L. M. Stinson (Mike)
Vice President

Southern Nuclear
Operating Company, Inc.
40 Inverness Center Parkway
Post Office Box 1295
Birmingham, Alabama 35201

Tel 205.992.5181
Fax 205.992.0341

SOUTHERN COMPANY
Energy to Serve Your World℠

September 7, 2005

Docket Nos.: 50-348
50-364

NL-05-1551

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

Joseph M. Farley Nuclear Plant
10 CFR 50.46 Annual ECCS Evaluation Model
Report for 2004 and Significant Error Report

Ladies and Gentlemen:

Pursuant to the reporting requirements of 10 CFR 50.46 (a)(3)(i), Southern Nuclear Operating Company (SNC) is submitting the emergency core cooling system (ECCS) evaluation model report for Farley Nuclear Plant Units 1 and 2 for the calendar year 2004.

The annual report provides information regarding the effects of the ECCS Evaluation Model modifications on the peak cladding temperature (PCT) results for the period from January 1, 2004 through December 31, 2004. Also, the attached annual report provides a summary of the plant changes performed under the provisions of 10 CFR 50.59 that also affect the PCT results. The report is in accordance with the Westinghouse Methodology for Implementation of 10 CFR 50.46 Reporting (WCAP-13451).

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 are combined with the effects of the ECCS Evaluation Model changes and errors applicable to Farley Units 1 and 2.

This annual report is also serving as a 30 day Significant Error Report for large-break LOCA PCT. The error is significant based on cumulative corrections such that the sum of the absolute magnitudes of the respective temperature changes is greater than 50 °F. SNC incorporated model changes which met this criterion into the FNP licensing basis on August 11, 2005. SNC will incorporate the information in the attached PCT rack-up sheets into the FNP licensing basis subsequent to this submittal. As shown in Table 1A and Table 1B, the large-break LOCA analysis PCT results for both units remain below the 10 CFR 50.46 limit of 2200 °F and therefore, no reanalysis is required. However, as a separate initiative, SNC has performed reanalysis of the large-break LOCA PCT using the ASTRUM methodology. SNC is in the process of preparing a package for submittal to the NRC in the near future.

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

Sincerely,

L. M. Stinson

L. M. Stinson

LMS/was/sdl

Enclosure: Joseph M. Farley Nuclear Plant 10 CFR 50.46 Annual ECCS Evaluation Model Report for 2004 and Significant Error Report

cc: Southern Nuclear Operating Company
Mr. J. T. Gasser, Executive Vice President
Mr. J. R. Johnson, General Manager – Plant Farley
RTYPE: CFA04.054; LC# 14328

U. S. Nuclear Regulatory Commission
Dr. W. D. Travers, Regional Administrator
Mr. R. E. Martin, NRR Project Manager – Farley
Mr. C. A. Patterson, Senior Resident Inspector – Farley

Enclosure

Joseph M. Farley Nuclear Plant
10 CFR 50.46 Annual ECCS Evaluation Model
Report for 2004 and Significant Error Report

Enclosure
Joseph M. Farley Nuclear Plant
10 CFR 50.46 Annual ECCS Evaluation Model
Report for 2004 and Significant Error Report

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, 2004 through December 31, 2004 for Farley Nuclear Plant (FNP) Units 1 and 2.

In addition, changes have been made and adopted into the FNP licensing basis in 2005 which meet the criteria for a Significant Error Report.

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 2003.

The following presents an assessment of the effects of modifications to the Westinghouse ECCS Evaluation Models on the Farley LOCA analysis results since the 2003 annual report (Reference 1) for the calendar year 2004. 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. 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 2004 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.

Enclosure
10 CFR 50.46 Annual ECCS Evaluation Model Significant Assessment Report for 2004 and Significant Error Report

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

$F\Delta H$ = 1.70

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.

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 95th percentile PCT for BE-LBLOCA (Reference 9). This caused an 8 °F penalty for Unit 1 and 2 on Reflood 1 only.

2004 10 CFR 50.46 PCT Assessments

The following three assessments are being reported in this PCT submittal.

Revised Blowdown Heatup Uncertainty Distribution

Correction of modeling inconsistencies and input errors in the LOFT input decks have resulted in a change in the predicted peak cladding temperature transients. The overall code uncertainty for blowdown was recalculated and programmed into a new version of MONTECF. This resulted in a 5 °F penalty for Unit 1 and 2 for both the first and second refloods.

PAD 4.0 Fuel Data

PAD 4.0 fuel data was used in evaluation of RHR pump surveillance testing. Use of the PAD 4.0 fuel data reduces the initial stored energy, therefore resulting in a PCT benefit during reflood. The PCT benefit of PAD 4.0 fuel data was determined to be 50 °F for Reflood 1 and 65 °F for reflood 2 for Unit 1 and 2.

RHR Test Configuration SI Flow Reduction

During surveillance testing of the RHR pumps, there would be a reduction in calculated SI flow should a LOCA occur while in the testing alignment. The PCT effects were determined to be negligible for reflood 1 and a 100 °F penalty for reflood 2.

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 2004 annual report (contained herein) is consistent with the changes implemented in the 2001 annual report.

Prior 10 CFR 50.59 Model Assessments

None.

2004 Planned Plant Changes

None.

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.

2004 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 2004 annual report (contained herein) is consistent with the change implemented in the 2001 annual report.

Prior 10 CFR 50.59 Model Assessments

None.

2004 Planned Plant Changes

None.

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 L. M. Stinson to USNRC (NL-04-1042), "Edwin I Hatch Nuclear Plant, Joseph M. Farley Nuclear Plant, Vogtle Electric Generating Plant 10 CFR 50.46 ECCS Evaluation Model Annual Reports for 2003," June 29, 2004.

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-05-23, "Southern Nuclear Operating Company, J. M. Farley Units 1 and 2, 10 CFR 50.46 Annual Notification and Reporting for 2004," April 14, 2005

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.

12. ALA-05-55, "Southern Nuclear Operating Company, Joseph M. Farley Nuclear Plant Units 1 and 2, Transmittal of Quarterly RHR Pump Testing Evaluation Revision 1," July 11, 2005

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*
3. Revised Blowdown Heatup Uncertainty Distribution	5#	5#
4. PAD 4.0 Fuel Data	-50**	-50**
5. RHR Test Configuration SI Flow Reduction (note 1)	0**	0**
D. CURRENT PLANNED PLANT CHANGE EVALUATIONS		
1. None	0	0
E. TOTAL RESULTANT LBLOCA PCT		
Total	**1987#**	**1987#**

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
# See Reference 4
** See Reference 12

Note 1 – Assessment applies during quarterly RHR Pump Testing Configuration only.

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*
3. Revised Blowdown Heatup Uncertainty Distribution	5#	5#
4. PAD 4.0 Fuel Data	-65**	-65**
5. RHR Test Configuration SI Flow Reduction (note 1)	100**	100**
D. CURRENT PLANNED PLANT CHANGE EVALUATIONS		
1. None	0	0
E. TOTAL RESULTANT LBLOCA PCT		
Total	**1988#**	**1988#**

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
# See Reference 4
** See Reference 12

Note 1 – Assessment applies during quarterly RHR Pump Testing Configuration only.

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