

September 7, 2006

Mr. Jeffery B. Archie
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Virgil C. Summer Nuclear Station
Post Office Box 88
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SUBJECT: VIRGIL C. SUMMER NUCLEAR STATION, ISSUANCE OF AMENDMENT
REGARDING BEST ESTIMATE LOSS-OF-COOLANT ACCIDENT ANALYSES
METHODOLOGY (TAC NO. MC8262)

Dear Mr. Archie:

The Nuclear Regulatory Commission has issued the enclosed Amendment No.176 to Renewed Facility Operating License No. NPF-12 for the Virgil C. Summer Nuclear Station. The amendment changes the Technical Specifications (TSs) in response to your application dated June 30, 2005, as supplemented on July 21, 2006.

This amendment revises TS 6.9.1.11 to permit the use of a best estimate methodology in performing loss-of-coolant accident analyses.

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

Sincerely,

/RA/

Robert E. Martin, Senior Project Manager
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-395

Enclosures:

1. Amendment No.176 to NPF-12
2. Safety Evaluation

cc w/enclosures: See next page

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SOUTH CAROLINA ELECTRIC & GAS COMPANY

SOUTH CAROLINA PUBLIC SERVICE AUTHORITY

DOCKET NO. 50-395

VIRGIL C. SUMMER NUCLEAR STATION, UNIT NO. 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 176
Renewed License No. NPF-12

1. The Nuclear Regulatory Commission (the Commission) has found that:
 - A. The application for amendment by South Carolina Electric & Gas Company (the licensee), dated June 30, 2005, as supplemented on July 21, 2006, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;
 - B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;
 - C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;
 - D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and
 - E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications, as indicated in the attachment to this license amendment; and paragraph 2.C.(2) of Renewed Facility Operating License No. NPF-12 is hereby amended to read as follows:

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 176, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. South Carolina Electric & Gas Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This amendment is effective as of its date of issuance and shall be implemented within 30 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Evangelos C. Marinos, Branch Chief
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical
Specifications

Date of Issuance: September 7, 2006

ATTACHMENT TO LICENSE AMENDMENT NO. 176
TO RENEWED FACILITY OPERATING LICENSE NO. NPF-12
DOCKET NO. 50-395

Replace page 3 of Renewed Facility Operating License No. NPF-12 with the attached revised page 3.

Replace the following page of the Appendix A Technical Specifications with the attached revised page. The revised page is identified by amendment number and contain marginal lines indicating the areas of change.

Remove Page

6-16a

Insert Page

6-16a

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 176 TO

RENEWED FACILITY OPERATING LICENSE NO. NPF-12

SOUTH CAROLINA ELECTRIC & GAS COMPANY

SOUTH CAROLINA PUBLIC SERVICE AUTHORITY

VIRGIL C. SUMMER NUCLEAR STATION

DOCKET NO. 50-395

1.0 INTRODUCTION

By application dated June 30, 2005, as supplemented on July 21, 2006, South Carolina Electric & Gas Company (SCE&G, the licensee) requested changes to the Technical Specifications (TSs) for the Virgil C. Summer Nuclear Station (VCSNS). The July 21, 2006, letter provided clarifying information that did not change the June 30, 2005 application and the initial proposed no significant hazards consideration determination.

The amendment will provide approval to apply the Nuclear Regulatory Commission (NRC)-approved Westinghouse best estimate (BE) large break loss-of-coolant accident (LBLOCA) methodology described in Reference 3 for VCSNS.

The NRC staff reviewed the licensee's analyses of the emergency core cooling system (ECCS) performance that were performed in accordance with the code qualification document (CQD) methodology described in Reference 3, at about 102 percent of the licensed core power of 2900 megawatts thermal (MWt). For VCSNS, the LOCA analyses were conducted assuming the plant uses Westinghouse Vantage+ fuel assemblies.

VCSNS is a 3-loop pressurized-water reactor of Westinghouse design, enclosed within a large, dry containment. The ECCS consists of low pressure safety injection flow and high head safety injection (HHSI) flow delivered to the cold legs, and three accumulators with a cover gas pressure of 628 ± 58 pounds per square inch absolute (psia), also injecting into the cold legs. The shut-off head of the HHSI pumps is about 1500 psia.

2.0 REGULATORY ANALYSIS

The LBLOCA analyses were performed to demonstrate that the system design would provide sufficient ECCS flow to transfer the heat from the reactor core following a LOCA at a rate such that (1) fuel and clad damage that could interfere with continued effective core cooling would be prevented, and (2) the clad metal-water reaction would be limited to less than enough to

compromise cladding ductility and would not result in excessive hydrogen generation. The NRC staff reviewed the analyses to assure that the analyses reflected suitable redundancy in components and features; and suitable interconnections, leak detection, isolation, and containment capabilities are available such that the safety functions could be accomplished, assuming a single failure, for LOCAs considering the availability of onsite power (assuming offsite electric power is not available, with onsite electric power available; or assuming onsite electric power is not available with offsite electric power available). The acceptance criteria for ECCS performance that were used by the NRC staff in assessing the acceptability of the CQD methodology for use at VCSNS are provided in Section 50.46 of Title 10 of the *Code of Federal Regulations* (10 CFR 50.46), specifically 50.46 (b)(1), Peak cladding temperature, 50.46 (b)(2) Maximum cladding oxidation, 50.46 (b)(3) Maximum hydrogen generation, 50.46 (b)(4) Coolable geometry, and 50.46 (b)(5) Long term cooling.

The NRC staff also reviewed the limitations and conditions stated in its safety evaluation (SE) supporting approval of the methodology and the range of parameters described in the WCAP-129450-P-A topical report (Reference 3) in its assessment of the acceptability of the methodology for VCSNS.

3.0 LBLOCA ANALYSIS

The licensee stated in its July 21, 2006, submittal that “Both SCE&G and its analysis vendor (Westinghouse) have ongoing processes such that the values and ranges of the BE LBLOCA analysis inputs for peak cladding temperature and oxidation sensitive parameters bound the values and ranges of the as-operated plants for those parameters in accordance with the approved methodology (WCAP-12945-P-A ...).” The NRC staff finds that this statement, along with the generic acceptance of the CQD methodology, provides assurance that the CQD and LBLOCA analyses performed using that methodology apply to VCSNS, operated at the current licensed power level of 2900 MWt.

In its submittal the licensee provided the results for the VCSNS BE LBLOCA analyses at 2958 MWt (about 102 percent of the operating power of 2900 MWt) performed in accordance with the CQD methodology. The licensee’s results for the calculated peak cladding temperatures (PCTs), the maximum cladding oxidation (local), and the maximum core-wide cladding oxidation are provided in the following table along with the acceptance criteria of 10 CFR 50.46(b).

TABLE 1: LARGE BREAK LOCA ANALYSIS RESULTS

| Parameter | V. C. Summer CQD Results | 10 CFR 50.46 Limits |
|--|--------------------------|---------------------------------|
| Limiting Break Size/Location | DEG/PD | N/A |
| Cladding Material | Zirlo | (Cylindrical) Zircaloy or Zirlo |
| Peak Clad Temperature | 1988 °F | 2200 °F (10 CFR 50.46(b)(1)) |
| Maximum Local Oxidation | 5.0% | 17.0% (10 CFR 50.46(b)(2)) |
| Maximum Total Core-Wide Oxidation (All Fuel) | 0.72 % | 1.0% (10 CFR 50.46(b)(3)) |

DEG/PD is a double ended guillotine break at the pump discharge.

In analyses for the VCSNS, the licensee considered the concern that present fuel (Vantage+, Zirlo) may have pre-existing oxidation that must be considered in its LOCA analyses. In the review of this issue, Westinghouse, has responded to the NRC staff's requests for additional information by stating that it considered in the analyses that the Zirlo clad fuel has both pre-existing oxidation and oxidation resulting from the LOCA (pre- and post-LOCA oxidation both on the inside and outside cladding surfaces). Westinghouse and licensees using Westinghouse designed plants have also noted that the fuel with the highest LOCA oxidation will likely not be the same fuel that has the highest pre-LOCA oxidation. Westinghouse and the licensee indicated that when the calculated pre-LOCA oxidation was factored into the licensee's BE LBLOCA analyses for the fuel, consistent with the previous LOCA methodology for VCSNS, that even during a fuel pin's final cycle in the core the sum of the calculated pre- and post-LOCA oxidation was sufficiently small that the total local oxidation remained less than the 17 percent acceptance criterion of 10 CFR 50.46(b)(2) as noted above.

The NRC staff finds that this appropriately addresses the issue with pre-LOCA oxidation because the VCSNS fuel is Westinghouse designed, and the computer code (COBRA/TRAC) used in the previous methodology is the same code used in the CQD methodology. The NRC staff also considered that the calculated LOCA oxidation is sufficiently low (5 percent) that the pre-accident oxidation would have to be incredibly high (greater than 12 percent) for any power-producing rod in the core to exceed the 10 CFR 50.46 (b)(2) total oxidation limit of 17 percent.

The concern with core-wide oxidation relates to the amount of hydrogen generated during a LOCA. Because hydrogen that may have been generated pre-LOCA (during normal operation) will be removed from the reactor coolant system throughout the operating cycle, the NRC staff notes that pre-existing oxidation does not contribute to the amount of hydrogen generated post-LOCA and therefore, it does not need to be addressed when determining whether the calculated total core-wide oxidation meets the 1.0 percent criterion of 10 CFR 50.46(b)(3).

As discussed previously, SCE&G requested Westinghouse to conduct the BE LBLOCA analyses for VCSNS at about 102 percent of the current licensed power level of 2900 MWt using the NRC-approved CQD methodology. The NRC staff concluded that the results of these analyses (see Table 1) demonstrated compliance with 10 CFR 50.46(b)(1) through (b)(3) for licensed power levels of up to 2900 MWt. Meeting these criteria provides reasonable assurance that at the current licensed power level the VCSNS core will remain amenable to cooling as required by 10 CFR 50.46(b)(4). The capability of VCSNS to satisfy the long-term cooling requirements of 10 CFR 50.46(b)(5) is assured by satisfaction of 10 CFR 50.46 (b)(1) through (b)(4) and the approved ECCS design.

3.1 OTHER TECHNICAL ISSUES

3.1.1 Breaks at the Top and Side of Cold Leg Pump Discharge Piping

A LOCA scenario of concern to the NRC staff is a break of a size that could result in extended core uncover requiring operator action to depressurize and establish low pressure recirculation cooling. The NRC staff's concern applies to plants with deep reactor coolant system pump (suction) loop seals (such as VCSNS). The break size for this scenario falls within the range of breaks analyzed by both small break (SB) and large break LOCA methodologies. To the extent

that the SBLOCA and LBLOCA analyses overlap on break size, the more conservative results are considered limiting by the NRC staff. A split break at the top or side of cold leg pump discharge piping could lead to this scenario.

In its July 21, 2006, letter, SCG&E provided information regarding the VCSNS reactor coolant system design that attenuates the conditions leading to core uncover, such as its T-cold upper head design. The licensee also referred to the VCSNS Emergency Operating Procedures (EOPs) that would instruct a timely cooldown, depressurization, and initiation of low pressure injection (LPI)/residual heat removal (RHR) cooling, that would be effective in averting extended core uncover for LOCAs of concern. Based on the existing emergency operating procedure guidance, the NRC staff concludes that the issue related to breaks at the top and side of cold leg pump discharge piping has been adequately addressed for VCSNS.

While the NRC staff found that this issue was addressed for VCSNS as far as it relates to the adoption of the CQD LBLOCA methodology, the NRC staff currently is evaluating the effect of this issue on long term cooling analyses. Resolution of the issues with long term cooling will be addressed generically with the industry.

3.1.2 Post-LOCA Boron Precipitation

This issue was considered in the original licensing of VCSNS. It is unlikely that the fuel in the VCSNS LBLOCA analysis methodology will affect this issue, since the potential for boron precipitation is affected by the power level and associated decay heat and the reactor coolant boron concentration and none of these parameters are affected by this licensing action. Therefore, the NRC staff considers that the status of this issue is unchanged by the use of the CQD LBLOCA methodology. The NRC staff may reconsider this if and when a VCSNS action is proposed that could affect boron precipitation findings, such as a power uprate.

3.1.3 Downcomer Boiling

The concern of this issue is that during the reflood stage of post-LOCA ECCS operation, latent heat from the reactor vessel, the core barrel, and other vessel internals would be transferred to the water in the downcomer of the vessel. The head of water in the downcomer provides the driving force for the reflooding of the core. The heat transferred to the downcomer water would reduce the density of the water, and boil water away. Both of these effects would reduce the driving head for reflooding of the core. Depending on the magnitude and timing of this heat addition to the downcomer water, the core reflood rate could be adversely affected, with adverse consequences to the fuel in the core. If the LOCA analysis methodology does not correctly model heat conduction in the reactor vessel wall, the magnitude and timing of vessel wall heat deposition to the fluid in the downcomer could be non-conservatively timed, such that oxidation and hydrogen generation would be underestimated due to the downcomer boiling effect.

This issue is a generic safety issue whose generic resolution is ongoing. VCSNS LBLOCA analyses were performed with the Westinghouse CQD methodology. WCOBRA-TRAC, the computer code used in the CQD methodology, is the same computer code used to identify the downcomer boiling concern and its treatment of downcomer boiling has been approved as part of the CQD methodology. Therefore, the VCSNS LBLOCA analyses using the Westinghouse CQD methodology acceptably accounts for downcomer boiling.

3.2 LBLOCA CONCLUSIONS

The NRC staff's review of the acceptability of the Westinghouse CQD LBLOCA methodology for VCSNS focused on assuring that the VCSNS specific input parameters or bounding values and ranges (where appropriate) were used to conduct the analyses, that the analyses were conducted within the conditions and limitations of the NRC-approved Westinghouse CQD methodology, and that the results satisfied the requirement of 10 CFR 50.46(b) based on a licensed power level of up to 2900 MWt.

This SE documents the NRC staff review and the bases of acceptance of the Westinghouse CQD, BE LBLOCA analysis methodology for application to the VCSNS, and of the results of the BE LBLOCA analyses discussed above that were performed with the Westinghouse CQD methodology for VCSNS.

Based on its review as discussed above, the NRC staff concludes that the Westinghouse BE LBLOCA methodology, as described in WCAP-12945(P)(A), is acceptable for use at VCSNS. Further, the NRC staff concludes that the results demonstrate, with a high probability level, that the acceptance criteria of 10 CFR 50.46(b)(1), (b)(2), and (b)(3), would not be exceeded during a LBLOCA at the current licensed power level. The NRC staff's conclusion was based on the assumed core power up to 2900 MWt (plus 2.0 percent measurement uncertainty or 2958 MWt).

4.0 VCSNS TECHNICAL SPECIFICATIONS

4.1 TS 6.9.1.11 Core Operating Limits Report

The revisions to TS 6.9.1.11 consist of adding a reference on page 6-16a relating to the VCSNS LBLOCA methodology:

- c. WCAP-12945-P-A, Volume 1 (Revision 2) - through Volumes 2 through 5 (Revision 1) "Code Qualification Document for Best Estimate LOCA Analysis," March 1998 (Westinghouse Proprietary)

Liparulo, N. (W) to NRC Document Control Desk, NSD-NRC-96-4746, "Reanalysis Work Plans Using Final Best Estimate Methodology" dated 6/13/1996.

WCAP-12945-P-A describes an acceptable methodology to apply to VCSNS for determination of PCT as discussed in section 3 of this safety evaluation, and therefore is an appropriate reference for the VCSNS LBLOCA analyses for core power up to 2900 MWt (plus 2.0 percent measurement uncertainty or 2958 MWt).

5.0 SUMMARY

In summary, the licensee has performed LBLOCA analyses for VCSNS using an NRC-approved Westinghouse methodology. The NRC staff concluded that the licensee's LBLOCA analyses were performed using an approved Westinghouse methodology that applies to VCSNS for core power up to 2900 MWt (plus 2.0 percent measurement uncertainty or 2958 MWt). The licensee's LBLOCA calculations demonstrate the following for operation at the current licensed power level:

- The calculated LBLOCA values for PCT, oxidation, and core-wide hydrogen generation provide a high probability that the acceptance criteria of 2200 °F, 17 percent, and 1.0 percent specified in 10 CFR 50.46(b)(1), (2), and (3), respectively, would not be exceeded during a LBLOCA.

Compliance with 10 CFR 50.46(b)(1) through (3) and (5) assures that the core will remain amenable to cooling as required by 10 CFR 50.46(b)(4). The NRC staff finds the licensee's LBLOCA analyses for VCSNS acceptable for core power up to 2900 MWt (plus 2.0 percent measurement uncertainty or 2958 Mwt).

Therefore, the NRC staff finds that use of the proposed methodology, as approved in this SE, will assure that operating limits will be determined such that all applicable limits (i.e., LBLOCA accident analyses limits) of the safety analysis are met. Therefore the proposed changes to TS 6.9 are acceptable.

6.0 STATE CONSULTATION

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

7.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 . The NRC staff has therefore determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (70 FR 59087, October 11, 2005). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

8.0 CONCLUSION

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

9.0 REFERENCES

1. Letter, J. B. Archie, South Carolina Electric & Gas Company, to NRC Document Control Desk, "Core Operating Limits Report - Reference for Best Estimate Loss of Coolant Accident (BELOCA)," June 30, 2005.

2. Letter, M. Fowlkes, South Carolina Electric & Gas Company, to NRC Document Control Desk, "Core Operating Limits report - Reference for best Estimate Loss of Coolant Accident (BELOCA)," July 21, 2006.
3. WCAP-129450-P-A, Volume 1, Revision 2, and Volumes 2 through 5, Revision 1, "Code Qualification Document for Best Estimate LOCA Analysis," S. M. Bajorek, et al., 1998.
4. Letter, N. Liparulo, Westinghouse, to NRC, NSD-NRC-96-4746, "Reanalysis Work Plans Using Final Best Estimate Methodology," June 13, 1996.

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Date: September 7, 2006

VIRGIL C. SUMMER NUCLEAR STATION

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