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10 CFR 50.46

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

Joseph M. Farley Nuclear Plant
10 CFR 50.46 Annual ECCS Evaluation Model Changes Report for 1995

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 1995.

The annual report provides information regarding the effects of the ECCS Evaluation Model modifications on the peak cladding temperature (PCT) results since the 1994 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.

If you have any questions, please advise.

Respectfully submitted,

Dave Morey

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Attachment

cc: Mr. S. D. Ebneter, Region II Administrator
Mr. B. L. Siegel, NRR Senior Project Manager
Mr. T. M. Ross, FNP Sr. Resident Inspector

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ATTACHMENT

Joseph M. Farley Nuclear Plant
10 CFR 50.46 ECCS Evaluation Model
1995 Annual Report

ATTACHMENT

JOSEPH M. FARLEY NUCLEAR PLANT
10 CFR 50.46 ECCS EVALUATION MODEL
1995 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 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 1994.

The following presents an assessment of the effects of modifications to the Westinghouse ECCS Evaluation Models on the Farley LOCA analysis results since the 1994 annual report for the calendar year 1995. The 1995 annual report also reflects the recent reanalysis of the Unit 2 large-break LOCA implemented in 1995 (Reference 5). This annual report has been prepared in accordance with the Westinghouse Methodology for Implementation of 10 CFR 50.46 Reporting (WCAP-13451, Reference 2). The results presented in the annual report as an analysis-of-record for the large-break LOCA and small-break LOCA PCTs reflect the use of VANTAGE-5 fuel in both units (Reference 3).

II. LARGE-BREAK LOCA

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

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 the 1981 version of the Westinghouse large-break LOCA ECCS Evaluation Model incorporating the BASH analysis technology (Reference 4). The large-break LOCA analysis for Unit 2 was revised and implemented in 1995 through the Cycle 11 reload safety evaluation process (Reference 5) to support increasing hot assembly average power, P-bar, from 1.42 to 1.514, increasing the nuclear enthalpy hot channel factor, $F^N_{\Delta H}$, from 1.65 to 1.70 (licensed value remained at 1.65 during 1995), and increasing the accumulator water temperature from 90°F to 120°F. As discussed in Reference 5, in order to gain additional PCT margin in the Unit 2 reanalysis, the steam generator tube plugging limit was reduced from 20% to 10% (10% average, 10% peak), administratively, in recognition of the fact that the actual plugging level was not expected to exceed 10% average or peak during Cycle 11 (see Table 1). As seen in Table 1, the reduction in the steam generator tube plugging limit was also adopted for Unit 1 through the Cycle 14 reload safety evaluation process (Reference 6).

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

Unit 1	Unit 2
Core Power = 1.02 x 2652 MWT	Core Power = 1.02 x 2652 MWT
17x17 VANTAGE-5 Fuel Assembly	17x17 VANTAGE-5 Fuel Assembly
F_Q = 2.45 for VANTAGE-5 Fuel F_Q = 2.32 for LOPAR Fuel	F_Q = 2.45 for VANTAGE-5 Fuel F_Q = 2.32 for LOPAR Fuel
$F\Delta H$ = 1.70 for VANTAGE-5 Fuel $F\Delta H$ = 1.55 for LOPAR Fuel	$F\Delta H$ = 1.70* for VANTAGE-5 Fuel $F\Delta H$ = 1.55 for LOPAR Fuel
SGTP** = 20%	SGTP** = 20%
Upflow Configuration	Downflow Configuration

* The licensed value remained at 1.65 during 1995.

** SGTP = Steam generator tube plugging limit assumed in the LOCA analysis. The limit was reduced administratively to 10% in 1995 in order to gain PCT margin (see Table l).

For Farley Units 1 and 2, the limiting size break analysis-of-record is a double-ended guillotine rupture of the cold leg piping with a discharge coefficient of $C_D = 0.4$. The limiting PCTs determined for the Unit 1 and Unit 2 large-break are shown in Table 1. The Unit 1 analysis-of-record limiting PCT value includes 3°F for containment mini-purge automatic isolation, 8°F for increased Tavg temperature uncertainty, and 6°F for combined safe shutdown earthquake (SSE) and LOCA events. The effects of containment mini-purge auto isolation and combined SSE plus LOCA events have been explicitly included in the Unit 2 revised analysis (Reference 5); however, the 8°F penalty for Tavg temperature uncertainty remains. It is noted that the 50°F transition core penalty has been removed by the Unit 1 Cycle 14 reload safety evaluation (Reference 6) since there are no LOPAR fuel assemblies loaded in the Unit 1 Cycle 14 core. However, the 50°F transition core penalty has been retained by the Unit 2 Cycle 11 reload safety evaluation (Reference 5) since there are still 33 LOPAR fuel assemblies loaded in the Unit 2 Cycle 11 core. In addition, Unit 1 Cycle 14 contains 1.5X IFBAs with 100 psi backfill pressure, which has shown to introduce a 7°F PCT penalty for Unit 1 (Reference 6). Unit 2 does not contain any 1.5X IFBA with 100 psi backfill pressure and, as such, is unaffected by the additional penalty (Reference 5).

II.B 1995 10 CFR 50.46 LOCA MODEL ASSESSMENTS

There are no changes and errors in the Westinghouse ECCS Evaluation Models found in 1995. Below are the previously reported changes and errors in the Westinghouse ECCS Evaluation Models affecting the BASH Evaluation Model large-break LOCA analysis-of-record results.

II.B.1 Prior Reported Assessments

The prior large-break LOCA PCT assessments given in Table 1 were submitted to the NRC in March 1995 as part of the 1994 Annual Report (Reference 1). It is noted in Table 1 that the previous changes and errors were corrected in the recent reanalysis of the large-break LOCA for Unit 2 (Reference 5).

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

As mentioned earlier and as noted in Table 1, the accumulator water temperature was increased from 90°F to 120°F for Unit 1 through the Cycle 14 reload safety evaluation process (Reference 6). For Unit 2, the accumulator temperature of 120°F was explicitly used in the reanalysis (Reference 5).

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 1, 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 BASH 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 7).

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-5 Fuel Assembly	17x17 VANTAGE-5 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-5 fuel analysis is a 3-inch diameter break in the cold leg. The limiting PCTs determined for the Unit 1 and Unit 2 17x17 VANTAGE-5 small-break are shown in Table 2. Both the Unit 1 and Unit 2 analysis-of-record limiting PCT values include a 20°F penalty due to the increased Tavg temperature uncertainty.

III.B 1995 10 CFR 50.46 LOCA MODEL ASSESSMENTS

The following changes and errors in the Westinghouse ECCS Evaluation Models would affect the NOTRUMP small-break LOCA analysis results obtained for the Farley VANTAGE-5 fuel analysis.

III.B.1 Prior Reported Assessments

The prior small-break LOCA PCT assessments shown in Table 2 were submitted to the NRC in Reference 1.

III.B.2 NOTRUMP Specific Enthalpy Error

A typographical error was found in a line of coding in the NOTRUMP code. Although the equation in the NOTRUMP topical report is correct, the coding represented the last term as a partial derivative with respect to the fluid node mixture region total energy instead of the mixture region total mass. The generic effect resulted in an estimated penalty of 20°F for both Unit 1 and Unit 2.

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

There have been no non-zero non-model PCT assessments under 10 CFR 50.59 made against the reference VANTAGE-5 LOCA analysis results to date. It should be noted that the effects of all of the applicable previous evaluations for both Farley Units 1 and 2 were incorporated into the VANTAGE-5 analysis.

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.

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, "Joseph M. Farley Nuclear Plant 10 CFR 50.46 Annual ECCS Evaluation Model Changes Report for 1994," March 20, 1995.

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

3. NRC Safety Evaluation Report, "Issuance of Amendment No. 92 to Facility Operating License No. NPF-2 and Amendment No. 85 to Facility Operating License No. NPF-8 Regarding the Use of VANTAGE-5 Fuel in Both Units and Allowing Removal and Replacement of the Resistance Temperature Detector Bypass Manifold System in Unit 2 - Joseph M. Farley Nuclear Plant, Units 1 and 2 (TAC Nos. M81025 and M81026)," March 11, 1992.

4. "The 1981 Version of the Westinghouse ECCS Evaluation Model Using the BASH Code," WCAP-10266-P-A, Rev. 2 (Proprietary), Young, M. Y., et. al, March 1987.

5. Joseph M. Farley Nuclear Plant Unit 2 Cycle 11 Reload Safety Evaluation - Revision 2 (10 CFR 50.59 Evaluation), letter CAF-NF-1455 dated April 7, 1995.

6. Joseph M. Farley Nuclear Plant Unit 1 Cycle 14 Reload Safety Evaluation (10 CFR 50.59 Evaluation), letter CAF-NF-1479 dated September 1, 1995.

7. "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.

TABLE 1

JOSEPH M. FARLEY NUCLEAR PLANT
TOTAL RESULTANT LARGE-BREAK LOCA PCT (°F)

	Unit 1, °F	Unit 2, °F
A. ANALYSIS-OF-RECORD (VANTAGE-5)		
1. ECCS Analysis	1896*	2120**
2. Containment Mini-Purge Auto Isolation	3	0**
3. Tavg Temperature Uncertainty	8	8**
4. Combined SSE and LOCA Events	6	0**
5. Transition Core Penalty	0[(a)]	50[(b)]**
6. SG Tube Plugging Margin of 10%	-40[(c)]	-40**
7. 1.5X IFBA	7	0
Total Analysis-of-Record PCT =	1880*	2138**
B. 1995 10 CFR 50.46 MODEL ASSESSMENTS		
1. Prior Reported Assessments	- 6***	0***
C. 10 CFR 50.59 PLANT MODIFICATIONS		
1. Increased Accumulator Water Temperature	48	0**
D. TOTAL RESULTANT LARGE-BREAK LOCA PCT	1922	2138**

(a) The Unit 1 Transition Core Penalty has been removed since the core contains all VANTAGE-5 fuel.
(b) Unit 2 still contains LOPAR fuel assemblies due to the redesign at EOC-10.
(c) To gain additional PCT margin, the steam generator tube plugging limit was reduced from 20% to 10%, similar to that of Unit 2 (Reference 5).

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

** The Unit 2 results correspond to the revised LOCA analysis performed as part of the increased peaking factors and accumulator water temperature (Reference 5).

*** The Structural Metal Heat Modeling correction (-25°F) and the LUCIFER error correction (-6°F) for Unit 2 and the LUCIFER error correction (-6°F) for Unit 1 were submitted to the NRC in March 1995 as part of the 1994 Annual Report (Reference 1). However, in the recent reanalysis of Unit 2 to increase the peaking factors and accumulator water temperature, the above corrections were explicitly accounted for in the large-break LOCA for Unit 2 (Reference 5).

TABLE 2

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

		Unit 1, °F	Unit 2, °F
A.	ANALYSIS-OF-RECORD (VANTAGE-5)		
	1. ECCS Analysis	1785*	1763*
	2. Tavg Temperature Uncertainty	20	20
	Total Analysis-of-Record PCT =	1805	1783
B.	1995 10 CFR 50.46 MODEL ASSESSMENTS		
	1. Prior Reported Assessments	171*	56*
	2. Boiling Heat Transfer Correlation Error	20	20
	3. Change in Burst and Blockage/Time in Life	17**	8**
C.	10 CFR 50.59 PLANT MODIFICATIONS		
	None	0	0
D.	TOTAL RESULTANT SMALL-BREAK LOCA PCT	2013	1867

* Reported to the NRC under 10 CFR 50.46 in Reference 1.

** For Burst and Blockage/Time in Life, penalties of 67°F for Unit 1 and 15°F for Unit 2 were included in B.1 above as previously reported to the NRC in Reference 1. Item B.3 reflects changes to the reported values due to the specific enthalpy error in B.2 and since the Burst and Blockage/Time in Life penalty is a function of PCT. Thus, the total penalties for change in Burst and Blockage/Time in Life are 84°F for Unit 1 and 23°F for Unit 2.