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SOUTHERN COMPANY
*Energy to Serve Your World*℠

June 2, 2003

Docket Nos.: 50-321 50-348 50-424
50-366 50-364 50-425

NL-03-0999

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

Edwin I. Hatch Nuclear Plant
Joseph M. Farley Nuclear Plant
Vogtle Electric Generating Plant
10 CFR 50.46 ECCS Evaluation Model Annual Reports for 2002

Ladies and Gentlemen:

Pursuant to the reporting requirements of 10 CFR 50.46 (a)(3)(ii), Southern Nuclear Operating Company (SNC) is submitting the emergency core cooling system (ECCS) evaluation model annual reports for Hatch Nuclear Plant Units 1 and 2, Farley Nuclear Plant Units 1 and 2, and Vogtle Electric Generating Plant Units 1 and 2.

These annual reports summarize the nature of and estimated effect of any changes or errors in the ECCS models for the period from January 1, 2002 through December 31, 2002.

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

Sincerely,

J. B. Beasley, Jr.

JBB/WAS/sdl

Enclosures:

1. Edwin I. Hatch Nuclear Plant 10 CFR 50.46 ECCS Evaluation Model Annual Report for 2002
2. Joseph M. Farley Nuclear Plant 10 CFR 50.46 ECCS Evaluation Model Annual Report for 2002
3. Vogtle Electric Generating Plant 10 CFR 50.46 ECCS Evaluation Model Annual Report for 2002

A001

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**Enclosure 1**

**Edwin I. Hatch Nuclear Plant**
**10 CFR 50.46 ECCS Evaluation Model**
**Annual Report for 2002**

**Enclosure 1**
**Edwin I. Hatch Nuclear Plant**
**10 CFR 50.46 ECCS Evaluation Model**
**Annual Report for 2002**

## BACKGROUND

In accordance with 10 CFR 50.46(a)(3)(ii), this annual report summarizes the nature of and estimated effect of any changes or errors in the emergency core cooling system (ECCS) model for the period from January 1, 2002 through December 31, 2002 for Hatch Nuclear Plant Units 1 and 2.

## DISCUSSION

Updated limiting licensing basis peak clad temperatures (PCTs) applicable to Hatch are provided in the following table.

Since the last Hatch report in accordance with 10 CFR 50.46, SNC has supplemented the Hatch ECCS loss-of-coolant accident (LOCA) evaluation results to reflect the loading of 10x10 GE14 fuel in Hatch Units 1 and 2 as of Cycles 21 and 18, respectively. This supplemental ECCS-LOCA evaluation (Reference 1) was performed using the SAFER/GESTR-LOCA licensing methodology as approved by the NRC, consistent with the previous ECCS-LOCA evaluation (Reference 2). Since GE13 fuel remains in the cores as of Cycles 21 and 18, the updated licensing basis PCTs are provided for both GE13 and GE14 fuel. As such, the following table begins by listing the baseline ECCS-LOCA evaluations (References 1 and 2).

The next section of the table lists the applicable changes or errors and their estimated effect on PCT that have previously been reported to the NRC (References 3 and 4).

The final section of the table lists those applicable changes or errors and their estimated effect on PCT which SNC has been notified of by GE during the period from January 1, 2002 through December 31, 2002. There have been no SNC changes to the ECCS model to report, other than the loading of 10x10 GE14 fuel in Hatch Units 1 and 2 as of Cycles 21 and 18, respectively, which resulted in the supplemental ECCS-LOCA evaluation (Reference 1) discussed previously. Additional information on the five applicable changes or errors identified during 2002 follows.

### SAFER Core Spray Injection Elevation Error

In GE 10 CFR 50.46 Notification Letter 2002-01 (Reference 5), GE reported that an error was found in the automation code that prepares the input basedecks for the SAFER analysis. This error resulted in too low a value being calculated for the core spray injection elevation for the lower core spray sparger. This error affects the buildup and draining of the pool in the upper plenum and also affects the amount of steam quenched by the core spray water. GE estimated the effect on PCT for Hatch to be N/A for GE14 and +5 °F for GE13.

### SAFER Bulk Water Level Error

In GE 10 CFR 50.46 Notification Letter 2002-02 (Reference 5), GE reported an error in the initial vessel water level used in some SAFE/REFLOOD and SAFER LOCA analyses. The effect of

the steam dryer pressure drop on the initial inventory of water in the vessel was not properly accounted for. The analyses used the indicated water level, the water level in the annular region between the dryer skirt and the vessel wall, as the initial water level. This resulted in too high an initial liquid inventory because the lower level inside the dryer skirt, due to the steam dryer pressure drop, was not addressed. This additional water in the vessel assumed in the analysis may delay the core uncovery, which may result in a non-conservative calculation of PCT. GE estimated the effect on PCT for Hatch to be N/A for GE14 and +10 °F for GE13.

### GESTR Input File Interpolation Error

In GE 10 CFR 50.46 Notification Letter 2002-03 (Reference 6), GE reported an error in the interpolation coding that determines initial gap conductance from GESTR input files consisting of tables of gap conductance and related fuel parameters as functions of Linear Heat Generation Rate (LHGR) and exposure. This error in the interpolation coding resulted in the initial gap conductance used in the SAFER calculations, for cases at or beyond the knee in the LHGR curve, being slightly lower than it should have been. GE estimated the effect on PCT for Hatch to be N/A for GE14 and 0 °F for GE13.

### SAFER04 Computer Platform Change

In GE 10 CFR 50.46 Notification Letter 2002-04 (Reference 6), GE reported the migration of the LOCA evaluation code SAFER04 from the VAX computer platform to the Alpha computer platform. This change in computer platform may affect the calculated PCT as a result of changes in the processor word size and FORTRAN compiler characteristics. GE estimated the effect on PCT for Hatch to be N/A for GE14 and 0 °F for GE13.

### WEVOL S1 Volume Error

In GE 10 CFR 50.46 Notification Letter 2002-05 (Reference 6), GE reported an error in the WEVOL code used to calculate the weight and volume inputs for jet pump plant SAFER analyses. Specifically, the free volume in the region of the shroud head is calculated incorrectly in that the volume of the standpipes inside the shroud head thickness is not properly accounted for. This resulted in the value for the free volume in the downcomer region being too small. GE estimated the effect on PCT for Hatch to be 0 °F for all fuel types.

## CONCLUSION

As documented in the following table, the updated Hatch limiting licensing basis PCTs for GE13 and GE14 remain in compliance with 10 CFR 50.46(b)(1), specifically requiring that the limiting licensing basis PCT shall not exceed 2200 °F. As such, there is no need for reanalysis or taking any other actions in accordance with 10 CFR 50.46(a)(3)(ii) because compliance with 10 CFR 50.46(b)(1) has been maintained.

| Report Period | Description of Change or Error in ECCS Evaluation | Estimated PCT Change ( °F) | | | | Updated PCT | |
|---|---|---|---|---|---|---|---|
| | | GE13 Fuel | | GE14 Fuel | | GE13 Fuel | GE14 Fuel |
| | | PCT Change | Absolute Value | PCT Change | Absolute Value | | |
| Baseline Evaluations | SAFER/ GESTR-LOCA Analysis dated March, 1997 (Ref. 2) | N/A | N/A | N/A | N/A | 1688 | N/A |
| | SAFER/ GESTR-LOCA Analysis dated March, 2002 (Ref. 1) | N/A | N/A | N/A | N/A | N/A | 1820 |
| Previously Reported Changes or Errors | Hatch 50.46 Annual Report for 2000 (Ref. 3) | 10 | 20 | N/A | N/A | 1698 | N/A |
| | Hatch 50.46 30 Day Report dated 5/21/01 (Ref. 4) | 100 | 100 | N/A | N/A | 1798 | N/A |
| 2002 Changes or Errors | GE 50.46 Notification Letter 2002-01 dated 6/13/02 (Ref. 5) | 5 | 5 | N/A | N/A | 1803 | N/A |
| | GE 50.46 Notification Letter 2002-02 dated 6/13/02 (Ref. 5) | 10 | 10 | N/A | N/A | 1813 | N/A |
| | GE 50.46 Notification Letter 2002-03 dated 8/26/02 (Ref. 6) | 0 | 0 | N/A | N/A | 1813 | N/A |
| | GE 50.46 Notification Letter 2002-04 dated 8/26/02 (Ref. 6) | 0 | 0 | N/A | N/A | 1813 | N/A |

| Report Period | Description of Change or Error in ECCS Evaluation | Estimated PCT Change ( °F) | | | | Updated PCT | |
|---|---|---|---|---|---|---|---|
| | | GE13 Fuel | | GE14 Fuel | | GE13 Fuel | GE14 Fuel |
| | | PCT Change | Absolute Value | PCT Change | Absolute Value | | |
| **2002 Changes or Errors Continued** | GE 50.46 Notification Letter 2002-05 dated 8/26/02 (Ref. 6) | 0 | 0 | 0 | 0 | 1813 | 1820 |

## REFERENCES

1. GE-NE-0000-0000-9200-02P, "Hatch Units 1 and 2 ECCS-LOCA Evaluation for GE14," dated March 2002.
2. NEDC-32720P, "Hatch Units 1 and 2 SAFER/GESTR Loss-of-Coolant Accident Analysis," dated March 1997.
3. SNC Letter HL-6028, H. L. Sumner, Jr. to NRC, "Reporting of Changes and Errors in ECCS Evaluation Models," dated January 31, 2001.
4. SNC Letter HL-6090, H. L. Sumner, Jr. to NRC, "Reporting of Changes and Errors in ECCS Evaluation Models," dated May 21, 2001.
5. E-mail from Glen A. Watford (Global Nuclear Fuel (GNF)) to Ken S. Folk (SNC), "10 CFR 50.46 Notification – 2002-01 – SAFER Core Spray Injection Elevation/2002-02 – SAFER Bulk Water Level Error – Southern," dated June 13, 2002.
6. E-mail from Glen A. Watford (GNF) to Ken S. Folk (SNC), "10 CFR 50.46 Notification – 2002-03, 2002-04, 2002-05," dated August 26, 2002.

**Enclosure 2**

**Joseph M. Farley Nuclear Plant**
**10 CFR 50.46 ECCS Evaluation Model**
**Annual Report for 2002**

**Enclosure 2**
**Joseph M. Farley Nuclear Plant**
**10 CFR 50.46 ECCS Evaluation Model**
**Annual Report for 2002**

## BACKGROUND

In accordance with 10 CFR 50.46(a)(3)(ii), this annual report summarizes the nature of and estimated effect of any changes or errors in the emergency core cooling system (ECCS) model for the period from January 1, 2002 through December 31, 2002 for Farley Nuclear Plant Units 1 and 2.

## DISCUSSION

In Reference 1, information was submitted to the NRC regarding modifications to the Westinghouse large-break and small-break Loss-of-Coolant Accident (LOCA) ECCS Evaluation Models as applicable to the Farley analyses for the calendar year 2001.

The following presents an assessment of the effects of modifications to the Westinghouse ECCS Evaluation Models on the Farley LOCA analysis results since the 2001 annual report (Reference 1) for the calendar year 2002. This annual report has been prepared in accordance with the Westinghouse Methodology for Implementation of 10 CFR 50.46 Reporting (WCAP-13451, Reference 2), with the exception of plant changes, as discussed in last year's annual report. Starting in 2001, a change in the Westinghouse reporting methodology was made to include the 50.59 Plant Change PCT values as a part of the 50 °F error reporting section. The 2002 annual report (contained herein) is consistent with the change implemented in the 2001 annual report.

Unit 2 implemented the Reactor Internals Upflow Conversion Program (Reference 3) in 2002, and as such a new PCT rack-up reflecting the new upflow configuration analysis is presented here for Unit 2.

### Large-Break LOCA

Table 1A shows the LBLOCA PCT rack-ups for both Unit 1 and Unit 2 for Reflood 1 (Reference 4). Table 1B shows the corresponding large-break LOCA PCT rack-ups for Reflood 2 (Reference 4).

#### LBLOCA ECCS MODEL ANALYSIS-OF-RECORD

The large-break LOCA analyses for Farley Units 1 and 2 were examined to assess the effects of the changes and errors in the Westinghouse large-break LOCA ECCS Evaluation Model on PCT results.

The large-break LOCA analysis-of-record results for Farley Units 1 and 2 were calculated using Westinghouse's BE-LOCA analysis (References 1 and 4).

The Unit 1 and Unit 2 analyses assumed the following information important to the large-break LOCA in the BE-LOCA analysis (References 1 and 4). One analysis was used to bound both Farley Unit 1 and Unit 2.

Core Power = 2775 MWT

17x17 VANTAGE+ Fuel Assembly

$F_Q$ = 2.50 for VANTAGE+ Fuel

$F\Delta H$ = 1.70 for VANTAGE+ Fuel

SGTP = 20%

For Farley Units 1 and 2, the limiting size break analysis-of-record is a split break of the cold leg piping with a discharge coefficient of $C_D = 1.0$. The limiting PCT values determined for the Unit 1 and Unit 2 large break LOCAs are shown in Table 1A (Reflood 1).

## PRIOR LBLOCA ECCS MODEL ASSESSMENTS

### Prior 10 CFR 50.46 Assessments Reported as Significant

There are no LBLOCA 10 CFR 50.46 prior assessments reported as significant.

### Prior 10 CFR 50.59 Assessments

The following two plant change assessments were reported in the last submittal (Reference 1) and occurred prior to 2001.

The addition of permanent storage boxes in containment was evaluated and found not to cause a change to PCT (Reference 6).

The finalization of Replacement Steam Generator Data was evaluated and found not to cause a change to PCT (Reference 1).

## CURRENT LBLOCA ECCS MODEL ASSESSMENTS

The following changes and errors in the Westinghouse ECCS Evaluation Model would affect the BE-LOCA Model.

### Prior 10 CFR 50.46 Reported Assessments

The following two assessments were reported in the last PCT submittal (Reference 1).

Accumulator line/Pressurizer Surge Line Data

It was determined that the design and actual plant accumulator line piping schedule were not the same. A Farley specific BE-LBLOCA sensitivity analysis resulted in a 41 °F benefit for the first reflood and a 9 °F benefit for the second reflood when actual plant data was modeled (Reference 7). This assessment is applicable to Unit 1 and Unit 2.

Decay Heat Uncertainty error in Monte Carlo Calculation

It was determined that an error existed in the calculation of decay heat uncertainty in the Monte Carlo calculation of the 95$^{th}$ percentile PCT for BE-LBLOCA (Reference 9). This caused an 8 °F penalty for Unit 1 and 2 on Reflood 1 only.

### 2002 10 CFR 50.46 PCT Assessments

None.

## CURRENT PLANNED PLANT CHANGE EVALUATIONS

Starting with the 2001 annual report (Reference 1), the 10 CFR 50.59 Plant Change PCT values have been considered to be a part of the 50 °F error reporting section. The 2002 annual report (contained herein) is consistent with the changes implemented in the 2001 annual report.

### Prior 10 CFR 50.59 Model Assessments

None.

### 2002 Planned Plant Changes

As noted earlier, the Reactor Internals Upflow Conversion was implemented in Unit 2 in 2002 (Reference 3). The original large-break LOCA analysis results were found to be bounding and conservative for this plant change.

TOTAL RESULTANT LBLOCA PCT

As discussed above, the changes and errors to the Westinghouse large-break LOCA ECCS Evaluation Model could affect the large-break LOCA analysis results by altering the PCT. As shown in Table 1A and Table 1B, the large-break LOCA analysis PCT results for both units are below the 10 CFR 50.46 limit of 2200 °F.

## Small-Break LOCA

Table 2 shows the small-break LOCA PCT rack-ups for both Unit 1 and Unit 2.

### SBLOCA ECCS MODEL ANALYSIS-OF-RECORD

The small-break LOCA analyses for Farley Units 1 and 2 were also examined to assess the effects of the changes and errors to the Westinghouse small-break LOCA ECCS Evaluation Models on PCT results. The small-break LOCA ECCS analysis results were calculated using the NOTRUMP small-break LOCA ECCS Evaluation Model (Reference 5). As noted earlier, the Unit 2 re-analysis reflects the Reactor Internals Upflow Conversion implemented in 2002 (Reference 3).

The Unit 1 and Unit 2 analyses assumed the following information important to the small-break LOCA analyses:

| Unit 1 | Unit 2 |
|---|---|
| Core Power = 1.02 X 2775 MWT | Core Power = 1.02 x 2775 MWT |
| 17x17 VANTAGE+ Fuel Assembly | 17x17 VANTAGE+ Fuel Assembly |
| $F_Q = 2.50$ | $F_Q = 2.50$ |
| $F\Delta H = 1.70$ | $F\Delta H = 1.70$ |
| Upflow Configuration | Upflow Configuration |

For Farley Units 1 and 2, the limiting size break analysis-of-record for the VANTAGE+ fuel analysis is a 3-inch diameter break in the cold leg. The limiting PCT values determined for the Unit 1 and Unit 2 17x17 VANTAGE+ small-break are shown in Table 2.

### PRIOR SLBLOCA ECCS MODEL ASSESSMENTS

#### Prior 10 CFR 50.46 Assessments Reported as Significant

The following SBLOCA 10 CFR 50.46 assessment was reported in March 2000 as significant.

An overall PCT benefit of 62 °F for Unit 1 for the "Burst and Blockage/Time in Life" penalty resulted from the SPIKE computer code correlation revision. (Reference 11)

Prior 10 CFR 50.59 Assessments

The following three plant change assessments were reported in the last submittal (Reference 1) and occurred prior to 2001.

The addition of permanent storage boxes in containment was evaluated and found not to cause a change to PCT (Reference 6).

The finalization of Replacement Steam Generator Data resulted in a 62 °F benefit for Unit 1 (Reference 10).

Annular pellets were determined to have a 10 °F penalty for SBLOCA results for Unit 1 (Reference 8).

Note that the Unit 2 result (in Table 2) is unaffected by these prior 50.59 plant changes. The reason is that the Unit 2 Upflow Conversion implemented in 2002 required a small-break LOCA re-analysis that included the above changes explicitly.

CURRENT SBLOCA ECCS MODEL ASSESSMENTS

The following changes and errors were identified:

Prior 10 CFR 50.46 Reported Assessments

The following assessments were reported in the last PCT submittal (Reference 1).

NOTRUMP Mixture Level Tracking/Region Depletion Errors

Several closely related errors have been discovered in how NOTRUMP deals with the stack mixture level transition across a node boundary in a stack of fluid nodes. As previously reported, the impact of this revision on the SBLOCA results has been determined to be a 13 °F penalty for Unit 1. In addition, the associated change in Burst and Blockage/Time in Life Components was an additional 12 °F for Unit 1. Thus, the total change was 25 °F for Unit 1. This error does not impact Unit 2's re-analysis result (see previously discussed Reactor Internals Upflow Conversion), since the re-analysis was performed with the corrected version of NOTRUMP.

2002 10 CFR 50.46 PCT Assessments

None.

CURRENT PLANNED PLANT CHANGE EVALUATIONS

Starting with the 2001 annual report (Reference 1), the 10 CFR 50.59 Plant Change PCT values have been considered to be a part of the 50 °F error reporting section. The 2002 annual report (contained herein) is consistent with the change implemented in the 2001 annual report.

Prior 10 CFR 50.59 Model Assessments

None.

2002 Planned Plant Changes

As noted earlier, the Reactor Internals Upflow Conversion was implemented in Unit 2 in 2002 (Reference 3). The new Unit 2 small-break LOCA analysis result (re-analysis) is reflected in Table 2.

TOTAL RESULTANT SBLOCA PCT

As discussed above, the changes and errors in the Westinghouse small-break LOCA ECCS Evaluation Model could affect the small-break LOCA analysis results by altering the PCT. As shown in Table 2, the small-break LOCA analysis PCT results for both units are below the 10 CFR 50.46 limit of 2200 °F.

## CONCLUSION

As documented in the following tables, the updated Farley large-break and small-break LOCA analyses PCTs remain in compliance with 10 CFR 50.46(b)(1), specifically requiring that the PCT shall not exceed 2200 °F. As such, there is no need for reanalysis or taking any other actions in accordance with 10 CFR 50.46(a)(3)(ii) because compliance with 10 CFR 50.46(b)(1) has been maintained.

## REFERENCES

1. Letter from D. N. Morey to USNRC (NEL-02-0067), "Joseph M. Farley Nuclear Plant 10 CFR 50.46 annual ECCS Evaluation Model Changes Report for 2001," March 25, 2002.

2. WCAP-13451, "Westinghouse Methodology for Implementation of 10 CFR 50.46 Reporting," October 1992.

3. ALA-02-039, "Transmittal of Reactor Internals Upflow Conversion Program Engineering Report, J. M. Farley Nuclear Plant Unit 2," June 2002 (also see WCAP-15974, November 2002).

4. ALA-03-016, "10 CFR 50.46 Annual Notification and Reporting for 2002," March 13, 2003.

5. "Westinghouse Small-break ECCS Evaluation Model Using the NOTRUMP Code," WCAP-10054-P-A (Proprietary), WCAP-10081-A (Non-Proprietary), Lee, N., et. al, August 1985.

6. SECL-97-062. Rev. 1, "Effects on LOCA PCT of Adding Permanent Storage Boxes and Lead Blankets Inside Containment," October 17, 1997.

7. ALA-00-037, "Final 10 CFR 50.46 Annual Notification and Reporting," March 8, 2000.

8. WCAP-15098, "Joseph M. Farley Nuclear Plant Units 1 and 2 RSG Program NSSS Licensing Report," November 1998.

9. ALA-01-008, "10 CFR 50.46 Annual Notification and Reporting for 2000," March 6, 2001.

10. ALA-01-01, "Southern Nuclear Operating Company, Joseph M. Farley Nuclear Plant Units 1 and 2, LBLOCA and SBLOCA Impacts Due to Final RSG Data for SGRP," February 11, 2000.

11. Letter from D. N. Morey to USNRC (NEL-00-0080), "Joseph M. Farley Nuclear Plant 10 CFR 50.46 Annual ECCS Evaluation Model Changes Report for 1999 and Significant Error Reports," March 29, 2000.

**TABLE 1A (Limiting for Unit 1 and Unit 2)**
**JOSEPH M. FARLEY NUCLEAR PLANT**
**TOTAL RESULTANT LARGE-BREAK LOCA PCT (°F) FOR REFLOOD 1**

| | UNIT 1 | UNIT 2 |
|---|---|---|
| **A. LBLOCA ECCS MODEL ANALYSIS-OF-RECORD** | | |
| 1. ECCS Analysis | 2056* | 2056* |
| 2. Increased Containment Spray Flow | 9* | 9* |
| Total Analysis-of-Record | 2065* | 2065* |
| **B. PRIOR LBLOCA ECCS MODEL ASSESSMENTS** | | |
| 1. Prior 10 CFR 50.46 Assessments Reported as Significant | 0 | 0 |
| 2. Prior 10 CFR 50.59 Assessments | | |
| A. Addition of Permanent Storage Boxes in Containment | 0 | 0 |
| B. Finalization of Replacement Steam Generator Data | 0 | 0 |
| Sum of Prior Assessments | 0 | 0 |
| **C. CURRENT LBLOCA ECCS MODEL ASSESSMENTS** | | |
| 1. Accumulator Line/Pressurizer Surge Line Data | -41* | -41* |
| 2. MONTECF Decay Heat Uncertainty Error | 8* | 8* |
| **D. CURRENT PLANNED PLANT CHANGE EVALUATIONS** | | |
| 1. None | 0 | 0 |
| **E. TOTAL RESULTANT LBLOCA PCT** | | |
| **Total** | **2032*** | **2032*** |

The PCT values are rounded up to the next highest integer number to avoid reporting in decimal points. The Analysis of Record PCT results reflect the Replacement Steam Generators analysis values.

* See References 1 and 4

## TABLE 1B
## JOSEPH M. FARLEY NUCLEAR PLANT
## TOTAL RESULTANT LARGE-BREAK LOCA PCT (°F) FOR REFLOOD 2

| | UNIT 1 | UNIT 2 |
|---|---|---|
| **A. LBLOCA ECCS MODEL ANALYSIS-OF-RECORD** | | |
| 1. ECCS Analysis | 1956* | 1956* |
| 2. Increased Containment Spray Flow | 1* | 1* |
| Total Analysis-of-Record | 1957* | 1957* |
| **B. PRIOR LBLOCA ECCS MODEL ASSESSMENTS** | | |
| 1. Prior 10 CFR 50.46 Assessments Reported as Significant | 0 | 0 |
| 2. Prior 10 CFR 50.59 Assessments | | |
| A. Addition of Permanent Storage Boxes in Containment | 0 | 0 |
| B. Finalization of Replacement Steam Generator Data | 0 | 0 |
| Sum of Prior Assessments | 0 | 0 |
| **C. CURRENT LBLOCA ECCS MODEL ASSESSMENTS** | | |
| 1. Accumulator Line/Pressurizer Surge Line Data | -9* | -9* |
| 2. MONTECF Decay Heat Uncertainty Error | 0* | 0* |
| **D. CURRENT PLANNED PLANT CHANGE EVALUATIONS** | | |
| 1. None | 0 | 0 |
| **E. TOTAL RESULTANT LBLOCA PCT** | | |
| **Total** | **1948*** | **1948*** |

The PCT values are rounded up to the next highest integer number to avoid reporting in decimal points. The Analysis of Record PCT results reflect the Replacement Steam Generators analysis values.

* See References 1 and 4

## TABLE 2
## JOSEPH M. FARLEY NUCLEAR PLANT
## TOTAL RESULTANT SMALL-BREAK LOCA PCT (°F)

| | UNIT 1 | UNIT 2 |
|---|---|---|
| **A. SBLOCA ECCS MODEL ANALYSIS-OF-RECORD** | | |
| 1. ECCS Analysis | 1883* | 1868** |
| 2. Burst and Blockage / Time in Life | 137* | 120** |
| Total Analysis-of-Record | 2020* | 1988* |
| **B. PRIOR SBLOCA ECCS MODEL ASSESSMENTS** | | |
| 1. Prior 10 CFR 50.46 Assessments Reported as Significant | -62* | 0 |
| 2. Prior 10 CFR 50.59 Assessments | | |
| A. Addition of Permanent Storage Boxes in Containment | 0* | 0 |
| B. Finalization of Replacement Steam Generator Data | -62# | 0 |
| C. Annular Pellet Blanket | 10* | 0 |
| Sum of Prior Assessments | -114* | 0 |
| **C. CURRENT SBLOCA ECCS MODEL ASSESSMENTS** | | |
| 1. NOTRUMP Mixture Level Tracking / Region Depl Errors | 13* | ** |
| 2. Associated change in Burst and Blockage | 12* | ** |
| **D. CURRENT PLANNED PLANT CHANGE EVALUATIONS** | | |
| 1. None | 0 | 0 |
| **E. TOTAL RESULTANT SBLOCA PCT** | | |
| **Total** | **1931*** | **1988**** |

The PCT values are rounded up to the next highest integer number to avoid reporting in decimal points.

* See References 1 and 4

** The revised analysis-of-record reflects the Unit 2's conversion of downflow to upflow configuration (see References 1 and 3).

\# See Reference 10

**Enclosure 3**

**Vogtle Electric Generating Plant**
**10 CFR 50.46 ECCS Evaluation Model**
**Annual Report for 2002**

**Enclosure 3**
**Vogtle Electric Generating Plant**
**10 CFR 50.46 ECCS Evaluation Model**
**Annual Report for 2002**

## BACKGROUND

In accordance with 10 CFR 50.46(a)(3)(ii), this annual report summarizes the nature of and estimated effect of any changes or errors in the emergency core cooling system (ECCS) model for the period from January 1, 2002 through December 31, 2002 for Vogtle Electric Generating Plant Units 1 and 2.

## DISCUSSION

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 Significant Change Report submitted on July 1, 2002 (Reference 10). This report has been prepared in accordance with the methodology presented in WCAP-13451 (Reference 1) and additional guidance provided by Westinghouse (Reference 2). The LBLOCA and SBLOCA analyses, Evaluation Model assessments, and planned plant change evaluation results reported herein will be included in a future VEGP Final Safety Analysis Report (FSAR) update.

### Large-Break LOCA

LBLOCA ECCS MODEL ANALYSIS-OF-RECORD

In the Significant Change Report submitted on July 1, 2002 (Reference 10), SNC reported a LBLOCA PCT of 2040.5 °F for both Unit 1 and Unit 2. This value is based on fuel designs containing 128 Integral Fuel Burnable Absorber (IFBA) rods.

The LBLOCA analysis was performed with the 1981 Version of the Westinghouse ECCS Evaluation Model using BASH (Reference 3) including changes in the methodology for execution of the model described in References 4 and 5, and the latest acceptable LOCBART model. The limiting size break analysis assumes the following information important to the LBLOCA analyses:

- 17x17 VANTAGE+ Fuel Assembly
- Core Power = 1.02 * 3565 MWt
- Vessel Average Temperature = 570.7 °F
- Steam Generator Plugging Level = 10%
- $F_Q = 2.50$
- $F\Delta H = 1.65$

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 2040.5 °F for both Unit 1 and Unit 2.

PRIOR LBLOCA ECCS MODEL ASSESSMENTS

There are no LBLOCA prior assessments.

CURRENT LBLOCA ECCS MODEL ASSESSMENTS

There are no LBLOCA current assessments.

CURRENT PLANNED PLANT CHANGE EVALUATIONS

There are no current planned plant changes that affect PCT for Unit 1 and Unit 2.

TOTAL RESULTANT LBLOCA PCT

For Unit 1, the absolute sum of the LBLOCA PCT assessments is 0 °F.

For Unit 2, the absolute sum of the LBLOCA PCT assessments is 0 °F.

## UNIT 1 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.** | **LBLOCA ECCS MODEL ANALYSIS-OF-RECORD** | |
| | LOCBART Analysis Result (128 IFBA) | 2040.5 °F |
| **B.** | **PRIOR LBLOCA ECCS MODEL ASSESSMENTS** | |
| | Combined assessments previously reported as significant | +0 °F |
| | Combined planned plant change evaluations | +0 °F |
| **C.** | **CURRENT LBLOCA ECCS MODEL ASSESSMENTS** | |
| | None | +0 °F |
| **D.** | **CURRENT PLANNED PLANT CHANGE EVALUATIONS** | |
| | None | +0 °F |
| **E.** | **TOTAL RESULTANT LBLOCA PCT** | |
| | **Total** | **2040.5 °F** |

## UNIT 2 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.** | **LBLOCA ECCS MODEL ANALYSIS-OF-RECORD** | |
| | LOCBART Analysis Result (128 IFBA) | 2040.5 °F |
| **B.** | **PRIOR LBLOCA ECCS MODEL ASSESSMENTS** | |
| | Combined assessments previously reported as significant | +0 °F |
| | Combined planned plant change evaluations | +0 °F |
| **C.** | **CURRENT LBLOCA ECCS MODEL ASSESSMENTS** | |
| | None | +0 °F |
| **D.** | **CURRENT PLANNED PLANT CHANGE EVALUATIONS** | |
| | None | +0 °F |
| **E.** | **TOTAL RESULTANT LBLOCA PCT** | |
| | **Total** | **2040.5 °F** |

**Small-Break LOCA**

SBLOCA ECCS MODEL ANALYSIS-OF-RECORD

In the Significant Change Report submitted on July 1, 2002 (Reference 10), SNC reported a SBLOCA PCT of 1138.0 °F for both Unit 1 and Unit 2.

The SBLOCA analysis was performed with the Westinghouse ECCS Evaluation Model using NOTRUMP (References 6 and 7), including changes to the methodology described in References 8 and 9, and the latest acceptable SBLOCTA model. The limiting size break analysis assumes the following information important to the SBLOCA analyses:

- 17x17 VANTAGE+ Fuel Assembly
- Core Power = 1.02 * 3565 MWt
- Vessel Average Temperature = 570.7 °F
- Steam Generator Plugging Level = 10%
- $F_Q$ = 2.58
- $F\Delta H$ = 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 SBLOCTA analysis-of-record calculated PCT value is 1138.0 °F for both Unit 1 and Unit 2.

PRIOR SBLOCA ECCS MODEL ASSESSMENTS

There are no SBLOCA prior assessments.

CURRENT SBLOCA ECCS MODELASSESSMENTS

There are no SBLOCA current assessments.

CURRENT PLANNED PLANT CHANGE EVALUATIONS

There are no current planned plant changes that affect PCT for Unit 1 and Unit 2.

TOTAL RESULTANT SBLOCA PCT

For Unit 1, the absolute sum of the SBLOCA PCT assessments is 0 °F.

For Unit 2, the absolute sum of the SBLOCA PCT assessments is 0 °F.

## UNIT 1 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 ECCS MODEL ANALYSIS-OF-RECORD

| | |
|---|---|
| SBLOCTA Analysis Result | 1138.0 °F |

### B. PRIOR SBLOCA ECCS MODEL ASSESSMENTS

| | |
|---|---|
| Combined assessments previously reported as significant | + 0 °F |
| Combined planned plant change evaluations | + 0 °F |

### C. CURRENT SBLOCA ECCS MODEL ASSESSMENTS

| | |
|---|---|
| None | + 0 °F |

### D. CURRENT PLANNED PLANT CHANGE EVALUATIONS

| | |
|---|---|
| None | + 0 °F |

### E. TOTAL RESULTANT SBLOCA PCT

| | |
|---|---|
| **Total** | **1138.0 °F** |

## 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 ECCS MODEL ANALYSIS-OF-RECORD** | |
| SBLOCTA Analysis Result | 1138.0 °F |
| **B. PRIOR SBLOCA ECCS MODEL ASSESSMENTS** | |
| Combined assessments previously reported as significant | + 0 °F |
| Combined plant change evaluations | + 0 °F |
| **C. CURRENT SBLOCA ECCS MODEL ASSESSMENTS** | |
| None | + 0 °F |
| **D. CURRENT PLANNED PLANT CHANGE EVALUATIONS** | |
| None | + 0 °F |
| **E. TOTAL RESULTANT SBLOCA PCT** | |
| **Total** | **1138.0 °F** |

## CONCLUSION

As documented in the preceding tables, the updated VEGP large-break and small-break LOCA analyses PCTs remain in compliance with 10 CFR 50.46(b)(1), specifically requiring that the PCT shall not exceed 2200 °F. As such, there is no need for reanalysis or taking any other actions in accordance with 10 CFR 50.46(a)(3)(ii) because compliance with 10 CFR 50.46(b)(1) has been maintained.

## REFERENCES

1. WCAP-13451, "Westinghouse Methodology for Implementation of 10 CFR 50.46 Reporting," October 1992.

2. Westinghouse letter GP-17337, "Southern Nuclear Operating Company, Inc., Vogtle Electric Generating Plant Units 1 and 2, 10 CFR 50.46 Annual Notification Reporting for 2001," March 1, 2002.

3. "The 1981 Version of the Westinghouse ECCS Evaluation Model Using the BASH Code," WCAP-10266-P-A, Revision 2 (Proprietary) and WCAP-11524-A, Revision 2 (Non-Proprietary), March 1987.

4. Westinghouse letter NTD-NRC-94-4143 from N. J. Liparulo to W. T. Russell (USNRC), "Change in Methodology for Execution of BASH Evaluation Model," May 23, 1994.

5. Westinghouse letter NTD-NRC-95-4540 from N. J. Liparulo to W. T. Russell (USNRC), "Change in Methodology for Execution of BASH Evaluation Model," August 29, 1995.

6. "NOTRUMP: A Nodal Transient Small Break and General Network Code," WCAP-10079-P-A (Proprietary) and WCAP-10080-A (Non-Proprietary), August 1985.

7. "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code," WCAP-10054-P-A, August 1985.

8. "Addendum to the Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code: Safety Injection into the Broken Loop and COSI Condensation Model," WCAP-10054-P-A, Addendum 2, Revision 1, July 1997.

9. "Model Changes to the Westinghouse Appendix K Small Break LOCA NOTRUMP Evaluation Model: 1988 – 1997," WCAP-15085, July 1998.

10. LCV-1629, "Vogtle Electric Generating Plant, 10 CFR 50.46 ECCS Evaluation Models Significant Change Report," letter from J. B. Beasley, Jr. (SNC) to USNRC, July 1, 2002.