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SOUTHERN COMPANY
Energy to Serve Your World℠

February 26, 2004

Docket Nos.: 50-424
50-425

NL-03-2177

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

Vogtle Electric Generating Plant
Request to Revise Technical Specifications and
Pressure and Temperature Limits Report

Ladies and Gentlemen:

In accordance with the provisions of 10 CFR 50.90, Southern Nuclear Operating Company (SNC) proposes to revise the Vogtle Electric Generating Plant (VEGP) Unit 1 and Unit 2 Technical Specifications (TS) related to the Pressure and Temperature Limits Report (PTLR). The proposed changes to the TS will revise Section 5.6.6, Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR), to facilitate future licensee-controlled changes to the PTLR in accordance with an NRC-approved methodology. In conjunction with this change, SNC is submitting a revised PTLR for review and approval by the NRC and a revision to Section 3.4.12 of the TS to change the RCS vent size in LCO 3.4.12 b. The revisions to the PTLR include new heatup and cooldown limits and Cold Overpressure Protection System (COPS) setpoints. In addition, SNC is submitting for your information TS Bases changes associated with the proposed changes to the TS.

By letter dated February 12, 1996, the NRC staff documented acceptance of the VEGP PTLR. However, because the pressure and temperature (P/T) limits and COPS limits were not consistent with the methodology contained in WCAP-14040-NP-A, Rev. 2, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," the NRC staff limited acceptance of the VEGP PTLR to the format only. As a result, the curves, setpoints, values, and parameters in the PTLR are those that had been previously approved by the staff with Amendments 87 and 65 to the VEGP TS. In a February 12, 1996 NRC letter, the staff stated that when new plant-specific P/T and COPS limits are developed based on an NRC-approved methodology, the staff will review and approve the new limits to ensure that these limits are acceptable based on the NRC-approved methodology.

At that time, the TS Administrative Controls section (Section 5.6.6) must be revised to reference the NRC-approved methodology. As stated above, this submittal includes appropriate changes to Section 5.6.6 of the VEGP TS to reference the NRC-approved methodology and will revise Section 3.4.12 to change the RCS vent size.

A PO1

Southern Nuclear Operating Company is submitting heatup and cooldown curves which have been prepared for 36 Effective Full Power Years (EFPY) for Vogtle Electric Generating Plant-Units 1 and 2. The PTLR is submitted per the guidance of Generic Letter 96-03, "Relocation of the Pressure Temperature Limit Curves and Low Temperature Overpressure Protection System Limits," for NRC approval to allow plant-specific application of the methodology used to calculate new plant heatup and cooldown limits and COPS setpoints and curves. Approval of this methodology will allow future changes to the PTLR to be performed by the licensee pursuant to 10 CFR 50.59, provided the methodology approved by the NRC is used to develop future PTLR changes. Enclosure 1 discusses the revised PTLR methodology.

These curves have been prepared using WCAP-14040-A, Revision 4, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," with one exception. The neutron fluence calculations used Equation 3 of Regulatory Guide 1.190 rather than Equation 4 to perform the flux synthesis. As discussed in Section 1.3.4 of Regulatory Guide 1.190, this approach tends to over predict the maximum flux at the pressure vessel, thereby resulting in slightly conservative calculated results.

ASME Boiler and Pressure Vessel (B&PV) Code, Section XI Code Case N-640, "Alternative Reference Fracture Toughness for Development of P-T Limit Curves for ASME Section XI, Division I," included as Enclosure 2, will be implemented to allow the revised P-T limit curves developed for Vogtle Units 1 and 2 to use the K_{IC}, fracture toughness curve shown on ASME Section XI, Appendix A, Figure A-4200-1, in lieu of the K_{Ia}, fracture toughness curve shown on ASME Section XI, Appendix G, Figure G-2210-1, as the lower bound for fracture toughness. The other margins involved with the ASME B&PV Code, Section XI, Appendix G method of determining P-T limit curves are unchanged. ASME Code Case N-640 has been approved for use without conditions by the NRC in Revision 13 to Regulatory Guide 1.147, "Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1," dated June 2003. Footnote 6 to 10 CFR 50.55a states "ASME Code cases that have been determined suitable for use by the commission staff are listed in...Regulatory Guide 1.147...."

An exemption request is required to eliminate the requirements for the closure head flange and vessel flange regions and is contained in Enclosure 3. The supporting justification for eliminating the requirements for the closure head flange and vessel flange regions is contained in WCAP-16142-P, Revision 1, "Reactor Vessel Closure Head/Vessel Flange Requirements Evaluation for Vogtle Units 1 and 2," which is Enclosure 4. Enclosure 4 contains information that is proprietary to Westinghouse Electric Company. Enclosure 5 contains the Affidavit for Withholding, the proprietary information notice, and the copyright notice in accordance with the requirements of 10 CFR 2.790. Enclosure 6 contains the non-proprietary version of Enclosure 4.

Enclosure 7 provides the proposed Technical Specification (TS) and Bases changes. Enclosure 8 contains the Significant Hazards Consideration Evaluation and the Environmental Impact Analysis. Enclosure 9 contains marked-up pages from the TS, Bases, and PTLR reflecting the proposed changes. Enclosure 10 contains clean-typed copies of the affected TS, Bases, and PTLR pages. Enclosure 11 includes a copy of

WCAP 15068, Revision 3, "Vogtle Electric Generating Plant Unit 1 Heatup and Cooldown Limit Curves for Normal Operation" and WCAP-15161, Revision 3, "Vogtle Electric Generating Plant Unit 2 Heatup and Cooldown Limit Curves for Normal Operation."

SNC requests approval of the proposed changes by February 15, 2005, so that the revised limits can be implemented for the start-up of Unit 1 following the refueling outage currently scheduled for March 2005.

Mr. J. T. Gasser 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.

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

Respectfully submitted,

SOUTHERN NUCLEAR OPERATING COMPANY

Jeffrey T. Gasser

Jeffrey T. Gasser

Sworn to and subscribed before me this 26th *day of* February*, 2004.*

Glenda C. Sparks

Notary Public

My commission expires: 11/10/06

JTG/DRG/daj

Enclosures: Enclosure 1 – PTLR Methodology
Enclosure 2 – ASME Section XI Code Case N-640
Enclosure 3 – Exemption Request to Allow the Elimination of the Reactor Vessel Flange Requirement from the RCS P-T Limits
Enclosure 4 – WCAP-16142-P, Rev. 1
Enclosure 5 – Affidavit for Withholding, Proprietary Information Notice, Copyright Notice
Enclosure 6 – Non-proprietary version of WCAP-16142, Rev. 1
Enclosure 7 – TS, Bases, and PTLR Amendment
Enclosure 8 – Significant Hazards Consideration Evaluation
Enclosure 9 – Pen and Ink Changes
Enclosure 10 – Final TS, Bases, and PTLR Changes
Enclosure 11 – WCAPs 15068 Rev. 3 and 15161 Rev. 3

cc: Southern Nuclear Operating Company
Mr. J. B. Beasley, Jr., Executive Vice President
Mr. W. F. Kitchens, General Manager – Plant Vogtle
Mr. M. Sheibani, Engineering Supervisor – Plant Vogtle
Document Services RTYPE: CVC7000

U. S. Nuclear Regulatory Commission
Mr. L. A. Reyes, Regional Administrator
Mr. C. Gratton, NRR Project Manager – Vogtle
Mr. J. Zeiler, Senior Resident Inspector – Vogtle

State of Georgia
Mr. L. C. Barrett, Commissioner – Department of Natural Resources

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Licensing Document Change Request

Plant ☐ Farley ☐ Hatch ☒ Vogtle

Activity/Document No. N/A LDCR No. 2003088 Unit No.

Activity/Doc. Rev. No. N/A LDCR Rev. No. 1 ☐ 1 ☐ 2 ☒ Shared

Title: Revision to Technical Specifications and Pressure and Temperature Limits

Attachments 1

Preparer:	D. Rick Graham (Print)	/ D. Rick Graham (Signature)	Date: 12-18-03
Reviewer:	T. D. Honeycutt (Print)	/ Tracy D. Honeycutt (Signature)	Date: 12/19/03
Reviewer: (As Needed)	N/A (Print)	/ N/A (Signature)	Date: N/A
LDO Coordinator:	N. J. Stringfellow (Print)	/ N J Stringfellow (Signature)	Date: 12/19/03

Impacted Licensing Documents: TS Sections 3.4.12 and 5.6.6, Bases Sections B 3.4.3 and B 3.4.12, and Units 1 and 2 PTLR.

Change: The proposed changes to the TS will revise Section 3.4.12 to change the RCS vent size in LCO 3.4.12 b and revise Section 5.6.6, Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR). The Bases Sections B 3.4.3 and B 3.4.12 will be revised to reflect the TS changes as well as some editorial changes. The proposed change also includes a revised PTLR.

Justification: The PTLR is submitted per the guidance of Generic Letter 96-03 "Relocation of the Pressure Temperature Limit Curves and Low Temperature Overpressure Protection System Limits," for NRC approval to allow plant-specific application of the methodology used to calculate new plant heatup and cooldown, and Cold Overpressure Protection System (COPS) setpoints and curves. Approval of this methodology will allow future changes to the PTLR to be performed by the licensee pursuant to 10 CFR 50.59; providing the methodology approved by the NRC is used to develop future PTLR changes.

An LDCR that is limited in scope to that described in Question 8 of the Applicability Determination Checklist does not require PRB review.

Yes ☒ No ☐ PRB Review required

Site LDCR Coordinator:	Tim Ruckman (Print)	/ Tim Ruckman (Signature)	Date: 1/22/04
IF NO: Licensing/Dept. Mngr:	N/A (Print)	(Signature)	Date: N/A
IF YES: PRB Meeting No.:	2004-03		Date: 1-28-04
NPGM Approval:	TOM TYNAN (Print)	/ Tom T. Tynan (Signature)	Date: 2/6/04

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Licensing Document Change Request

Plant ☐ Farley ☐ Hatch ☒ Vogtle

Activity/Document No. N/A LDCR No. 2003088 Unit No.

Activity/Doc. Rev. No. N/A LDCR Rev. No. ______ ☐ 1 ☐ 2 ☒ Shared

Title: Revision to Technical Specifications and Pressure and Temperature Limits

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Yes ☒ No ☐ Implementation Pending-See Comments (e.g. NRC prior approval required, DCP/MDC/RER completion required)

Comments: NRC Approval required for this Tech Spec. Change.

Licensing Document Implementation

Incorporated: ☐ As Requested ☐ Not As Requested

Comments: ______

LDCR Rev. No. ______

Site LDCR Coordinator: ______ / ______ Date: ______
Print Signature

SOUTHERN COMPANY
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Applicability Determination Checklist

Plant ☐ Farley ☐ Hatch ☒ Vogtle

Activity/Document No. N/A Unit No.

Activity/Doc. Rev. No. N/A AD Rev. No.________ ☐ 1 ☐ 2 ☒ Shared

Title: Revision to Technical Specifications and Pressure and Temperature Limits Report

Section I – Activity Summary

Preparer: D. Rick Graham (Print) D. Rick Graham (Signature) Date: 12-18-03

Reviewer: T. D. Honeycutt (Print) Tracy D. Honeycutt (Signature) Date: 12/19/03

Change: The proposed changes to the TS will revise Section 3.4.12 to change the RCS vent size in LCO 3.4.12 b and revise Section 5.6.6, Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR). The change also includes a revised PTLR.

Justification: The PTLR is submitted per the guidance of Generic Letter 96-03 "Relocation of the Pressure Temperature Limit Curves and Low Temperature Overpressure Protection System Limits," for NRC approval to allow plant-specific application of the methodology used to calculate new plant heatup and cooldown, and Cold Overpressure Protection System (COPS) setpoints and curves. Approval of this methodology will allow future changes to the PTLR to be performed by the licensee pursuant to 10 CFR 50.59; providing the methodology approved by the NRC is used to develop future PTLR changes.

Section II – Applicable Regulation Determination

1. ☒ Yes ☐ No **Does the activity require a change to the (identify which):**
   - ☐ **Operating License/Renewed Operating License,**
   - ☒ **Technical Specifications**
   - ☐ **Environmental Protection Plan?**

   If the answer to question (1) is yes, perform a 10 CFR 50.92 evaluation.

2. ☐ Yes ☒ No **Does the activity require a change to the Quality Assurance Program?**

   If the answer to question (2) is yes, perform a 10 CFR 50.54(a) evaluation.

3. ☐ Yes ☒ No **Does the activity require a change to the Security Plan, Contingency Plan, or the Security Training and Qualification Plan?**

   If the answer to question (3) is yes, perform a 10 CFR 50.54(p) evaluation.

4. ☐ Yes ☒ No **Does the activity require a change to the Emergency Plan?**

   If the answer to question (4) is yes, perform a 10 CFR 50.54(q) evaluation.

5. ☐ Yes ☒ No **Does the activity require a change to the Inservice Inspection Program or to the Inservice Testing Program, including relief requests?**

   If the answer to question (5) is yes, refer this activity to Material and Inspection Services for evaluation in accordance with 10 CFR 50.55a.

Applicability Determination Checklist

Plant ☐ Farley ☐ Hatch ☒ Vogtle

Activity/Document No. N/A

Unit No.

Activity/Doc. Rev. No. N/A AD Rev. No.

☐ 1 ☐ 2 ☒ Shared

Title: Revision to Technical Specifications and Pressure and Temperature Limits Report

6. ☐ Yes ☒ No **Does the activity require a change to the Fire Protection Program and/or implementing procedures?**

   If the answer to question (6) is yes, perform a fire protection evaluation in accordance with the applicable Operating License condition.

7. ☐ Yes ☒ No **Does the activity require a managerial or administrative procedure change?**

   If the answer to question (7) is yes, the change is subject to the controls of 10 CFR 50, Appendix B. Process the change in accordance with procedures and/or LDCR processing procedures.

8. ☐ Yes ☒ No **Does the activity require a change to the Updated FSARs/TSAR or 10 CFR 72.212 Report (including documents incorporated by reference) that is excluded from the requirements to perform a 10 CFR 50.59 or 10 CFR 72.48 review in accordance with NEI 96-07, Revision 1 or NEI 98-03, Revision 1 such as (identify which):**

   - ☐ **Editorial Changes,**
   - ☐ **Clarifications to improve reader understanding,**
   - ☐ **Correction of inconsistencies within the Updated FSARs/TSAR or 10 CFR 72.212 Report which are clearly discernible (e.g., between sections),**
   - ☐ **Designation of information as historical,**
   - ☐ **Minor corrections to drawings (e.g., correcting mislabeled valves), or**
   - ☐ **Similar changes that do not change the meaning or substance of information presented (e.g., reformatting or removing detail).**
   - ☐ **Incorporation of information submitted to and approved by the NRC as a result of a license amendment.**

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Applicability Determination Checklist

Plant ☐ Farley ☐ Hatch ☒ Vogtle

Activity/Document No. N/A

Unit No. ☐ 1 ☐ 2 ☒ Shared

Activity/Doc. Rev. No. N/A

AD Rev. No.

Title: Revision to Technical Specifications and Pressure and Temperature Limits Report

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9. ☐ Yes ☒ No **Does the activity involve (identify which):**

☐ **A temporary plant alteration to support maintenance (e.g., jumpering terminals, lifting leads, lead shielding, HVAC, scaffolding, and blocking doors) which:**

- **will be restored to the as-designed condition prior to startup if shutdown, or**

will be restored to the as-designed condition within 90 days if at power (Modes 1 and 2)?

☐ **A temporary plant alteration that supports the installation and post-modification testing of an approved plant change which:**

- **will be restored to the as-designed condition prior to startup if shutdown, or**
- **will be restored to the as-designed condition within 90 days if at power (Modes 1 and 2)?**

If the answer to question (9) is yes, perform an assessment of the risk associated with this temporary plant alteration and manage in accordance with 10 CFR 50.65(a)(4).

10. ☐ Yes ☒ No **Does the activity involve a change to a regulatory commitment not covered by another regulation based change process?**

If the answer to question (10) is yes, perform an evaluation consistent with NEI 99-04.

11. ☐ Yes ☒ No **Does the activity involve a change, test, or experiment associated with the Independent Spent Fuel Storage Installation (ISFSI) or spent fuel cask design?**

If the answer to question (11) is yes, perform a 10 CFR 72.48 screen based on the following:

Has the proposed activity been evaluated by the Certificate of Compliance (COC) holder?

☐ **Yes** **Evaluate the proposed activity against the ISFSI as described in the 10 CFR 72.212 Report.**

☐ **No** **Evaluate the proposed activity against the ISFSI as described in the applicable dry storage FSAR/TSAR and the 10 CFR 72.212 Report.**

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Applicability Determination Checklist

Plant ☐ Farley ☐ Hatch ☒ Vogtle

Activity/Document No. N/A

Unit No.

Activity/Doc. Rev. No. N/A AD Rev. No.________

☐ 1 ☐ 2 ☒ Shared

Title: Revision to Technical Specifications and Pressure and Temperature Limits Report

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12. ☐ Yes ☒ No **Does the activity impact other plant specific programs which are different from those already identified above and are excluded from the scope of 10 CFR 50.59 and 10 CFR 72.48, and are controlled by (identify which):**
    - ☐ **Another regulation (e.g., 10 CFRs 20, 26, 50.12, 50.46, and 72.7), (If marked identify):** ________
    - ☐ **Operating License/Renewed Operating License condition (e.g., maximum power level), or**
    - ☐ **Technical Specifications or Environmental Protection Plan (e.g., the ODCM)?**

    If the answer to question (12) is yes, perform the activity in accordance with the applicable requirement.

13. **Does the activity constitute a matter which could result in adverse environmental impact (either direct or indirect)? Check (a) or (b)**

    a. ☒ **No The nature of this change is such that it will not produce conditions which could result in significant adverse environmental impact.**

    b. ☐ **Possibly** (Explain briefly):________

    If question (13.b) is checked, refer this activity to SNC Environmental Services for preparation of an Environmental Evaluation.

14. ☐ Yes ☒ No **Are there any aspects of the activity not controlled by the processes described in items 1 – 13 above?**

    If the answer to question (14) is yes, perform a 10 CFR 50.59 screen. Question (14) must be answered yes if items 1 – 13 are all answered no. Question (14) must also be answered yes for all Design Change Packages (DCPs).

Section III – NRC Approval / LDCR Determination

15. ☒ Yes ☐ No **Is NRC approval required prior to implementation of this activity?**

    If the answer to question (15) is yes, forward the activity to SNC corporate licensing for preparation of a submittal to NRC.

16. ☒ Yes ☐ No **Does this activity require a change to a licensing document(s)?**

    If the answer to question (16) is yes, process the change in accordance with the applicable procedure.

If an LDCR is required, enter the LDCR number. LDCR Number: 2003088

Enclosure 1

Vogtle Electric Generating Plant Units 1 and 2
PTLR Methodology

Enclosure 1

Vogtle Electric Generating Plant
Request to Revise Technical Specifications and
Pressure and Temperature Limits Report

PTLR Methodology

Pressure-temperature (P-T) limit curves were developed for Vogtle Electric Generating Plant (VEGP) Units 1 and 2 for normal operation at 36 EFPY using the methodology from the 2000 Adenda to the 1998 Edition of the ASME Boiler and Pressure Vessel Code, Section XI, Appendix G. Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," was used for the calculation of Adjusted Reference Temperature (ART) values at the 1/4T and 3/4T locations. The 1/4T and 3/4T values were calculated using the intermediate shell plate B8805-2 for Unit 1, and lower shell plate R8-1 for Unit 2 (i.e., the limiting beltline region material). The P-T limit curves were generated without margins for instrumentation errors for heatup rates of 60 and 100°F/hr and cooldown rates of 0, 20, 40, 60 and 100° F/hr. The Vogtle Units 1 and 2 heatup and cooldown P-T limit curves utilize ASME Code Case N-640, which allows the use of the K_{Ic} methodology, and do not include the reactor vessel flange temperature requirement as justified in WCAP-16142, Revision 1.

The heatup and cooldown P-T limit curves were calculated using the adjusted RT_{NDT} (reference nil-ductility temperature) corresponding to the limiting beltline region material of the reactor vessel. The adjusted RT_{NDT} of the limiting material in the core region of the reactor vessel is determined by using the unirradiated reactor vessel material fracture toughness properties, estimating the radiation-induced ΔRT_{NDT}, and adding a margin. The unirradiated RT_{NDT} is designated as the higher of either the drop weight nil-ductility transition temperature (NDTT) or the temperature at which the material exhibits at least 50 ft-lb of impact energy and 35-mil lateral expansion (normal to the major working direction) minus 60°F.

The RT_{NDT} increases as the material is exposed to fast-neutron radiation. Therefore, to find the most limiting RT_{NDT} at any time period in the reactor's life, ΔRT_{NDT} due to the radiation exposure associated with that time period must be added to the unirradiated RT_{NDT} (IRT_{NDT}). The extent of the shift in RT_{NDT} is enhanced by certain chemical elements (such as copper and nickel) present in reactor vessel steels. The Nuclear Regulatory Commission (NRC) has published a method for predicting radiation embrittlement in Regulatory Guide 1.99, Revision 2. Regulatory Guide 1.99, Revision 2, is used for the calculation of ART values ($IRT_{NDT} + \Delta RT_{NDT}$ + margins for uncertainties) at the 1/4T and 3/4T locations, where T is the thickness of the vessel at the beltline region measured from the clad/base metal interface. The most limiting ART values are used in the generation of heatup and cooldown P-T limit curves for normal operation. The calculated capsule and vessel fluence projections were used to determine the most limiting ART values. The fluence evaluation used the ENDF/B-VI scattering cross-section data set. This fluence evaluation is consistent with the methods presented in WCAP-14040-NP-A, Revision 4, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," with one exception. The neutron fluence calculations used Equation 3 of Regulatory Guide 1.190 rather than Equation 4 to perform the flux synthesis. As discussed in Section 1.3.4 of Regulatory Guide 1.190, this approach tends to overpredict the maximum flux at the pressure vessel, thereby resulting in slightly conservative calculated results.

The heatup and cooldown P-T limit curves developed for Vogtle Units 1 and 2 were generated using the most limiting ART values and the NRC-approved methodology documented in WCAP-14040-NP-A, Revision 4, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves." The K_{Ic} critical stress intensities were used instead of the K_{Ia} critical stress intensities. The K_{Ic} methodology is taken from ASME Code Case N-640. The reactor vessel flange temperature requirement has been eliminated from the heatup and cooldown curves. The justification for eliminating the reactor vessel flange requirement is contained in WCAP-16142, Revision 1.

The Vogtle Unit 1 and 2 heatup and cooldown P-T limit curves were generated without margins for instrumentation errors. The curves include a hydrostatic leak test limit curve from 2485 to 2000 psig.

Revised COPS maximum allowable PORV setpoints were determined based on the revised P-T limit curves. The revised COPS maximum allowable PORV setpoints ensure that the Appendix G reactor vessel limits are not exceeded during any potential RCS overpressurization transients that could occur during low temperature operation. The revised setpoints are based on a constant ΔP between the reactor vessel mid-plane and the wide range pressure transmitter of 74 psi, based on all four RCPs in operation. A pressure uncertainty of 46.6 psi and a temperature uncertainty of 18.2°F are included in the PORV setpoints. The COPS maximum allowable PORV setpoints included the effect of 50°F thermal transport. The COPS PORV maximum allowable setpoints were developed in accordance with the methodology contained in WCAP-14040-NP-A, Revision 4, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves."

The revised VEGP Unit 1 and 2 PTLRs are consistent with NRC Generic Letter 96-03, "Relocation of the Pressure Temperature Limit Curves and Low Temperature Overpressure Protection System Limits."

Enclosure 2

Vogtle Electric Generating Plant Units 1 and 2
ASME Section XI Code Case N-640

Enclosure 2

Vogtle Electric Generating Plant
Request to Revise Technical Specifications and
Pressure and Temperature Limits Report

ASME Section XI Code Case N-640, "Alternative Reference Fracture Toughness for Development of P-T Limit Curves for ASME Section XI, Division I"

CASE
N-640

CASES OF ASME BOILER AND PRESSURE VESSEL CODE

Approval Date: February 26, 1999

See Numeric Index for expiration and any reaffirmation dates.

Case N-640
Alternative Reference Fracture Toughness for Development of P-T Limit Curves
Section XI, Division 1

Inquiry: May the reference fracture toughness curve K_{Ic}, as found in Appendix A of Section XI, be used in lieu of Fig. G-2210-1 in Appendix G for the development of P-T Limit Curves?

Reply: It is the opinion of the Committee that the reference fracture toughness K_{Ic} of Fig. A-4200-1 of Appendix A may be used in lieu of Fig. G-2210-1 in Appendix G for the development of P-T Limit Curves. When this Case is employed, LTOP Systems shall limit the maximum pressure in the vessel to 100% of the pressure allowed by the P-T Limit Curves.

Enclosure 3

Vogtle Electric Generating Plant Units 1 and 2
Exemption Request to Allow the Elimination of
the Reactor Vessel Flange Requirement from the RCS P-T Limits

Enclosure 3

Vogtle Electric Generating Plant
Request to Revise Technical Specifications and
Pressure and Temperature Limits Report

Justification for Exemption Request to Allow the Elimination of the Reactor Vessel Flange Requirement from the RCS P-T Limits

In accordance with 10 CFR 50.12, "Specific exemptions," SNC is requesting an exemption from the requirements of 10 CFR 50.60, "Acceptance criteria for fracture prevention measures for lightwater nuclear power reactors for normal operation." The exemption would permit the use of WCAP-16142, Revision 1, "Reactor Vessel Closure Head/Vessel Flange Requirements Evaluation for Vogtle Units 1 and 2," in lieu of the methodology required by 10 CFR 50, Appendix G, footnote 2 to Table 1. WCAP-16142, Revision 1 demonstrates that the flange region can tolerate assumed flaws of 0.1 T (thickness) at 20°F. Additionally, it can be concluded that flaws are unlikely to initiate in the flange region, since there is no known degradation mechanism for the flange region and the fatigue usage in the flange region is less than 0.1. Therefore, eliminating the requirement for the reactor vessel/head flange region when determining pressure-temperature (P-T) limits for the reactor vessel is justified. The proposed exemption meets the criteria of 10 CFR 50.12 as discussed below.

10 CFR 50.12(a) Requirements

10 CFR 50.12 states that the Commission may grant an exemption from the requirements contained in 10 CFR 50 provided that the following is met:

1. The requested exemption is authorized by law.

   No law exists which precludes the activities covered by this exemption request. 10 CFR 50.60 (b) allows the use of alternatives to 10 CFR 50, Appendix G, when an exemption is granted by the Commission under 10 CFR 50.12.

2. The requested exemption does not present an undue risk to the public health and safety.

   The revised P-T limit curves developed for VEGP Units 1 and 2 use the methodology described in WCAP-16142, Revision 1. The WCAP methodology uses a higher material fracture toughness, K_{Ic} instead of K_{Ia}, which results in higher allowable pressures. 10 CFR 50, Appendix G addresses the metal temperature of the closure head flange and vessel flange regions. The regulation states that the metal temperature of the closure flange regions must exceed the material unirradiated RT_{NDT} by at least 120°F for normal operation when the pressure exceeds 20 percent of the preservice hydrostatic test pressure. Implementing the P-T limit curves that use the K_{Ic} material fracture toughness without eliminating the flange requirement of 10 CFR 50, Appendix G, would place a restricted operating window in the temperature range associated with the flange/closure head, i.e., flange RT_{NDT} + 120°F. In accordance with WCAP-16142, Revision 1, the K_{Ic} toughness has been shown to provide significant margin between the applied stress intensity factor and the fracture toughness of the flange/closure head. WCAP-16142, Revision 1, concludes that the integrity of the closure head/flange is not a concern for safe unit operation and testing. Therefore, the VEGP Units 1 and 2 P-T limit curves were developed without the flange requirements.

3. The requested exemption is consistent with the common defense and security.

Justification for Exemption Request to Allow the Elimination of the Reactor Vessel Flange Requirement from the RCS P-T Limits

This exemption request does not affect the national defense or security issues. The common defense and security are not impacted by the approval of this exemption request.

4. Special circumstances are present which necessitate the request for an exemption to the regulations of 10 CFR 50.60.

In accordance with 10 CFR 50.12(a) (2), the NRC will consider granting an exemption to the regulations if special circumstances are present. This requested exemption meets the special circumstances of 10 CFR 50.12(a) (2) (ii) that states:

"Application of the regulation in the particular circumstances would not serve the underlying purpose of the rule or is not necessary to achieve the underlying purpose of the rule;"

The underlying purpose of 10 CFR 50.60 and 10 CFR 50, Appendix G is to protect the integrity of the reactor coolant pressure boundary. WCAP-16142, Revision 1 demonstrated that significant margin exists between the applied stress intensity factor and the material fracture toughness when using the K_{Ic} toughness, which has been endorsed by the ASME B&PV Code, Section XI for developing Pressure-Temperature Limit Curves. Another purpose of the requirements of 10 CFR 50, Appendix G is to assure that fracture margins are maintained to protect against service induced cracking due to environmental effects. Since the governing flaw is on the outside surface (the inside surface is in compression) where there are no environmental effects, there is a greater assurance that the fracture margin is maintained. Therefore, it can be concluded that the integrity of the closure head/flange region is not a concern for VEGP Units 1 and 2 using the K_{Ic} toughness. Additionally, there are no known degradation mechanisms for this region, other than fatigue. The calculated design fatigue usage for this region is less than 0.1; therefore, it can be concluded that flaws are unlikely to initiate in this region.

The use of WCAP-16142, Revision 1, achieves the underlying intent of 10 CFR 50.60 and 10 CFR 50, Appendix G.