Georgia Power Company
40 Inverness Center Parkway
Post Office Box 1295
Birmingham, Alabama 35201
Telephone 205 877-7122

C. K. McCoy
Vice President, Nuclear
Vogtle Project

Georgia Power
the southern electric system

February 23, 1993

ELV-05255
003075

Docket Nos. 50-424
50-425

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

Vogtle Electric Generating Plant
10 CFR 50.46 ECCS MODEL 1992 ANNUAL REPORT

Gentlemen:

Attached is Georgia Power Company's 10 CFR 50.46 emergency core cooling system (ECCS) 1992 Model Annual Report based on WCAP-13451 and in compliance with the reporting requirements of 10 CFR 50.46(a)(3)(ii). It is based on information provided by Westinghouse on January 29, 1993, of errors and changes assessed against the (VEGP) ECCS models for the calendar year 1992.

The attached annual report provides information on the ECCS Evaluation Model errors/changes on peak clad temperature (PCT) based on an ECCS reanalysis that incorporated the T_{hot} reduction safety evaluation and the fuel rod model update, IMP database errors, and miscellaneous input changes. These items were previously identified and assessed in the 1991 annual report (ELV-03439 dated February 13, 1992) as reanalyzed assessments. Also, the attached annual report provides a summary of the plant change safety evaluations performed under the provisions of 10 CFR 50.59 that also affect the PCT results. This revised ECCS analysis, assessments, and safety evaluations will be incorporated in the next Final Safety Analysis Report (FSAR) update.

It has been determined that compliance with the requirements of 10 CFR 50.46 continues to be maintained when the effects of plant design changes performed under 10 CFR 50.59, which could affect the large break loss of coolant accident (LOCA) and small break LOCA analyses results, are combined with the effects of the ECCS Evaluation Model errors/changes applicable to VEGP Units 1 and 2.

010032

A001

Georgia Power

If you have any questions regarding this report, please contact this office.

Sincerely,

C.K.McC

C. K. McCoy

CKM/BCA/HWM:gps

Attachment

cc: Georgia Power Company
Mr. W. B. Shipman
Mr. M. Sheibani
NORMS

U.S. Nuclear Regulatory Commission
Mr. S. D. Ebneter, Regional Administrator
Mr. D. S. Hood, Licensing Project Manager, NRR
Mr. B. R. Bonser, Senior Resident Inspector, Vogtle

ATTACHMENT

10 CFR 50.46 ECCS MODEL 1992 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. Reference 1 defines a significant error or change as 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 as 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 an assessment of the effects of the significant errors and changes to the Westinghouse ECCS Evaluation Models on revised loss of coolant accident (LOCA) ECCS analyses performed for Vogtle Electric Generating Plant (VEGP) Units 1 and 2. The ECCS reanalysis results were reported in the 1991 Annual Report (reference 2) and are being incorporated in this 1992 Annual Report based on the format presented in WCAP-13451 (reference 3). The ECCS reanalysis, assessments, and safety evaluations reported herein will be included in the next VEGP Final Safety Analysis Report (FSAR) Update.

LARGE BREAK LOCA

ECCS EVALUATION MODEL

A large break LOCA ECCS reanalysis for VEGP Units 1 and 2 was performed to incorporate the effect of errors and changes to the Westinghouse large break LOCA ECCS evaluation model on PCT results reported since the 1991 Annual Report (reference 2). The large break LOCA ECCS reanalysis results were calculated using the same Westinghouse BASH large break LOCA ECCS evaluation model (reference 4) approved by NRC for VEGP specific application (reference 5). The limiting size break analysis assumed the following information important to the large break LOCA 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 revised ECCS analysis calculated PCT is 2025°F. This ECCS reanalysis value incorporates the fuel rod model update, IMP database errors, and miscellaneous input changes assessment and a T_{hot} reduction (reference 2).

The transition core penalty, containment purge, and T_{avg} uncertainty items (reference 2, attachment D, items 2a, b, and c) continue to be listed separately per the format of WCAP-13451 (reference 3). The items are listed separately because these items are not explicitly modeled. The PCT assessment values on these items are 10, 11, and 50°F, respectively.

NEW LOCA MODEL ASSESSMENTS

The following errors and changes to the Westinghouse ECCS Evaluation Models would affect the BASH Evaluation Model large break LOCA reanalysis results.

Structural Metal Heat Modeling

A discrepancy was discovered during review of the finite element heat conduction model used in the WREFLOOD-INTERIM code to calculate heat transfer from structural metal in the vessel during the reflood phase. It was noted that the material properties available in the code corresponded to those of stainless steel. While this is correct for the internal structures, it is inappropriate for the vessel wall which consists of carbon steel with a thin stainless internal clad. This was defined as a nondiscretionary change per Section 4.1.2 of WCAP-13451, since there was thought to be a potential for increased PCT with a more sophisticated composite model. The model was revised by replacing it with a more flexible one that allows detailed specification of structures. The estimated effect of this correction is a 25°F PCT benefit.

NEW 10 CFR 50.59 SAFETY EVALUATION ASSESSMENTS

No new 10 CFR 50.59 safety evaluations affecting the large break LOCA reanalysis results have been identified. The safety evaluation for reduced full power operating temperature, T_{hot} reduction, was explicitly modeled in the revised ECCS analysis results (reference 2).

TOTAL RESULTANT LARGE BREAK LOCA PCT

Based on the above discussions concerning the VEGP specific application of the Westinghouse large break LOCA ECCS Evaluation Model, the 1992 year end total resultant large break LOCA PCT is as follows:

1. 1991 Annual Report ECCS results (reference 2)		2037.0°F
2. ECCS reanalysis results - (incorporates reference 2, attachment C, item 3, LOCA model assessments and reference 2, attachment D, item 3, safety evaluation assessments)		2025.0°F
3. Prior evaluations which are not explicitly modeled -		
a. Evaluation for containment purging (reference 2)	+	10.0°F
b. Evaluation for +/- 6°F uncertainty band (reference 2)	+	11.0°F
c. Evaluation for transition cycle penalty (reference 2)	+	50.0°F
4 Prior LOCA model assessments - steam generator flow area application (reference 2)	+	10.0°F
5. New LOCA model assessments - structural metal heat modeling	-	25.0°F
6. 10 CFR 50.59 safety evaluation - none	+	0.0°F
Total Resultant PCT =		2081.0°F

CONCLUSION

An evaluation of the effect of errors and changes to the Westinghouse large break BASH ECCS Evaluation Model was performed on the large break LOCA ECCS reanalysis results. When the effects of the large break ECCS model errors/changes and safety evaluation were combined with the VEGP ECCS reanalysis results, it was determined that VEGP Units 1 and 2 are in compliance with the requirements of 10 CFR 50.46(b).

SMALL BREAK LOCA

ECCS EVALUATION MODEL

A small break LOCA ECCS reanalysis for VEGP Units 1 and 2 was performed to incorporate the effect of the errors and changes to the Westinghouse small break LOCA ECCS Evaluation Model on PCT results reported since the 1991 Annual Report (reference 2). The small break LOCA ECCS reanalysis results were calculated using the same Westinghouse NOTRUMP small break LOCA ECCS Evaluation Model (reference 6) approved by NRC for VEGP specific application (reference 5). The analysis assumed the following information important to the small break LOCA analyses:

- o 17x17 VANTAGE-5 Fuel Assembly
- o Core Power = 1.02 * 3565 MWt
- o Vessel Average Temperature = 571.9^{o}F
- o Steam Generator Plugging Level = 10%
- o F_Q = 2.48 at 9.5 ft
- o F-delta-H = 1.70

For VEGP Units 1 and 2, the limiting size small break continues to be a 3-inch equivalent diameter break in the cold leg. The revised ECCS analysis calculated PCT is 1809^{o}F. This ECCS reanalysis value incorporates the fuel rod model update, IMP database errors, and miscellaneous input changes assessment and T_{hot} reduction (reference 2).

The steam generator lower level tap relocation and T_{avg} uncertainty items (reference 2, attachment D, items 2a and b) continue to be listed separately per the format of WCAP-13451 (reference 3). The items are listed separately because these items are not explicitly modeled. The PCT assessment values on these items are 15 and 4^{o}F, respectively.

NEW LOCA MODEL ASSESSMENTS

The following errors and changes to the Westinghouse ECCS Evaluation Models would affect the NOTRUMP small break LOCA reanalysis results.

Bessel Function Error

An error was discovered in SUBROUTINE BESSJO which led to the calculation of incorrect values for the zeroth order Bessel function of the first kind. This calculation is used in the algorithm designed to limit heat transfer out of a quenching fuel rod to the theoretical conduction limit. This error existed only in one cycle of the NOTRUMP computer code (Cycle 23) and therefore only affects analyses performed with that version. Cycle 23 of NOTRUMP was in use from February 1991 until the error was corrected in February 1992. This error correction returned the NOTRUMP code to consistency with the applicable section of WCAP-10079-P-A and therefore is not a change to the Evaluation Model. This was determined to be a nondiscretionary change in accordance with Section 4.1.2 of WCAP-13451 and was corrected in accordance with Section 4.1.3 of WCAP-13451.

The effect of this change on the small break LOCA analysis PCT calculations was determined using the 1985 small break LOCA Evaluation Model (Reference 5) by performing computer analysis calculations. The calculations showed that the PCT increased by $25^{\circ}F$. Therefore, a $25^{\circ}F$ penalty has been assessed against the VEGP small break LOCA PCT results.

NEW 10 CFR 50.59 SAFETY EVALUATION ASSESSMENTS

No new 10 CFR 50.59 safety evaluations affecting the small break LOCA reanalysis results have been identified. The safety evaluation for reduced full power operating temperature, T_{hot} reduction, was explicitly modeled in the revised ECCS analysis results (reference 2).

TOTAL RESULTANT SMALL BREAK LOCA PCT

Based on the above discussions concerning the VEGP specific application of the Westinghouse small break LOCA ECCS Evaluation Model, the 1992 year end total resultant small break LOCA PCT is as follows:

1. 1991 Annual Report ECCS results (reference 2)	$2037.0^{\circ}F$
2. ECCS reanalysis results - (incorporates reference 2, attachment C, item 3, LOCA model assessments and reference 2, attachment D, item 3, safety evaluation assessments)	$\underline{1809.0}^{\circ}F$
3. Prior evaluations which are not explicitly modeled -	
a. Evaluation for steam generator lower level tap relocation (reference 2)	+ $15.0^{\circ}F$
b. Evaluation for +/- $6^{\circ}F$ uncertainty band (reference 2)	+ $4.0^{\circ}F$
4. Prior LOCA model assessments - none	+ $0.0^{\circ}F$
5. New LOCA model assessments Bessel function error	+ $25.0^{\circ}F$
6. 10 CFR 50.59 safety evaluation assessments - none	+ $0.0^{\circ}F$
Total Resultant PCT =	$\underline{1853.0}^{\circ}F$

CONCLUSION

An evaluation of the effect of errors and changes to the Westinghouse small break NOTRUMP ECCS Evaluation Model was performed on the small break LOCA ECCS reanalysis results. When the effects of the small break ECCS model errors/changes and safety evaluations were combined with the VEGP ECCS reanalysis results, it was determined that VEGP Units 1 and 2 are in compliance with the requirements of 10 CFR 50.46(b).

REFERENCES

1. "Emergency Core Cooling Systems; Revisions to Acceptance Criteria," Federal Register, Vol. 53, No. 180, pp. 35996-36005, dated September 16, 1988.

2. ELV-03439, "Vogtle Electric Generating Plant, 10 CFR 50.46 ECCS Model Annual Report," letter from C. K. McCoy (GPC) to USNRC, dated February 13, 1992.

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-11524-A (Non-Proprietary), March 1987.

5. ELV-02166, "Vogtle Electric Generating Plant, Request for Technical Specifications Changes VANTAGE-5 Fuel Design," letter from W. G. Hairston, III, to USNRC, dated November 29, 1990.

6. "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code," WCAP-10081-A (Non-Proprietary).