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

Joseph M. Farley Nuclear Plant
10 CFR 50.46 Annual ECCS Evaluation Model Changes Report for 2001

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, attached is Southern Nuclear Operating Company's report for Joseph M. Farley Nuclear Plant Units 1 and 2 for the calendar year 2001.

The 2001 annual report provides information regarding the effects of the ECCS evaluation model modifications on the peak cladding temperature (PCT) results since the 2000 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), with the exception of plant changes, as discussed here. Beginning with the 2001 calendar year 10 CFR 50.46 report, Westinghouse changed reporting methodology. This change moves the 50.59 plant change PCT values to the 50°F error reporting section. Since FNP has carried these 10 CFR 50.59 changes and has previously reported them, they are being categorized as "Prior 10 CFR 50.59 Assessments." FNP intends to report future PCT changes resulting from plant changes as errors.

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.

If there are any questions, please advise. This letter contains no NRC commitments.

Respectfully submitted,

DN Morey

D. N. Morey

DNM: CMC/slp

A001

Attachment: Joseph M. Farley Nuclear Plant 10 CFR 50.46 ECCS Evaluation Model 2001 Annual Report

cc: Southern Nuclear Operating Company
Mr. D. N. Morey, Vice President - Farley
Mr. D. E. Grissette, General Manager - Farley

U. S. Nuclear Regulatory Commission, Washington, D. C.
Mr. F. Rinaldi, 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

Joseph M. Farley Nuclear Plant
10 CFR 50.46 ECCS Evaluation Model
2001 Annual Report

JOSEPH M. FARLEY NUCLEAR PLANT
10 CFR 50.46 ECCS EVALUATION MODEL
2001 ANNUAL REPORT

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 (LOCA) ECCS evaluation models as applicable to the Farley Nuclear Plant (FNP) analyses for the calendar year 2000.

The following presents an assessment of the effects of modifications to the Westinghouse ECCS evaluation models on the Farley LOCA analysis results since the 2000 annual report (Reference 1) for the calendar year 2001. 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 here. Beginning with the 2001 calendar year 10 CFR 50.46 report, Westinghouse changed reporting methodology. This change moves the 50.59 plant change PCT values to the 50°F error reporting section. Since FNP has previously reported them, they are being categorized as "Prior 10 CFR 50.59 Assessments."

Unit 2 implemented the Replacement Steam Generators in mid-2001, and as such, a new PCT rack-up reflecting the RSG is presented here for Unit 2.

II. 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).

II.A LARGE-BREAK LOCA 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 methodology.

The Unit 1 and Unit 2 analyses assumed the following information important to the large-break LOCA in the BE-LOCA analysis. 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).

II.B 2001 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 10 CFR 50.46 Reported Assessments

The following 10 CFR 50.46 assessments were reported in the last 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 (MONTECF)

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.

II.B.2 2001 10 CFR 50.46 PCT Assessments

None

II.C PLANNED PLANT CHANGES (formerly 10 CFR 50.59 Safety Evaluation for Non-Model Impacts)

Beginning with the 2001 calendar year 10 CFR 50.46 report, Westinghouse changed reporting methodology. This change moves the 50.59 plant change PCT values to the 50°F error reporting section. Since FNP has previously reported them, they are being categorized as "Prior 10 CFR 50.59 Assessments."

II.C.1 Prior 10 CFR 50.59 Model Assessments

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

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 3).

II.C.2 Planned Plant Changes

None

II.D TOTAL RESULTANT LARGE-BREAK LOCA 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.

II.E LARGE-BREAK LOCA CONCLUSIONS

An evaluation of the effects of changes and errors in the Westinghouse large-break BE-LOCA ECCS evaluation model was performed on the large-break LOCA applicable to the Farley reference analysis. When the effects of the large-break ECCS evaluation model changes and errors were combined with those of plant changes and the large-break LOCA 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. SMALL-BREAK LOCA

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

III.A SMALL-BREAK LOCA 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).

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	Downflow 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.

III.B 2001 10 CFR 50.46 LOCA MODEL ASSESSMENTS

The following changes and errors were identified.

III.B.1 Prior Reported Assessments

The following assessment(s) were reported in the last submittal in Reference 1.

NOTRUMP Mixture Level Tracking/Region Depletion Error

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 have been determined to be an 13°F penalty for Unit 1 and Unit 2. In addition, the associated change in burst and blockage/time in life components were an additional 12°F for Unit 1 and 15°F for Unit 2. Thus, the total changes were 25°F for Unit 1 and 28°F for Unit 2.

III.B.2 2001 PCT Assessments

None

III.C PLANNED PLANT CHANGES (formerly 10 CFR 50.59 Safety Evaluation for Non-Model Impacts)

Beginning with the 2001 calendar year 10 CFR 50.46 report, Westinghouse changed reporting methodology. This change moved the 50.59 plant change PCT values to the 50°F error reporting section. Since FNP has previously reported them, they are being categorized as "Prior 10 CFR 50.59 Assessments."

II.C.1 Prior 10 CFR 50.59 Model Assessments

The following three plant change assessments were reported in the last submittal (Reference 1).

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 3).

Annular pellets were determined to have a 10°F penalty for SBLOCA results for Unit 1 and Unit 2 (Reference 8).

II.C.2 Planned Plant Changes

None

III.D TOTAL RESULTANT SMALL-BREAK LOCA 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.

III.E SMALL-BREAK LOCA CONCLUSIONS

An evaluation of the effects of changes and errors to the Westinghouse ECCS evaluation model was performed for the small-break LOCA analysis results. When the effects of the small-break ECCS evaluation model changes and errors were combined with those of plant changes and the small-break LOCA 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.

IV. REFERENCES

1. Letter from D. N. Morey to USNRC (NEL-01-0073), "Joseph M. Farley Nuclear Plant 10 CFR 50.46 Annual ECCS Evaluation Model Changes Report for 2000 and Significant Error Report," March 30, 2001.

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

3. ALA-00-018, "Southern Nuclear Operating Company, J. M. Farley Units 1 and 2, LBLOCA and SBLOCA Impacts Due to Final RSG Data for SGRP," February 11, 2000.

4. ALA-02-005, "10 CFR 50.46 Annual Notification and Reporting for 2001," March 5, 2002.

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.

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

	Unit 1	Unit 2
A. LBLOCA ANALYSIS OF RECORD		
1. ECCS Analysis	2056*	2056*
2. Increased Containment Spray Flow	9	9
Total Analysis-of-Record	2065*	2065*
B. PRIOR LBLOCA MODEL ASSESSMENTS		
1. Prior 10 CFR 50.46 Assessments *	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 10 CFR 50.46 LBLOCA MODEL ASSESSMENTS ***		
1. Accumulator Line/Pressurizer Surge Line Data	-41**	-41**
2. MONTECF Decay Heat Uncertainty Error	8**	8**
D. PLANNED PLANT CHANGE EVALUATIONS		
1. None	0	0
E. TOTAL RESULTANT LBLOCA PCT A+B+C+D		
Total Resultant LBLOCA	**2032**	**2032**

* The PCT values are rounded up to the next highest integer number to avoid reporting in decimal points. These values were calculated using the BE-LOCA methodology as submitted with the Farley Uprate Analysis.

** See Reference 4

*** Reported in previous annual report but not as a significant error.

TABLE 1B

JOSEPH M. FARLEY NULCEAR PLANT

TOTAL RESULTANT LARGE-BREAK LOCA PCT (°F) FOR REFLOOD 2

	Unit 1	Unit 2
A. LBLOCA ANALYSIS OF RECORD		
1. ECCS Analysis	1956*	1956*
2. Increased Containment Spray Flow	1	1
Total Analysis-of-Record	1957*	1957*
B. PRIOR LBLOCA MODEL ASSESSMENTS		
1. Prior 10 CFR 50.46 Assessments *	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 10 CFR 50.46 LBLOCA MODEL ASSESSMENTS ***		
1. Accumulator Line/Pressurizer Surge Line Data	-9**	-9**
2. MONTECF Decay Heat Uncertainty Error	0**	0**
D. PLANNED PLANT CHANGE EVALUATIONS		
1. None	0	0
E. TOTAL RESULTANT LBLOCA PCT A+B+C+D		
Total Resultant LBLOCA	**1948**	**1948**

* The PCT values are rounded up to the next highest integer number to avoid reporting in decimal points. These values were calculated using the BE-LOCA methodology as submitted with the Farley Uprate Analysis.

** See Reference 4

*** Reported in previous annual report but not as a significant error.

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

	Unit 1	Unit 2
A. SBLOCA ANALYSIS OF RECORD		
1. ECCS Analysis	1883*	1847*
2. Burst and Blockage / Time in Life	137	137
Total Analysis-of-Record	2020*	1984*
B. PRIOR SBLOCA MODEL ASSESSMENTS		
1. Prior 10 CFR 50.46 Assessments *	-62**	-14**
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**	-5
C. Annular Pellet Blanket	10**	10
Sum of Prior Assessments	-114	-9
C. CURRENT 10 CFR 50.46 SBLOCA MODEL ASSESSMENTS ****		
1. NOTRUMP Mixture Level Tracking / Region Depletion Errors	13**	***
2. Associated change in Burst and Blockage	12**	***
D. PLANNED PLANT CHANGE EVALUATIONS		
1. None	0	0
E. TOTAL RESULTANT SBLOCA PCT		
A+B+C+D		
Total Resultant SBLOCA	**1931**	**1975**

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

** See Reference 4

*** The NOTRUMP and Associated Change in Burst Blockage was reported in a 30 Day error report for Farley 2 SBLOCA last year. Therefore, these values are now in the prior 10 CFR 50.46 Assessments.

**** Reported in previous annual report but not as a significant error.