November 20, 2007

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

SUBJECT: VOGTLE ELECTRIC GENERATING PLANT, UNITS 1 AND 2, REQUEST FOR ADDITIONAL INFORMATION (RAI) REGARDING THE MEASUREMENT UNCERTAINTY RECAPTURE POWER UPRATE AMENDMENT (TAC NOS. MD6625 AND MD6626)

Dear Mr. Tynan:

By letter dated August 28, 2007, and supplemented by letter dated October 9, 2007, Southern Nuclear Operating Company, Inc. (the licensee), submitted an application requesting to increase the licensed core power level by 1.7 percent with the installation of the Caldon Check-Plus ultrasonic feedwater flow element for the Vogtle Electric Generating Plant, Units 1 and 2. The Nuclear Regulatory Commission (NRC) staff is reviewing the submittal and has determined that additional information is required to complete its evaluation.

The NRC staff's RAI is enclosed. The licensee is required to provide a response to the RAI within 30 days.

Sincerely,

/RA/

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

Docket Nos. 50-424 and 50-425

Enclosure:
RAI

cc w/encl: See next page

November 20, 2007

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

SUBJECT: VOGTLE ELECTRIC GENERATING PLANT, UNITS 1 AND 2, REQUEST FOR ADDITIONAL INFORMATION (RAI) REGARDING THE MEASUREMENT UNCERTAINTY RECAPTURE POWER UPRATE AMENDMENT (TAC NOS. MD6625 AND MD6626)

Dear Mr. Tynan:

By letter dated August 28, 2007, and supplemented by letter dated October 9, 2007, Southern Nuclear Operating Company, Inc. (the licensee), submitted an application requesting to increase the licensed core power level by 1.7 percent with the installation of the Caldon Check-Plus ultrasonic feedwater flow element for the Vogtle Electric Generating Plant, Units 1 and 2. The Nuclear Regulatory Commission (NRC) staff is reviewing the submittal and has determined that additional information is required to complete its evaluation.

The NRC staff's RAI is enclosed. The licensee is required to provide a response to the RAI within 30 days.

Sincerely,

/RA/

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

Docket Nos. 50-424 and 50-425

Enclosure:
RAI

cc w/encl: See next page

DISTRIBUTION: Public
LPL2-1 R/F
RidsNrrDorlLpl2-1 (EMarinos)
RidsNrrPMSLingam (hard copy)
RidsRgn2MailCenter (SShaeffer)
RidsNrrDciCvib (MMitchell)
RidsNrrDciCvib (TMclellan)
RidsNrrLAMO'Brien (hard copy)
RidsOgcRp
RidsAcrsAcnwMailCenter
RidsNrrDciCsgb (AJohnson)
RidsNrrDciCsgb (AHiser)
RidsNrrDraAadb (MHart)
RidsNrrDprPgcb (TAlexion)
RidsNrrDeEicb (WKemper)
RidsNrrDirsItsb (TKobetz)
RidsNrrDirsIolb (NSalgado)
RidsNrrDeEeeb (GWilson)
RidsNrrDraAfpb (AKlein)
RidsNrrDraAadb (NGreene)
RidsNrrDprPgcb (JO'Driscoll)
RidsNrrDeEicb (IAhmed)
RidsNrrDirsItsb (CSchulten)
RidsNrrDirsIolb (GArmstrong)
RidsNrrDeEeeb (SRay)
RidsNrrDraAfpb (NIqbal)
RidsNrrDraAadb (JShea)
RidsNrrDprPgcb (MMurphy)

ADAMS Accession No.ML073200227 *transmitted by memo dated

OFFICE	NRR/LPL2-1/PM	NRR/LPL2-1/LA	NRR/EICB/BC	NRR/ITSB/BC	NRR/IOLB/BC
NAME	SLingam	MO'Brien	WKemper	TKobetz	NSalgado
DATE	11 /19/07	11/20/07	11/7/07*	10/2/07*	11/1/07*
NRR/EEEB/BC	NRR/AFPB/BC	NRR/CVIB/BC	NRR/AADB/BC	NRR/CSGB/BC	NRR/LPL2-1/BC
GWilson	AKlein	MMitchell	MHart	AHiser	EMarinos
10/18/07*	10/26/07*	11/9/07*	11/8/07*	11/9/07*	11 /20/07

OFFICIAL RECORD COPY

REQUEST FOR ADDITIONAL INFORMATION

REGARDING THE MEASUREMENT UNCERTAINTY

RECAPTURE POWER UPRATE

VOGTLE ELECTRIC GENERATING PLANT, UNITS 1 AND 2

DOCKET NOS. 50-424 AND 50-425

By letter dated August 28, 2007 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML072470691), as supplemented by letter dated October 9, 2007 (ADAMS Accession No. ML072850108), Southern Nuclear Operating Company, Inc., (the licensee), submitted a request for changes to the Vogtle Electric Generating Plant, Units 1 and 2 (Vogtle 1 and 2) Technical Specifications (TSs). The requested changes are to obtain a power uprate on the basis of a plant modification that would result in improved accuracy of feedwater flow measurement, which is used to calculate reactor thermal power. Installation of a Caldon Check-Plus ultrasonic flow meter (UFM) system to measure feedwater flow will allow the licensee to operate the plant with a reduced margin of 0.3% for instrumentation uncertainty and an increased power level of 1.7% of the licensed thermal power. In order to complete its review, the Nuclear Regulatory Commission (NRC) staff needs the following additional information:

Instrumentations and Controls Branch

(1) Criterion 3 in NRC Staff's safety evaluation (SE) on the Caldon Topical Reports ER-80P and ER-157P requested licensees to confirm that the methodology used to calculate UFM uncertainty is based on accepted plant setpoint methodology. Usage of an alternative should be justified and applied to both the venturi and UFM for comparison.

The licensee's response in Enclosure 5 of letter dated August 28, 2007, neither confirmed that the methodology used is based on the NRC staff's accepted plant setpoint methodology (SE reference should be provided) nor justified using an alternate methodology applying to both instruments for comparison.

(2) Section G in Enclosure 5 provides justification for the proposed 48 hours allowed outage time (AOT). It is stated in the first bullet that the 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. Is this a qualitative statement or is there transmitter drift data for this conclusion? Provide the calculated effect of the known transmitter drift on the power calorimetric calculation during the AOT.

(3) Provide confirmation that the UFM mass flow uncertainty used in the total thermal power uncertainty determination includes uncertainty for the actual location of the transducers within the housing as identified in Cameron Customer Information Bulletin CIB 125, Rev. 0 dated April 23, 2007.

(4) Section 7.11.2.5 in WCAP-16736 indicates that the P-9 setpoint and the associated allowable values (AV) are changed to lower values to accommodate non-availability of pressurizer spray flow, three steam dump valves out-of-service, and manual rod control conditions. Confirm that the current setpoint and allowable value of P-8 will not need adjustment to reflect the P-9 changes. The current TS P-9 setpoint and AV are higher than the P-8 values. The proposed change will reverse the difference.

(5) To support NRC assessment of the acceptability of the license amendment request (LAR) in regard to the setpoint change, the licensee is requested to provide the following:

A. Provide documentation of the methodology used for establishing the limiting setpoint (or non-limiting setpoint) and the limiting acceptable values for the as-found and as-left setpoints as measured in periodic surveillance testing. Indicate the related analytical limits and other limiting design values (and the sources of these values).

B. Provide a statement as to whether or not the P-9 setpoint is a limiting safety system setting on which a safety limit (SL) has been placed as discussed in Title 10 of the *Code of Federal Regulations* (10 CFR), Part 50, Section 50.36(c)(1)(ii)(A). If the P-9 setpoint is not SL-related, explain the basis for this.

C. If the P-9 setpoint is determined to be SL-related, please refer to the NRC letter to the NEI Setpoint Methods Task Force (SMTF) dated September 7, 2005 (ADAMS Accession No. ML052500004), which describes setpoint-related TSs (SRTSs) that are acceptable to the NRC for instrument settings associated with SL-related setpoints. Specifically: Part “A” of the enclosure to the letter provides limiting condition for operation (LCO) notes to be added to the TS, and Part “B” includes a check list of the information to be provided in the TS bases related to the proposed TS change.

1. Describe whether and how the SRTS suggested in the September 7, 2005, letter will be implemented. If you do not plan to adopt the suggested SRTS, then explain how compliance with 10 CFR 50.36 will be assured by addressing items C2 and C3, below.

2. Describe how surveillance test results and associated TS limits are used to establish operability of the safety system. Show that this evaluation is consistent with the assumptions and results of the setpoint calculation methodology. Discuss the plant corrective action processes (including plant procedures) for restoring channels to operable status when channels are determined to be “inoperable” or “operable but degraded.” If the criteria for determining operability of the instrument being tested are located in a document other than the TSs (e.g. plant test procedure), explain how the requirements of 10 CFR 50.36 are met.

3. Describe the controls employed to ensure that the instrument setpoint is, upon completion of surveillance testing, consistent with the assumptions of the associated analyses. If the controls are located in a document other than the TSs (e.g. plant test procedure), explain how the requirements of

10 CFR 50.36 are met.

D. For setpoints that are determined to be non-SL-related, describe the measures to be taken to ensure that the associated instrument channel is capable of performing its specified safety functions in accordance with applicable design requirements and associated analyses. Include in your discussion information on the controls you employ to ensure that the as-left trip setting after completion of periodic surveillance is consistent with your setpoint methodology. Also, discuss the plant corrective action processes (including plant procedures) for restoring channels to operable status when channels are determined to be "inoperable" or "operable but degraded." If the controls are located in a document other than the TSs (e.g., plant test procedure), describe how it is ensured that the controls will be implemented.

Technical Specifications Branch

1. Demonstrate compliance with 10 CFR 50.36(c)(2) and (c)(3) for plant operating conditions when the Caldon Check-Plus ultrasonic feedwater flow element cannot perform its specified support function in performing the calorimetric heat balance.

The LAR revises the rated thermal power (RTP) MWt limit in TSs Section 1.1, Definitions. Additionally, the changes include revising the Power Range Neutron Flux, P-9 Interlock nominal trip setpoint and AV setpoint related to operating at 3625.6 MWt. Regulation 10 CFR 50.36(c)(2) specifies when an LCO is not met, the plant must be shutdown or follow any remedial action specified by the TS. Regulation 10 CFR 50.36(c)(3) specifies that surveillance requirements assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits and that the limiting conditions for operation will be met. The proposed LAR TS changes do not specify TS required actions that must be followed if the Caldon Check-Plus ultrasonic feedwater flow element inputs to the calorimetric heat balance are not available for meeting surveillance requirements to demonstrate the LCO is met.

Further, RIS 2002-03, "Guidance on the Content of Measurement Uncertainty Recapture Power Uprate Applications," Attachment 2, "Evaluation of Feedback Received during the Public Workshop on August 23, 2001 (Arranged by Guidance Section)," provides the following guidance:

I.5. What should a licensee do when the instrument is out of service?

NRC staff approvals of topical reports for the feedwater measurement technique identify what information is appropriate for addressing this comment (typically included as the first criterion). Therefore, this information is covered by Items I.1.C. and I.1.D. of the draft guidance. However, as a result of this comment, the NRC staff has modified Section I. to provide more explicit guidance in this area. ***Specifically, a licensee should propose an allowed outage time for the instrument, similar to the allowed outage times contained in the technical specifications for other equipment. If an approved allowed outage time is exceeded, the licensee should reduce the power level of the plant to ensure that it appropriately accounts for the uncertainty in the instrumentation being relied upon.***

Item I.1.G. and H. of the guidance now address the NRC staff's information needs for this case. (emphasis added)

Therefore, the licensee must describe what TSs should be created or modified to address the requirements of 10 CFR 50.36(c)(2), including Required Actions and Completion Times, or state why no additional changes are needed to the TSs when the Caldon Check-Plus ultrasonic feedwater flow element inputs are not available for the heat balance calorimetric algorithm.

Operator Licensing and Human Performance Branch

1. Operator Actions (RIS 2002-03 Section VII.1)

   a. Has the licensee identified any additional design bases events that will require any revisions to existing operator manual actions or available times?

2. Emergency and Abnormal Operating Procedures (RIS 2002-03, Section VII.2.A)

   a. What will be revised in the Vogtle 1 and 2 emergency operating procedures (EOPs) and abnormal operating procedures (AOPs) to accommodate the measurement uncertainty recapture (MUR) power uprate? How will the operators be made aware of these changes?

3. Control Room Controls, Displays (Including the Safety Parameter Display System), and Alarms (RIS 2002-03 Section VII.2.B)

   a. How will the installation of the new Caldon Leading Edge Flowmeter (LEFM) annunicator and associated computer display modifications impact the operators' ability to operate Vogtle 1 and 2 after implementation of the MUR power uprate? How will the operators diagnose and address functionality errors of the Caldon LEFM system?

4. Control Room Plant Reference Simulator and Operator Training Program (RIS 2002-03 Section VII.2.C and D)

   a. Has the licensee identified any additional changes to the plant simulator to address the effects of the MUR power uprate?

   b. What other aspects of the MUR power uprate will be incorporated by the licensee's design change process in order to ensure that operators will be made aware of all plant modifications prior to implementation of the MUR power uprate?

Electrical Engineering Branch

1. In Section V, Part D of the LAR, the licensee states that the stability impact of the power uprate was evaluated and concludes that the proposed electrical output uprate for the units will not cause any stability problems. Provide the grid stability study and discuss in depth the assumptions, methodology, cases studied, and evidence to support the conclusion.

Fire Protection Branch

1. LAR, Section II, "Accidents and Transients for Which the Existing Analyses of Record Bound Plant Operation at the Proposed Uprated Power Level," mentions safe-shutdown fire analysis. The results of the Appendix R evaluation for the MUR power uprate are provided in Table II-1. However this section does not discuss the time necessary for the repair of systems required to achieve and maintain cold shutdown nor the increase in decay heat generation following plant trips. The NRC staff requests the licensee to verify that, with the increased reactor power level of 3625.6 megawatts thermal, the safe-shutdown equipment for Vogtle 1 and 2, would remain in compliance with 10 CFR Part 50, Appendix R.

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

3. The NRC staff notes that LAR, Section III, "Accidents and Transients for Which the Existing Analyses of Record do not Bound Plant Operation at the Proposed Uprated Power Level," does not include any discussion regarding changes to the fire protection program or other operating conditions that may adversely impact the post-fire safe shutdown capability in accordance with 10 CFR Part 50, Appendix R. Clarify whether this request involves changes to the fire protection program or other operating conditions that may adversely impact the post-fire safe-shutdown capability in accordance with 10 CFR, Part 50, Appendix R. Provide the technical justification for whether and, if so, why, existing analyses do not bound any impact on accidents or transients resulting from any changes.

Vessels and Internal Integrity Branch

1. In the Final Safety Analysis Report Update, Revision 13, dated April 2006, under Section 5.3.1, "Reactor Vessel Materials," Table 5.3.1-8 and Table 5.3.1-9, "Reactor Vessel Material Surveillance Program Withdrawal Schedule," for Vogtle 1 and 2, respectively, the capsule withdrawal time is listed as effected full power years (EFPY) from plant start up. Please provide the dates and refueling outage numbers that correspond with the EFPY listed in Table 5.3.1-8 and Table 5.3.1-9 for Vogtle 1 and 2.

2. In WCAP-16736-NP, Revision 1 dated May 2007, Section 8, "Other Evaluations," Sub-Section 8.1.1, "Introduction," Page 8-1, it was stated in the first paragraph: "In both cases, the analyses included plant-specific evaluations for Cycles 1 through 11 and future projections were based on operation with low leakage fuel management and core power level of 3,565 MWt."

   In the second paragraph it was stated "... an analyses is discussed that uses the data from References 1 and 2 as a base and updates the future projections based on core power level of 3636 MWt with low leakage... "

Please clarify the above statements as they appear inconsistent with each other regarding the core power level used in determining the future fluence projections.

3. Please provide a detailed discussion on the impact of the 1.7-percent MUR on the projected upper-shelf energy (USE) values for the reactor vessel (RV) beltline limiting plate materials, including plate identifications, based on USE data from surveillance capsules for Vogtle 1 and 2.

4. In Tables 6.1.2-1 and 6.1.2-2 of the MUR analysis report, it appears that the clad/metal neutron fluence values were used for calculations of ΔRT_{NDT}. American Society of Mechanical Engineers (ASME), *Boiler and Pressure Vessel Code* (Code), Section XI, Appendix G, Article 2120 notes that a postulated flow to the 1/4T (thickness) is to be used in determining facility pressure-temperature (P-T) limits. Therefore, neutron fluence values for the 1/4T depth should be used in lieu of fluence values located at the clad/metal interface for P-T limit determination. In the licensee's "Revision to the Pressure Temperature Report," dated July 1, 2005 (ADAMS Accession No. ML051870322), Table 5-5 notes that neutron fluence values located at 1/4T were used in calculating ΔRT_{NDT} values. Please clarify the differences between Table 6.1.2-1 and Table 6.1.2-2 of the MUR submittal and Table 5-5 of your July 1, 2005 report, and revise Tables 6.1.2-1 and 6.1.2-2, if necessary.

5. Please discuss the effects of the MUR on the integrity of ferritic Class 1 components, specifically the RV, steam generators, and pressurizer, in Section 6.3 of the MUR analysis report; e.g, provide the postulated flaw depth values and the ratio of the calculated stress intensity factor (K_I) to the reference fracture toughness (K_{IR}) for the ferritic components.

6. Discuss the impact of the MUR on the Vogtle 1 and 2 reactor vessel pressurized thermal shock (PTS) assessments in Section 6.1.2.3, *Description of Analyses/Evaluation Performed* for sub-paragraph *Pressurized Thermal Shock*, of the MUR analysis report and how your conclusions regarding this issue relate to the values in Tables 6.1.2-1 through 6.1.2-7 of the report.

7. Discuss in detail the impact of the MUR on the structural integrity of the Vogtle 1 and 2 internal components in Section 6.2, *Reactor Internals*, of the MUR; e.g, the effect of changes due to the MUR evaluations of the RV internals for loading due to structure deadweight, temperature differences, flow loads, fuel assembly pre-load, control rod assembly dynamic loads, vibratory loads, and earthquake accelerations. Identify the Code or Standard (including applicable edition) which was used in the structural evaluation of the Vogtle 1 and 2 internals.

8. The RV internals of pressurized water reactor-designed light-water reactors may be susceptible to the following aging effects:

   (a) cracking induced by thermal cycling (fatigue-induced cracking);
   (b) stress corrosion cracking (SCC);
   (c) irradiation-assisted stress corrosion cracking (IASCC)
   (d) loss of fracture toughness properties induced by radiation exposure for all stainless steel grades;
   (e) the synergistic effects of radiation exposure and thermal aging for cast austenitic stainless steel (CASS) grades;

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

Table Matrix-1 of NRC Review Standard RS-001, Revision 0, "Review Standard for Extended Power Uprate," provides the NRC staff's basis for evaluating the potential for extended power uprates to induce these aging effects. In Table Matrix-1, the NRC staff states that guidance on the neutron irradiation-related threshold levels for inducing IASCC in RV internal components are given in WCAP-14577, Revision 1-A. However, the industry, through the Materials Reliability Program (MRP), is in the process of developing comprehensive inspection and evaluation guidelines for the management of PWR internals degradation due to all of the effects listed above.

In order to ensure that the functionality of the Vogtle 1 and 2 RV internals is maintained over the remaining licensed life of the facility, the NRC staff requests that the licensee provide a commitment to follow and participate in the MRP's development of these comprehensive guidelines. In addition, the NRC staff requests that the licensee commit to evaluating the applicability of these guidelines, upon their completion, to Vogtle 1 and 2 and implementing any recommended inspections/evaluations to manage the effect noted above.

Accident Dose Branch

1. Regulatory Guide (RG) 1.195, "Methods and Assumptions for Evaluating Radiological Consequences of Design Basis Accidents (DBA's) at Light-Water Nuclear Power Reactors," is the U.S. Nuclear Regulation Commission (NRC) guidance document for performing accident dose consequence re-analysis for light water reactor power plants that utilize the Technical Information Document (TID)-14844 source term methodology. In regulatory position 1.3.1, "Design Basis Radiological Analyses" of RG 1.195 states in principal that all the applicable dose consequence DBA's listed in the RG and the final safety analysis report (FSAR) need to be re-analyzed for control room (CR) dose consequence. This is done to identify the limiting event for the general design criterion (GDC)-19 CR dose design criterion.

   Please verify that you performed this analysis for all dose consequence DBA's other than the loss-of-coolant accident (LOCA) and the fuel handling accident (FHA) with no containment isolation. List the accidents analyzed for CR dose consequences and the resulting accident dose values. Also provide the analysis methods and assumptions, including the values used for the CR atmospheric dispersion factor (χ/Q) for each accident.

   Please confirm that the LOCA is the bounding CR dose DBA by presenting your analysis results above or by justifying that your change in analysis assumptions would not cause any other DBA to be limiting for the CR dose consequence.

2. The CR χ/Q values presented in Table III-2 of Enclosure 5 - Section III to Vogtle 1 and 2 MUR power uprate submittal represent Vogtle 2 containment hatch door releases to Vogtle 2 CR emergency air intakes (i.e., Output File: VGC2R2X.out) during a design-basis LOCA. These values are characterized as the most limiting χ/Q values for releases from each of the two units to each of the CR emergency air intakes. However, χ/Q values presented on the enclosed compact disk with supplemental letter dated October 9, 2007 (ADAMS Accession No. ML072850108), shows more conservative atmospheric dispersion factors for the releases from the Vogtle 1 Fuel Handling Building (FHB) to the Vogtle 1 CR (i.e., Output File: VGC1FHB.out) at all time periods (0-720 hours).

Please verify that the estimated CR doses derived using the revised χ/Q values for the MUR power uprate presented in the submittal dated August 28, 2007 (ADAMS Accession No. ML072470691), bound the CR dose estimates derived using χ/Q values presented in the VGC1FHB.out file.

3. Pursuant to NRC Regulatory Issue Summary (RIS) 2002-03, *Guidance on the Content of Measurement Uncertainty Recapture Power Uprate Applications*, a change in licensed power level requires that the radiological consequences of the accidents analyzed in the Chapter XV of the Vogtle 1 and 2 UFSAR be reanalyzed due to increase in source terms. Accordingly, new χ/Q values were presented for the "limiting" LOCA DBA for releases to Vogtle 1 and 2 CR (i.e., onsite). However, no offsite χ/Q values were presented. Please confirm that the χ/Q values for releases to the exclusion area boundary and outer boundary of the low-population zone are bound by the values currently listed in the Vogtle 1 and 2 UFSARs. Please provide these calculations to confirm your determination.

4. In Section 7.8.1, "Radiation Source Terms," of WCAP-16736-P, you have generally described the basis for the re-calculation of your reactor core source term. Describe in detail the analysis methods and the specific assumptions including the release fractions used for the reactor core source term recalculation. Describe in detail any deviations from the guidance in Section 3.1 of RG 1.195.

5. In Section 7.8.1, "Radiation Source Terms," of WCAP-16736-P, you have generally described the basis for the re-calculation of your reactor coolant source term. Describe in detail the analysis methods and the specific assumptions including the change in your iodine appearance rates. Also provide the regulatory basis for the changes that you cited in your application.

6. In Table 7.8-3, "Assumptions Used in the Calculation of revised Iodine Appearance Rates", you changed the percentage of Iodine removal by the Letdown Heat Exchanger from 90-percent to 100-percent removal. Please provide the regulatory basis or the justification for this non-conservative change in the iodine appearance rate assumption.

7. What was the basis for the increased control room unfiltered in-leakage to 835 cfm for unfiltered in-leakage and the 130 cfm for the pressurized control room condition?

8. For the accidents listed in RG 1.195 describe the re-analysis dose consequence calculation input assumptions and methods details. For any calculation assumptions or methods that are not in conformance with the guidance provide the safety justification for these differences. The applicable accidents include the following:

   a) LOCA

   b) Fuel-Handling Accident

   c) PWR Steam Generator Tube Rupture

   d) Main Steam Line Break

   e) Locked Rotor

   f) Rod Ejection

9. Table 7.8-3, “Summary of Revised MUR-PU Control Room Doses for the FHA with No Isolation of Containment,” provides a range of time to switch the CR Heating, Ventilation and Air Conditioning (HVAC) to emergency mode. Explain the mechanism for the CR HVAC switch to “emergency mode,” for the FHA and if this would be accomplished in time to meet regulatory dose limits to control room personnel or if the operator(s) would need to take any actions to meet the required CR personnel dose limits.

SG Tube Integrity and Chemical Engineering Branch

1. The Case 3 analysis in Table 2-1 does not list the maximum value used for steam outlet moisture. Please confirm that the maximum value used was 0.25 percent. If 0.25 percent was not the maximum value used, state the maximum value used and discuss why a different value was used from the other cases.

2. In Section 6.6.9, “Key Input Parameters and Assumptions,” indicates that the reactor vessel outlet temperature for the MUR-PU is 618.2 °F, that minimum T_{hot} is 603.8 °F, that maximum T_{hot} is 620.0 °F, and that the maximum T_{hot} is a 1.8 °F increase from the current operating conditions. Please clarify whether the reactor vessel outlet temperature of 618.2 °F is the reactor vessel outlet temperature for the current operating conditions or for the MUR power uprate conditions.

3. Section 6.6.9, “Description of Analyses and Evaluations and Results,” for tube integrity uses temperatures (T_{hot} = 620 °F) and pressures (steam outlet pressure = 961 psia) from Case 3 in Table 2-1, which is the 0 percent plugging case. Discuss why temperature and pressure parameters from Case 4 in Table 2-1, which is the 10-percent plugging case, were not used, since Case 4 results in a lower steam outlet pressure of 941 psia.

4. In Section 6.6.9, “Effect of T_{hot} Temperature Increase on Steam Generator Tube Degradation,” verify that the provided calculation for crack initiation and propagation of outside diameter stress corrosion cracking is correct. By using the supplied equation, energy of activation value and temperatures resulted in an increase of 4.59 percent as compared to the 4.88 percent value shown in the paper.

5. Confirm that the original coating qualification temperature and pressure profile, used by Vogtle 1 and 2 to qualify Service Level I coatings, remains bounding in light of the power uprate pressures and temperatures. If the original coating qualification pressure and temperature profile is no longer bounding, discuss the conditions to be used and corrective actions that will be taken to assure that Service Level I containment coatings will be qualified.

Vogtle Electric Generating Plant, Units 1 & 2

cc:

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

Mr. N. J. Stringfellow
Manager, Licensing
Southern Nuclear Operating Company, Inc.
P.O. Box 1295
Birmingham, AL 35201-1295

Mr. Jeffrey T. Gasser
Executive Vice President
Southern Nuclear Operating Company, Inc.
P.O. Box 1295
Birmingham, AL 35201-1295

Mr. Steven M. Jackson
Senior Engineer - Power Supply
Municipal Electric Authority of Georgia
1470 Riveredge Parkway, NW
Atlanta, GA 30328-4684

Mr. Reece McAlister
Executive Secretary
Georgia Public Service Commission
244 Washington St., SW
Atlanta, GA 30334

Mr. Harold Reheis, Director
Department of Natural Resources
205 Butler Street, SE, Suite 1252
Atlanta, GA 30334

Attorney General
Law Department
132 Judicial Building
Atlanta, GA 30334

Mr. Laurence Bergen
Oglethorpe Power Corporation
2100 East Exchange Place
P.O. Box 1349
Tucker, GA 30085-1349

Arthur H. Domby, Esquire
Troutman Sanders
Nations Bank Plaza
600 Peachtree Street, NE
Suite 5200
Atlanta, GA 30308-2216

Resident Inspector
Vogtle Plant
8805 River Road
Waynesboro, GA 30830

Office of the County Commissioner
Burke County Commission
Waynesboro, GA 30830