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April 4, 2000

LCV-1436

Docket Nos: 50-424
50-425

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

Ladies and Gentlemen:

**VOGTLE ELECTRIC GENERATING PLANT
10 CFR 50.46 ECCS EVALUATION MODELS SIGNIFICANT ERROR
AND 1999 ANNUAL REPORT**

Pursuant to the requirements of 10 CFR 50.46 (a)(3)(ii), Southern Nuclear Operating Company (SNC) is submitting a significant error report to report errors in the Small-Break Loss of Coolant Accident (SBLOCA) Emergency Core Cooling System (ECCS) model for Unit 1 only. This report also satisfies the annual reporting requirement for 1999 pursuant to 10 CFR 50.46 (a)(3)(ii). Attached is a description of the errors along with a revised assessment of the SBLOCA peak clad temperature (PCT). The report is based on information provided by Westinghouse of changes and errors assessed against the Vogtle Electric Generating Plant (VEGP) ECCS Evaluation Model and has been prepared in accordance with the guidance in WCAP-13451.

The sum of the absolute magnitudes of additional assessments against the VEGP ECCS Evaluation Model for SBLOCA has resulted in a total error greater than 50 °F for Unit 1 only, which is considered to be significant per 10 CFR 50.46 (a)(3)(i). This report is being submitted pursuant to 10 CFR 50.46 (a)(3)(ii).

In the 1998 annual report (LCV-1342, April 29, 1999), SNC reported a Large-Break Loss of Coolant Accident (LBLOCA) PCT of 2046 °F for Unit 1 and 1996 °F for Unit 2. During 1999, SNC submitted a significant error report (LCV-1388, October 19, 1999) which resulted in a LBLOCA PCT of 2159 °F for both Unit 1 and Unit 2. Subsequent to this report, but during 1999, an additional 10 °F LBLOCA PCT allowance was made for both units for future additions of metal mass inside the containment building. The net LBLOCA PCT for 1999 is 2169 °F for both Unit 1 and Unit 2.

In the 1998 annual report referenced above, SNC reported a SBLOCA PCT of 1845 °F for Unit 1 and 1843 °F for Unit 2. For 1999, the SBLOCA PCT for Unit 1 and Unit 2 remained unchanged.

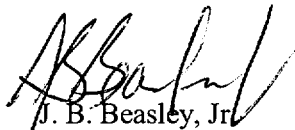
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In March 2000, SNC received notification from Westinghouse of an error in the SPIKE computer program used to estimate fuel rod burst PCT penalties for SBLOCA analyses. The magnitude of the error is 31 °F for Unit 1 and 29 °F for Unit 2. The absolute sum of the errors since the last SBLOCA significant error report (LCV-0327-B, December 8, 1994) is 51 °F for Unit 1 and 49 °F for Unit 2. The net SBLOCA PCT to-date is 1876 °F for Unit 1 and 1872 °F for Unit 2.

The resultant LBLOCA and SBLOCA PCT for each unit remains in compliance with the criterion set forth in 10 CFR 50.46 (b)(1). The criterion requires that the PCT does not exceed 2200 °F.

Per 10 CFR 50.46 (a)(3)(ii), reanalysis or taking other action is not required because compliance with 10 CFR 50.46 (b)(1) has been demonstrated for both LBLOCA and SBLOCA. The results of this report will be incorporated in a future Final Safety Analysis Report (FSAR) update.

Sincerely,



J. B. Beasley, Jr.

JBB/RJF

Attachment

cc: Southern Nuclear Operating Company
Mr. J. T. Gasser
Mr. M. Sheibani
SNC Document Management

U. S. Nuclear Regulatory Commission
Mr. L. A. Reyes, Regional Administrator
Mr. Ramin R. Assa, Vogtle Project Manager, NRR
Mr. J. Zeiler, Senior Resident Inspector, Vogtle

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

ATTACHMENT
VOGTLE ELECTRIC GENERATING PLANT
10 CFR 50.46 ECCS EVALUATION MODELS SIGNIFICANT ERROR REPORT
AND 1999 ANNUAL REPORT

BACKGROUND

Provisions in 10 CFR 50.46 require applicants and holders of operating licenses or construction permits to notify the Nuclear Regulatory Commission (NRC) of errors and changes in the Emergency Core Cooling System (ECCS) Evaluation Models on an annual basis when the errors and changes are not significant, and within 30 days of discovery when the errors and changes are significant. A significant error or change, as defined by 10 CFR 50.46, is one which results in a calculated fuel peak cladding temperature (PCT) different by more than 50 °F from the temperature calculated for the limiting transient using the last acceptable model, or a cumulation of changes and errors such that the sum of the absolute magnitudes of the respective temperature changes is greater than 50 °F.

The following presents a summary of the effects of errors and changes to the Westinghouse ECCS Evaluation Models on the Vogtle Electric Generating Plant (VEGP) Units 1 and 2 loss of coolant accident (LOCA) analyses since the 1998 annual report (Reference 1) and significant error report (Reference 10). This report has been prepared in accordance with the methodology presented in WCAP-13451 (Reference 3). The LBLOCA and SBLOCA analyses, Evaluation Model assessments, and safety evaluation results reported herein will be included in a future VEGP Final Safety Analysis Report (FSAR) update.

LARGE-BREAK LOCA

ECCS Evaluation Model

Since the 1998 annual report (Reference 1), SNC submitted a significant error report in 1999 (Reference 10). The significant error report reported the cumulation of errors since the last LBLOCA significant error report in 1997 (Reference 2). Subsequent to the significant error report submitted in 1999, an additional 10 °F LBLOCA allowance has been made for both units for future additions of metal mass inside the containment buildings.

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The LBLOCA analysis results are based on the Westinghouse BASH large-break ECCS Evaluation Model (Reference 4), as approved by the NRC for VEGP-specific application (References 5 and 6), and the latest acceptable LOCBART model. The limiting size break analysis continues to assume the following information important to the LBLOCA analyses:

- o 17x17 VANTAGE-5 Fuel Assembly
- o Core Power = 1.02 * 3565 MWT
- o Vessel Average Temperature = 571.9 °F
- o Steam Generator Plugging Level = 10%
- o $F_Q = 2.50$
- o $F_{\Delta H} = 1.65$

For VEGP Units 1 and 2, the limiting size break continues to be the double-ended guillotine rupture of the cold leg piping with a discharge coefficient of $C_D = 0.6$. The LBLOCA LOCBART analysis-of-record calculated PCT value is 1915 °F.

The Analysis-of-Record category continues to include an assessment of - 4 °F for the LOCBART clad creep and burst error.

The containment purge, T_{avg} uncertainty, and transition core penalty items continue to be listed separately. The items are listed separately because these items are not explicitly modeled. The PCT assessment values of the containment purge and T_{avg} penalties remain at 10 and 11 °F, respectively. The cycle-specific transition core penalty may be used in subsequent cycles, depending on core design, so it remains a line item but is reported as having a value of 0 °F.

VEGP cores contain ZIRLO clad IFBA fuel rods with a backfill pressure of 100 psig. The ZIRLO clad IFBA rods result in a penalty of 21 °F PCT as calculated by the latest acceptable LOCBART model.

VEGP cores contain ZIRLO clad fuel rods. The use of ZIRLO clad fuel rods results in a penalty of 5 °F PCT as calculated by the latest acceptable LOCBART model.

VEGP cores continue to contain a mix of Zircaloy and ZIRLO clad fuel rods and IFBA and non-IFBA rods. VEGP will continue to show an analysis-of-record LOCBART calculated PCT value based on non-IFBA, Zircaloy fuel rods (1915 °F), and will apply PCT penalties to address the use of ZIRLO clad fuel rods.

Prior 10CFR50.46 BASH Large-Break ECCS Evaluation Model Assessments

As reported in the significant error report in Reference 2, four prior model assessments have been combined into a single assessment of -6°F . These assessments are: (1) Steam Generator Flow Area Application, (2) Structural Metal Heat Modeling, (3) LUCIFER Error Correction, and (4) Translation of Fluid Conditions from SATAN to LOCTA.

In the significant error report in Reference 10, three model assessments were reported. Their combined assessment is 206°F . These assessments are: (1) Increased Accumulator Line Resistances, (2) LOCBART Spacer Grid Single-Phase Heat Transfer Error, and (3) LOCBART Zirc-Water Oxidation Error.

Because these two sets of assessments have been previously reported in significant error reports, they have been combined into a single assessment of 200°F .

1999 10CFR50.46 BASH Large-Break ECCS Evaluation Model Assessments

Since the significant error report in Reference 10, there were no additional assessments in 1999.

LBLOCA 10CFR50.46 ECCS Evaluation Model Assessment Summary

The absolute sum of the LBLOCA PCT assessments since the last LBLOCA significant error report (Reference 10) is 0°F .

10 CFR 50.59 Evaluation Assessments

There are three plant modifications pursuant to 10 CFR 50.59 which affect the LBLOCA analysis results. The combined PCT effects from the two evaluations for the permanent radiation shield and for the trisodium phosphate baskets result in only a 1°F PCT assessment. The third plant modification is the addition of metal mass in containment. An allowance of 10°F has been made for both units for future additions of metal mass.

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Licensing Basis LBLOCA PCT

Based on the above discussions concerning the VEGP-specific application of the Westinghouse BASH large-break ECCS Evaluation Model, the licensing basis LBLOCA PCT is as follows:

A. 1999 Annual Report LBLOCA BASH ECCS Model Analysis-of-Record

1. LOCBART Analysis Result	1915 °F
2. LOCBART Clad Creep and Burst Error	- 4 °F
3. Evaluation for Containment Purging	+ 10 °F
4. Evaluation for +/- 6 °F Uncertainty Band	+ 11 °F
5. Evaluation for Transition Cycle Penalty	+ 0 °F
6. 100 psig Backfill Pressure IFBA with ZIRLO Clad	+ 21 °F
7. ZIRLO Clad Fuel Rods	+ 5 °F

B. Prior 10CFR50.46 BASH Large-Break ECCS Model Assessments

Combined assessments previously reported in References 2 and 10	+200 °F
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C. 1999 10CFR50.46 BASH Large-Break ECCS Model Assessment

No assessments since last significant error report in Reference 10	0 °F
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D. 10 CFR 50.59 Evaluations

1. Permanent Radiation Shield/TSP Baskets	+ 1 °F
2. Addition of Metal Mass in Containment	+ 10 °F

$$\text{Licensing Basis LBLOCA PCT (Unit 1 and Unit 2)} = \underline{2169} \text{ °F}$$

Conclusion

When the effects of assessments to the BASH ECCS Evaluation Model and of safety evaluations were combined with the VEGP LBLOCA analysis results, it was determined that compliance with the requirements of 10 CFR 50.46 is being maintained for both Units 1 and 2.

SMALL-BREAK LOCA

ECCS Evaluation Model

Since the 1998 annual report (Reference 1), no new assessments were identified against the small-break LOCA (SBLOCA) analysis PCT for VEGP Units 1 and 2 during 1999. However, in March 2000, SNC received notification from Westinghouse of an error in the SPIKE computer program used to estimate fuel rod burst PCT penalties for SBLOCA analyses. This error has resulted in a significant error for Unit 1 only. The current SBLOCA analysis results are based on the earlier Westinghouse NOTRUMP small-break ECCS Evaluation Model (Reference 7), as approved by the NRC for VEGP-specific application (References 5 and 6), and the latest acceptable SBLOCA model. The limiting size break analysis continues to assume the following information important to the SBLOCA analyses:

- o 17x17 VANTAGE-5 Fuel Assembly
- o Core Power = 1.02 * 3565 MWT
- o Vessel Average Temperature = 571.9 °F
- o Steam Generator Plugging Level = 10%
- o $F_Q = 2.58$
- o $F_{\Delta H} = 1.70$

For VEGP Units 1 and 2, the limiting size small-break continues to be a three-inch equivalent diameter break in the cold leg. The SBLOCA analysis-of-record SBLOCA calculated PCT value is 1770 °F.

The Analysis-of-Record category continues to include an assessment of +8 °F for the SBLOCA fuel rod initialization error.

The steam generator lower level tap relocation and T_{avg} uncertainty items continue to be listed separately. The items are listed separately because these items are not explicitly modeled. The PCT assessment values on these items are 15 °F and 4 °F, respectively. A PCT assessment of 30 °F is also listed separately for Burst and Blockage/Time in Life.

VEGP cores contain ZIRLO clad fuel rods. The use of ZIRLO clad fuel rods results in a penalty of 3 °F PCT as calculated in the latest acceptable SBLOCA model. This penalty applies to both IFBA and non-IFBA rods.

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Prior 10CFR50.46 NOTRUMP Small-Break ECCS Evaluation Model Assessments

Five prior model assessments have been combined into a single assessment of -17°F (Reference 8) since the last SBLOCA significant error report submitted in 1994 (Reference 9). These assessments are: (1) Safety Injection (SI) Flow into the Broken RCS Loop/Improved Steam Condensation Model, (2) Drift Flux Flow Regime Error, (3) LUCIFER Error Corrections, (4) Boiling Heat Transfer Correlation Error, and (5) Steam Line Isolation Logic Error.

The NOTRUMP specific enthalpy error continues to be listed separately in accordance with WCAP-13451 since it was not combined with the prior model assessments (see Reference 10).

1999 10CFR50.46 NOTRUMP Small-Break ECCS Evaluation Model Assessments

No new assessments were identified in 1999.

2000 To-Date 10CFR50.46 NOTRUMP Small-Break ECCS Evaluation Model Assessments

The SPIKE computer program is used to predict fuel rod burst peak cladding temperature penalties for SBLOCA analyses. The SPIKE computer program has been revised. The revision has resulted in a net PCT penalty of 31°F for Unit 1 and 29°F for Unit 2.

SBLOCA 10CFR50.46 ECCS Model Assessment Summary

The absolute sum of the SBLOCA PCT assessments since the last SBLOCA significant error report (Reference 9) is 51°F for Unit 1 and 49°F for Unit 2. This results in a significant error for Unit 1 only.

10 CFR 50.59 Evaluation Assessments

There are two plant modifications pursuant to 10 CFR 50.59 which affect the SBLOCA analysis results for VEGP Unit 1. These are: (1) annular pellet blankets, and (2) loose part in the RCS. The PCT penalty on annular pellet blankets is the only one of the two which is applicable to VEGP Unit 2.

Licensing Basis SBLOCA PCT

Based on the above discussions concerning the VEGP-specific application of the Westinghouse NOTRUMP small-break ECCS Evaluation Model, the licensing basis SBLOCA PCT is as follows:

A. SBLOCA NOTRUMP ECCS Model Analysis-of-Record

1. SBLOCA Analysis Result	1770 °F
2. SBLOCA Fuel Rod Initialization Error	+ 8 °F
3. Evaluation for Steam Generator Lower Level Tap Relocation	+ 15 °F
4. Evaluation for +/- 6 °F Uncertainty Band	+ 4 °F
5. Burst and Blockage/Time in Life	+ 30 °F
6. ZIRLO Clad Fuel Rods	+ 3 °F

B. Prior 10CFR50.46 NOTRUMP Small-Break ECCS Model Assessments

1. Combined assessments previously reported in References 8 and 9	- 17 °F
2. NOTRUMP Specific Enthalpy Error	+ 20 °F

C. Current 10CFR50.46 NOTRUMP Small-Break ECCS Model Assessments

1. 1999 Model Assessments	0 °F
2. Burst and Blockage/Time in Life SPIKE Correlation Revision	
(Unit 1)	+ 31 °F
(Unit 2)	+ 29 °F

D. 10 CFR 50.59 Evaluations

1. Annular Pellet Blankets	+ 10 °F
2. Loose Part (Unit 1 only)	+ 2 °F

Licensing Basis SBLOCA PCT (Unit 1) =	<u>1876 °F</u>
(Unit 2) =	<u>1872 °F</u>

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Conclusion

When the effects of assessments to the NOTRUMP ECCS Evaluation Model and the effects of safety evaluations were combined with the VEGP SBLOCA analysis results, it was determined that compliance with the requirements of 10 CFR 50.46 is being maintained for both Units 1 and 2.

REFERENCES

1. LCV-1342, "Vogtle Electric Generating Plant, 10CFR50.46 ECCS Evaluation Models 1998 Annual Report," letter from J. B. Beasley, Jr. (SNC) to USNRC, dated April 29, 1999.
2. LCV-0998, "Vogtle Electric Generating Plant, 10 CFR 50.46 ECCS Evaluation Models 1996 Annual Report and Significant Error Report," letter from C. K. McCoy (SNC) to USNRC, dated March 31, 1997.
3. WCAP-13451, "Westinghouse Methodology for Implementation of 10 CFR 50.46 Reporting," dated October 1992.
4. "The 1981 Version of the Westinghouse ECCS Evaluation Model Using the BASH Code," WCAP-10266-P-A, Rev. 2, (Proprietary), March 1987.
5. Safety Evaluation by the Office of Nuclear Reactor Regulation Related to Amendment Nos. 43 and 44 to Facility Operating License NPF-68 and Amendment Nos. 23 and 24 to Facility Operating License NPF-81, attachment to letter from Hood (USNRC) to Hairston (GPC), dated September 19, 1991.
6. Safety Evaluation by the Office of Nuclear Reactor Regulation Related to Amendment No. 60 to Facility Operating License NPF-68 and Amendment No. 39 to Facility Operating License NPF-81, attachment to letter from Hood (USNRC) to Hairston (GPC), dated March 22, 1993.
7. "Westinghouse Small-Break ECCS Evaluation Model Using the NOTRUMP Code," WCAP-10054-P-A (Proprietary) and WCAP-10081-A (Non-Proprietary), August 1985.
8. LCV-0579, "Vogtle Electric Generating Plant, 10 CFR 50.46 ECCS Evaluation Models 1994 Annual Report," letter from C. K. McCoy (GPC) to USNRC, dated March 17, 1995.
9. LCV-0327-B, "Vogtle Electric Generating Plant, 10 CFR 50.46 ECCS Evaluation Models Significant Change Report," letter from C. K. McCoy (GPC) to USNRC, dated December 8, 1994.
10. LCV-1388, "Vogtle Electric Generating Plant, 10 CFR 50.46 ECCS Evaluation Models Significant Error Report," letter from J. B. Beasley, Jr. (SNC) to USNRC, dated October 19, 1999.