the P-9 (Power Range Neutron Flux) interlock.
(l) For the P-10 input to P-7, the Allowable Value is ≤ 12.3% RTP and the Nominal Trip Setpoint is 10% RTP.
(m) For the Power Range Neutron Flux, P-10, the Allowable Value is ≥ 7.7% RTP and the Nominal Trip Setpoint is 10% RTP.
(n) A channel is OPERABLE with an actual Trip Setpoint value outside its calibration tolerance band provided the Trip Setpoint value is conservative with respect to its associated Allowable Value and the channel is readjusted to within the established calibration tolerance band of the Nominal Trip Setpoint. A Trip Setpoint may be set more conservative than the Nominal Trip Setpoint as necessary in response to plant conditions

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

c. Power Range Neutron Flux, P-8 (continued)

when greater than approximately 48% power. On decreasing power, the reactor trip on low flow in any loop is automatically blocked.

The LCO requires four channels of Power Range Neutron Flux, P-8 interlock to be OPERABLE in MODE 1.

In MODE 1, a loss of flow in one RCS loop could result in DNB conditions, so the Power Range Neutron Flux, P-8 interlock must be OPERABLE. In MODE 2, 3, 4, 5, or 6, this Function does not have to be OPERABLE because the core is not producing sufficient power to be concerned about DNB conditions.

d. Power Range Neutron Flux, P-9

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The Power Range Neutron Flux, P-9 interlock (NI-0041B & C, NI-0042B & C, NI-0043B & C, NI-0044B & C) is actuated at approximately $\boxed{50}$% power as determined by two-out-of-four NIS power range detectors. The LCO requirement for this Function ensures that the Turbine Trip — Low Fluid Oil Pressure and Turbine Trip — Turbine Stop Valve Closure reactor trips are enabled above the P-9 setpoint. Above the P-9 setpoint, a turbine trip will cause a load rejection beyond the capacity of the Steam Dump System. A reactor trip is automatically initiated on a turbine trip when it is above the P-9 setpoint, to minimize the transient on the reactor.

The LCO requires four channels of Power Range Neutron Flux, P-9 interlock to be OPERABLE in MODE 1.

In MODE 1, a turbine trip could cause a load rejection beyond the capacity of the Steam Dump System, so the Power Range Neutron Flux interlock must be OPERABLE. In MODE 2, 3, 4, 5, or 6, this Function does not have to be OPERABLE because the

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.2 (continued)

contributor to the instrument uncertainty for a secondary side power calorimetric measurement is the feedwater flow measurement which is typically a ΔP measurement across a feedwater venturi. While the measurement uncertainty remains constant in ΔP as power decreases, when translated into flow, the uncertainty increases as a square term. Thus a 1% flow error at 100% RTP can approach a 10% error at 30% RTP even though the ΔP error has not changed. An evaluation of extended operation at part-power conditions would conclude that it is prudent to administratively adjust the setpoint of the Power Range Neutron Flux – High bistables to ≤ 90% RTP for a calorimetric power determined below 50% RTP, and to ≤ 75% RTP for a calorimetric power determined below 20% RTP when: 1) the power range channel output is adjusted in the decreasing power direction due to a part-power calorimetric; or 2) for a post-refueling startup.

Insert B

Before the Power Range Neutron Flux – High bistables are reset to the nominal value in Table 3.3.1-1 of Specification 3.3.1, the power range channel adjustment must be confirmed based on a calorimetric performed at a power level ≥ 50% RTP.

The Note clarifies that this Surveillance is required only if reactor power is ≥ 15% RTP and that 12 hours is allowed for performing the first Surveillance after reaching 15% RTP. At lower power levels, calorimetric data are inaccurate.

The Frequency of every 24 hours is adequate. It is based on unit operating experience, considering instrument reliability and operating history data for instrument drift. Together these factors demonstrate that a difference between the calorimetric heat balance calculation and the power range channel output of more than +2% RTP is not expected in any 24 hour period.

In addition, control room operators periodically monitor redundant indications and alarms to detect deviations in channel outputs.

SR 3.3.1.3

SR 3.3.1.3 compares the incore system to the NIS channel output every 31 EFPD. If the absolute difference is ≥ 3%, the NIS channel is still OPERABLE, but must be readjusted. If the NIS channel cannot be properly readjusted, the channel is declared

(continued)

INSERT "B"

REVISION TO BASES FOR SR 3.3.1.2

While the part-power calorimetric uncertainty based on a feedwater flow measurement from the leading-edge flow meter (LEFM) is less than that based on the feedwater venturi, it is prudent to continue to apply the same adjustments to the setpoint.

BASES

REFERENCES (continued)

2. FSAR, Chapter 6.

3. FSAR, Chapter 15.

4. IEEE-279-1971.

5. 10 CFR 50.49.

6. WCAP-11269, Westinghouse Setpoint Methodology for Protection Systems; as supplemented by:

   - Amendments 34 (Unit 1) and 14 (Unit 2), RTS Steam Generator Water Level – Low Low, ESFAS Turbine Trip and Feedwater Isolation SG Water Level – High High, and ESFAS AFW SG Water Level – Low Low.
   - Amendments 48 and 49 (Unit 1) and Amendments 27 and 28 (Unit 2), deletion of RTS Power Range Neutron Flux High Negative Rate Trip.
   - Amendments 60 (Unit 1) and 39 (Unit 2), RTS Overtemperature ΔT setpoint revision.
   - Amendments 57 (Unit 1) and 36 (Unit 2), RTS Overtemperature and Overpower ΔT time constants and Overtemperature ΔT setpoint.
   - Amendments 43 and 44 (Unit 1) and 23 and 24 (Unit 2), revised Overtemperature and Overpower ΔT trip setpoints and allowable values.
   - Amendments 104 (Unit 1) and 82 (Unit 2), revised RTS Intermediate Range Neutron Flux, Source Range Neutron Flux, and P-6 trip setpoints and allowable values.
   - Amendments 127 (Unit 1) and 105 (Unit 2), revised Overtemperature ΔT trip setpoint to limit value of the compensated temperature difference and revised the modifier for axial flux difference.
   - Amendments 128 (Unit 1) and 106 (Unit 2), revised Overtemperature ΔT and Overpower ΔT trip setpoints to increase the fundamental setpoints K_1 and K_4, and to modify coefficients and dynamic compensation terms.

Insert C

7. Westinghouse Letter GP-16696, November 5, 1997.

8. WCAP-14333-P-A, Rev. 1, October 1998.

(continued)

INSERT "C"

REVISION TO REFERENCE 6 OF BASES B3.3.1

- Amendments ___ (Unit 1) and ___ (Unit 2), revised P-9 setpoint and allowable value.

B 3.7 PLANT SYSTEMS

B 3.7.1 Main Steam Safety Valves (MSSVs)

BASES

BACKGROUND

The primary purpose of the MSSVs is to provide overpressure protection for the secondary system. The MSSVs also provide protection against overpressurizing the reactor coolant pressure boundary (RCPB) by providing a heat sink for the removal of energy from the Reactor Coolant System (RCS) if the preferred heat sink, provided by the Condenser and Circulating Water System, is not available.

Five MSSVs are located on each main steam header, outside containment, upstream of the main steam isolation valves, as described in the FSAR, Section 10.3 (Ref. 1). The actual MSSV capacity is ~~117~~% of rated steam flow at 110% of the steam generator design pressure. This meets the requirements of the ASME Code, Section III (Ref. 2). The MSSV design includes staggered setpoints, according to Table 3.7.1-2 in the accompanying LCO, so that only the needed valves will actuate. Staggered setpoints reduce the potential for valve chattering that is due to steam pressure insufficient to fully open all valves following a turbine reactor trip.

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APPLICABLE SAFETY ANALYSES

The design basis requirement is that secondary system pressure is limited to 110% of design pressure which is specified in Reference 2. The actual design basis applied for the MSSVs comes from Reference 6 and its purpose is to limit the secondary system pressure to ≤ 110% of design pressure when passing 105% of design steam flow. This design basis is sufficient to cope with any anticipated operational occurrence (AOO) or accident considered in the Design Basis Accident (DBA) and transient analysis.

The events that challenge the relieving capacity of the MSSVs, and thus RCS pressure, are those characterized as decreased heat removal events, which are presented in the FSAR, Section 15.2 (Ref. 3). Of these, the full power turbine trip without steam dump is the limiting AOO. This event also terminates normal feedwater flow to the steam generators.

(continued)

Enclosure 4

Vogtle Electric Generating Plant
Request to Change Licensed Maximum Power Level
Clean Typed Operating License, Technical Specifications, and Bases Pages

B. Subject to the conditions and requirements incorporated herein, the Commission hereby licenses:

(1) Southern Nuclear, pursuant to Section 103 of the Act and 10 CFR Part 50, to possess, manage, use, maintain, and operate the facility at the designated location in Burke County, Georgia, in accordance with the procedures and limitations set forth in this license;

(2) Georgia Power Company, Oglethorpe Power Corporation, Municipal Electric Authority of Georgia, and City of Dalton, Georgia, pursuant to the Act and 10 CFR Part 50, to possess, but not operate the facility at the designated location in Burke County, Georgia, in accordance with the procedures and limitations set forth in this license;

(3) Southern Nuclear, pursuant to the Act and 10 CFR Part 70, to receive, possess, and use at any time special nuclear material as reactor fuel, in accordance with the limitations for storage and amounts required for reactor operation, as described in the Final Safety Analysis Report, as supplemented and amended;

(4) Southern Nuclear, pursuant to the Act and 10 CFR Parts 30, 40, and 70 to receive, possess, and use at any time any byproduct, source and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

(5) Southern Nuclear, pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess, and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components;

(6) Southern Nuclear, pursuant to the Act and 10 CFR Parts 30, 40, and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility authorized herein.

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

(1) <u>Maximum Power Level</u>

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.

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. , 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) Southern Nuclear Operating Company shall be capable of establishing containment hydrogen monitoring within 90 minutes of initiating safety injection following a loss of coolant accident.

(4) DELETED

(5) DELETED

(6) DELETED

(7) DELETED

(8) DELETED

(9) DELETED

(10) Additional Conditions

The Additional Conditions contained in Appendix D, as revised through Amendment No. 102, are hereby incorporated into this license. Southern Nuclear shall operate the facility in accordance with the Additional Conditions.

D. The facility requires exemptions from certain requirements of 10 CFR Part 50 and 10 CFR Part 70. These include (a) an exemption from the requirements of 10 CFR 70.24 for two criticality monitors around the fuel storage area, and (b) an exemption from the requirements of Paragraph III.D.2(b)(ii) of Appendix J of 10 CFR 50, the testing of containment air locks at times when containment integrity is not required. The special circumstances regarding exemption b are identified in Section 6.2.6 of SSER 5.

An exemption was previously granted pursuant to 10 CFR 70.24. The exemption was granted with NRC materials license No. SNM-1967, issued August 21, 1986, and relieved GPC from the requirement of having a criticality alarm system. GPC and Southern Nuclear are hereby exempted from the criticality alarm system provision of 10 CFR 70.24 so far as this section applies to the storage of fuel assemblies held under this license.

These exemptions are authorized by law, will not present an undue risk to the public health and safety, and are consistent with the common defense and security. The exemptions in items b and c above are granted pursuant to 10 CFR 50.12. With

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

(1) Maximum Power Level

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.

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. , 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.

The Surveillance Requirements (SRs) contained in the Appendix A Technical Specifications and listed below are not required to be performed immediately upon implementation of Amendment No. 74. The SRs listed below shall be successfully demonstrated prior to the time and condition specified below for each:

a) DELETED

b) DELETED

c) SR 3.8.1.20 shall be successfully demonstrated at the first regularly scheduled performance after implementation of this license amendment.

(3) Southern Nuclear Operating Company shall be capable of establishing containment hydrogen monitoring within 90 minutes of initiating safety injection following a loss of coolant accident.

(4) Additional Conditions

The Additional Conditions contained in Appendix D, as revised through Amendment No. 80, are hereby incorporated into this license. Southern Nuclear shall operate the facility in accordance with the Additional Conditions.

D. The facility requires exemptions from certain requirements of 10 CFR Part 50 and 10 CFR Part 70. These include (a) an exemption from the requirements of 10 CFR 70.24 for two criticality monitors around the fuel storage area, and (b) an exemption from the requirements of Paragraph III.D.2(b)(ii) of Appendix J of 10 CFR 50, the testing of containment air locks at times when containment integrity is not required. The special circumstances regarding exemption b are identified in Section 6.2.6 of SSER 8.

1.1 Definitions (continued)

CHANNEL CHECK	A CHANNEL CHECK shall be the qualitative assessment, by observation, of channel behavior during operation. This determination shall include, where possible, comparison of the channel indication and status to other indications or status derived from independent instrument channels measuring the same parameter.
CHANNEL OPERATIONAL TEST (COT)	A COT shall be the injection of a simulated or actual signal into the channel as close to the sensor as practicable to verify the OPERABILITY of required alarm, interlock, and trip functions. The COT shall include adjustments, as necessary, of the required alarm, interlock, and trip setpoints so that the setpoints are within the required range and accuracy.
CORE ALTERATION	CORE ALTERATION shall be the movement of any fuel, sources, or other reactivity control components within the reactor vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe position.
CORE OPERATING LIMITS REPORT (COLR)	The COLR is the unit specific document that provides cycle specific parameter limits for the current reload cycle. These cycle specific parameter limits shall be determined for each reload cycle in accordance with Specification 5.6.5. Unit operation within these limits is addressed in individual Specifications.
DOSE EQUIVALENT I-131	DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries/gram) that alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in EPA Federal Guidance Report No. 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," EPA-520/1-88-020, September 1988.

(continued)

1.1 Definitions

PHYSICS TESTS (continued)	a. Described in Chapter 14 of the FSAR; b. Authorized under the provisions of 10 CFR 50.59; or c. Otherwise approved by the Nuclear Regulatory Commission.
PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)	The PTLR is the unit specific document that provides the reactor vessel pressure and temperature limits, including heatup and cooldown rates, Cold Overpressure Protection System (COPS) arming temperature and the nominal PORV setpoints for the COPS, for the current reactor vessel fluence period. These pressure and temperature limits shall be determined for each fluence period in accordance with Specification 5.6.6. Unit operation within these operating limits is addressed in individual specifications.
QUADRANT POWER TILT RATIO (QPTR)	QPTR shall be the ratio of the maximum upper excore detector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of the maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs, whichever is greater.
RATED THERMAL POWER (RTP)	RTP shall be a total reactor core heat transfer rate to the reactor coolant of 3625.6 MWt.
REACTOR TRIP SYSTEM (RTS) RESPONSE TIME	The RTS RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its RTS trip setpoint at the channel sensor until loss of stationary gripper coil voltage. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and the methodology for verification have been previously reviewed and approved by the NRC.

(continued)

Table 3.3.1-1 (page 5 of 9)
Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	NOMINAL TRIP SETPOINT(n)
14. Turbine Trip						
a. Low Fluid Oil Pressure	1(j)	3	O	SR 3.3.1.10 SR 3.3.1.16	≥ 500 psig	580 psig
b. Turbine Stop Valve Closure	1(j)	4	P	SR 3.3.1.10 SR 3.3.1.14	≥ 90% open	96.7% open
15. Safety Injection (SI) Input from Engineered Safety Feature Actuation System (ESFAS)	1,2	2 trains	Q	SR 3.3.1.13	NA	NA
16. Reactor Trip System Interlocks						
a. Intermediate Range Neutron Flux, P-6	2(d)	2	R	SR 3.3.1.11 SR 3.3.1.12	≥ 1.2E-5% RTP	2.0E-5% RTP
b. Low Power Reactor Trips Block, P-7	1	1 per train	S	SR 3.3.1.5	NA	NA
c. Power Range Neutron Flux, P-8	1	4	S	SR 3.3.1.11 SR 3.3.1.12	≤ 50.3% RTP	48% RTP
d. Power Range Neutron Flux, P-9	1	4	S	SR 3.3.1.11 SR 3.3.1.12	≤ 40.6% RTP	40% RTP
e. Power Range Neutron Flux, P-10 and input to P-7	1,2	4	R	SR 3.3.1.11 SR 3.3.1.12	(l,m)	(l,m)
f. Turbine Impulse Pressure, P-13	1	2	S	SR 3.3.1.10 SR 3.3.1.12	≤ 12.3% Impulse Pressure Equivalent turbine	10% Impulse Pressure Equivalent turbine

(continued)

(d) Below the P-6 (Intermediate Range Neutron Flux) interlocks.
(j) Above the P-9 (Power Range Neutron Flux) interlock.
(l) For the P-10 input to P-7, the Allowable Value is ≤ 12.3% RTP and the Nominal Trip Setpoint is 10% RTP.
(m) For the Power Range Neutron Flux, P-10, the Allowable Value is ≥ 7.7% RTP and the Nominal Trip Setpoint is 10% RTP.
(n) A channel is OPERABLE with an actual Trip Setpoint value outside its calibration tolerance band provided the Trip Setpoint value is conservative with respect to its associated Allowable Value and the channel is readjusted to within the established calibration tolerance band of the Nominal Trip Setpoint. A Trip Setpoint may be set more conservative than the Nominal Trip Setpoint as necessary in response to plant conditions

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

c. Power Range Neutron Flux, P-8 (continued)

when greater than approximately 48% power. On decreasing power, the reactor trip on low flow in any loop is automatically blocked.

The LCO requires four channels of Power Range Neutron Flux, P-8 interlock to be OPERABLE in MODE 1.

In MODE 1, a loss of flow in one RCS loop could result in DNB conditions, so the Power Range Neutron Flux, P-8 interlock must be OPERABLE. In MODE 2, 3, 4, 5, or 6, this Function does not have to be OPERABLE because the core is not producing sufficient power to be concerned about DNB conditions.

d. Power Range Neutron Flux, P-9

The Power Range Neutron Flux, P-9 interlock (NI-0041B & C, NI-0042B & C, NI-0043B & C, NI-0044B & C) is actuated at approximately 40% power as determined by two-out-of-four NIS power range detectors. The LCO requirement for this Function ensures that the Turbine Trip — Low Fluid Oil Pressure and Turbine Trip — Turbine Stop Valve Closure reactor trips are enabled above the P-9 setpoint. Above the P-9 setpoint, a turbine trip will cause a load rejection beyond the capacity of the Steam Dump System. A reactor trip is automatically initiated on a turbine trip when it is above the P-9 setpoint, to minimize the transient on the reactor.

The LCO requires four channels of Power Range Neutron Flux, P-9 interlock to be OPERABLE in MODE 1.

In MODE 1, a turbine trip could cause a load rejection beyond the capacity of the Steam Dump System, so the Power Range Neutron Flux interlock must be OPERABLE. In MODE 2, 3, 4, 5, or 6, this Function does not have to be OPERABLE because the

(continued)

SURVEILLANCE REQUIREMENTS

SR 3.3.1.2 (continued)

contributor to the instrument uncertainty for a secondary side power calorimetric measurement is the feedwater flow measurement which is typically a ΔP measurement across a feedwater venturi. While the measurement uncertainty remains constant in ΔP as power decreases, when translated into flow, the uncertainty increases as a square term. Thus a 1% flow error at 100% RTP can approach a 10% error at 30% RTP even though the ΔP error has not changed. An evaluation of extended operation at part-power conditions would conclude that it is prudent to administratively adjust the setpoint of the Power Range Neutron Flux – High bistables to ≤ 90% RTP for a calorimetric power determined below 50% RTP, and to ≤ 75% RTP for a calorimetric power determined below 20% RTP when: 1) the power range channel output is adjusted in the decreasing power direction due to a part-power calorimetric; or 2) for a post-refueling startup. While the part-power calorimetric uncertainty based on a feedwater flow measurement from the leading-edge flow meter (LEFM) is less than that based on the feedwater venturi, it is prudent to continue to apply the same adjustments to the setpoint.

Before the Power Range Neutron Flux – High bistables are reset to the nominal value in Table 3.3.1-1 of Specification 3.3.1, the power range channel adjustment must be confirmed based on a calorimetric performed at a power level ≥ 50% RTP.

The Note clarifies that this Surveillance is required only if reactor power is ≥ 15% RTP and that 12 hours is allowed for performing the first Surveillance after reaching 15% RTP. At lower power levels, calorimetric data are inaccurate.

The Frequency of every 24 hours is adequate. It is based on unit operating experience, considering instrument reliability and operating history data for instrument drift. Together these factors demonstrate that a difference between the calorimetric heat balance calculation and the power range channel output of more than +2% RTP is not expected in any 24 hour period.

In addition, control room operators periodically monitor redundant indications and alarms to detect deviations in channel outputs.

SR 3.3.1.3

SR 3.3.1.3 compares the incore system to the NIS channel output

(continued)

SURVEILLANCE REQUIREMENTS

SR 3.3.1.3 (continued)

every 31 EFPD. If the absolute difference is ≥ 3%, the NIS channel is still OPERABLE, but must be readjusted. If the NIS channel cannot be properly readjusted, the channel is declared inoperable. This surveillance is primarily performed to verify the (AFD) input to the overtemperature ΔT function.

SR 3.3.1.3 compares the incore system to the NIS channel output every 31 EFPD. If the absolute difference is ≥ 3%, the NIS channel is still OPERABLE, but must be readjusted. If the NIS channel cannot be properly readjusted, the channel is declared inoperable. This surveillance is primarily performed to verify the f(AFD) input to the overtemperature ΔT function.

The Note clarifies that the Surveillance is required only if reactor power is ≥ 15% RTP and that 24 hours is allowed for performing the first Surveillance after reaching 15% RTP.

Axial offset is the difference between the power in the top half of the core and the bottom half of the core expressed as a fraction (percent) of the total power being produced by the core. Mathematically, it is expressed as:

$$AO = 100 \times \frac{(Flux_T - Flux_B)}{(Power)(Flux_T + Flux_B)}$$

where $Flux_T$ = neutron flux at the top of the core, and

$Flux_B$ = neutron flux at the bottom of the core

The relationship between AFD and axial offset is:

$$AFD = AO \times (Power(\%)/100)$$

AFD as displayed on the main control board and as determined by the plant computer use inputs from the power range NIS detectors which are located outside the reactor vessel. Axial offset is measured using incore detectors.

The surveillance assures that the AFD as displayed on the main control board and as determined by the plant computer is within 3% of the AFD as calculated from the axial offset equation. Agreement is required so that the reactor is operated within the bounds of the safety analysis regarding axial power distribution.

(continued)

BASES

REFERENCES (continued)

2. FSAR, Chapter 6.

3. FSAR, Chapter 15.

4. IEEE-279-1971.

5. 10 CFR 50.49.

6. WCAP-11269, Westinghouse Setpoint Methodology for Protection Systems; as supplemented by:

   - Amendments 34 (Unit 1) and 14 (Unit 2), RTS Steam Generator Water Level – Low Low, ESFAS Turbine Trip and Feedwater Isolation SG Water Level – High High, and ESFAS AFW SG Water Level – Low Low.

   - Amendments 48 and 49 (Unit 1) and Amendments 27 and 28 (Unit 2), deletion of RTS Power Range Neutron Flux High Negative Rate Trip.

   - Amendments 60 (Unit 1) and 39 (Unit 2), RTS Overtemperature ΔT setpoint revision.

   - Amendments 57 (Unit 1) and 36 (Unit 2), RTS Overtemperature and Overpower ΔT time constants and Overtemperature ΔT setpoint.

   - Amendments 43 and 44 (Unit 1) and 23 and 24 (Unit 2), revised Overtemperature and Overpower ΔT trip setpoints and allowable values.

   - Amendments 104 (Unit 1) and 82 (Unit 2), revised RTS Intermediate Range Neutron Flux, Source Range Neutron Flux, and P-6 trip setpoints and allowable values.

   - Amendments 127 (Unit 1) and 105 (Unit 2), revised Overtemperature ΔT trip setpoint to limit value of the compensated temperature difference and revised the modifier for axial flux difference.

   - Amendments 128 (Unit 1) and 106 (Unit 2), revised Overtemperature ΔT and Overpower ΔT trip setpoints to increase the fundamental setpoints K_1 and K_4, and to modify coefficients and dynamic compensation terms.

   - Amendments ___ (Unit 1) and ___ (Unit 2), revised P-9 setpoint and allowable value.

7. Westinghouse Letter GP-16696, November 5, 1997.

(continued)

BASES

REFERENCES (continued)

8. WCAP-14333-P-A, Rev. 1, October 1998.

9. WCAP-10271-P-A, Supplement 1, May 1986.

10. WCAP-10271-P-A, Supplement 2, Rev. 1, June 1990.

11. WCAP-15376, Rev. 0, October 2000.

12. FSAR, Chapter 16.

13. WCAP-13632-P-A Revision 2, "Elimination of Periodic Sensor Response Time Testing Requirements," January 1996.

14. WCAP-14036-P-A Revision 1, "Elimination of Periodic Protection Channel Response Time Tests," October 1998.

B 3.7 PLANT SYSTEMS

B 3.7.1 Main Steam Safety Valves (MSSVs)

BASES

BACKGROUND

The primary purpose of the MSSVs is to provide overpressure protection for the secondary system. The MSSVs also provide protection against overpressurizing the reactor coolant pressure boundary (RCPB) by providing a heat sink for the removal of energy from the Reactor Coolant System (RCS) if the preferred heat sink, provided by the Condenser and Circulating Water System, is not available.

Five MSSVs are located on each main steam header, outside containment, upstream of the main steam isolation valves, as described in the FSAR, Section 10.3 (Ref. 1). The actual MSSV capacity is 114% of rated steam flow at 110% of the steam generator design pressure. This meets the requirements of the ASME Code, Section III (Ref. 2). The MSSV design includes staggered setpoints, according to Table 3.7.1-2 in the accompanying LCO, so that only the needed valves will actuate. Staggered setpoints reduce the potential for valve chattering that is due to steam pressure insufficient to fully open all valves following a turbine reactor trip.

APPLICABLE SAFETY ANALYSES

The design basis requirement is that secondary system pressure is limited to 110% of design pressure which is specified in Reference 2. The actual design basis applied for the MSSVs comes from Reference 6 and its purpose is to limit the secondary system pressure to $\leq$ 110% of design pressure when passing 105% of design steam flow. This design basis is sufficient to cope with any anticipated operational occurrence (AOO) or accident considered in the Design Basis Accident (DBA) and transient analysis.

The events that challenge the relieving capacity of the MSSVs, and thus RCS pressure, are those characterized as decreased heat removal events, which are presented in the FSAR, Section 15.2 (Ref. 3). Of these, the full power turbine trip without steam dump is the limiting AOO. This event also terminates normal feedwater flow to the steam generators.

(continued)

Enclosure 5

Vogtle Electric Generating Plant
Request to Change Licensed Maximum Power Level
Summary of Engineering Reviews

SECTION I

FEEDWATER FLOW MEASUREMENT TECHNIQUE AND POWER MEASUREMENT UNCERTAINTY

SECTION I

FEEDWATER FLOW MEASUREMENT TECHNIQUE AND POWER MEASUREMENT UNCERTAINTY

A. Approved Topical Reports on Feedwater Flow Measurement Technique

The reference Topical Reports are as follows:

1. ER-80P, "Improving Thermal Power Accuracy and Plant Safety While Increasing Operating Power Level Using the LEFM Check System," dated March 1997.
2. ER-160P, "Supplement to Topical Report ER-80P: Basis for a Power Uprate with the LEFM Check System," dated May 2000.
3. ER-157P, "Supplement to Topical Report ER-80P: Basis for a Power Uprate with the LEFM Check or CheckPlus System," dated October 2001.

B. NRC Approval of Feedwater Flow Measurement Technique

The NRC approved the subject Topical Reports referenced in Item A above on the following dates:

1. ER-80P, NRC SER dated March 8, 1999
2. ER-160P, NRC SER dated January 19, 2001
3. ER-157P, NRC SER dated December 20, 2001

The LEFM CheckPlus System will be permanently installed at VEGP in accordance with the requirements of ER-80P, ER-160P and ER-157P.

C. VEGP-Specific Implementation of Guidelines in Staff's Safety Evaluation

The feedwater flow measurement system to be installed in each VEGP unit is a Caldon LEFM CheckPlus ultrasonic multi-path transit time flow meter as described in Caldon Topical Reports ER-80P, "Improving Thermal Power Accuracy and Plant Safety While Increasing Operating Power Level Using the LEFM Check System," dated March 1997 and approved by NRC on March 8, 1999, and ER-157P, "Supplement to Topical Report ER-80P: Basis for a Power Uprate with the LEFM Check or CheckPlus System," dated October 2001 and approved by NRC on December 20, 2001.

Each LEFM CheckPlus System at VEGP will consist of one spool piece measurement section integrally welded into each unit's common feedwater header upstream of the four feedwater loops. Each measurement section will be installed in a 31 foot-10 inch (31'-10") length of vertical section of feedwater piping approximately 96 inches (96") downstream of an upstream 90° elbow. The header length and elbow configuration is the same for each of the VEGP units. The installation location of each measurement section conforms to the requirements in Caldon Topical Reports ER-80P and ER-157P. Each measurement section will be installed in accordance with approved VEGP procedures and work controls processes to achieve an installation tolerance within the bounds stated in the Caldon uncertainty analyses. A Caldon LEFM CheckPlus 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.

The Caldon LEFM CheckPlus Systems will be permanently installed at VEGP in accordance with the requirements of Topical Reports ER-80P and ER-157P and approved VEGP procedures. The systems will determine feedwater parameters to be used for continuous thermal power calorimetric measurement and communicate these parameters to the VEGP 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 CheckPlus System will communicate with the IPC via an Ethernet digital communications interface, and feedwater data will be transmitted to the IPC via fiber optic cables and data converters. Dual data outputs from the LEFM CheckPlus cabinet will provide redundancy and fault tolerance for IPC communications. The communication links will provide raw and conditioned data for input into the secondary calorimetric algorithm resident on the IPC. Raw data will include diagnostic information and unconditioned process data (i.e., normalized path velocities, acoustic gains, speed of sound, path status and data rejects). The conditioned data includes the following intermediate and final calculation results to be used in the calorimetric algorithm, alarm status, and operational interface.

- Total mass flow
- Acoustically derived feedwater temperature
- Feedwater Pressure
- System maintenance and failure status

The diagnostic and signal quality data will be communicated to the IPC to allow monitoring for degradation of LEFM CheckPlus System parameters as well as triggering of a control room alarm when conditions are at a state which could impact the flow measurement uncertainty.

The VEGP LEFM CheckPlus measurement sections were calibrated in a site specific model test at Alden Research Laboratories (October 30 through November 3, 2006) with all calibration standards traceable to National Institute of Standards and Technology (NIST) standards. The test report is included in Enclosure 8 of this submittal. Each measurement section was installed in a common piping section which modeled the actual header configuration (the same flow model was used to test the Unit 1 and Unit 2 metering sections). The Alden VEGP specific test plan was developed to provide meter factor calibration data over a wide range of hydraulic test conditions. The tests included plant piping modeling and parametric variations of those models, straight pipe testing, and inducement of extreme swirl conditions. The Meter Factor data, determined by comparing the Alden Lab reference standard to the flow as measured by the Caldon

FEEDWATER FLOW MEASUREMENT TECHNIQUE AND POWER MEASUREMENT UNCERTAINTY

LEFM CheckPlus System, were collected for each piping configuration at various flow rates. Measurements of the hydraulic profile, called Flatness Ratio, were also collected at each flow rate. The Meter Factor versus Flatness Ratio was plotted for all conditions and all flow rates and was compared to analytically derived expected performance curves for quality control purposes. These data provide a quantitative measure of Caldon LEFM CheckPlus Meter Factor versus actual velocity profile encountered and determines the meter uncertainty used in the overall calorimetric uncertainty analysis discussed in Section E below.

Following installation of the LEFM CheckPlus System at VEGP, in-situ measurements of feedwater flow velocity profile will be compared to the reference measurements that were collected during the Alden Lab testing. It is expected that the range of velocity flow profiles encountered during the VEGP Alden Laboratory testing will envelope the velocity profiles encountered at the plant and will require no extrapolation. However, in the event swirl conditions encountered in the plant exceed the limits set in the uncertainty analysis, the profile factor value will be revisited, taking into account the high swirl data. The high swirl tests conducted at Alden Laboratories represent conditions that are not expected to exist with the plant at full power. A preliminary estimate of the uncertainty associated with applying the calibration data in the plant was made as part of the Caldon calibration test report submitted to SNC.

A final verification of the feedwater mass flow measurement uncertainty provided in the Caldon uncertainty analysis reports (Enclosures 6 and 7 for Units 1 and 2 respectively) will be made following installation and commissioning of the LEFM CheckPlus meter based on in-plant measurements of the hydraulic velocity profile flatness. These data will be collected and the comparison to laboratory data will be made per Caldon Engineering Field Procedure EFP-61. This procedure will include, but not be limited to, verification of ultrasonic sound quality, hydraulic velocity profile, swirl, etc.

The LEFM CheckPlus System parameters that will be controlled, displayed, or alarmed in the control room are presented below.

<u>LEFM CheckPlus System Controls</u>

- There are no LEFM CheckPlus System controls available in the main control room. All control functions reside locally at the LEFM CheckPlus System cabinets to be located in the Turbine building.
- The control room operators will have the ability to select the LEFM CheckPlus System output as the source of input data for the Integrated Plant Computer (IPC) secondary calorimetric calculation via a control room display.

SECTION I

FEEDWATER FLOW MEASUREMENT TECHNIQUE AND POWER MEASUREMENT UNCERTAINTY

LEFM CheckPlus System Displays

- The calculated core power level (MWt) based on the LEFM CheckPlus System outputs will be available for display on the IPC computer for use by the control room staff.
- Detailed LEFM CheckPlus System process and diagnostic data, communicated to the IPC, will also be available via display in the main control room for use by the operations staff for diagnosis of system alarms.

LEFM CheckPlus System Alarms

The following conditions will cause the IPC to trigger a common LEFM CheckPlus System alarm on the main control board (MCB) annunciator panel. There are no hardwired alarms from the LEFM CheckPlus System cabinet to the control room. An operations annunciator response procedure (ARP) will be written to give the operations staff clear direction to determine the cause of the alarm and necessity to start the 48-hour Out of Service (OOS) clock for corrective action. In some cases, an operator may be directed to the local LEFM cabinet display to determine the specific cause of the failure alarm.

The following conditions will trigger the MCB alarm:

LEFM CheckPlus System Meter Status Not Normal

The meter status (Normal, Alert, or Failed) will be communicated to the IPC, and the MCB alarm will be triggered if the status is detected in an alert or failed mode which indicates a condition that adversely affects the uncertainty of the LEFM CheckPlus System mass flow determination and will notify operations to start the 48-hour OOS clock for corrective action.

IPC Data Link Failure

A communications failure from the LEFM CheckPlus cabinet to the IPC results in a MCB alarm and would require start of the 48-hour OOS clock for corrective action.

LEFM CheckPlus System Cabinet High Temperature

The LEFM CheckPlus cabinet temperature will be communicated to the IPC and appropriate alarm limits established for corrective action. The alarm alert limit will be set to allow monitoring and timely corrective action (i.e., air conditioner maintenance) prior to exceeding the temperature limit for LEFM CheckPlus operability. If the maximum limit is exceeded, operations will be required to start the 48-hour OOS clock for corrective action.

SECTION I

FEEDWATER FLOW MEASUREMENT TECHNIQUE AND POWER MEASUREMENT UNCERTAINTY

D. Disposition of Criteria in Staff's Safety Evaluation

In approving Topical Reports ER-80P and ER-157P, the NRC established four criteria to be addressed by each licensee. The four criteria and a discussion of how SNC will address each is discussed 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.

Response to Criterion 1

Implementation of the power uprate license amendment will include developing the necessary procedures and documents required for operation, maintenance, calibration, testing, and training at the uprate power level with the new Caldon LEFM CheckPlus System. Plant procedures will be revised to incorporate Caldon's maintenance and calibration requirements prior to declaring the Caldon LEFM CheckPlus System operational and raising reactor core power above 3565 MWt. The incorporation of, and continued adherence to, these requirements will assure that the Caldon LEFM CheckPlus System is properly maintained and calibrated.

System maintenance and contingency plans for operation of the plant with the Caldon LEFM CheckPlus out of service are described in detail in Sections F and G below.

Criterion 2

For plants that currently have LEFMs installed, provide an evaluation of the operational and maintenance history of the installed installation 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.

Response to Criterion 2

This Criterion is not applicable to VEGP, as the plant 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 applied to both venturi and ultrasonic flow measurement instrumentation installations for comparison.

Response to Criterion 3

The total power calorimetric uncertainty 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 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.

Criterion 4

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

Response to Criterion 4

Criterion 4 does not apply to VEGP. 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 VEGP feedwater header hydraulic geometry and tests in a straight pipe. The same Alden test model was used for testing of both the Unit 1 and Unit 2 measurement sections. An Alden data report for these tests (Enclosure 8 of this submittal) and Caldon uncertainty reports (Enclosures 6 and 7 for Units 1 and 2 respectively) evaluating the test data were provided to Southern Nuclear. The calibration meter factor and the uncertainty in the calibration factor used for the Caldon LEFM CheckPlus system at VEGP 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 VEGP MUR per QA record retention requirements. Final acceptance of the VEGP 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 CheckPlus Systems is expected to be completed following the Unit 1 refueling outage in Spring 2008 and the Unit 2 refueling outage in Fall 2008.

E. Calculation of Total Power Measurement Uncertainty

The thermal power calorimetric uncertainty to support the VEGP uprate is presented in the Caldon reports in Enclosure 6 for Unit 1 and Enclosure 7 for Unit 2. Table I-1 shows that the uncertainty results for Unit 1 and Unit 2 are identical. While the uncertainty reports show that there are minute differences between the Unit 1 and Unit 2 spool piece meter factors, transducer path spacing, transducer installed angles, and electronic timing, (contributes to Items 1-3 in Table I-1), these differences are not large enough to cause a variance in the final calculated thermal power uncertainty values for Unit 1 and Unit 2. The overall thermal power calorimetric uncertainty for both VEGP units is shown to be 0.30%.

The uncertainties shown in Table I-1 were determined utilizing the calculation methodology described in Caldon Topical Report ER-80P, as amended by ER-157P.

<u>Other Process Inputs</u>

In addition to the calorimetric inputs provided by the Caldon LEFM CheckPlus System for determination of feedwater mass flow rate and enthalpy, the VEGP plant computer uses the process inputs listed below to calculate the contribution of steam enthalpy and other gains/losses (Items 7 and 8 in Table I-1) in the calorimetric algorithm for calculation of reactor core thermal power. An uncertainty calculation was performed for each of the parameters below to determine a bounding instrument loop uncertainty for Unit 1 and Unit 2. The thermal power sensitivity of each parameter was multiplied by the instrument loop uncertainty to determine the total impact on the thermal power calorimetric for each loop. The effects from all loops were combined via SRSS, since the uncertainties are random. The result was combined appropriately with the Caldon LEFM system uncertainties to determine an overall thermal power uncertainty. The results are incorporated in the Caldon uncertainty reports in Enclosures 6 and 7 for Units 1 and 2, respectively.

Process inputs:

- Charging Flow
- Letdown Flow

- Steam Generator Blowdown Flow
- Letdown Pressure
- Charging Pump Discharge Pressure
- Steam Line Pressure
- Charging Temperature
- Volume Control Tank Temperature
- Letdown Temperature
- Feedwater Pressure
- RCS Wide Range Temperature
- Pressurizer Pressure

With the exception of the steam line pressure instrumentation, no changes will be required for existing plant process instrumentation to support the Caldon calorimetric uncertainties as documented in Enclosures 6 and 7. The existing steam line pressure transmitters on both Units will be replaced prior to the uprate with new Rosemount transmitters to achieve a lower uncertainty than the existing devices and to meet the assumptions made for steam line pressure uncertainty. The new feedwater pressure instrumentation to support the uprate will be integrally attached to the Caldon LEFM CheckPlus measurement sections and have specifications which are bounded by the assumed uncertainty in Enclosures 6 and 7.

F. Calibration and Maintenance Procedures

The following information addresses specific aspects of calibration, maintenance, and corrective action pertaining to all instruments that affect the calorimetric. (Those that will pertain to the Caldon LEFM CheckPlus System and to the instrumentation for other calorimetric process inputs are listed in Section E.

i) <u>Maintaining Calibration</u>

Calibration and maintenance for the Caldon LEFM CheckPlus System hardware and instrumentation will be performed using procedures which have been developed based on the appropriate Caldon LEFM CheckPlus technical manuals. The other calorimetric process instrumentation and computer points are maintained and periodically calibrated in accordance with approved procedures. Preventative Maintenance (PM) 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 VEGP work control processes and procedures.

- Instrumentation and Control (I&C) personnel will be trained and qualified, per the VEGP I&C training program, on the LEFM CheckPlus System prior to performing maintenance or calibration on the system. Formal training on the system operation and maintenance by Caldon will be provided to the appropriate engineering and maintenance personnel. Operations personnel will be trained by qualified VEGP instructors using materials which have been developed per VEGP Training department guidelines and procedures. Training will be completed prior to Caldon LEFM CheckPlus System commissioning.

- Routine PM activities for the Caldon LEFM CheckPlus 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 CheckPlus electronics unit. Ultrasonic signal verification and alignment is performed automatically by the LEFM CheckPlus System. Signal verification will be determined by review of signal quality measurements performed and displayed by the LEFM CheckPlus System. Selected I&C personnel in the maintenance department will be trained and qualified per the Institute for Nuclear Power Operations (INPO) accredited Southern Nuclear training program before maintenance or calibration is performed and prior to increasing power above 3565 MWt. This training will also include industry Operational Experience and formal training provided by Caldon.

ii) <u>Controlling Software and Hardware Configuration</u>

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

iii) <u>Performing Corrective Actions</u>

Plant instrumentation that affects the power calorimetric, including the Caldon LEFM CheckPlus inputs, will be monitored by the VEGP System Engineering personnel. These same instruments are included in the VEGP PM program and/or the Technical Specification surveillance program for periodic calibration. Problems that are detected are documented per the VEGP corrective action process and necessary actions are planned and implemented. Corrective action procedures, which ensure compliance with the requirements of 10 CFR 50, Appendix B, include instructions for notification of deficiencies and error reporting.

The LEFM CheckPlus System performance will be monitored and documented per the requirements of the VEGP 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 VEGP corrective action program and needed actions will be controlled by the VEGP work control process.

iv) Reporting Deficiencies to the Manufacturer

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

v) Receiving and Addressing Manufacturer Deficiency Reports

VEGP has processes in place for addressing manufacturer's deficiency reports. Such deficiencies will be documented in the VEGP corrective action program and actions will be controlled by the VEGP work control process.

G. Proposed Allowed Outage Time

Each of the Caldon LEFM CheckPlus Systems to be installed at VEGP will consist of a single measurement section installed in a common section of feedwater piping. Each measurement section consists of two planes of transducers (four paths in each plane) as described in ER-157P. Each plane provides input to its own subsystem of electronics hardware. The electronics for the two subsystems, while electrically separated, are housed in a single processing cabinet. The LEFM CheckPlus System performs on line self-diagnostics to verify that the system is operating within its design basis uncertainty limits. In general, a failure of one or more transducer paths or failure of communications from the Caldon LEFM CheckPlus cabinet to the plant computer will result in generation of a failure alarm on the main control board. A control room annunciator response procedure (ARP) will be developed to provide guidance to the control room operations staff for initial alarm diagnosis and response. Depending on the alarm condition, the "clock" may be started for the allowed outage time for the LEFM CheckPlus System.

The proposed allowed outage time for operation at any power level in excess of the current licensed power level (3565 MWt) with the Caldon LEFM CheckPlus System out of service is 48 hours provided steady state conditions persist (i.e., no power changes in excess of 10 percent) throughout the 48-hour period. The bases for the proposed allowed outage time are as follows:

- A back-up calorimetric algorithm which receives input from alternate plant instruments for the calculation of feedwater mass flow rate (feedwater venturis and RTDs) will be available. Specifically, the total feedwater flow from the four venturis will be normalized to the Caldon LEFM CheckPlus feedwater mass flow rate so that the alternate calorimetric closely matches the primary LEFM based calorimetric. Alternate instrumentation accuracy could degrade over time as a result of nozzle fouling or transmitter drift, but this degradation will not result in significant uncertainty associated with the calorimetric measurement over a 48-hour period. Operations procedures will direct the use of the back-up calorimetric in the event of LEFM CheckPlus failure.

- Most repairs to the Caldon LEFM CheckPlus System are expected to be complete within an 8-hour shift. Forty eight hours gives plant personnel time to plan and package the work orders, make repairs, and verify normal operation of the LEFM system within its original uncertainty bounds. Repairs and verification of normal LEFM CheckPlus operation will be required to be completed within 48 hours of failure, or the plant will have to be derated to the pre-uprate licensed power level of 3565 MWt prior to expiration of the 48 hours.

- If 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 power level of 3565 MWt, since a plant transient may impact the calibration accuracy of the alternate calorimetric instrumentation.

- As described in ER-157P, the Caldon LEFM CheckPlus System will consist of two planes (eight paths total) of transducers. While a malfunction in a single plane results in only a minimal increase in feedwater flow uncertainty, Operators will conservatively respond to a failure in a single section in the same manner as that for a complete system failure (two planes failed). While conservative, this approach will simplify the control room response and minimize the chance for misdiagnosis of the failure mode.

For the Caldon LEFM CheckPlus system out-of-service condition or loss of plant computer communication, the 48-hour out-of-service time will start at the time of the failure annunciated in the main control room. Operations procedures will be developed to allow an initial diagnosis for the cause of the alarm. The method of identifying the status of the LEFM CheckPlus System data and cause of the alarm is described in the Caldon vendor documentation and software design descriptions for the data link. The Caldon documentation will be used to develop specific procedures for operator and maintenance response actions.

SECTION I

FEEDWATER FLOW MEASUREMENT TECHNIQUE AND POWER MEASUREMENT UNCERTAINTY

The Caldon LEFM CheckPlus electronic unit and central processing unit will continuously monitor, test, and/or verify the following attributes of system performance. A failure alarm on the main control board will be initiated if any of these parameters is outside the allowed bounds for acceptable flow measurement:

- Profile Factor
- Spool Piece Dimensions
- Clock Accuracy
- Transmitter/Receiver Reciprocity
- Signal-to-Coherent noise ratio
- Signal-to-Noise Ratio
- Non Fluid Delays
- Pressure

H. Proposed Actions to Reduce Power

The Caldon LEFM CheckPlus System to be installed at VEGP (one system per unit) will consist of a single measurement spool piece installed in the common feedwater header and a single electronics cabinet. Both components will be located on Level 1 (essentially ground level) of the VEGP turbine buildings. Failure of the Caldon LEFM CheckPlus System will result in a calculation of thermal power based on operation of the feedwater venturis and RTDs in the feedwater lines. With the Caldon LEFM CheckPlus System out of service for greater than 48 hours, the thermal power uncertainty increases such that the justifiable power level is reduced from 3625.6 MWt to 3565 MWt. Plant procedures will be revised to ensure that the core power level will be at or below the current licensed core power level of 3565 MWt within 48 hours in the event of a loss of the Caldon LEFM CheckPlus System.

In the event the Caldon LEFM CheckPlus System is out of service, the feedwater mass flow rate and enthalpy inputs into the calorimetric must be determined by alternate instrumentation. The feedwater mass flow rate will be determined from venturi-based flow rates, and the feedwater enthalpy will be based on RTDs installed in the feedwater piping. To ensure that the venturi-based calorimetric is consistent with the LEFM based calorimetric, the venturi-based mass flow rate will be normalized to the Caldon LEFM CheckPlus flow, i.e., adjusted by a constant factor which represents the normal ratio of Caldon flow to the venturi flow. A loss of the Plant Computer will be treated as a loss of both the Caldon LEFM CheckPlus System and the ability to obtain a corrected calorimetric power using the alternate plant instrumentation. 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.

SECTION I

FEEDWATER FLOW MEASUREMENT TECHNIQUE AND POWER MEASUREMENT UNCERTAINTY

Table I-1 Total Thermal Power Uncertainty Determination

Parameter	**ER-157P Rev 5 Uncertainty**	**Enclosure 6 U1 VEGP Specific Uncertainty (1)**	**Enclosure 7 U2 VEGP Specific Uncertainty (1)**
1. Hydraulics: Profile factor	0.25%	0.20%	0.20%
2. Geometry: Spool dimensions Spool piece alignment Spool piece thermal expansion Material properties	0.10%	0.10%	0.10%
3. Time Measurements: Time of flight measurements Non Fluid Delay	0.05%	0.09%	0.09%
4. Feedwater Density: (2) Feedwater Density/Correlation Feedwater Density/Temperature Feedwater Density/Pressure (4)	0.07%	0.07%	0.07%
5. Subtotal: Mass flow uncertainty	0.28%	0.25%	0.25%
6. Feedwater Enthalpy (Pressure and Temperature) (3) Feedwater Enthalpy/Temperature Feedwater Enthalpy/Pressure Power Uncertainty, Thermal Expansion	0.08% 0.12%	0.07% 0.13%	0.07% 0.13%
7. Steam Enthalpy: (Pressure and moisture)	0. 07%	0.06%	0.06%
8. Gains/ Losses	0.07%	0.07%	0.07%
Total Thermal Power Uncertainty Square Root Sum Square (SRSS)	**0.33%**	**0.30%**	**0.30%**

Notes

1. Items 1-6 are directly associated with the Caldon LEFM CheckPlus system device. Items 7 and 8 are based on VEGP-specific process inputs.
2. Density errors due to the density correlation, the LEFM feedwater temperature determination and the feedwater pressure measurement.
3. Enthalpy errors due to the enthalpy correlation, the LEFM feedwater temperature determination and the feedwater pressure measurement.
4. The bounding uncertainties in pressure and temperature are ± 15 psi and 0.6 °F, respectively.

SECTION II

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

SECTION II

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

ACCIDENTS

Table II-1 summarizes the accidents for Chapter 15 of the VEGP UFSAR that were reviewed for the MUR power uprate. As indicated in the table, the summary of the reviews is included in Section 7.3 of Enclosures 9 and 10.

OTHER ANALYSES

Table II-1 summarizes other analyses listed in Section II.1 of RIS 2002-03. The table references the enclosures and sections of this submittal containing the summaries. Summaries not included in other enclosures are described below.

Containment Response

The containment response was evaluated for the following cases:

1. LOCA long-term mass and energy release
2. LOCA short-term mass and energy release (sub-compartments)
3. MSLB mass and energy release inside containment
4. Inadvertent containment spray actuation

For each of these cases, the NSSS vendor evaluation concluded that the current licensing-basis mass and energy releases are bounding for the MUR power uprate. The containment responses for Cases 1, 3, and 4 were evaluated by the NSSS vendor. The containment response for Case 2 was evaluated by the architect-engineer. The evaluations concluded that the current licensing-basis analyses are bounding for the MUR power uprate.

Post-LOCA Hydrogen Generation

The post-LOCA hydrogen generation analysis-of-record is based on the current core power level of 3565 MWt.

Following a Loss of Coolant Accident (LOCA), hydrogen gas may be generated inside the containment by reactions such as zirconium metal with water, corrosion of materials of construction, and radiolysis of aqueous solution in the core and in the sump. The MUR power uprate is for a licensed core power level increase of 1.7%. The impacts of MUR power uprate on the hydrogen generation analysis are addressed below:

1. Zirconium-metal reaction is the core wide oxidation (CWO) of the zirconium fuel cladding and the reactor coolant. In the analysis, the hydrogen from zirconium-metal reaction is determined based directly on the fuel weight/cladding surface

area. The hydrogen generation from this reaction is not dependent on the power level. Therefore, the MUR power uprate has no impact on the hydrogen generation from this source, since there is no change to the fuel.

2. The hydrogen generation due to corrosion is based on corrosion rates which are a function of post-LOCA containment temperatures, which affect the corrosion of metals in containment by solutions used for containment spray. Oxidation of metals such as aluminum (Al) and zinc (Zn) in aqueous solution results in the generation of hydrogen gas as one of the corrosion products. Addition of these hydrogen-producing materials inside the containment would impact hydrogen generation due to corrosion; however, any change to the Al and Zn inventories is not directly due to the MUR power uprate. The MUR power uprate can only impact corrosion if the LOCA containment temperatures change, but the MUR power uprate has no effect on the post-LOCA containment pressure and temperature response (Section 7.5.1 of Enclosures 9 and 10). Therefore, the MUR power uprate has no impact on hydrogen generation due to corrosion.

3. Hydrogen generation due to radiolysis is potentially affected by the MUR power uprate. The slightly higher MUR power level determines the fission product energy which affects the radiolysis of the coolant in the core as well as the radiolysis of the water in the sump. Water radiolysis is a complex process involving reactions of numerous intermediates. As the emergency core cooling solution flows through the core, it is subjected to gamma radiation by decaying fission products. This energy deposition results in solution radiolysis and the production of molecular hydrogen and oxygen. Another potential source of hydrogen assumed for the post-accident period arises from water contained in the reactor containment sump being subjected to radiolytic decomposition by fission products.

   In the existing post-LOCA hydrogen generation analysis, the amount of hydrogen generated due to radiolysis using the original Al and Zn surface areas is less than 50% of the total hydrogen generated from all sources. Hydrogen generation from radiolysis is proportional to reactor power level. For a 1.7% MUR power uprate, the amount of hydrogen generated due to radiolysis would increase less than 50% of 1.7% or less than 1% of the total hydrogen generated from all sources. This small increase in hydrogen due to radiolysis for MUR power uprate would have a minimal impact on the total post-LOCA hydrogen generation.

Based on the discussion above, it is concluded that the post-LOCA hydrogen generation analysis-of-record is still applicable for the MUR power uprate condition.

SECTION II

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

Station Blackout (SBO) Coping Evaluation

This item is addressed in Section V, Item B, of this enclosure.

Analyses to Determine Equipment Qualification (EQ) Parameters

Critical EQ parameters include temperature, pressure, radiation (integrated dose), and relative humidity.

Mass and energy releases for long-term LOCA and steamline break for current analyses remain bounding for the MUR power uprate conditions as the analyses of record assume a core thermal power of 102% of 3565 MWt. These are discussed in Sections 7.4.1, 7.6.1, and 7.6.2 of Enclosures 9 and 10. The MUR power uprate causes the gamma and beta doses inside containment to increase by less than 10% and the gamma doses outside containment to increase by 5%. These increases are within the accuracy of the analyses. Radiological doses used in the EQ evaluations bound the increase in doses due to the MUR power uprate.

Therefore, it is concluded that the current EQ parameters remain bounding for the MUR power uprate.

Safe Shutdown Fire Analyses

The Fire Protection System (FPS) is considered as one integrated system that involves all aspects of power plant fire prevention, detection, extinguishment, and the mitigation of the consequences of a fire. 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 and added heat load to the plant environment is not significant. No physical changes are being made to buildings in the plant that would affect fire loading or add combustibles or fire initiation sources as a result of the MUR power uprate. There is no change in the fire detection and protection systems for the safe shut down aspect of the Fire Protection Water System (FPWS) that could affect its safe shut down capability. Evaluation of the FPWS concluded that the system parameters are not changed significantly nor adversely affected by the MUR power uprate and are bounded by the existing design basis and analyses.

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

Spent Fuel Pool Cooling

Spent Fuel Pool Bulk Heat Load and Temperature Limit

The bulk temperature limit for both the Unit 1 and Unit 2 spent fuel pools is not changing. The pool bulk water temperature limit is maintained by managing the spent fuel pool heat load. The heat load is maintained within limit through administrative controls. Since the heat load limit for the pools is not changing, there is no impact on the spent fuel pool cooling requirements.

Spent Fuel Pool Local Temperature Analysis

The current licensing-basis local temperature analysis is non-bounding at the MUR power uprate conditions. The revision of the analysis is discussed in this section for completeness.

This analysis determined the maximum local water temperature and the maximum fuel clad temperature in the spent fuel racks containing the hottest fuel assemblies. The local water temperature analysis for the MUR power uprate was performed employing a Computational Fluid Dynamic (CFD) technique using the Holtec FLUENT fluid flow and heat transfer modeling program. Since the decay heat assumed in the local temperature analysis is dependent on the core power level, the analyses for both units were revised using 102% of the current core power level which bounds the MUR power uprate. Using the maximum local water temperature, the maximum fuel clad temperature was calculated using the principles of laminar flow heat transfer. This approach for determining the maximum local water temperature and the maximum fuel clad temperature in the spent fuel racks has been previously approved for use for VEGP Unit 1 (Amendment 102 for Unit 1 and Amendment 80 for Unit 2, June 29, 1998).

The results demonstrated that considerable margin exists between the saturation temperature at the top of the active fuel and the maximum local water and maximum fuel clad temperatures.

Time-to-Boil Analysis

As described above, the bulk water temperature and heat load limits are not affected by the MUR power uprate. Therefore, there is no impact on the time-to-boil. Refer also to Section VI.D of this Enclosure.

SECTION II

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

Flooding

The design bases for flooding analysis inside and outside the containment building were evaluated. For the MUR power uprate, the piping systems with increased flow rates (e.g., condensate and main feedwater piping and main steam piping) were evaluated to determine any impact on the flooding analysis. Based on a review of flooding analysis calculations it was determined that the current flood levels in the flooding analyses are not affected by the MUR power uprate.

Therefore, it is concluded that the current flooding analyses remain bounding for the MUR power uprate.

SECTION II

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

TABLE II-1 (Sheet 1 of 13)
ACCIDENTS/TRANSIENTS AND OTHER ANALYSES

ACCIDENTS

Accident/Transient	UFSAR Section	Bounding? Yes/No	CLB Assumed Reactor Power Level (%)	LAR Section	Prior NRC Review or 10CFR50.59	Reanalysis or Evaluation
Decrease in FW Temperature	15.1.1	Yes (Note 1)	N/A	Encl. 9 & 10 §7.3.1	Reference 2 Reference 5	Reanalysis Evaluation
Increase in FW Flow (HZP)	15.1.2	Yes	0	Encl. 9 & 10 §7.3.1	Reference 2 Reference 5	Reanalysis Evaluation
Increase in FW Flow (HFP)	15.1.2	Yes	100	Encl. 9 & 10 §7.3.1	Reference 2 Reference 5	Reanalysis Evaluation
Excessive Increase in Steam Flow	15.1.3	Yes	100	Encl. 9 & 10 §7.3.1	Reference 2 Reference 5	Reanalysis Evaluation
Inadvertent Opening of SG Relief or Safety Valve	15.1.4	Yes	0	Encl. 9 & 10 §7.3.1	Reference 5	Reanalysis
Steam System Piping Failure (HZP)	15.1.5	Yes	0	Encl. 9 & 10 §7.3.1	Reference 5 50.59	Reanalysis Reanalysis

SECTION II

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

TABLE II-1 (Sheet 2 of 13)

ACCIDENTS (CONTINUED)

Accident/Transient	UFSAR Section	Bounding? Yes/No	CLB Assumed Reactor Power Level (%)	LAR Section	Prior NRC Review or 10CFR50.59	Reanalysis or Evaluation
Steam System Piping Failure (HFP)	15.1.5	Yes	102	Encl. 9 & 10 §7.3.1	Reference 9	Reanalysis
Steam Pressure Regulator Malfunction	15.2.1	N/A for VEGP	N/A	Encl. 9 & 10 §7.3.1	N/A	N/A
Loss of Electrical Load	15.2.2	Yes (Note 1)	N/A	Encl. 9 & 10 §7.3.1	Reference 9	Reanalysis
Turbine Trip with Pressure Control	15.2.3	Yes	100	Encl. 9 & 10 §7.3.1	Reference 9	Reanalysis
Turbine Trip without Pressure Control	15.2.3	Yes	102	Encl. 9 & 10 §7.3.2	Reference 9	Reanalysis
Inadvertent Closure of MSIV	15.2.4	Yes (Note 1)	N/A	Encl. 9 & 10 §7.3.1	OLB Reference 2 Reference 5	 Evaluation Evaluation
Loss of Condenser Vacuum	15.2.5	Yes (Note 1)	N/A	Encl. 9 & 10 §7.3.1	OLB Reference 2 Reference 5	 Evaluation Evaluation

SECTION II

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

TABLE II-1 (Sheet 3 of 13)

ACCIDENTS (CONTINUED)

Accident/Transient	UFSAR Section	Bounding? Yes/No	CLB Assumed Reactor Power Level (%)	LAR Section	Prior NRC Review or 10CFR50.59	Reanalysis or Evaluation
Loss of Non-Emergency AC	15.2.6	Yes	102	Encl. 9 & 10 §7.3.2	Reference 5 50.59	Reanalysis Reanalysis
Loss of Normal Feedwater Flow	15.2.7	Yes	102	Encl. 9 & 10 §7.3.2	Reference 5 50.59	Reanalysis Reanalysis
Feedwater System Pipe Break	15.2.8	Yes	102	Encl. 9 & 10 §7.3.2	Reference 1 Reference 2 Reference 5 50.59	Reanalysis Evaluation Evaluation Reanalysis
Partial Loss of Reactor Coolant Flow	15.3.1	Yes	100	Encl. 9 & 10 §7.3.1	Reference 2 Reference 5	Reanalysis Evaluation
Complete Loss of Forced Reactor Coolant Flow	15.3.2	Yes	100	Encl. 9 & 10 §7.3.1	Reference 2 Reference 5	Reanalysis Evaluation
Locked Rotor - Rods in DNB	15.3.3	Yes	100	Encl. 9 & 10 §7.3.1	Reference 2 Reference 5	Reanalysis Evaluation

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

TABLE II-1 (Sheet 4 of 13)

ACCIDENTS (CONTINUED)

Accident/Transient	UFSAR Section	Bounding? Yes/No	CLB Assumed Reactor Power Level (%)	LAR Section	Prior NRC Review or 10CFR50.59	Reanalysis or Evaluation
Locked Rotor - Peak Pressure	15.3.3	Yes	102	Encl. 9 & 10 §7.3.2	Reference 2 Reference 5	Reanalysis Evaluation
Reactor Coolant Pump Shaft Break	15.3.4	Yes (Note 1)	N/A	Encl. 9 & 10 §7.3.1	OLB Reference 2 Reference 5	 Evaluation Evaluation
Uncontrolled RCCA Bank Withdrawal from Subcritical or Low Power Startup	15.4.1	Yes	0	Encl. 9 & 10 §7.3.2	Reference 2 Reference 5	Reanalysis Evaluation
Uncontrolled RCCA Bank Withdrawal at Power	15.4.2	Yes (Note 2)	10/60/100	Encl. 9 & 10 §7.3.1	Reference 9	Reanalysis
RCCA Misalignment	15.4.3	Yes	100	Encl. 9 & 10 §7.3.1	Reference 3 Reference 5	Reanalysis Evaluation
Startup of Inactive Loop	15.4.4	Yes (Note 3)	About 72%	Encl. 9 & 10 §7.3.1	Reference 2 Reference 5	Reanalysis Evaluation

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

TABLE II-1 (Sheet 5 of 13)

ACCIDENTS (CONTINUED)

Accident/Transient	**UFSAR Section**	**Bounding? Yes/No**	**CLB Assumed Reactor Power Level (%)**	**LAR Section**	**Prior NRC Review or 10CFR50.59**	**Reanalysis or Evaluation**
BWR Accident	15.4.5	N/A	N/A	N/A	N/A	
CVCS Malfunction - Decrease in RCS Boron Concentration	15.4.6	Yes (Note 4)	0/100	Encl. 9 & 10 §7.3.1	Reference 2 Reference 4 Reference 5 Reference 9	Reanalysis Evaluation Evaluation Evaluation
Inadvertent Loading of Fuel Assembly	15.4.7	Yes	N/A	Encl. 9 & 10 §7.3.3	OLB Reference 2	 Evaluation
Spectrum of RCCA Ejection Accidents (HZP)	15.4.8	Yes	0	Encl. 9 & 10 §7.3.2	Reference 2 Reference 5	Reanalysis Evaluation
Spectrum of RCCA Ejection Accidents (HFP)	15.4.8	Yes	102	Encl. 9 & 10 §7.3.2	Reference 2 Reference 5	Reanalysis Evaluation
Steamline Break/Concurrent RCCA Withdrawal	15.4.9	N/A	N/A	Encl. 9 & 10 §7.3.1	50.59	Deleted from Licensing Basis
Inadvertent Operation of ECCS at Power (DNB)	15.5.1	Yes	100	Encl. 9 & 10 §7.3.1	Reference 2 Reference 5	Reanalysis Evaluation

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

TABLE II-1 (Sheet 6 of 13)

ACCIDENTS (CONTINUED)

Accident/Transient	UFSAR Section	Bounding? Yes/No	CLB Assumed Reactor Power Level (%)	LAR Section	Prior NRC Review or 10CFR50.59	Reanalysis or Evaluation
Inadvertent Operation of ECCS at Power (PZR Fill)	15.5.1	Yes	102	Encl. 9 & 10 §7.3.2	Reference 6 50.59	Reanalysis Reanalysis
CVCS Malfunction - Increase in RCS Inventory	15.5.2	Yes (Note 1)	N/A	Encl. 9 & 10 §7.3.1	OLB Reference 2 Reference 5	 Evaluation Evaluation
BWR Transients	15.5.3	N/A	N/A	N/A	N/A	N/A
Inadvertent Opening of Pressurizer Safety or Relief Valve	15.6.1	Yes	100	Encl. 9 & 10 §7.3.1	Reference 9	Reanalysis
Steam Generator Tube Failure (Overfill)	15.6.3	Yes	102	Encl. 9 & 10 §7.7	Reference 5 50.59	Reanalysis Reanalysis
BWR Transients	15.6.4	N/A	N/A	N/A	N/A	N/A
Small-Break LOCA - Thermal-hydraulic transient	15.6.5	Yes	102	Encl. 9 & 10 §7.2.1	Reference 5 Reference 8	Reanalysis Reanalysis
Large-break LOCA - Thermal-hydraulic transient	15.6.5	Yes	102	Encl. 9 & 10 §7.2.1	Reference 5 Reference 8	Reanalysis Reanalysis

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

TABLE II-1 (Sheet 7 of 13)

OTHER ANALYSES

Other Analyses	UFSAR Section	Bounding? Yes/No	CLB Assumed Reactor Power Level (%)	LAR Section	Prior NRC Review or 10CFR50.59	Reanalysis or Evaluation
LOCA Hydraulic Forces	3.6.2.2.1.1	Yes	102	Encl. 9 & 10 §6.3.1 and 7.1	Reference 5	Reanalysis
Containment - LOCA Long-Term Mass and Energy Release	6.2.1.3	Yes	102	Encl. 9 & 10 §7.4.1	Reference 5	Reanalysis
Containment - LOCA Long-Term Containment Response	6.2.1.1.3	Yes	102	Encl. 9 & 10 §7.5.1	Reference 5	Reanalysis
Containment - LOCA Short-Term Mass and Energy Release	6.2.1.2.3.2	Yes	100	Encl. 9 & 10 §7.4.2	OLB Reference 5	Evaluation
Containment - LOCA Short-Term Containment Sub-compartments Response	6.2.1.2	Yes	102	Encl. 5 §II	OLB Reference 5	Evaluation

SECTION II

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

TABLE II-1 (Sheet 8 of 13)

OTHER ANALYSES

Other Analyses	UFSAR Section	Bounding? Yes/No	CLB Assumed Reactor Power Level (%)	LAR Section	Prior NRC Review or 10CFR50.59	Reanalysis or Evaluation
Containment - MSLB Mass and Energy Releases Inside Containment	6.2.1.4	Yes	0/30/70/102	Encl. 9 & 10 §7.6.1	Reference 5	Reanalysis
Containment - MSLB Inside Containment Response	6.2.1.4	Yes	0/30/70/102	Encl. 9 & 10 §7.6.1	Reference 5	Reanalysis
Post-LOCA Long-Term Cooling	15.6.5 6.3	Yes	102	Encl. 9 & 10 §7.2.2	OLB Reference 2 Reference 5	 Evaluation Evaluation
Hot Leg Switchover	15.6.5 6.3.2.5.4	Yes	102	Encl. 9 & 10 §7.2.3	OLB Reference 2 Reference 5 50.59	 Evaluation Evaluation Reanalysis
Post-LOCA Hydrogen Generation	6.2.5	Yes	100	Encl. 5 §II	OLB 50.59 Reference 5	 Reanalysis Evaluation

SECTION II

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

TABLE II-1 (Sheet 9 of 13)

OTHER ANALYSES (CONTINUED)

Other Analyses	**UFSAR Section**	**Bounding? Yes/No**	**CLB Assumed Reactor Power Level (%)**	**LAR Section**	**Prior NRC Review or 10CFR50.59**	**Reanalysis or Evaluation**
Containment -MSLB Mass and Energy Releases Outside Containment	3.11.B.1 3.F.4.2	Yes	0/30/70/100/ 102	Encl. 9 & 10 §7.6.2	Reference 9	Reanalysis
Containment - Inadvertent Spray Actuation	6.2.1.1.3.3	Yes	N/A	Encl. 9 & 10 §7.5.2	OLB	
Natural Circulation Cooldown	5.4.2.3.2 15.2.6	Yes	102	Encl. 9 & 10 §7.9	OLB	
Anticipated Transients Without Scram	4.3.1.7 15.8	Yes	Note 5	Encl. 9 & 10 §7.3.4	OLB	
Station Blackout Coping Evaluation	8.4.1.1	Yes	100	Encl. 5 §V.B	OLB Reference 5	Evaluation
EQ Parameters - EQ Doses	3.11	Yes	100	Encl. 5 §II	OLB Reference 5 Reference 9	Evaluation Reanalysis

SECTION II

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

TABLE II-1 (Sheet 10 of 13)

OTHER ANALYSES (CONTINUED)

Other Analyses	UFSAR Section	Bounding? Yes/No	CLB Assumed Reactor Power Level (%)	LAR Section	Prior NRC Review or 10CFR50.59	Reanalysis or Evaluation
EQ Parameters - Pressure and Temperature Inside Containment	3.11	Yes	0/30/70/102	Encl. 5 §II	Reference 5	Reanalysis
EQ Parameters - Pressure and Temperature Outside Containment	3.11	Yes	0/30/70/102	Encl. 5 §II	Reference 9	Reanalysis
Safe Shutdown Fire Analyses	9A	Yes	N/A	Encl. 5 §II	OLB	
Spent Fuel Pool Cooling - Bulk Cooling	9.1.3	Yes	N/A	Encl. 5 §II	Reference 7	Reanalysis
Flooding	3F.2.4	Yes	N/A	Encl. 5 §II	OLB Reference 5 50.59	Evaluation Evaluation and Reanalysis

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

TABLE II-1 (Sheet 11 of 13)

References:

1. Issuance of Amendment 34 for Unit 1 and Amendment 14 for Unit 2, August 30, 1990, for Relocation of Steam Generator Level Taps
2. Issuance of Amendments 43 and 44 for Unit 1 and Amendments 23 and 24 for Unit 2, September 19, 1991 for Vantage-5 Fuel
3. Issuance of Amendment 48 and 49 for Unit 1 and Amendment 27 and 28 for Unit 2, November 1, 1991, for Deletion of Reactor Trip System Power Range Neutron Flux High Negative Rate Trip
4. Issuance of Amendment 57 for Unit 1 and Amendment 36 for Unit 2, March 10, 1993, for Overtemperature and Overpower ΔT Time Constants and Overpower ΔT Setpoint
5. Issuance of Amendment 60 for Unit 1 and Amendment 39 for Unit 2, March 22, 1993, for Stretch Power Uprate
6. Issuance of Amendment 98 for Unit 1 and Amendment 76 for Unit 2, August 26, 1997, for Pressurizer Safety Valve Setpoint
7. Issuance of Amendment 102 for Unit 1 and Amendment 80 for Unit 2, June 29, 1998, for Unit 1 Spent Fuel Pool Reracking
8. Results of reanalysis reported to NRC by letter LCV-1629, "Vogtle Electric Generating Plant, 10 CFR 50.46 ECCS Evaluation Models Significant Change Report," July 1, 2002. UFSAR revised under 10 CFR 50.59.
9. Issuance of Amendment 128 for Unit 1 and Amendment 106 for Unit 2, June 4, 2003, for Setpoint Margin Recovery

TABLE II-1 (Sheet 12 of 13)

Abbreviations:

CLB - Current Licensing Basis
RCS - Reactor Coolant System
PZR - Pressurizer
OLB - Original Licensing Basis
HZP - Hot Zero Power
HFP - Hot Full Power
LOCA - Loss of Coolant Accident
MSLB - Main Steam Line Break
MUR-PU - Measurement Uncertainty Recapture Power Uprate
UFSAR - Updated Final Safety Analysis Report
LAR - License Amendment Request
NRC - Nuclear regulatory Commission
FW - Feedwater
SG - Steam generator
MSIV - Main steam isolation valve
AC - Alternating-current
EQ - Equipment qualification
DNB - Departure from Nucleate Boiling
RCCA - Rod cluster control assembly
BWR - Boiling water reactor
CVCS - Chemical and volume control system
ECCS - Emergency core cooling system

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

TABLE II-1 (Sheet 13 of 13)

Notes:

1. Bounded by another event.
2. 60% case limiting.
3. Part-power case bounds Mode 3.
4. Mode 3, 4, 5, and 6 cases not affected.
5. Generic ATWS 4-Loop PWR analysis with Model F steam generators applicable. Power level assumed is 3427 MWt.

SECTION III

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

SECTION III

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

RADIOLOGICAL ANALYSES

Table III-1 lists the radiological analyses performed for the MUR power uprate. The table references the enclosures and sections of this submittal containing the summaries of the radiological analyses. Some of the summaries are documented in Section 7.8.2 of Enclosures 9 and 10. The remaining analyses are discussed below.

Radiological Consequences

The increase in licensed core power results in an increase in the core radioactive inventory. To provide the maximum level of consistency for radiological analyses in support of the MUR power uprate, the source terms and analytical models and assumptions used are those described in Section 7.8 of Enclosures 9 and 10. The following information is provided as an addition to or modification of the discussions in Section 7.8 of Enclosures 9 and 10.

Radiation Source Terms

Power level, source terms and dose conversion factors are as described in Sections 7.8.1 and 7.8.2 of Enclosures 9 and 10.

Accident Analyses

The following were re-analyzed in addition to the accident re-analyses described in Section 7.8.2 of Enclosures 9 and 10:

- UFSAR Section 15.1.5 – Steam System Piping Failure
- UFSAR Section 15.6.3 – Steam Generator Tube Failure
- UFSAR Section 15.6.5 – Loss-of-Coolant Accidents
- UFSAR Section 15.7.3 – Postulated Radioactive Release Due to Liquid Tank Failure (Ground Release)

In addition to the revised power level, source terms and dose conversion factors, the following adjustments were factored into the above re-analyzed accident doses:

- The density of primary-to-secondary leakage for the MSLB and SGTR was based on cold water, that is 62.4 lb/ft^3.

- For the MSLB and SGTR, primary-to-secondary leakage and steam generator steaming is extended to 20 hours until the residual heat removal system can remove decay heat.

- The modeling of accident-initiated iodine spike model for the MSLB and SGTR events takes into account the iodine appearance rates from Section 7.8.1 of Enclosures 9 and 10.

For the LOCA, the following additional adjustments were modeled:

- Control room χ/Q values were recalculated using ARCON96 and site meteorological data for the three year period from 1998-2000 as shown below in Table III-2.

- Control room unfiltered inleakage increased from 755 cfm and 5 cfm shown in UFSAR Table 15A-1 to 835 cfm and 130 cfm (including 10 cfm ingress/egress) for unpressurized and pressurized control room respectively.

- Containment spray and plateout removal rates were recalculated in conformance with NUREG-0800, Section 6.5.2, Revision 2, as described in Regulatory Guide 1.195 and shown below in Table III-3.

- The chemical form of iodine for elemental, organic and particulate was changed from 95.5%, 2%, and 2.5% shown in UFSAR Table 15.6.5-9 to 91%, 4% and 5% respectively in accordance with Regulatory Guide 1.195.

Conclusion

The revised results and acceptance limits for the above accidents are presented in Table III-4. For each accident re-analyzed, the current UFSAR values are exceeded, i.e., the results are not bounded; however, all accidents re-analyzed meet the acceptance criteria.

RELOAD ANALYSES

No accident or transient reviews are being deferred to the first reload design at MUR power uprate conditions.

SECTION III

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

TABLE III-1 (Sheet 1 of 2)

RADIOLOGICAL ANALYSES

Radiological Analyses	UFSAR Section	Bounding? Yes/No	MUR-PU Assumed Reactor Power Level (%/MWt)	License Amendment Request Section
Accidental Release of Liquid Effluents in Ground and Surface Water	2.4.13 (Refer to 15.7.3.4)	No	102/3636	Encl. 5 Sec. III
Steam System Piping Failure	15.1.5.3	No	102/3636	Encl. 5 Sec. III
Loss of Non-Emergency AC	15.2.6.3	No	102/3636	Encl. 9 and 10 Sec. 7.8.2
Locked Rotor	15.3.3.3	No	102/3636	Encl. 9 and 10 Sec. 7.8.2
Spectrum of RCCA Ejection Accidents	15.4.8.3	No	102/3636	Encl. 9 and 10 Sec. 7.8.2
Break in Instrumentation Line	15.6.2	No	102/3636	Encl. 9 and 10 Sec. 7.8.2
Steam Generator Tube Failure	15.6.3.4	No	102/3636	Encl. 5 Sec. III

SECTION III

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

TABLE III-1 (Sheet 2 of 2)

RADIOLOGICAL ANALYSES

Radiological Analyses	UFSAR Section	Bounding? Yes/No	MUR-PU Assumed Reactor Power Level (%/MWt)	License Amendment Request Section
LOCA	15.6.5.4	No	102/3636	Encl. 5 Sec. III
Radioactive Waste Gas Decay Tank Failure	15.7.1.5	No	102/3636	Encl. 9 and 10 Sec. 7.8.2
Radioactive Liquid Waste System Leak or Failure	15.7.2.5	No	102/3636	Encl. 9 and 10 Sec. 7.8.2
Liquid Tank failure (Ground Release)	15.7.3.4 (2.4.13)	No	102/3636	Encl. 5 Sec. III
Fuel Handling Accidents	15.7.4.5	No	102/3636	Encl. 9 and 10 Sec. 7.8.2

SECTION III

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

TABLE III-2 - Control Room χ/Q (sec / m^3)

Location Type/ Time Interval	UFSAR Table 15A-2	Revised for MUR Power Uprate
0-2 hours	5.7 E-03	1.04 E-03
2-8 hours	5.7 E-03	7.10 E-04
8-24 hours	3.8E-03	3.08 E-04
24-96 hours	2.3 E-03	2.69 E-04
96-720 hours	1.1 E-03	2.08 E-04

TABLE III-3 - LOCA Containment Spray and Plateout Removal Constants

Time Constant (hr^{-1})	UFSAR Table 15.6.5-9	Revised for MUR Power Uprate
Spray Iodine Removal Constant - Elemental	1.0 (DF $\leq$ 18.9)	10[(a)] (DF = 21.4)
Spray Iodine Removal Constant - Particulate	5.3 (DF $\leq$ 100)	4.19 (DF $\leq$ 50)
Spray Iodine Removal Constant - Particulate	0.53 (DF > 100)	0.419 (DF > 50)
Deposition Iodine Removal Constant - Elemental Iodine Only	13 (DF $\leq$ 100)	4.76 (DF = 200)

(a) Calculated value of 22.5 hr^{-1} is conservatively limited to 10 hr^{-1}

DF = Decontamination Factor

SECTION III

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

Table III-4 – Summary of Revised Accident Analyses

		UFSAR 2.4.13 Conc. (μCi/cc)	Revised MUR Conc. (μCi/cc)	Concentration Limit (μCi/cc)
UFSAR 15.7.3 Liquid Tank Failure (Ground Release)	H^3	3.2 E-05	3.3 E-05	3.0 E-03
	Sr^{90}	1.8 E-25	1.7 E-25	3.0 E-07
	Cs^{137}	2.9 E-196	1.3 E-183	2.0 E-05

	UFSAR Dose (REM)	Revised MUR Dose (REM)	Limit (REM)
UFSAR 15.1.5 MSLB			
Pre-accident Iodine Spike	Table 15.1.5-3		
EAB thyroid	1.0	1.0	300
EAB whole body	<0.01	<0.1	25
LPZ thyroid	0.8	1.0	300
LPZ whole body	<0.01	<0.1	25
Accident Initiated Iodine Spike			
EAB thyroid	1.1	1.2	30
EAB whole body	<0.01	<0.1	2.5
LPZ thyroid	1.1	6.7	30
LPZ whole body	<0.01	<0.1	2.5
UFSAR 15.6.3 SGTR	Table 15.6.3-11		
Pre-accident Iodine Spike			
EAB thyroid	21.0	20.4	300
EAB whole body	0.1	0.1	25
LPZ thyroid	8.7	8.6	300
LPZ whole body	<0.1	<0.1	25
Accident Initiated Iodine Spike			
EAB thyroid	12.8	12.3	30
EAB whole body	0.1	0.1	2.5
LPZ thyroid	5.4	5.7	30
LPZ whole body	<0.1	<0.1	2.5
UFSAR 15.6.5 LOCA	Table 15.6.5-11		
Offsite			
EAB thyroid	117.3	84.6	300
EAB whole body	1.5	2.0	25
EAB beta skin	0.7	4.2	-
LPZ thyroid	72.5	124.0	300
LPZ whole body	1.0	1.5	25
LPZ beta skin	0.6	3.5	-
Control Room			
Thyroid	19.4	29.7	50
Whole body	4.9	1.0	5
Beta skin	66.4[a]	16.7	50[a]

(a) The operator will take appropriate action to ensure that the resultant doses are within the limits established by General Design Criterion 19.

SECTION IV

MECHANICAL/STRUCTURAL/MATERIAL COMPONENT INTEGRITY AND DESIGN

SECTION IV

MECHANICAL/STRUCTURAL/MATERIAL COMPONENT INTEGRITY AND DESIGN

A. COMPONENTS

i. Reactor Vessel, Nozzles, and Supports

These items are discussed in Sections 6.1.1 and 6.3.1 of Enclosures 9 and 10.

ii. Reactor Core Support Structures and Vessel Internals

These items are discussed in Sections 6.2 and 8.2 of Enclosures 9 and 10.

iii. Control Rod Drive Mechanisms

This item is discussed in Section 6.4 of Enclosures 9 and 10.

iv. Nuclear Steam Supply System (NSSS) Piping, Pipe Supports, and Branch Nozzles

These items are discussed in Section 6.3 of Enclosures 9 and 10.

v. Balance-of-Plant (BOP) Piping

Piping for NSSS Interface Systems, Containment Systems, and Safety-Related Cooling Water Systems is discussed in Sections VI.A, VI.B, and VI.C, respectively, of this enclosure under the headings for individual systems.

vi. Steam Generator (SG) Tubes, Secondary Side Internal Support Structures, Shell, and Nozzles

These items are discussed in Sections 6.3.1 and 6.6 of Enclosures 9 and 10.

vii. Reactor Coolant Pumps

This item is discussed in Section 6.5 of Enclosures 9 and 10.

viii. Pressurizer Shell, Nozzles, and Surge Line

These items are discussed in Sections 6.3.1 and 6.7 of Enclosures 9 and 10.

ix. Safety-Related Valves

The following safety-related valves were reviewed from the perspective of function, setpoints, and capacity:

1. Main Steam Safety Valves (Section 5.2.2.1 of Enclosures 9 and 10)

2. Steam Generator Atmospheric Relief Valves (Section 5.2.2.2 of Enclosures 9 and 10)
3. Main Steam Isolation and Bypass Valves (Section 5.2.2.3 of Enclosures 9 and 10)
4. Main Feedwater Isolation and Regulating Valves (Section 5.2.4.1 of Enclosures 9 and 10)
5. Main Feedwater Regulating and Bypass Valves (Section 5.2.4.2 of Enclosures 9 and 10)

The evaluation of Class 1 and Class 2 auxiliary system valves is described in Section 6.8.4.2 of Enclosures 9 and 10.

Systems which have valves maintained within the AOV program, GL 89-10 MOV program, and pressure locking/thermal binding valves programs were reviewed. The review concluded that the MUR power uprate does not impact program valves.

B. ASPECTS OF COMPONENT DESIGN POTENTIALLY AFFECTED BY THE MUR POWER UPRATE

i. Stresses

This item is discussed in Sections 5.2.2.3, 5.2.4.1, 6.1.1, 6.2.1.2, 6.3.1, 6.5.1, 6.6.2, 6.6.3, 6.6.4, 6.6.9, and 6.7 of Enclosures 9 and 10 for the NSSS systems and components.

The review for each individual BOP piping system concluded that there was no detrimental impact on system parameters due to MUR that would have any adverse effect on piping stresses.

BOP piping systems for which additional stress review was performed included the main steam and main feedwater piping systems (for impact on dynamic transient forcing functions), and containment spray piping systems (for temperature impact).

The MUR power uprate has no adverse impact on the water hammer forcing functions for the event of feedwater check valve closure upon a postulated line break. It was concluded that the forcing functions for the original power conditions remain conservative and bounding for the MUR power uprate condition. Thus, there is no adverse impact on BOP piping stresses for the main feedwater system.
The steam hammer forcing functions for the main steam turbine stop valve closure event may experience a 5.2% increase due to the MUR power uprate. It was concluded that the impact on pipe stresses for main steam piping outside containment is acceptable.

For main steam piping inside containment, adequate margins exist to accommodate the minor increase in steam hammer stress. The review included consideration of impact to actual measured test results for the steam hammer event. The existing analyses include a conservative manual analytical adjustment to loads, the effect from MUR being either insignificant or having no adverse impact on the qualification of the piping and system components. Thus, there is no adverse impact on BOP piping stresses for the main steam piping system.

The impact of containment sump temperatures for VEGP Units 1 and 2 due to MUR power uprate on the containment spray pipe stress calculations was investigated. The maximum sump temperature for MUR power uprate is 252.7 °F. This temperature of 252.7 °F is enveloped by the existing design temperature (253.0 °F) used in the current pipe stress analysis of the containment spray piping for both trains of VEGP Units 1 and 2. Therefore, the power uprate temperature for the piping increased slightly, but the stress analysis temperature that was considered for evaluation enveloped the power uprate temperature. Therefore, the VEGP MUR power uprate has no impact on existing Containment Spray piping stress analysis for both units.

ii. Cumulative Usage Factors

This item is discussed in Sections 6.1.1, 6.2.1.2, 6.3.1, 6.5.1, 6.6.2, 6.6.4, and 6.8 of Enclosures 9 and 10 for the NSSS components.

The piping reviewed for the BOP scope is non-Class 1 piping, and may be qualified to non-Class 1 rules. Therefore, the calculation of Cumulative Usage Factors is not required.

iii. Flow Induced Vibration

This item is discussed in Sections 6.2.1, 6.6.4, 6.6.7, 6.6.8, and 6.6.9 of Enclosures 9 and 10 for the NSSS components.

The VEGP MUR power uprate will be increasing the licensed core thermal power level for VEGP Units 1 and 2 by 1.7%. To accomplish the power uprate, the mass and volumetric flow rates for the main steam and main feedwater systems will increase. The impact of the MUR power uprate on main steam and feedwater piping has been assessed to be not significant, based on the relatively small changes in kinetic energy and no known piping and support modifications needed as a result of the MUR power uprate.

SNC will perform a screening of secondary side systems that will experience a change in flow due to the MUR power uprate. Within the systems affected, specific locations that are deemed to be most susceptible to flow induced vibration will be selected, using drawings, operating experience (OE) from other similar plants that have uprated, experience, and plant history. Based on the results of the screening, locations on the secondary side of the plant will be selected for pre- and post-uprate vibration monitoring.

After the pre-uprate (baseline) vibration data is obtained, projections of the expected increase in vibration for the planned uprate will be developed, along with acceptance criteria against which post-modification measurements can be compared to confirm that there will be no adverse vibration effects. The flow induced vibration plan will be part of the functional test/start-up plan developed for the MUR power uprate.

In addition, the residual heat removal system loop suction valve bypass line, which has already shown a susceptibility to flow induced vibration, will be monitored to confirm the post-uprate vibration levels remain within established acceptance criteria.

iv. Changes in Temperature (pre- and post-uprate)

There will be no change in the current licensing-basis range for the reactor vessel average temperature of 570.7 ^{o}F to 588.4 ^{o}F as a result of the MUR power uprate. However, the vessel outlet temperature will increase by about 0.6 ^{o}F, and the vessel inlet temperature will decrease by about 0.6 ^{o}F at normal full-power conditions as a result of the MUR power uprate. There will be no change in both the current licensing-basis vessel outlet temperature upper limit of 620.0 ^{o}F and vessel inlet temperature lower limit of 538.3 ^{o}F as a result of the MUR power uprate. There will be no change to the RCS average temperature limit in Technical Specification 3.4.1.

Changes in feedwater and main steam system temperatures are discussed in Section VI, Item A, of this enclosure.

v. Changes in Pressure (pre- and post-uprate)

There will be no change in the reactor coolant system (RCS) operating pressure as a result of the MUR power uprate. The nominal operating pressure is 2250 psia (Table 2-1 of Enclosures 9 and 10). There will be no change to the RCS pressure limit in Technical Specification 3.4.1.

Changes in main steam and feedwater system pressure are discussed in Section VI, Item A, of this enclosure.

vi. Changes in Flow Rates (pre- and post-uprate)

There will be no change in the RCS thermal design flow rate as a result of the MUR power uprate. The thermal design flow rate is 93,600 gpm (Table 2-1 of Enclosures 9 and 10). There will be no change to the RCS flow limit in Technical Specification 3.4.1.

Changes in feedwater and main steam system flow rates are discussed in Section VI, Item A, of this enclosure.

MECHANICAL/STRUCTURAL/MATERIAL COMPONENT INTEGRITY AND DESIGN

vii. High-Energy Line Break Locations

This evaluation considers three different aspects associated with High Energy Line Break (HELB): a) pipe whip restraints design and jet impingement barriers, b) effects on the structure through which the High Energy Line passes including design bases flooding, and c) containment sub-compartment pressurization due to HELB. Pipe break locations within high energy lines were reviewed. The review concluded that there are no new pipe break locations as a result of the MUR power uprate.

The following systems are high energy systems:

1. Reactor Coolant
2. Safety Injection
3. Residual Heat Removal
4. Chemical Volume and Control
5. Nuclear Sampling
6. Main Steam
7. Auxiliary Feedwater
8. Condensate and Main Feedwater
9. Auxiliary Steam
10. Steam Generator Blowdown
11. Waste Processing
12. Turbine-Generator
13. Auxiliary Gas

From a civil/structural design point of view, the design parameters of interest are the pressure, temperature, mass flow rate and the changes associated with the MUR uprate. Pressure values are used in computing the pipe whip thrust and jet impingement forces, and the temperatures are used in computing the pipe whip restraint gaps, which in turn may influence the pipe whip loading, and the mass flow rate influences the steady state thrust coefficient.

Among the above-listed systems, the following systems undergo no change in pressure or temperature: a) Safety Injection System, b) Residual Heat Removal, c) Auxiliary Feedwater System, d) Auxiliary Steam System, and e) Auxiliary Gas System.

The following systems experience either reductions in both parameters, or no change in one parameter and a reduction in the other parameter: a) Reactor Coolant System, b) Chemical and Volume Control System, c) Nuclear Sampling System, d) Main Steam System, e) Waste Processing System, f) Turbine Generator System, and g) Steam Generator Blowdown System. For the main steam system, the temperature decreases from 543.8 °F to 543.0 °F, and for the steam generator blowdown system, the temperature decreases from 544.0 °F to 521.7 °F. Therefore, for these systems, the changes due to power uprate cause no impact on the pipe whip restraints.

For the condensate and main feedwater system, the temperature at the steam generator inlet increases from 446.3 °F to 448.2 °F and the pressure increases from 1148.5 psia to 1156.5 psia. The effects of these changes in design parameters are discussed below.

A. Pipe Whip Restraints Design and Jet Impingement

<u>Changes in Temperature</u>

1. Pipe-line temperature is significant for high energy piping systems in consideration of pipe-line movement (position) relative to the position when installed (cold position). The change in pipe-line temperature and associated movement at the points of pipe whip restraints has an effect on the gap between the whip restraint and the pipe. The gap between the pipe and whip restraint affects the resulting forces upon the restraint when calculating the forces due to pipe break.

2. Among all high energy pipe systems with pipe whip restraints, only the condensate and main feedwater system, and main steam system experience changes in temperature. The steam generator blowdown system experiences a change in temperature. However, there are no active pipe whip restraints on the high energy pipes in this system. Therefore, this change in temperature is not relevant to this evaluation. The condensate and main feedwater pipe whip forces are computed for an operating temperature of 445 °F. The operating temperature will increase to 448.2 °F following the MUR power uprate. This increase is very small; for example, assuming a conservative pipe length of 100 feet, for carbon steel, the associated temperature increase will only cause an additional linear pipe expansion of 0.025 inch and a maximum radial expansion of 0.0004 inch for 36-inch diameter pipes. For the main steam system, where the temperature decreases by 0.8 °F due to MUR power uprate, the carbon steel piping will experience linear and radial contractions of 0.006 inch and 0.0001 inch respectively, for 100-foot long, 38-inch diameter pipes. The effect of these axial and radial expansions/contractions on the pipe whip restraint gaps (+ or -) is too small to cause significant additional forces onto the pipe whip restraints.

Therefore, it is concluded that the changes in temperature associated with the MUR power uprate produce insignificant changes in gaps between restraints and pipes, and the resulting condition is acceptable.

Changes in Pressure

The condensate and feedwater system is the only high energy pipe system that experiences an increase in pressure due to the power uprate.

It was stated earlier that the operating pressure in the condensate and feedwater system pipe-lines due to power uprate is 1156.5 psia. This is less than the minimum operating pressure of 1165 psia used in the design. Additionally, a conservative estimate of the steady state thrust coefficient, enveloping the changes in mass flow rate, has been used in computation of the pipe whip/ jet thrust forces. Therefore, the jet thrust forces for the existing design are greater than the forces for the MUR power uprate as would be applied to the high energy pipe whip restraints.

B. Flooding Inside Structures Due to HELB

Flooding is discussed in Section II of this enclosure. The review concluded that the current flooding analyses remain bounding for the MUR power uprate.

C. Containment Sub-Compartment Pressurization

Containment sub-compartment response is discussed in Section II of this enclosure. The review concluded that the current sub-compartment analyses remain bounding for the MUR power uprate.

D. Conclusion

The changes in high energy piping system pressures, temperatures, and mass flow due to the MUR power uprate do not result in conditions that adversely affect the pipe whip restraints, jet impingement barriers, or other structural components.

viii. Jet Impingement and Thrust Forces

Jet impingement and thrust forces are discussed in Item vii above.

C. REACTOR PRESSURE VESSEL (RPV) INTEGRITY

i. Pressurized Thermal Shock (PTS) Calculations

This item is discussed in Section 6.1.2 of Enclosures 9 and 10.

ii. Fluence Evaluations

This item is discussed in Sections 6.1.2, 8.1, and 8.2 of Enclosures 9 and 10.

iii. Heatup and Cooldown Pressure-Temperature Limit Curves

This item is discussed in Section 6.1.2 of Enclosures 9 and 10.

iv. Low-Temperature Overpressure Protection

This item is discussed in Section 7.11.3 of Enclosures 9 and 10.

v. Upper Shelf Energy

This item is discussed in Section 6.1.2 of Enclosures 9 and 10.

vi. Surveillance Capsule Withdrawal Schedule

This item is discussed in Section 6.1.2 of Enclosures 9 and 10.

D. CODE OF RECORD

Enclosures 9 and 10 address the ASME code requirements for the following:

1. Steam Generator Main Steam Safety Valves (Section 5.2.2.1 of Enclosures 9 and 10)
2. Reactor Vessel Structural Evaluation (Section 6.1.1 of Enclosures 9 and 10)
3. Nuclear Steam Supply System Piping (Section 6.3.1 of Enclosures 9 and 10)
4. Reactor Coolant Pumps (Structural) (Section 6.5.1 of Enclosures 9 and 10)
5. Steam Generator (Sections 6.6.2, 6.6.3, and 6.6.4 of Enclosures 9 and 10)
6. Pressurizer (Section 6.7 of Enclosures 9 and 10)
7. Auxiliary Equipment (Section 6.8 of Enclosures 9 and 10)
8. Reactor Internals Heat Generation Rates (Section 8.2 of Enclosures 9 and 10)

E. COMPONENT INSPECTION, TESTING, AND EROSION/CORROSION PROGRAMS

Component In-Service Inspection (ISI) Program

Inservice Inspection of Class 1 and 2 vessels is performed per the 2001 Edition of ASME Section XI with Addenda through 2003, using a sampling system. This sampling system does not use physical parameters (e.g., flow, pressure, temperature) in the determination of welds to examine. Therefore, the uprate does not affect the selection of welds on vessels.

Inservice Inspection of Class 1 and 2 piping welds uses the methodology prescribed in WCAP-14572, Revision 1-NP-A, "Westinghouse Owner's Group Application of Risk-

Informed Methods to Piping Inservice Inspection Topical Report." This methodology utilizes Westinghouse provided "Structural Reliability and Risk Assessment Model (SRRA)" software to determine the failure probability of piping welds. SRRA input includes the normal operating pressure and temperature parameters at the piping weld. Using these parameters, plus data on pipe size, material, stress, etc. a "point value" failure probability of the weld is generated. Experience with the use of the SRRA software indicates that small variations in pressure and temperature have minimal effect on the "point value" results. In addition, due to uncertainties in the calculation of the failure probability, this "point value" has a distribution applied to it that could affect the results of the SRRA failure probabilities by a factor of five to twenty. Therefore, small changes in the pressure and temperature used as inputs to SRRA will have no effect on the selection of piping welds for examination.

Pressure testing of Program specific safety-related Class 2 components will be performed at nominal operating pressure prior to and after the uprate; therefore, pressure testing will not be impacted.

Component In-Service Testing (IST) Program

The VEGP In-Service Testing (IST) of pumps and valves currently uses the 1990 Edition of the ASME OM Code and will be based on the 2001 Edition through 2003 Addenda after MUR Power Uprate implementation. The Code requires performance of a comprehensive pump test at ± 20% of plant system accident analyzed flow. The plant uprate accident analysis required flows are not changing, specific Nuclear Class valve response times are not changing, and Nuclear grade pump and valve components are not physically changing to support the MUR power uprate, therefore the IST Program will not be affected.

Erosion/Corrosion Program

To support the MUR power uprate, a new high-pressure turbine is being installed to accept the new increased main steam flow. As a result, most of the secondary plant fluid systems will experience flow increases less than 10% with insignificant changes in fluid temperatures. The largest areas of increase are in the high-pressure extraction steam lines. These lines have been reviewed, and only minimal increases in wear rates are predicted. Other system pressures and temperatures are not changing significantly; thus, corrosion is not expected to increase significantly.

As part of implementation of the MUR power uprate, the MUR uprate fluid conditions will be utilized as input to Flow Accelerated Corrosion (FAC) predictive models to establish and validate component erosion/wear rates. Major changes to the scope of FAC inspections or their 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.

F. SG TUBE HIGH CYCLE FATIGUE (NRC BULLETIN 88-02)

This item is discussed in Section 6.6.9 of Enclosures 9 and 10.

OTHER HAZARDS AND PROGRAMS

Missiles

There is no impact of the missile analyses as a result of the MUR power uprate. The high pressure (HP) turbine is being replaced in both units. Missile production was considered by the turbine vendor.

Brittle fracture failure mechanism is only applicable to conventional rotors with shrunk-on wheels and axial keyways. The new turbine rotors have monoblock forgings. Therefore, brittle fracture associated with stress corrosion cracking at keyway slots for shrunk-on wheels does not apply. Rather, the probability of ductile failure is considered to be a function of speed, temperature, and material tensile strength. With stresses below the ultimate strength, the probability of a ductile fracture is negligible. For this failure mechanism, the overspeed capability of the HP rotor is well in excess of the low pressure (LP) rotor capability. Therefore, the HP rotor does not represent a missile risk as it is directly coupled to the LP rotors. The brittle and ductile failure modes are statistically independent.

External Hazards

VEGP UFSAR Chapters 2 and 3 describe the design and features providing protection against hazards originating outside systems, structures, and components (SSCs) important to safety, including:

- nearby facilities
- severe weather
- site flooding
- high winds
- tornados and tornado missiles.

Neither the frequency nor the severity of any of these events is impacted by MUR power uprate. The MUR power uprate does not change or add any SSCs important to safety in areas outside of, nor change or require a change to the exterior walls or roof of, any structure which provides protection against these hazards.

SECTION V

ELECTRICAL EQUIPMENT DESIGN

SECTION V

ELECTRICAL EQUIPMENT DESIGN

A. EMERGENCY DIESEL GENERATORS

The primary purpose of the emergency diesel generator (EDG) system is to be able to generate onsite electrical power to feed the standby power system which provides alternating current (AC) power for a safe shutdown of the plant in the event of loss of offsite power. The MUR power uprate will not add any new Class 1E electrical loads. Also, because the current licensing-basis LOCA analyses remain bounding for the MUR uprate conditions, there are no changes in the existing Class 1E LOCA loading requirements for the EDG and no changes to the sequencing of these loads. There are some changes in the EDG operating conditions such as the jacket water temperature. However, the evaluation of the EDG system concluded that the system parameters are not adversely affected by the MUR power uprate.

B. STATION BLACKOUT EQUIPMENT

For the station blackout (SBO) duration, the plant is required by 10 CFR 50.63 to be capable of maintaining core cooling and appropriate containment integrity. The minimum acceptable SBO coping duration is 4 hours.

The design criteria for the SBO have been reviewed for potential impact from a MUR power uprate. 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. 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. No design changes are required as a result of the MUR power uprate.

The condensate storage tank (CST) volume requirements in Technical Specification 3.7.6 envelope the required storage capacity for dedicated safety grade water for a SBO at MUR power uprate conditions. The minimum required useable inventory is based on a reactor trip from 102 percent of the current power level of 3565 MWt.

The MUR power uprate neither increases loads required to cope with the SBO nor does it impact the load shedding requirement. Therefore, the MUR power uprate will have no affect on station battery capacity.

No air operated valves are relied upon to cope with a SBO for four (4) hours.

The areas containing equipment required to cope with a SBO were reviewed for any impact related to the loss of ventilation at MUR power uprate conditions. None of the areas evaluated will see any increase in temperatures due to the SBO under MUR power uprate conditions. Therefore, the MUR power uprate has no impact on the effects of loss of ventilation during a SBO.

Valves identified as containment isolation valves of concern for SBO that would be required to be operable during an SBO event are not adversely affected by the MUR power uprate. All containment isolation valves are designed to fail in their safe position. Therefore, the MUR power uprate has no impact on the requirements of containment isolation during an SBO.

C. ENVIRONMENTAL QUALIFICATION OF ELECTRICAL EQUIPMENT

The impact of the MUR power uprate on Equipment Qualification (EQ) parameters was discussed in Section II of this enclosure. It was concluded that the current EQ parameters remain bounding for the MUR power uprate. Therefore, the qualification of electrical equipment subject to EQ requirements is not impacted.

D. GRID STABILITY

The stability impact of uprating the electrical output of each unit by 25 MWe has been evaluated. The review was made for a light load system load level and for a valley system load level. Previous studies have shown that the stability of the plant is worse for lower system load levels. The review was conducted with an uprate of 33 MWe for each unit (the initial uprate estimate, giving a total of 1280 MWe gross output for colder ambient temperatures). The results showed that with the proposed uprate for the units, the critical breaker failure clearing time for the studied contingencies would be reduced by some degree. This reduction could be as much as 3.75 cycles from what was obtained with the existing electrical outputs from the units. However, the current actual clearing times are much lower than the critical clearing times obtained in the study. With an uprate of only 25 MWe, the stability will be better than the review indicated. Therefore, the proposed electrical output uprate for the units will not cause any stability problems.

The reactive power output capability of the VEGP units was reviewed to determine if it meets the system reactive policy after the MUR power uprate. This review indicated that the units will meet the system reactive policy if the units were uprated by 33 MW (a total of 1233 MW gross output for peak system summer conditions). With the units only being uprated by 25 MW, the units will be able to produce even more reactive power and will thus meet the system reactive policy.

The reactive output evaluations are summarized below:

1. Transmission planning studies were performed under several different conditions. The most demanding MVAR output for the VEGP units for the cases studied is 413 MVAR for Unit 2 assuming Unit 1 is offline and the VEGP-Scherer 500 kV

line out. In all of the cases considered, the minimum required 230kV switchyard voltage is maintained within the acceptable range of 105% to 100 % of 230kV.

2. The Transmission Operator expects all system generating units to each be able to carry MVAR load up to the unit's capability curve unless limited by transmission system voltage and plant station service voltage. In the case of VEGP Units 1 and 2, the units are not limited by station service and can provide up to their capability curve MVAR output. At the present gross output (at generator terminals), the MVAR output is 575 MVAR for Unit 1 and essentially the same for Unit 2.

3. Following the uprate, at the new Unit 1 generator gross output (at generator terminals), the MVAR output will be approximately 567 MVAR. Unit 2 is essentially the same. The increased 25 MWe output from the proposed uprate results in an 8 MVAR decrease. Transmission planning studies were performed under several different conditions. The most demanding MVAR output for the VEGP units for the cases studied is 440 MVAR for Unit 2 assuming Unit 1 is offline and the VEGP-Scherer 500 kV line out. In all of the cases considered, the minimum required 230kV switchyard voltage is maintained within the acceptable range of 105% to 100 % of 230kV.

4. The 25 MWe power uprate resulted in a decrease in generator reactive capability of approximately 8 MVAR per unit. As discussed in Item 3 above, the highest expected MVAR need post-uprate is 440 MVAR compared to the unit's capability of 567 MVAR. In addition to the reactive capability of the generating units, there are two 90 MVAR capacitor banks that can be individually switched onto the 230kV bus by transmission system operators. These capacitor banks were OFF in the studies performed to provide the results presented in Items 1 and 3 above. Therefore all local reactive power needs are met and no compensatory measures are needed.

5. The power uprate and associated reduction in generator MVAR production does not result in 230kV post trip voltages outside of the expected allowable range. In the worst case contingency with one VEGP unit still on-line, there was still approximately 125 MVAR of remaining capacity (567 – 440 MVAR) in that running unit. In addition, if needed, the transmission system operators can switch on up to 180 MVAR of 230kV capacitors located on the VEGP 230kV bus. These capacitors have the capability to increase the 230kV bus voltage by approximately 1.3%.

OTHER ELECTRICAL SYSTEMS

Generator

The generators convert mechanical energy from the turbines into electrical energy. An evaluation of the capability of the Unit 1 and Unit 2 generators at MUR power uprate conditions was performed. The evaluation concluded that the existing generators can support the MUR power uprate. Both generators are rated at 1350 MVA with a power factor of 1.0. For the MUR power uprate, the existing generator ratings will not change. For the main generator, the pre-uprate power factor is estimated to be 0.91 which corresponds to an electrical output of 1,225.3 MWe. At MUR power uprate conditions, the power factor is estimated to be approximately 0.93. which corresponds to an electrical output of 1,249.8 MWe. The generator electrical power output for the MUR power uprate falls within the range of operation on the Generator Reactive Capability Curve. Evaluation of the generators has demonstrated that the increase in electrical output remains bounded by their design ratings.

The generator stator cooling water system removes heat generated by electrical losses from the generator stator and the exciter rectifier. There will be an increase in the cooling water temperature and flow rate as a result of the MUR power uprate. Evaluation of this system has demonstrated that the increase in temperature is bounded by the system design basis.

The generator hydrogen system controls the temperature rise of the rotor components by circulating hydrogen gas through water-cooled hydrogen coolers using single-stage fans attached to both ends of the field. Evaluation of this system has demonstrated that there is no change in the hydrogen pressure and flow rate as a result of the MUR power uprate.

The turbine-generator carbon dioxide (CO_2) system purges hydrogen or air from the turbine-generator before and after opening for inspection or repairs, using CO_2 as the purging medium. The generator purging requirements have not changed as a result of MUR power uprate.

Offsite Power Systems

The primary function of the offsite power system is to provide two independent, preferred power sources to feed Class 1E station auxiliary systems in compliance with Nuclear Regulatory Commission (NRC) Generic Design Criterion 17 of 10 CFR 50 Appendix A. The system supplies power to Class 1E loads during normal operation and accident conditions. The system also supplies power to the non Class 1E loads during startup or shutdown of unit 1 or unit 2 and whenever one or both unit auxiliary transformers for a generating unit are out of service.

SECTION V

ELECTRICAL EQUIPMENT DESIGN

The offsite power sources are connected via the switchyard to the 230-kV and the 500-kV transmission system. The 230-kV switchyard supplies power through two 230/13.8/4.16-kV reserve auxiliary transformers (RAT) per unit (preferred power source) to the Class 1E buses and to the non Class 1E buses under certain operating conditions. There is also a "swing" 13.8/4.16-kV, 10/12.5 MVA standby auxiliary transformer (SAT) which may be manually connected to supply power to the Class 1E buses and to a portion of the non Class 1E loads.

The two systems that directly receive power from the offsite power are the 4.16-kV and the 13.8-kV systems. Both of these systems are evaluated individually for the impact of the MUR power uprate on their operational requirements and components and found to be acceptable.

The offsite power system interfaces with the 4.16-kV and the 13.8-kV systems through the reserve auxiliary transformers (RATs). The RATs are sized to feed Class 1E loads and non Class 1E under certain operating conditions. The ratings of the existing RATs are:

230-kV (H winding) – 60 MVA
13.8-kV (X winding) – 35 MVA
4.16-kV (Y winding) – 25 MVA

The load on the offsite power is increased due to the MUR power uprate, because the heater drain pump motors are changing from a 1500 hp motor to a 1800 hp motor. The heater drain pump motors are normally fed from the unit auxiliary transformers (UATs). If the UATs are not available, non-1E loads are fed from RATS. RATs are sized to feed whole 1E and non-1E loads under the worst operating conditions.

There are two RATs per unit, and two heater drain pumps are installed per unit. One heater drain pump is fed from each RAT. The additional load added to each RAT from this motor change is approximately 500-kVA. The total 1E and non 1E load on each RAT including this additional load is less than the RAT rating.

Based on the above, the existing RATS are capable of operation at MUR power uprate conditions, and no changes are required to the RATs.

Since offsite power system is not required to function in a manner different during MUR power uprate conditions and it's interaction with the 4.16-kV and the 13.8-kV systems does not change, the offsite power system is acceptable for operation at MUR power uprate conditions.

SECTION V

ELECTRICAL EQUIPMENT DESIGN

25 kV AC System

The primary function of the 25-kV AC system is to connect the main generator output to the low-voltage terminals of the main power transformer (MPT)/generator step-up transformer and to the high-voltage terminals of the unit auxiliary transformers (UAT). The 25-kV system is energized whenever the main generator is delivering output and uses an isolated phase bus to interconnect the main generator, MPT, and UATs.

The main generator voltage is stepped up to 230 kV by the Unit 1 main power transformer and 500 kV by the Unit 2 main power transformer. The 25-kV AC system provides power directly to the UATs which step down the voltage for distribution in the 13.8 kV and 4.16 kV station auxiliary systems.

The UATs provide power to the 4.16 kV system which is affected by the MUR power uprate, because the heater drain pump motors are being changed from 1500 horsepower up to 1800 horsepower motors. There are two UATs per unit, and two heater drain pumps are installed per unit. One heater drain pump is fed from each UAT. The additional load added to each UAT is approximately 500-kVA. The total load on each UAT, including this additional load is less than the UAT rating.

The MPT has three single phase units. Each unit is rated at 452 MVA. The three phase rating of the MPT is 1356 MVA. The generator maximum rating is 1350 MVA. With the MUR power uprate, the maximum generator output is 1343 MVA which is below the MPT rating.

Based on the above, the existing UATS and MPT are capable of operation at MUR power uprate conditions, and no changes are required to the UATs or the MPT.

Iso-Phase Bus System

The primary function of the isophase bus system is to connect the output of the main generator to the low-voltage terminals of the main step-up transformer bank and to the high-voltage terminals of the of the UATs.

The MUR uprate is expected to yield an increase in electrical output of about 25 MWe due to the 1.7% MWt increase in reactor output. This electrical output increase is being achieved without requiring a change to the existing main generator nameplate rating and without a change to the existing generator Capability Curve. The existing main generator nameplate rating is 1350 MVA, 25,000 volts, 31,177 amperes, and 75 psig hydrogen gas pressure. The continuous forced cooled current rating of the isophase bus is 32,818 amperes and was derived using the 1350 MVA generator power rating and 95% of the 25,000 V generator voltage rating (assuming worse case grid variation). Since the main generator will continue to be operated within the existing rating following the MUR power uprate, the existing isophase bus continuous current rating will not be challenged, and the existing analyses performed to establish the fault and continuous ratings for the isophase bus and interface components remain bounding.

The turbine plant cooling water (TPCW) system provides cooling water to the isophase bus duct cooling unit heat exchangers. Since the TPCW pump suction is from the circulating water basin, the inlet water temperature to the isophase bus cooler is expected to be essentially the same as the cooling tower cold water return temperature. The design value for the cooling tower return temperature is 95 °F and is the value that corresponds to the most extreme summer time ambient weather conditions. The post uprate value for maximum expected circulating water return temperature has been evaluated to be 94 °F which remains within the design value of 95 °F.

Trend data show that the isophase bus conductor temperature increases approximately 20 °C (36 °F) from winter conditions to summer conditions. This variance is expected since the circulating water average return temperature (cold water temperature) increases by about the same amount over the same time period. The maximum observed conductor temperature was less than 70 °C (158 °F) during the summer time period. Based on this trend, there is currently about a 20 °C (36 °F) margin to the design limit of 90 °C (194 °F). Since the uprate will result in a relatively small increase in electrical output of about 25MWe, the existing margin in isophase bus cooling will be adequate for the anticipated MUR power uprate conditions.

The isophase bus system operating procedure and annunciator response procedure limit the isophase bus conductor temperatures to 85 °C (185 °F) which is 5 °C (9 °F) below the design value. The procedures require a reduction in electrical load as necessary to maintain bus conductor temperature below 85 °C (185 °F).

<u>High-Voltage Switchyard</u>

The primary function of the high-voltage switchyard is to distribute the generated power to the transmission grid. It also supplies two independent, immediately available, offsite sources to the reserve auxiliary transformers (RATs). The system includes all 230 kV and 500 kV autotransformers, transmission line terminations, breakers, buses and switching equipment between the transmission lines and the Unit 1 and Unit 2 main power transformers (MPT)/generator step-up transformers and RATs.

The high-voltage switchyard is not a Class 1E system; however, it supplies two independent and immediately available offsite sources of AC power to the RATs, which are the preferred power source for the Class 1E safety-related AC distribution system and provide startup power to the plant auxiliaries. The two offsite sources are not affected by the MUR power uprate.

The high-voltage switchyard interfaces with the offsite power system, non-Class 1E DC system, and switchyard interfaces to the plant. The evaluations for these systems determined that their operation at MUR power uprate conditions is bounded by the current design basis. The other functions of the high voltage switchyard (such as short circuit protection, lightning protection, etc.) are not affected by power level.

There are no generator output breakers on the Vogtle units. The generators are disconnected from the system by disconnecting the high voltage switchyard breakers, i.e., associated 230-kV breakers are opened to disconnect the Unit 1 generator, and the 500-kV breakers are opened to disconnect the Unit 2 generator. The 500 and 230-kV breakers are sized with ample margin based on the present and future transmission system conditions, including the Vogtle generators. Presently, the 230-kV and 500-kV breakers have adequate margins. The breaker interrupting studies are conducted regularly to assure that breakers have adequate short circuit and continuous current rating margins. One change to the system loads, due to the power uprate, is the increase from 1500 hp to 1800 hp of two heater drain pump motors per unit on the 4.16-kV system. The impact of this change on 230-kV and 500-kV switchyard short circuit current fault levels was reviewed. It was concluded that the short circuit fault level limits remain within acceptable criteria. The review also examined steady voltages and concluded that they remain within acceptable limits. Therefore, there are no changes required to the 230 and 500-kV breakers which disconnect the generator from the system.

The other functions of the high-voltage switchyard (such as short circuit protection, lightning protection, etc.) are not affected by power level.

Based on the above discussions, the functions of the high-voltage switchyard to support the transmission lines and supply power to various breakers and other equipment in the high-voltage switchyard are not changed by the MUR power uprate.

Electrical Loads

In order to support operation at MUR power uprate conditions, the installation of a new ultrasonic feedwater measurement system in both Vogtle units is required. As discussed in Enclosure 1 of this submittal, SNC also plans to replace the heater drain pumps and motors on both units to improve system performance. The existing heater drain pumps (two per unit) have 1500 HP motors. The new pumps will have 1800 HP motors. The new Caldon ultrasonic feedwater flow measurement system is being installed to support operation at MUR power uprate conditions. This system will add two new electrical loads, including an electronics control cabinet and an air conditioning unit (for the cabinet).

The new Caldon electronics cabinet power supply will be connected to a non-1E 120 volt AC inverter with a rating of 3 kVA. The existing load for the inverter is 0.192 kVA . An existing load of 0.200 kVA will be assumed. The load being added is 5A, which is 0.600 kVA. Therefore the new loading will be 0.800 kVA. The 3 kVA rating is sufficient for the new load.

SECTION V

ELECTRICAL EQUIPMENT DESIGN

The new Caldon electronics cabinet will require an air conditioner. The new air conditioner will be fed from an existing 120 volt AC panel. The existing maximum loading for this panel is 7.2 kVA. The air conditioning unit will add 1.2 kVA. Therefore, the new loading will be 8.4 kVA. The 120 volt AC panel is powered from a 480V motor control center (MCC) by a 15 kVA transformer. The 15 kVA supply transformer is sufficient for the new load.

The size of both heater drain pumps will increase to support operation at MUR power uprate conditions. A motor size of 1800 HP (from existing 1500 HP) was used in the electrical load evaluation. This represents a load increase of 0.300 MVA. The heater drain pumps are powered from different non-1E 4.16 kV buses.

Power for non-1E 4.16 kV buses is normally from the respective UAT. Under normal configuration, the maximum expected load on any (Unit 1 or Unit 2) 4.16 kV winding is 12.39 MVA. The addition of 0.300 MVA will result in a loading of 12.69 MVA. The 20 MVA rating for the 4.16 kV winding of any UAT is sufficient for the new load.

Under certain conditions, power for the 4.16 kV buses (non-1E and 1E) is supplied by the respective RAT. Under this configuration, the maximum expected load on any (Unit 1 or Unit 2) 4.16 kV winding is 18.77 MVA. The addition of 0.300 MVA will result in a loading of 19.07 MVA. The 25 MVA rating for the 4.16 kV winding of any RAT is sufficient for the new load.

SECTION VI

SYSTEM DESIGN

SECTION VI

SYSTEM DESIGN

A. NSSS INTERFACES

Main Steam System

The primary function of the main steam system (MSS) is to supply relatively dry, high-pressure steam from the steam generators (SG) to the high-pressure turbine (HPT).

The MSS has been reviewed for potential impact from a MUR power uprate. Full-load steam flow is increasing from 15.9 E6 lbm/hr to approximately 16.3 E6 lbm/hr. System temperature will decrease from 543.8°F to 543.0°F and is bounded by the system design temperature of 600°F. SG pressure will decrease from 993.7 psia to 986.9 psia and is bounded by the system design pressure of 1185 psig. Review of this system has shown that system parameter changes due to the MUR power uprate are bounded by the system design basis and analyses and do not adversely affect system components or functions.

The major safety-related components of the MSS include the main steam safety valves (MSSVs) and the atmospheric relief valves (ARVs). The setpoints of the MSSVs are determined based on the 1185 psig design pressure of the steam generators. Since this design pressure is not impacted, these setpoints will not be impacted due to MUR power uprate conditions. The flow capacity of the MSSVs and ARVs has been verified to be sufficient and is bounded by the analyses of record for the range of NSSS design parameters associated with the MUR power uprate conditions.

It is concluded that the MSS will continue to perform its intended design function at the stated conditions of the MUR power uprate. The MSS design bases have been reviewed and bound the MUR power uprate conditions.

Extraction Steam System

The primary function of the extraction steam system (ESS) is to supply turbine extraction steam to the feedwater heaters to increase plant thermal cycle efficiency. The extraction steam system is not safety-related and has no safety design basis.

The design of the non-return valves, motor operated isolation valves, and pertinent instrumentation for prevention of turbine water induction was evaluated and found acceptable for MUR power uprate conditions.

The ESS and its components are capable of supporting the MUR power uprate with the exception of the 5^{th} stage extraction steam drainage piping. The high pressure extraction steam drain flows will increase as a result of high pressure turbine replacement. The 5^{th} stage extraction steam drain lines are predicted to be undersized, so the 2-inch diameter piping will be changed to 3-inch diameter piping to accommodate the increased drain flow. In addition, the 5^{th} stage drain orifice to the condenser will be changed to a manual valve to allow for adjustment of the drain flow, and the 2-inch bypass air operated valve (AOV) will be replaced with a 3-inch bypass AOV to accommodate the increased drain flow. The feedwater heater and moisture separator reheater (MSR) performance has been evaluated, and these components have sufficient heat transfer capabilities to perform as predicted at MUR power uprate conditions.

In conclusion, all requirements for the ESS have been reviewed and, given the planned modifications, the system will perform its design function at MUR power uprate conditions.

Main Steam Line Isolation System (Main Steam Isolation)

The functions of the main steam line isolation system are to limit blowdown to one steam generator in the event of a steam line break, to limit related effects on the reactor core, and to limit the containment pressure to a value less than 90% of the design pressure.

The main steam line isolation system has been reviewed for potential impact from a MUR power uprate. Critical system parameters reviewed include flow, temperature, and pressure. Full-load steam flow is increasing from 15.9E6 lbm/hr to 16.3E6 lbm/hr. Main steam temperature will decrease from 543.8°F to 543.0°F. Steam generator pressure will decrease from 993.7 psia to 986.9 psia. The actuator designs for the main steam isolation valves (MSIVs) and MSIV bypass valves are based on design conditions which bound the MUR power uprate temperature and pressure conditions.

The required closure times for the MSIV and MSIV bypass valves will not be affected at MUR power uprate conditions.

It is concluded that the main steam isolation system will continue to perform its intended design function at the stated conditions for the MUR power uprate. The design bases have been reviewed and bound the MUR power uprate conditions.

Turbine Steam Bypass System

The primary function of the turbine steam bypass system (TBS) is to reduce the magnitude of nuclear system transients following large turbine load reductions. The turbine bypass system has the capability to bypass main steam from the steam generators to the main condenser through 12 steam dump valves, in a controlled manner, to minimize transient effects on the reactor coolant system of startup, hot shutdown and cooldown, and the step-load reductions in generator load.

The capacity of the TBS is affected by the steam flow properties entering the system. Under MUR power uprate conditions, the steam mass flow rate is increasing and the system pressure is decreasing. The steam volumetric flow rate is increasing because of the increased mass flow and the lower pressure. Under the current licensing basis, the full power reactor vessel average temperature range is from 570.7°F to 588.4°F. This temperature range is maintained for NSSS analyses supporting the MUR power uprate. An evaluation has determined that, for this temperature range and a bounding reactor power level of 3636 MWt, the steam generator outlet pressure ranges from 825 psia to 961 psia and the corresponding steam dump flow capacity ranges from 31.8% to 37.5% of the nominal steam flow at MUR power uprate conditions. For steam pressures higher than 961 psia, the steam dump capacity will be greater than 37.5% of the full load steam flow. At the MUR reactor power level of 3625.6 MWt and a planned vessel average operating temperature of 586.4°F, the predicted full power steam pressure is 986.9 psia. At these conditions, the steam dump capacity will be greater than 37.5%.

A 50% load rejection and a turbine trip from full power were analyzed considering the above described steam dump capacities. The analysis of these transients demonstrated that for a 50% load rejection, the overtemperature and overpower delta-temperature (OTΔT and OPΔT) setpoints were not challenged. In the case of a reactor trip, the pressurizer level remained above the pressurizer heater cutout level and the steam generator relief and safety valves were not challenged (Reference section 7.11 of Enclosures 9 and 10).

It is concluded that the turbine bypass system will continue to perform its intended design function at the stated conditions for the MUR power uprate.

CONDENSATE

Condensate System

The condensate system is discussed in the section titled "Condensate and Feedwater System."

Circulating Water System

The primary function of the circulating water system (CWS) is to supply cooling water to the main condenser and the turbine plant cooling water system and reject the waste heat to the atmosphere through pumps, piping, and a natural draft cooling tower.

Evaluations show that MUR power uprate will increase the heat loads placed on the circulating water tower by 161E6 btu/hr and will increase the hot water return temperature to the tower by 0.6°F. The heat load and return temperature will remain within tower design values. The estimated impact of increased heat load due to the MUR power uprate is less than 0.1°F in basin water temperature at full power plant operation. Consequently, the MUR power uprate has minimal impact on the CWS ability to deliver cold water to the main condenser and turbine plant cooling water (TPCW) system. The estimated decrease in condenser vacuum at full power operation is less than 0.1 inch Hg. These minimal operational changes in basin temperature and condenser vacuum will not significantly reduce plant operating margins.

In conclusion, the CWS will support continued operation of the main condensers and turbine plant cooling water system at MUR power uprate conditions.

SECTION VI

SYSTEM DESIGN

FEEDWATER

Feedwater Heater and MSR Drain System

The primary function of the feedwater heater and MSR drain system is to take the liquid drains from the feedwater heaters, moisture separators, and reheaters and route these drains to the condensate and feedwater system. The feedwater heater and MSR drain system principal design function is to provide feedwater heater drainage to prevent water induction into the main plant turbine.

Critical system design and operating parameters of this system have been reviewed and include pressure, temperature, fluid flow, fluid velocity, component liquid level, electrical amperage, voltage and motor speed, as well as requirements for instrumentation set point changes.

The heater drain pumps (HDPs) and motors are planned for replacement to add flow margin for the expected MUR power uprate conditions. The MUR power uprate operating flow is 6450 gpm per pump with MUR power uprate parameters bounded by existing heater drain system design parameters.

The moisture separator drain tank and MSR drain tank control valves as well as the HDP discharge control valves have been evaluated to be adequately sized to manage the MUR power uprate drain flows.

In conclusion, all requirements for the feedwater heater and MSR drain system have been reviewed, and the system will continue to perform its design function at MUR power uprate conditions.

Condensate and Feedwater System

The condensate and feedwater system provides for condensation of high-pressure and low-pressure turbine extraction and exhaust steam and steam generator feedwater pump turbine exhaust steam. The system also provides for the collection of condensate in the condenser hotwells and maintains the steam generator (SG) water level by supplying preheated feedwater during all power operation modes of the plant. The Condensate and Feedwater System is not required for safe shutdown of the plant, except for feedwater piping common to the auxiliary feedwater system.

The condensate and feedwater system has been reviewed for potential impact from a MUR power uprate. System temperature at the SG inlet will increase from 446.3°F to 448.2°F and is bounded by the system design temperature of 460°F. System temperature at the suction of the condensate pumps will not change. System pressure at the SG inlet will increase from 1148.5 psia to 1156.5 psia and is bounded by the system design pressure of 1800 psig. System flow at the SG inlet will increase from 16.0E6 lb/hr to

16.4E6 lb/hr. System flow at the suction of the condensate pumps will increase from 10.7E6 lb/hr to 10.9E6 lb/hr. No design changes are required to this system as a result of the MUR power uprate, other than the installation of the ultrasonic flow measurement device.

In conclusion, all requirements for the condensate and feedwater system have been reviewed and the system will continue to perform its design function at MUR power uprate conditions.

Feedwater Pump Turbine Drive Steam System

The function of the feedwater pump turbine (FPT) drive steam system is to supply steam to the SG FPT and exhaust it to the condenser. The FPT drive steam system is not required for the safe shutdown of the plant.

The FPT drive steam system critical system parameters have been reviewed for potential impact from a MUR power uprate. The critical system parameters reviewed include system flow, pressure, and temperature. Steam flow to the FPT will increase by about 9.4% and will be delivered at a slightly higher pressure and at a slightly lower temperature. The efficiency of the feedwater pump will decrease from 80.00% to 76.51% as a result of the MUR power uprate. This review has confirmed that the FPT speed control valves (control and stop valves), which regulate steam flow to the FPT, were sized for a flow rate which bounds the expected flow under MUR power uprate conditions.

The lift of the feedwater regulating valve at MUR uprate full power will increase by as much as 7% (from 92% to 99%). This lift does not provide adequate margin to meet the required delivery during a design bases load rejection transient. An evaluation was performed to determine that a feedwater regulating valve position of 80% at the MUR full load conditions is an optimum position for control during steady-state and transient conditions. Therefore, the feed pump speed controller will be recalibrated to achieve optimum valve control position at MUR power uprate load conditions.

In conclusion, all requirements of the FPT drive system have been reviewed and, with the recalibration of the FPT speed controller, the system will continue to perform its design function at MUR power uprate conditions.

Feedwater Line Isolation

The function of the main feedwater line isolation system is to automatically and positively isolate safety Class 2 piping from non-safety Class 4 piping. In the event of a secondary side pipe rupture inside containment, the system limits the quantity of high energy fluid that enters containment through the break and provides a pressure boundary for the controlled addition of auxiliary feedwater (AFW) flow to the intact loops.

The main feedwater line isolation system has been reviewed for potential impact from a MUR power uprate. Critical system parameters reviewed included flow, temperature, and pressure. Feedwater temperature at the SG inlet will increase from 446.3°F to 448.2°F. Feedwater pressure at the SG inlet will increase from 1148.5 psia to 1156.5 psia. Feedwater flow at the SG inlet will increase from 16.0E6 lb/hr to 16.4E6 lb/hr. These values are bounded by their respective design values. This review has confirmed that the closure requirements of the main feedwater isolation valves (MFIVs) and bypass feedwater isolation valves (BFIVs) are bounded by the analyses of record.

It is concluded that the feedwater line isolation system will continue to perform its design function at the MUR power uprate conditions. The design bases have been reviewed and bound the MUR power uprate conditions.

AUXILIARY FEEDWATER

Auxiliary Feedwater System

The auxiliary feedwater (AFW) system provides an independent means of supplying feedwater to the steam generators in addition to the main feedwater system. Its function is to maintain water inventory in the steam generators for reactor residual heat removal (RHR) during those plant conditions when the main feedwater system is inoperable. During emergency operating conditions, the AFW system provides a sufficient reserve of feedwater to permit the plant to operate at hot standby followed by orderly plant cooldown to the point where the RHR system may be placed in service.

The AFW system has been reviewed for potential impact from a MUR power uprate. Critical system parameters reviewed include minimum flow rate and inventory required for the steam generators during an emergency. Since the minimum AFW flow requirement analyses of record are based on the safety analysis and the MUR power uprate impacts the safety analysis performed at the nominal 100% power rating of 3565 MWt, an evaluation was performed and confirmed that the existing safety analyses bound plant operations at the MUR power uprate conditions and remain applicable at the uprate power level. The analyses that credit the AFW system confirmed that the AFW system performance is acceptable at the MUR power uprate conditions (Reference section 5.2.5 of Enclosures 9 and 10). The condensate storage tank (CST) volume requirements in Technical Specification 3.7.6 are bounding for the MUR power uprate conditions as the analyses of record are based on a power level of 102% of the current power level of 3565 MWt (i.e., 3636 MWt) (Reference section 5.2.5 of Enclosures 9 and 10).

The impact of increased decay heat in reactor fuel on the AFW system has been evaluated and found to be acceptable.

It is concluded that the AFW System will continue to perform intended design functions at the MUR power uprate conditions. The design bases have been reviewed and bound the MUR power uprate conditions.

B. CONTAINMENT SYSTEMS

Containment Spray System

The primary function of the containment spray system (CSS) is to reduce post-accident containment building iodine concentrations as necessary to limit calculated offsite doses to less than 10 CFR 100 guidelines. The CSS also functions in conjunction with the safety injection system (SIS) and containment coolers to limit the maximum calculated post-accident pressure in the containment.

The ability of the CSS to reduce iodine concentration and limit the maximum calculated pressure in the containment building post-accident conditions will not be impacted, as current analyses bound MUR power uprate conditions.

The MUR power uprate will result in an increase in containment sump temperature and, therefore, an increase in the containment spray temperature. The maximum containment sump temperature at MUR power uprate conditions is expected to increase from 249.6 °F to 252.7 °F. This increase is bounded by the design temperature of 253.0 °F used in the latest pipe stress analysis of the CSS piping for both trains of both units. Therefore, the increase in temperatures does not affect the current pipe stress analysis for the containment spray system. A review indicates that there is no effect on the long-term post-LOCA containment response/containment integrity analysis and the VEGP FSAR conclusions, since the analysis of record presently assumes a value of 102% of the current nominal rated thermal power.

It is concluded that the CSS will continue to perform design functions at the MUR power uprate conditions. The design bases have been reviewed and bound the MUR power uprate conditions.

Containment Building Air Cooling System

The containment building (CTB) air-cooling system removes thermal energy from the containment atmosphere caused by heat losses of operating equipment during normal power generation, reduces the containment pressure to 50% or less of the containment peak pressure in 24 hours or less during a design basis accident (DBA), and supports the RCS leak detection function during normal operation.

The MUR power uprate will not significantly increase containment heat loads which remain well within the existing containment heat load design margin. CTB cooler heat loads contain a 10% design margin. The analysis of record for the long-term post-LOCA containment response presently assumes a core thermal power of 3565 MW_{th} with an additional 2% for calorimetric uncertainty.

In conclusion, all requirements of the CTB air-cooling system have been determined to conform to the appropriate design criteria. The system safety function to reduce containment pressure during a DBA is bounded by the current analyses.

Containment Isolation System

The primary function of the containment isolation system is to allow the normal or emergency passage of fluids through the containment boundary, while preserving the ability of the boundary to prevent or limit the escape of fission products from postulated accidents. The containment isolation system interfaces with several systems which are impacted by MUR power uprate.

The systems which interface with the containment isolation system have each been reviewed. Since the affects of system parameters (i.e., temperature, heat load, pressure, flow) are not changed or are bounded by current design bases, it can be concluded that the MUR power uprate will not impact the containment isolation system.

For accident conditions inside containment, the containment heat load, temperature, and pressure at MUR power uprate conditions are bounded by current analyses. Also, temperature and pressure outside containment due to postulated pipe breaks are bounded by current analyses. Therefore, the environmental conditions for containment isolation system components are not impacted by the MUR power uprate conditions.

In summary, the containment isolation system has been reviewed for potential impact from the MUR power uprate. Critical system parameters which could impact the containment isolation functions, including the ability of the system to isolate containment during normal and accident conditions, have been reviewed. The evaluation has concluded that system parameters are not changed significantly and are bounded by the existing system design basis. This system and its major components will continue to perform their design functions under the MUR power uprate conditions.

Air Cooling / Circulating Systems

The following air cooling / circulating systems have been reviewed for potential impact from a MUR power uprate:

<u>Containment Building Lower Level Air-Circulating System</u>

The primary function of the CTB lower level air-circulation system is to improve air mixing and air movement to minimize high temperature gradients within the lower elevation levels of the containment during normal operation.

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SYSTEM DESIGN

Containment Building Preaccess Filter System

The primary function of the CTB preaccess filter system is to provide air circulation and filtration of containment areas to reduce airborne radioactivity levels in the containment atmosphere prior to access during normal operation or after reactor shutdown.

Containment Building Cavity Cooling System

The primary function of the CTB cavity cooling system is to provide cool air during normal and loss-of-offsite power (LOSP) conditions to prevent the reactor cavity primary shield concrete from exceeding its maximum temperature limits, to prevent the nuclear instrumentation system from exceeding its maximum temperature limits, and to provide containment cooling during refueling outages.

Containment Building Reactor Support Cooling System

The primary function of the CTB reactor support cooling system is to exhaust cooling air from the reactor vessel supports to maintain the concrete within its operating temperature limit during normal and LOSP conditions.

Containment Building Auxiliary Air-Cooling System

The primary function of the CTB auxiliary air-cooling system is to remove the excess thermal energy from the containment atmosphere due to heat losses of operating equipment during normal power operation.

The functions and critical parameters of the above air cooling and circulating systems have been reviewed for potential impact from a MUR power uprate. Review of these systems has concluded that the parameters for MUR power uprate conditions are bounded by the design bases; therefore, these systems are capable of performing their design functions at MUR power uprate conditions.

Purge Systems

The following purge systems have been reviewed for potential impact from a MUR power uprate:

Containment Building Minipurge Supply and Normal Preaccess Purge Supply Systems

The primary function of the CTB minipurge and normal preaccess purge supply systems is to supply filtered outside air during different modes of plant operations. The CTB minipurge supply system supplies the filtered outside air during plant startup, normal plant operation, hot shutdown, and hot standby. The CTB normal preaccess purge supply system supplies filtered outside air during cold shutdown and refueling.

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Containment Building Minipurge Exhaust and Normal Preaccess Purge Exhaust Systems

The primary function of the CTB minipurge and normal preaccess purge exhaust systems is to provide the necessary containment ventilation air exhaust and filtration in support of the CTB normal minipurge and preaccess supply system. The CTB minipurge exhaust system provides ventilation and filtration during startup, normal operations, cold shutdown, cooldown, hot standby, and refueling. The CTB normal preaccess purge exhaust system provides ventilation and filtration during cold shutdown and refueling.

Containment Building Post-LOCA Purge Exhaust System

The primary function of the CTB post-loss-of-coolant accident (post-LOCA) purge exhaust system is to allow containment purging as a backup to the hydrogen recombiner system to maintain the post-LOCA hydrogen concentration below required limits.

Containment Building Post-LOCA Cavity Purge System

The primary function of the CTB post-LOCA cavity purge system is to prevent hydrogen pocketing in the reactor cavity after a LOCA by supplying air to the reactor cavity for dilution below the limit of flammability. The system does not function during normal plant operation.

The functions and critical parameters of the containment purge systems have been reviewed for potential impact from a MUR power uprate. Review of these systems has concluded that the parameters for MUR power uprate conditions are bounded by the design bases; therefore, these systems are capable of performing their functions at MUR power uprate conditions.

Control Rod Drive Mechanism (CRDM) Ventilation System

This system is discussed in section 6.4.2 of Enclosures 9 and 10.

Hydrogen Recombiner and Monitoring System

The hydrogen recombiner and monitoring system was designed to ensure that the containment hydrogen concentration post-LOCA is maintained at a level low enough (4 volume %) to preclude endangering containment integrity. This function was to be accomplished by means of an electric hydrogen recombiner. However, VEGP TS amendment numbers 134 (Unit 1) and 113 (Unit 2) eliminated the requirements regarding containment hydrogen recombiners and relaxed the requirements for hydrogen monitors.

Review of the hydrogen monitoring system has concluded that the system will continue to perform its design function at MUR power uprate conditions.

C. SAFETY-RELATED COOLING WATER SYSTEMS

Reactor Coolant System

The evaluation of the reactor coolant system is discussed in section 5.1.2 of Enclosures 9 and 10.

Nuclear Service Cooling Water System

The primary function of the nuclear service cooling water (NSCW) system is to provide essential cooling to safety-related equipment and to some non-safety-related auxiliary components. The system functions during normal plant operations and also during abnormal and accident conditions. The NSCW system rejects heat directly to the nuclear service cooling towers (NSCT). The NSCW system is composed of two redundant, completely independent, full-capacity flow trains. The function of the NSCT system is to remove heat from the NSCW system and reject it to the atmosphere.

The MUR power uprate will result in a slight increase in NSCW temperatures during normal operations. This increase is bounded, however, by the system design supply temperature of 95°F, and the ultimate heat sink (UHS) will continue to be maintained at less than 90°F during normal operation as required by TS.

An assessment of the system design calculations confirmed that the basin temperature is relatively insensitive to large changes in heat load due to the mass of the basin. The postulated emergency of a LOSP with three NSCW fans in service (one taken from service by a tornado missile) serves as the bounding case for NSCW supply temperatures. In the original analysis, NSCW temperatures were analyzed to exceed the 95°F design supply temperature by a few degrees. New evaluations show that the slight increase in heat load due to the MUR power uprate and the selection of conservative initial basin temperatures in original NSCW calculations result in an increase in NSCW temperature of less than 1°F.

The assessment also confirmed that for the limiting scenario described in the FSAR, the UHS will be capable of providing the required cooling without normal makeup for 30 days following a postulated LOCA at the MUR power uprate conditions as required by Regulatory Guide 1.27.

In conclusion, the NSCW and NSCT systems have been evaluated, and it has been determined that their operation at MUR power uprate conditions will be bounded by the system design bases. These systems will be capable of performing their design functions as described in the FSAR at the MUR power uprate conditions, and no design changes to NSCW are required as a result of the uprate.

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Component Cooling Water

The component cooling water (CCW) system removes heat from heat exchangers which handle radioactive fluids during normal operation and are necessary for safe operation of the reactor facility. The CCW system also removes heat from the RHR heat exchangers and pumps during accident conditions. In the CCW system, cooling is accomplished through an intermediate closed loop, which is cooled by the NSCW system.

The CCW system has been reviewed for potential impact from a MUR power uprate. Critical system parameters reviewed include flow, pressure, heat load, and temperature. System temperature and heat load will increase; however, flow and pressure will not change. During 100% power generation, the most significant heat load on the CCW system is the spent fuel pool (SFP) heat exchangers. The expected heat load of 23.6E6 btu/hr for the MUR power uprate on the CCW system from the SFP heat exchangers is bounded by the analyzed value of 28.6E6 btu/hr.

It is concluded that the CCW system will continue to perform its intended design function at the stated conditions of the MUR power uprate. The CCW system analyses of record have been reviewed and bound the MUR power uprate conditions.

Safety Injection System

The evaluation of the safety injection system is discussed in section 5.1.3.2 of Enclosures 9 and 10.

Residual Heat Removal System

The RHR system removes heat energy from the reactor core and the reactor coolant system during plant cooldown and refueling operations. The RHR system transfers refueling water between the refueling water storage tank and the refueling cavity at the beginning and end of refueling operations and provides emergency reactor core cooling and long-term recovery as part of the safety injection system (SIS).

The RHR system has been reviewed for potential impact from a MUR power uprate. Critical system parameters reviewed include flow, pressure, and temperature. System flow and pressure will not change. The impact of increased decay heat in reactor fuel on RHR due to the MUR power uprate has been reviewed and found to be acceptable.

A reanalysis of the RHR one and two train cooldown demonstrated that the time for RHR cooldown increases for both units but remains within system design bases limits. From these results, it is concluded that the MUR power uprate will not adversely affect the capability of the RHR system, with a single train operable, to cool the RCS to Mode 5 in the time required by the plant TS.

The RHR system has been reviewed to demonstrate it has the capacity to pass the increased heat of reactor fuel decay onto the CCW system during normal, abnormal, and accident conditions after the MUR power uprate.

The system review has verified that the requirements of 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors," and Regulatory Guide 1.139, "Guidance for Residual Heat Removal," as applicable to long-term cooling and to bringing the reactor to a cold-shutdown condition within 36 hours following shutdown, with only offsite or onsite power available, continue to be met at MUR power uprate conditions.

It is concluded that the RHR system will continue to perform its intended design function at the stated conditions of the MUR power uprate.

D. SPENT FUEL STORAGE AND COOLING SYSTEMS

The primary function of the spent fuel pool cooling system (SFCS) is to remove decay heat generated by the stored fuel assemblies during all modes of plant operation. The heat load on the SFCS is limited by procedural controls governing fuel movement as described in section 9.1.3 of the FSAR. Since the heat load limit for the pools is not changing, there is no impact on the SFCS as a result of the MUR power uprate.

The spent fuel pool cooling system is also discussed in Section II of this enclosure. The evaluation of the criticality analyses for the storage of spent fuel is discussed in section 7.10.7 of Enclosures 9 and 10.

E. RADIOACTIVE WASTE SYSTEMS

Backflushable Filter System

The backflushable filter system consists of the following two major subsystems which function to filter and transport radioactive crud:

Backflushable Filters Subsystem

The backflushable filters function to filter crud in the following systems during normal operation:

- Chemical and volume control system (CVCS).
- Boron recycle system (BRS).
- Liquid waste processing system (LWPS).
- Spent fuel cooling and purification system (SFCPS).

- Steam generator blowdown system (SGBS).

Crud Collection Subsystem

The crud collection subsystem consists of a backflushable filter crud tank and two crud tank pumps which function to collect and transport the crud solution to the radwaste processing facility for disposal.

The backflushable filter system performs no safety function and is not required for safe shutdown of the plant.

The functions and critical parameters of the backflushable filter system have been reviewed for potential impact from a MUR power uprate. Review of this system has concluded that the system will continue to perform its design function at MUR power uprate conditions.

Steam Generator Blowdown System

The function of the steam generator blowdown system (SGBS) is to control the concentration of chemical impurities and radioactive materials in the secondary side of the steam generator.

The blowdown flow rate required to control chemistry and the buildup of solids in the steam generators is a function of allowable condenser in-leakage, total dissolved solids in the plant service water, and the allowable primary to secondary leakage. Since these variables are not impacted by the MUR power uprate, the blowdown flow rate required to control secondary chemistry and steam generator solids will not be impacted.

The inlet pressure to the SGBS varies with steam generator operating pressure, and system control valves modulate as necessary to maintain flow rate. Based on the revised range of NSSS design parameters for power uprate, no-load steam pressure remains the same and the minimum allowable full-load steam pressure also remains the same. Accordingly, the range of NSSS design parameters for the power uprate will not impact blowdown flow control.

The functions and critical parameters of the SGBS have been reviewed for potential impact from a MUR power uprate. Review of this system has concluded that the system will continue to perform its design function at MUR power uprate conditions.

Liquid Waste Processing System

The primary function of the liquid waste processing system (LWPS) is to control, collect, process, handle, store and dispose of liquid radioactive waste generated as a result of normal plant operation, including anticipated operational occurrences. The operation and duty of the system is primarily influenced by the rate of volume increase and isotopic makeup of liquid waste collected in the waste monitor tanks. Neither of these parameters is expected to be impacted by the MUR power uprate; therefore, the capability of the liquid waste processing system to process collected waste volumes is not impacted.

The functions and critical parameters of the Liquid Waste Processing System have been reviewed for potential impact from a MUR power uprate. Review of this system has concluded that the system will continue to perform its design function at MUR power uprate conditions.

Radwaste Processing Facility and Systems

The primary functions of the radwaste processing facility and systems (RPF) are the following:

- To receive and process liquid waste from various plant operations, such that the liquids may be recycled or safely returned to the environment per standard operating procedures.
- To receive, dewater, dry, and provide for safe packing of fluidized, wet solid wastes such as spent resins and filter crud.
- To receive, process, and provide facilities for safe, permanent and temporary short-term storage and packaging of dry or wet radioactive solid wastes, such as spent filter cartridges, which are collected in process shields.
- The RPF also serves the function of containment of liquid and airborne contamination, which may result from RPF operations, to prevent uncontrolled and off-normal releases of radioactivity to the environment.
- The RPF systems serve a shielding function for protection of personnel and equipment from radiation emitted from process functions and equipment.

The functions and critical parameters of the RPF have been reviewed for potential impact from a MUR power uprate. Review of this system has concluded the system will continue to perform its design function at MUR power uprate conditions.

Gaseous Waste Processing System

The primary function of the gaseous waste processing system (GWPS) is to remove fission product gases generated by plant operations, including anticipated operational occurrences.

The functions and critical parameters of the GWPS have been reviewed for potential impact from a MUR power uprate. The sources and production volume of radioactive gas is not expected to change due to the MUR power uprate conditions; therefore, waste gas decay tank storage capacity is not impacted. The MUR power uprate operating conditions do not impact the ability of the GWPS to perform as designed.

Review of this system has concluded that the GWPS will continue to perform its design function at MUR power uprate conditions.

F. ESF HVAC

CONTROL ROOM HABITABILITY SYSTEM

Control Room HVAC System

The functions and operating parameters of the control room heating, ventilation, and air conditioning (HVAC) system have been reviewed. Critical parameters such as operating pressure, temperature, HVAC load and air flow have been reviewed and are not impacted by MUR power uprate conditions. The control room HVAC system remains bounded by the existing design basis and is capable of performing its design function at MUR power uprate conditions.

ESF ATMOSPHERE CLEANUP SYSTEMS (SRP/FSAR 6.5.1)

Systems include the following:

- Control Room HVAC System
- Fuel Handling Building (FHB) Post-Accident Exhaust System
- Piping Penetration Filter Exhaust System

The functions and operating parameters of the above systems have been reviewed. Critical parameters such as operating pressure, temperature, HVAC load and air flow have been reviewed and are not impacted by MUR power uprate conditions. These systems remain bounded by the existing design bases and are capable of performing their design functions at MUR power uprate conditions.

CONTROL ROOM AREA VENTILATION SYSTEM

Systems include the following:

- Control Room HVAC System

- Control Building Wing Area, Levels A, B, 1 and 2 Normal HVAC System
- Control Building Lab Hood and Lab Area Vent System
- Control Building Cable Spreading Room HVAC System
- Technical Support Center HVAC System

The functions and operating parameters of the above systems have been reviewed. Critical parameters such as operating pressure, temperature, HVAC load and air flow have been reviewed and are not impacted by MUR power uprate conditions. These systems remain bounded by the existing design bases and are capable of performing their design functions at MUR power uprate conditions.

SPENT FUEL POOL AREA VENTILATION SYSTEM

Systems include the following:

- FHB Normal HVAC
- FHB Post-Accident Exhaust System

The functions and operating parameters of the above systems have been reviewed. Critical parameters such as operating pressure, temperature, HVAC load and air flow have been reviewed and are not impacted by MUR power uprate conditions. These systems remain bounded by the existing design bases and are capable of performing their design functions at MUR power uprate conditions.

ESF VENTILATION SYSTEM

Systems include the following:

- Control Building Safety Feature Electrical Equipment Room
- Control Building Cable Spreading Rooms HVAC Systems

The functions and operating parameters of the above systems have been reviewed. Critical parameters such as operating pressure, temperature, HVAC load and air flow have been reviewed and are not impacted by MUR power uprate conditions. These systems remain bounded by the existing design bases and are capable of performing their design functions at MUR power uprate conditions.

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1. OPERATOR ACTIONS

The Operator actions and times have been evaluated for the following design bases events included in the Westinghouse Safety Analyses:

- Steam Line Break M&E Releases outside containment
- Steam Line Break M&E Releases inside containment
- Post-LOCA Hot Leg Switchover
- Steam Generator Tube Rupture
- Non-LOCA Transient Analysis:

  - Boron dilution
  - Inadvertent operation of the ECCS
  - Loss of non-emergency AC power (with CVCS injection case)

All of the analyses for these events were performed at 102% power (with exception of the boron dilution event) and therefore bound the uprate conditions; thus, no operator actions and action times are impacted for these events. The boron dilution event was evaluated, and it was determined that the operator actions and times for this event remain valid for the uprate.

The uprate is being implemented under the administrative controls of the design change process. As part of this process, other potential impacts on operator actions and action times in plant procedures will be identified and evaluated during the design change impacts review. The design change process will ensure that impacted procedures will be revised prior to the implementation of the uprate.

2. PROCEDURES, CONTROL ROOM, SIMULATOR, AND TRAINING

A. Emergency and Abnormal Operating Procedures

The EOP and AOP procedures have been reviewed for power uprate impacts, and potentially impacted procedures have been identified. The uprate is being implemented under the administrative controls of the design change process. The design change process will ensure that the impacted procedures will be revised prior to the implementation of the power uprate.

B. Control Room Controls, Displays, and Alarms

The physical modifications to the plant required to support the Vogtle MUR uprate include the Caldon LEFM Check-Plus feedwater measurement system and the new HP turbine. As described in Enclosure 1, the feedwater heater drain pumps and motors are also being replaced to improve system performance. The Caldon feedwater flow system will incorporate a new annunciator on the main control board and new diagnostic functions that will be displayed on the existing plant computer control room monitors. Other computer display functions will be added or modified as required to support the revised calorimetric algorithm. The new HP turbine and heater drain pump modifications require no changes to the control room controls, displays, or alarms. Also, no changes are required to the Safety Parameter Display System monitoring panel.

C. Control Room Plant Reference Simulator

The plant modifications discussed above will require the following changes to the plant reference simulator:

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

The uprate is being implemented under the administrative controls of the design change process. As part of this process, other potential simulator modifications will be identified during the design change impacts review. The design change process will ensure that the simulator modifications will be made prior to the implementation of the uprate.

D. Operator Training Program

The Operations Training department has been involved in the design review process for the modifications required to support the MUR. The Operations staff will be trained on the modifications, technical specification changes, and procedure changes prior to implementation of the MUR. Training on the operation of the Caldon LEFM Check-Plus system and calorimetric impacts will be developed and completed prior to implementation of the MUR.

3. INTENT TO COMPLETE MODIFICATIONS

All modifications discussed above that are required to support the MUR will be completed prior to MUR implementation. The modifications will be evaluated to ensure that changes in Operator actions do not adversely affect defense in depth or safety margins.

4. TEMPORARY OPERATION ABOVE LICENSED POWER LEVEL

The unit operating procedure for Mode 1 operation includes precautions for temporary operation above the licensed power level for certain periods of time. These precautions will be revised to account for the MUR power level.

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5. 10 CFR 51.22 CATEGORICAL EXCLUSION

A. Amounts and Types of Effluents

Non-Radioactive Effluents

The non-radiological environmental evaluation performed for the MUR power uprate is discussed in Enclosure 1. The evaluation concluded that the proposed MUR power uprate will not result in a change in the types of non-radioactive effluents, nor will it result in a significant increase in the amount of non-radioactive effluents released off-site.

Radioactive Effluents

Radioactive liquid discharges to the Savannah River will continue to be monitored and maintained within the limits specified in the Offsite Dose Calculation Manual (ODCM). The review of the operation of liquid waste processing and waste water effluent systems concluded that there will be no changes to the volume or types of liquid radioactive effluents released to the river as a result of the MUR power uprate.

The actual non-gaseous activity of the liquid effluent releases, excluding tritium, is less than 3% of the limit specified in Docket RM-50-2, incorporated into 10 CFR 50, Appendix I. The calculated non-gaseous activity of the liquid release, excluding tritium, increases by less than 0.5% due to the uprated power level. Therefore, the activity of the liquid effluent releases is expected to remain well below the limit at MUR power uprate conditions.

Radioactive gaseous discharges will continue to be monitored and maintained within the limits specified in the ODCM. The review of the gaseous waste system concluded that there will be no changes to the volume or types of gaseous radioactive effluents released to the atmosphere as a result of the MUR power uprate.

There is no detectable I-131 in the gaseous effluent releases. The calculated activity of I-131 in the gaseous effluent releases increases by less than 25% due to the uprated power level. Therefore, the activity of I-131 in the gaseous effluent releases is expected to remain well below the limit specified in Docket RM-50-2, incorporated into 10 CFR 50, Appendix I, at MUR power uprate conditions.

Conclusion

Based on the above reviews, it is concluded that there is no significant change in the types or significant increase in the amounts of any effluents that may be released offsite.

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B. Effects on Individual and Cumulative Occupational Exposure

Individual Exposure

The yearly liquid doses reported in the 2005 annual radioactive effluent release report for both VEGP units are less than 2% of the limits in 10 CFR 50, Appendix I, and Docket RM-50-2, incorporated into 10 CFR 50, Appendix I. The calculated liquid doses increase by less than 20% due to the MUR power uprate. Therefore, the yearly liquid doses are expected to remain well below the limits at MUR power uprate conditions.

The yearly gaseous doses reported in the 2005 annual radioactive effluent release report for both VEGP units are less than 0.01% of the limits in 10 CFR 50, Appendix I, and Docket RM-50-2, incorporated into 10 CFR 50, Appendix I. The calculated gaseous doses increase by less than 25% due to the MUR power uprate. Therefore, the yearly gaseous doses are expected to remain well below the limits at MUR power uprate conditions.

Cumulative Occupational Exposure

The 2003 - 2005 three-year rolling average occupational dose from both VEGP units combined is approximately 106 rem per year. This is expected to increase by less than 2 rem due to the MUR power uprate. This is not considered to be a significant increase when compared to the estimated annual dose of 780 rem in VEGP UFSAR Table 12.4.1-13.

Conclusion

Based on the above reviews, it is concluded that there is no significant increase in individual or cumulative occupational radiation exposure.

SECTION VIII

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

SECTION VIII

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

CHANGES TO TECHNICAL SPECIFICATIONS

The proposed Technical Specification changes are discussed in Enclosure 1.

CHANGES TO PROTECTION SYSTEM SETPOINTS

The only change to protection system setpoints is the change in the value of the power range neutron flux permissive P-9 setpoint and associated allowable value. This is discussed in Enclosure 1.

CHANGES TO EMERGENCY SYSTEM SETPOINTS

As discussed in Section 3.3 of Enclosures 9 and 10, there are no changes to the Engineered Safety Feature Actuation System (ESFAS) system setpoints.