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SOUTHERN COMPANY
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U. S. Nuclear Regulatory Commission
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Joseph M. Farley Nuclear Plant
10 CFR 50.46 Annual ECCS Evaluation Model Changes Report for
1999 and Significant Error Reports

Ladies and Gentlemen:

Provisions in 10 CFR 50.46 require applicants and holders of operating licenses or construction permits to annually notify the Nuclear Regulatory Commission (NRC) of changes and errors in the emergency core cooling system (ECCS) evaluation models. In compliance with this requirement, enclosed is the Southern Nuclear Operating Company's report for Joseph M. Farley Nuclear Plant Units 1 and 2 for the calendar year 1999.

The annual report (Attachment 1) provides information regarding the effects of the ECCS evaluation model modifications on the peak cladding temperature (PCT) results since the 1998 annual report. 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 report also serves as a 30 day significant error report for the Unit 2 small break loss of coolant accident (SBLOCA). This is due to an error in the Burst and Blockage/Time in Life (SPIKE Correlation Revision), which resulted in a total PCT impact greater than 50°F, as reported by Westinghouse. The SBLOCA has been re-analyzed for replacement steam generators (RSG). The NRC has approved the re-analysis (Reference 3 of Attachment 1).

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Also enclosed (Attachment 2) is a 30 day significant error report for SBLOCA with regards to RSG. RSG is planned to be implemented for Farley 1 Cycle 17 that is scheduled for startup May 2000. The 30 day report is being issued due to an error in the Burst and Blockage/Time in Life (SPIKE correlation revision), which resulted in a total PCT impact greater than 50°F, as reported by Westinghouse.

Southern Nuclear will incorporate model changes at the next licensing action requiring reanalysis of the SBLOCA and for which NOTRUMP is used as the evaluation model. This item will be tracked as an NRC commitment. Southern Nuclear proposes not to reanalyze the SBLOCA analysis at this time because:

1) The current magnitude of the PCT for the SBLOCA event maintains considerable margin to the 2200°F acceptance criteria.
2) The NRC has approved the overall conservatism in the SBLOCA evaluation model.
3) The assessments performed by Westinghouse to address new issues and phenomena since the previous analyses were conservative.

There is one NRC commitment as stated above. If there are any questions, please advise.

Respectfully submitted,

SOUTHERN NUCLEAR OPERATING COMPANY

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Dave Morey

DNM:EWC/maf 1999pcterrorreport.doc

Attachments:

1. 10 CFR 50.46 ECCS Evaluation Model 1999 Annual Report and Significant Error Report for Unit 2 SBLOCA
2. Unit 1 10 CFR 50.46 SBLOCA Significant Error Report Replacement Steam Generators

cc: Southern Nuclear Operating Company
Mr. L. M. Stinson, General Manager - Farley

U. S. Nuclear Regulatory Commission, Washington, D. C.
Mr. L. M. Padovan, Licensing Project Manager – Farley

U. S. Nuclear Regulatory Commission, Region II
Mr. L. A. Reyes, Regional Administrator
Mr. T. P. Johnson, Senior Resident Inspector – Farley

ATTACHMENT 1

Joseph M. Farley Nuclear Plant
10 CFR 50.46 ECCS Evaluation Model
1999 Annual Report and Significant Error Report for
Unit 2 SBLOCA

JOSEPH M. FARLEY NUCLEAR PLANT
10 CFR 50.46 ECCS EVALUATION MODEL
1999 ANNUAL REPORT AND SIGNIFICANT ERROR REPORT FOR UNIT 2 SBLOCA

I. 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. 10 CFR 50.46 also requires that significant errors or changes in the ECCS evaluation model be reported to the NRC within 30 days with a proposed schedule for providing a reanalysis or taking other action as may be needed to show compliance with 10 CFR 50.46 requirements. 10 CFR 50.46 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.

In Reference 1, information was submitted to the NRC regarding modifications to the Westinghouse large-break and small-break Loss-of-Coolant Accident (SBLOCA) ECCS evaluation models as applicable to the Farley Nuclear Plant (FNP) analyses for the calendar year 1998.

The report presents an assessment of the effects of modifications to the Westinghouse ECCS evaluation models on the Farley LOCA analysis results since the 1998 annual report (Reference 1) for the calendar year 1999. This annual report has been prepared in accordance with the Westinghouse Methodology for Implementation of 10 CFR 50.46 Reporting (WCAP-13451, Reference 2).

This annual report also serves as a 30 day Significant Error Report for the Unit 2 SBLOCA. The Burst and Blockage/Time in Life error (SPIKE Correlation Revision) resulted in a total PCT impact greater than 50°F, as reported by Westinghouse's official notification (ALA-00-020, March 8, 2000). Since Farley Unit 2 remains in compliance with 10 CFR 50.46 requirements as demonstrated in Table 2, and because Farley Unit 2 plans to replace steam generators in 2001 (Reference 3), no reanalysis of SBLOCA considering the existing steam generators is planned.

II. LARGE-BREAK

Tables 1A and 1B show the LBLOCA PCT rack-ups for both Unit 1 and Unit 2. Due to a limiting PCT change from Reflood 1 (Table 1A) to Reflood 2 (Table 1B) both rack-ups are shown for comparison.

II.A LBLOCA ANALYSIS-OF-RECORD

The LBLOCA analyses for Farley Units 1 and 2 were examined to assess the effects of the changes and errors in the Westinghouse LBLOCA ECCS evaluation model on PCT results. The LBLOCA analysis-of-record results for Farley Units 1 and 2 were calculated using Westinghouse BE-LOCA analysis.

The Unit 1 and Unit 2 analyses assumed the following information important to the LBLOCA in the BE-LOCA analysis (Reference 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 LBLOCAs are shown in Table 1B (Reflood 2).

II.B 1999 10 CFR 50.46 LOCA MODEL ASSESSMENTS

The following changes and errors in the Westinghouse ECCS evaluation model affect the BE-LOCA Model.

II.B.1 Prior Reported Assessments

The following two 10 CFR 50.46 assessments were reported in the last PCT submittal (Reference 1)

Vessel Channel DX Error (impacts vessel channel flow modeling)

In the gap flow wall friction and interfacial drag coefficient calculation, the incorrect cell height was used. Rather than using cell specific heights (DX) at each level only one DX value was used (Reference 9).

Increased Containment Spray Flow

An evaluation of RWST uncertainties and increase in containment spray flow effects was performed. In this evaluation, it was determined that there was an effect on PCT (Reference 8).

II.B.2 1999 PCT Assessments

The following two 10 CFR 50.46 assessments impact the 1999 PCT report.

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 when actual plant data was modeled (Reference 7).

Inconsistent Guidance for HOTSPOT Outputs in BE-LBLOCA Methodology

Procedures for the selection of the range of the second reflood using HOTSPOT outputs were inconsistent. As a result, the wrong time was chosen for the second reflood peak PCT. This affected the time, magnitude, and elevation of the second reflood PCT. A site-specific analysis was performed and Farley received a 20°F benefit for the second reflood only (Reference 7). The accumulator line/pressurizer surge line data error and the inconsistent guidance for HOTSPOT outputs caused the LBLOCA PCT results to be limiting at the second reflood rather than the first. Therefore, the net change in PCT in the second reflood is 29°F.

II.C 10 CFR 50.59 SAFETY EVALUATIONS FOR NON-MODEL PCT IMPACTS

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

II.D TOTAL RESULTANT LBLOCA PCT

As discussed above, the changes and errors to the Westinghouse LBLOCA ECCS evaluation model could affect the LBLOCA analysis PCT results. As shown in Table 1A and Table 1B, The LBLOCA analysis PCT results for both units are below the 10 CFR 50.46 limit of 2200^{o}F.

II.E LBLOCA CONCLUSIONS

An evaluation of the effects of changes and errors in the Westinghouse large-break BE-LOCA ECCS evaluation model was performed on the LBLOCA applicable to the Farley reference analysis. When the effects of the LBLOCA ECCS evaluation model changes and errors were combined with those of plant changes and the LBLOCA analysis-of-record results, it was determined that Farley Units 1 and 2 were in compliance with the requirements of 10 CFR 50.46.

III. SBLOCA

Table 2 shows the SBLOCA PCT rack-ups for both Unit 1 and Unit 2.

III.A SBLOCA ANALYSIS-OF-RECORD

The SBLOCA analyses for Farley Units 1 and 2 were examined to assess the effects of the changes and errors to the Westinghouse SBLOCA ECCS evaluation models on PCT results. The SBLOCA ECCS analysis results were calculated using the NOTRUMP SBLOCA ECCS evaluation model (Reference 5).

The Unit 1 and Unit 2 analyses assumed the following information important to the SBLOCA 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	Downflow Configuration

For Farley Units 1 and 2, the limiting size break analysis-of-record is a 3-inch diameter break in the cold leg. The limiting PCT values determined for Unit 1 and Unit 2 are shown in Table 2. An analysis for each Unit was performed due to Farley power uprate on Unit 1 and Unit 2 (Reference 4).

III.B 1999 10 CFR 50.46 SBLOCA MODEL ASSESSMENTS

The following changes and errors were identified:

III.B.1 Prior Reported Assessments

There are no prior reported assessments (Reference 1).

III.B.2 1999 PCT Assessments

Burst and Blockage/Time in Life (SPIKE Correlation Revision)

An update to the SPIKE code was made to reflect more recent data generated using the current SBLOCA evaluation model and methodology. This new version of SPIKE resulted in a 96°F penalty for Farley Unit 2 (Reference 7). SBLOCTA was used for the Unit 1 analysis.

The SPIKE Correlation Revision caused a 30 day error report for Farley Unit 2 only. Because Farley Unit 2 remains in compliance with 10 CFR 50.46 requirements, and because Farley Unit 2 will replace steam generators (Reference 3) in 2001, no reanalysis of SBLOCA considering existing steam generators is planned.

III.C 10 CFR 50.59 SAFETY EVALUATIONS FOR NON-MODEL PCT IMPACTS

The re-analysis for Farley Unit 1 mentioned above calculated a new Farley specific annular pellet penalty. Annular pellets were found to have a 3°F penalty for Farley Unit 1. (A re-analysis was not performed for Farley Unit 2; therefore the 10°F penalty still remains valid for Unit 2.)

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

III.D TOTAL RESULTANT SBLOCA PCT

As discussed above, the changes and errors in the Westinghouse SBLOCA ECCS evaluation model could affect the SBLOCA analysis PCT results. As shown in Table 2, the SBLOCA analysis PCT results for both units are below the 10 CFR 50.46 limit of 2200°F.

III.E SBLOCA CONCLUSIONS

An evaluation of the effects of changes and errors to the Westinghouse ECCS evaluation model was performed for the SBLOCA analysis results. When the effects of the SBLOCA ECCS evaluation model changes and errors were combined with those of plant changes and the SBLOCA analysis-of-record results, it was determined that compliance with the requirements of 10 CFR 50.46 would be maintained for both Units 1 and 2.

This annual report also serves as a 30 day Significant Error Report for the Unit 2 SBLOCA. This is due to the SPIKE correlation revision, which resulted in a total PCT impact greater than 50°F, as reported by Westinghouse to SNC on March 8, 2000. Since Farley Unit 2 remains in compliance with 10 CFR 50.46 requirements, and because Farley Unit 2 will replace steam generators (Reference 3) in 2001, no reanalysis of SBLOCA considering existing steam generators is planned.

IV. REFERENCES

1. Letter from D. N. Morey to USNRC, "Joseph M. Farley Nuclear Plant 10 CFR 50.46 annual ECCS Evaluation Model Changes Report for 1998 and Significant Error Report," March 19, 1999.

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

3. "Issuance of Amendment RE: Steam Generator Replacement," (TAC MA4393 & MA4394) December 29, 1999.

4. Letter from D. N. Morey to USNRC, "Joseph M. Farley Nuclear Plant Facility Operating Licenses and Technical Specifications Change Request for Power Uprating," June 20, 1997.

5. "Westinghouse SBLOCA 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, October 20, 1997.

7. ALA-00-020, "Final 10 CFR 50.46 Annual Notification and Reporting," March 8, 2000.

8. ALA-98-333, "Evaluation of RWST Uncertainties and Increase in Containment Spray Flow Effects," December 16, 1998.

9. ALA-99-041, "10 CFR 50.46 Annual Notification and Reporting for 1998," March 5, 1999.

TABLE 1A (non-limiting)

JOSEPH M. FARLEY NUCLEAR PLANT
TOTAL RESULTANT LBLOCA PCT (°F) FOR REFLOOD 1

	Unit 1, °F	Unit 2, °F
A. ANALYSIS-OF-RECORD (VANTAGE-+)		
1. ECCS Analysis	2004	2004
Total Analysis-of-Record PCT =	2004	2004
B. 1999 10 CFR 50.46 MODEL ASSESSMENTS		
1. Prior Reported Assessments	65	65
2. Accumulator Line /Pressurizer Surge Line Data	-41*	-41*
C. 10 CFR 50.59 PLANT MODIFICATIONS		
1. Addition of Permanent Storage Boxes in Containment	0**	0**
D. TOTAL RESULTANT LBLOCA PCT	2028	2028

* An accumulator/pressurizer surge line data error was reported. Due to differences in modeled and actual plant specifications a sensitivity analysis was performed using Farley specific data resulting in a change to the calculated PCT.

** The addition of permanent storage boxes in containment did not increase PCT.

TABLE 1B (limiting)

JOSEPH M. FARLEY NUCLEAR PLANT
TOTAL RESULTANT LBLOCA PCT (°F) FOR REFLOOD 2

	Unit 1, °F	Unit 2, °F
A. ANALYSIS-OF-RECORD (VANTAGE-+)		
1. ECCS Analysis	2064	2064
Total Analysis-of-Record PCT =	2064	2064
B. 1999 10 CFR 50.46 MODEL ASSESSMENTS		
1. Prior Reported Assessments	-3	-3
2. Accumulator Line/Pressurizer Surge Line Data	-9*	-9*
3. Inconsistent Guidance for HOTSPOT Outputs in BE-LBLOCA Methodology	-20**	-20**
C. 10 CFR 50.59 PLANT MODIFICATIONS		
1. Addition of Permanent Storage Boxes in Containment	0***	0***
D. TOTAL RESULTANT LBLOCA PCT	2032	2032

* An accumulator/pressurizer surge line data error was reported. Due to differences in modeled and actual plant specifications a sensitivity analysis was performed using Farley specific data resulting in a change to the calculated PCT.

** Resulted from inconsistent guidance for HOTSPOT outputs error. A site-specific evaluation was performed and Farley's PCT was found to have changed.

*** The addition of permanent storage boxes in containment did not increase PCT.

TABLE 2

JOSEPH M. FARLEY NUCLEAR PLANT
TOTAL RESULTANT SBLOCA PCT (^{o}F)

		Unit 1, °F	Unit 2, °F
A.	ANALYSIS-OF-RECORD (VANTAGE-+)		
	1. ECCS Analysis	1923*	1891*
	2. Annular Pellets (Farley 2 Only)	0	10**
	3. Burst and Blockage/Time in Life	117****	61
	Total Analysis-of-Record PCT =	2040	1962
B.	1999 10 CFR 50.46 MODEL ASSESSMENTS		
	1. Prior Reported Assessments	0	0
	2. Burst and Blockage/Time In Life (SPIKE Correlation Revision)	0	96***
C.	10 CFR 50.59 PLANT MODIFICATIONS		
	1. Additional Permanent Storage Boxes in Containment	0	0
	2. Annular Fuel Pellets (Farley 1 Only)	3****	0
D.	TOTAL RESULTANT SBLOCA PCT	2043	2058

* First Analysis since uprate approval

** The 10 CFR 50.59 plant modification for annular pellets was originally reported to the NRC in the PCT Error Report sent September 10, 1998.

*** A new version of the SPIKE code was used to analyze Farley 2 to correct an error in SPIKE. The analysis resulted in a change to the PCT reported for Farley 2. Farley 1 was reanalyzed with SBLOCTA; therefore the SPIKE penalty is not applicable

**** The SBLOCTA margin recovery performed for Farley 1 resulted in a revised annular pellet penalty. In addition, the burst and blockage/time in life value is revised as a result of the reanalysis. Since SBLOCTA represents a reanalysis, these changes are not reported as model assessments.

ATTACHMENT 2

Joseph M. Farley Nuclear Plant - Unit 1
10 CFR 50.46 SBLOCA Significant Error Report
Replacement Steam Generators

Based on the information provided by Westinghouse (ALA-00-020, March 8, 2000), we have determined that a significant error has occurred that is applicable to Farley Unit 1 (only) SBLOCA for Replacement Steam Generators (RSG). This error did not cause a penalty greater than 50°F for Farley Unit 2; therefore only Unit 1 data is reported in Table 1. The PCT assessment for Unit 1 is -62°F. The absolute magnitude of the PCT assessments exceeds the NRC's 50°F significant error criteria; therefore, the error is reportable to the NRC within 30 days. In addition, it has been determined that Unit 1 will continue to be in compliance with all LOCA acceptance criteria.

Although RSG has not yet been implemented at Farley Unit 1, Southern Nuclear is choosing to report this error now. SNC plans to implement RSG on Unit 1 in 2000.

Southern Nuclear will incorporate model changes at the next licensing action requiring reanalysis of the SBLOCA and for which NOTRUMP would be used as the evaluation model. Southern Nuclear proposes not to reanalyze the SBLOCA analysis at this time because:

1) The current magnitude of the PCT for the SBLOCA event maintains considerable margin to the 2200°F acceptance criteria.
2) The overall conservatism in the SBLOCA evaluation model has been approved by the NRC.
3) The assessments performed by Westinghouse to address new issues and phenomena since the previous analyses were conservative.

Discussion of SBLOCA Model Error

<u>Burst and Blockage/Time in Life (SPIKE Correlation Revision)</u>

The SPIKE computer program and the associated methodology are used as an evaluation tool in the 10 CFR 50.46 plant licensing process to estimate fuel rod burst peak cladding temperature penalties for SBLOCA analyses. The SPIKE code has been revised to reflect more recent data generated using the current SBLOCA evaluation model and methodology. The revision could result in a net PCT penalty for SBLOCA analyses which have utilized the SPIKE code.

The current SBLOCA evaluation model was employed, and a series of plant types were considered at varying beginning-of-life non-burst PCTs to develop a new database of bust "data" points. The evaluation tool was updated and validated to reflect the new database information. SBLOCA analyses which include burst and blockage effects based on direct burnup studies are not impacted by the revisions to SPIKE.

For the current "Burst and Blockage/Time In Life" penalty assessment, the new code version (SPIKE V4.01) was used. The SPIKE V4.01 calculation for Farley was conducted using the BOL PCTs documented in WCAP-15098 (FNP Units 1 and 2 Replacement Steam Generator Program NSSS Licensing Report, November 1988) plus the annular fuel pellet penalty (10°F). Also included in the new calculation was a benefit of 61°F for Unit 1 that was the result of assessing finalized Replacement Steam Generator Data. This SPIKE V4.01 calculation resulted in a revised "Burst and Blockage/Time in Life" penalty of 75°F for Unit 1. This yields an overall PCT benefit of 62°F for Unit 1 for the "Burst and Blockage/Time in Life" penalty over last year's assessment. These benefits are considered the result of the SPIKE correlation revision, and have been added to the Unit 1 SBLOCA PCT rack-up (Table 1) as such. This is due to the fact that since the corrected version of SPIKE was not released at the time of last year's annual report, a temporary assessment of the projected SPIKE re-correlation was assessed for Farley. Note that the reason for the net PCT benefit relative to the new SPIKE calculation was the benefit received from the finalized RSG data.