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10 CFR 50.46

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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 1992

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 insignificant errors and changes 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 1992.

The annual report provides information regarding the effects of the ECCS Evaluation Model modifications on the peak cladding temperature (PCT) results performed as part of the Farley VANTAGE-5 fuel analysis. Also, the attached annual report indicates that no plant change safety evaluations performed under the provisions of 10 CFR 50.59 in 1992 affected the PCT results. The report is in accordance with the Westinghouse Methodology for Implementation of 10 CFR 50.46 Reporting (WCAP-13451).

In addition to the annual reporting requirement, this letter also serves as notification under 10 CFR 50.46 (a)(3)(ii) of a greater than 50°F change in small-break LOCA PCT for Farley Unit 2. On February 15, 1993, it was determined that a significant error which had occurred in the NOTRUMP Computer Code (Cycle 23) is applicable to the Farley VANTAGE-5 analysis. The effects of this error on the Farley VANTAGE-5 fuel small-break LOCA analysis have been determined to not result in a penalty for Unit 1 (upflow configuration) but a 72°F penalty must be assigned for Unit 2 (downflow configuration). The above assessments have been substantiated by the reanalysis of the limiting transient for each unit with the corrected version of the NOTRUMP Code (Cycle 24). For Unit 1, the resulting PCT was calculated to be 0.2°F lower than the current Unit 1 analysis-of-record PCT (when rounded, the same value is obtained). The impact of this error on the Unit 2 small-break LOCA analysis results is determined to be significant and reportable within 30 days as required by 10 CFR 50.46. However, due to the proximity of the annual reporting date, the significant error has been incorporated into the attached annual report and is being

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reported within 30 days (from February 15, 1993) as required by 10 CFR 50.46. It is further noted that since an assessment of the significant error was performed by Westinghouse through a reanalysis of the limiting transient, a new reference analysis-of-record has been established and the corresponding FSAR updates will be implemented under the provisions of 10 CFR 50.59. Therefore, the requirement for providing a reanalysis schedule (normally required by 10 CFR 50.46 for reporting significant errors) is deemed satisfied.

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 performed under 10 CFR 50.59 are combined with the effects of the ECCS Evaluation Model modifications applicable to Farley Units 1 and 2.

If there are any questions, please advise.

Respectfully submitted,

J. D. Woodard

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JDW:AA/gps

Attachment

cc: Mr. S. D. Ebneter
Mr. T. A. Reed
Mr. G. F. Maxwell

ATTACHMENT

JOSEPH M. FARLEY NUCLEAR PLANT
10 CFR 50.46 ECCS EVALUATION MODEL
1992 ANNUAL REPORT

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 when the errors and changes are not significant, and within 30 days of determining the impact on the plant when the errors and changes are significant. Reference 1 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 2, 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 1991.

The following presents an assessment of the effects of modifications to the Westinghouse ECCS Evaluation Models on the Farley LOCA analysis results for the calendar year 1992. The 1992 annual report has been prepared in accordance with the Westinghouse Methodology for Implementation of 10 CFR 50.46 Reporting (WCAP-13451, Reference 3). 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 and were reviewed and approved by the NRC in Reference 4.

LARGE-BREAK LOCA

<u>ECCS EVALUATION MODEL</u>

The large-break LOCA analyses for Farley Units 1 and 2 were examined to assess the effects of the applicable modifications (errors and changes) to the Westinghouse large-break LOCA ECCS Evaluation Model on PCT results obtained for the Farley VANTAGE-5 fuel analysis approved by the NRC in Reference 4. The large-break LOCA analyses 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 5).

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

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 = 1.65 for VANTAGE-5 Fuel F-delta-H = 1.55 for LOPAR Fuel	F-delta-H = 1.65 for VANTAGE-5 Fuel F-delta-H = 1.55 for LOPAR Fuel
SGTP* = 20%	SGTP* = 20%
Upflow Configuration	Downflow Configuration

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* SGTP = Steam generator tube plugging limit assumed in the LOCA analysis

For Farley Units 1 and 2, the limiting size break for the VANTAGE-5 fuel analysis 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 17x17 VANTAGE-5 large-break are 2033°F and 2141°F, respectively. Both the Unit 1 and Unit 2 limiting PCT values include 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. Also included in the limiting PCT values for both units is the addition of a 50°F transition penalty due to the mixed core conditions during the transition to VANTAGE-5 fuel. However, the above penalties are listed below according to the format of WCAP-13451 (Reference 3) and are listed separately because they are not explicitly modeled in the ECCS analysis.

1992 10 CFR 50.46 LOCA MODEL ASSESSMENTS

The following error in the Westinghouse ECCS Evaluation Models would affect the BASH Evaluation Model large-break LOCA analysis results obtained for the Farley VANTAGE-5 fuel analysis.

Structural Metal Heat Modeling

An error was discovered during the review of the finite element heat conduction model used in the WREFLOOD-INTERIM code to calculate the heat transfer from the structural metal in the vessel during the reflood phase. It was noted that the material properties available in the code corresponded to those of stainless steel. While this is correct for the internal structures, it is inappropriate for the vessel wall which consists of carbon steel with a thin stainless internal clad. This was defined as a

non-discretionary change per Section 4.1.2 of WCAP-13451 since there was thought to be a potential for increased PCT with a more sophisticated composite model. The model was revised by replacing it with a more flexible one that allows detailed specification of structures. The estimated effect of this correction is a 25°F PCT benefit.

10 CFR 50.59 SAFETY EVALUATIONS FOR NON-MODEL IMPACTS

No 10 CFR 50.59 safety evaluations for non-model impacts have been assessed against the reference VANTAGE-5 large-break 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 as reviewed and approved by the NRC in Reference 4.

TOTAL RESULTANT LARGE-BREAK LOCA PCT

As discussed above, modifications (errors and changes) to the Westinghouse large-break LOCA ECCS Evaluation Model could affect the large-break LOCA analysis results by altering the PCT as shown below:

	Unit 1	Unit 2
A. ANALYSIS-OF-RECORD (VANTAGE-5)		
1. ECCS Analysis Result	1966°F*	2074°F*
2. Containment Mini-Purge Auto Isolation	3°F	3°F
3. Tavg Temperature Uncertainty	8°F	8°F
4. Combined SSE and LOCA Events	6°F	6°F
5. Transition Core Penalty	50°F	50°F
Total Analysis-of-Record PCT =	2033°F*	2141°F*
B. 1992 10 CFR 50.46 MODEL ASSESSMENTS		
Structural Metal Heat Modeling	- 25°F	- 25°F
C. 10 CFR 50.59 SAFETY EVALUATIONS		
None	+ 0°F	+ 0°F
Total Resultant PCT =	2008°F	2116°F

* The PCT values are rounded up to the next higher integer number to avoid reporting in decimal points. In the VANTAGE-5 fuel analysis submittal to the NRC which was subsequently approved in Reference 4, these values were presented in decimal points.

CONCLUSION

An evaluation of the effects of modifications (errors and changes) to the Westinghouse large-break BASH ECCS Evaluation Model was performed on the large-break LOCA applicable to the Farley VANTAGE-5 fuel analysis. When the effects of the large-break ECCS model errors/changes and safety evaluations were combined with the Farley ECCS reanalysis results, it was determined that Farley Units 1 and 2 were in compliance with the requirements of 10 CFR 50.46(b).

SMALL-BREAK LOCA

ECCS EVALUATION MODEL

The small-break LOCA analyses for Farley Units 1 and 2 were also examined to assess the effects of the applicable modifications (errors and changes) to the Westinghouse small-break LOCA ECCS Evaluation Models on PCT results obtained for the Farley VANTAGE-5 fuel analysis approved by the NRC in Reference 4. The small-break LOCA ECCS analysis results were calculated using the NOTRUMP small-break LOCA ECCS Evaluation Model (Reference 6).

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 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 1805^{o}F and 1711^{o}F, respectively. Both the Unit 1 and Unit 2 limiting PCT values include a 20^{o}F penalty due to the increased Tavg temperature uncertainty. However, the above penalties are listed below according to the format of WCAP-13451 (Reference 3) and are listed separately because they are not explicitly modeled in the ECCS analysis.

1992 10 CFR 50.46 LOCA MODEL ASSESSMENTS

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

Bessel Function Error

An error was discovered in the SUBROUTINE BESSJO which led to the calculation of incorrect values for the zeroth order Bessel function of the first kind. This calculation is used in the algorithm designed to limit the heat transfer out of a quenching fuel rod to the theoretical conduction limit. This error existed only in one cycle of the NOTRUMP computer code (Cycle 23) and therefore only affects analyses performed with that version. Cycle 23 of NOTRUMP was in use from February 1991 until the error was corrected in February 1992. This error correction returned the NOTRUMP code to consistency with the applicable section of WCAP-10079-P-A and therefore is not a change to the Evaluation Model. This was determined to be a non-discretionary change in accordance with Section 4.1.2 of WCAP-13451 and was corrected in accordance with Section 4.1.3 of WCAP-13451.

The effects of this error on the Farley VANTAGE-5 fuel small-break LOCA analysis have been determined to be a no penalty for Unit 1 (upflow configuration) but a 72°F penalty for Unit 2 (downflow configuration). The penalty assessments on Units 1 and 2 were substantiated by the reanalysis of the limiting transient for each unit with the corrected version of the NOTRUMP Code (Cycle 24). For Unit 1, the resulting PCT was calculated to be 0.2°F lower than the current Unit 1 analysis-of-record PCT (when rounded, the same value is obtained). The impact of this error on the Farley Unit 2 small-break LOCA analysis results is considered to be significant and reportable to the NRC within 30 days as required by 10 CFR 50.46.

10 CFR 50.59 SAFETY EVALUATIONS FOR NON-MODEL IMPACTS

No 10 CFR 50.59 safety evaluations for non-model impacts have been assessed 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 as reviewed and approved by the NRC in Reference 4.

TOTAL RESULTANT SMALL-BREAK LOCA PCT

As discussed above, modifications (errors and changes) to the Westinghouse small-break LOCA ECCS Evaluation Model could affect the small-break LOCA analysis results by altering the PCT as shown below:

	Unit 1	Unit 2
A. ANALYSIS-OF-RECORD (VANTAGE-5)		
1. ECCS Analysis	1785°F	1691°F
2. Tavg Temperature Uncertainty	20°F	20°F
Total Analysis-of-Record	1805°F	1711°F
B. 1992 10 CFR 50.46 MODE ASSESSMENT		
Bessel Function Error	+ 0°F	+ 72°F*
C. 10 CFR 50.59 SAFETY EVALUATIONS		
None	+ 0°F	+ 0°F
Total Resultant PCT =	1805°F	1783°F

* As stated above, the error in the Unit 2 small-break LOCA PCT results is considered to be a significant error and is reportable under 10 CFR 50.46.

CONCLUSION

An evaluation of the effects of modifications (errors and changes) to the Westinghouse ECCS Evaluation Model was performed for the small-break LOCA analysis results. When the effects of the ECCS Evaluation Model changes were combined with the VANTAGE-5 fuel analysis results, it was determined that compliance with the requirements of 10 CFR 50.46 would be maintained. A significant error report on the Farley Unit 2 small-break LOCA ECCS analysis results is required to be submitted to the NRC within 30 days (under 10 CFR 50.46). However, this annual report is deemed to fulfill the reporting requirement.

REFERENCES

1. "Emergency Core Cooling Systems; Revisions to Acceptance Criteria," Federal Register, Vol. 53, No. 180, pp. 35996-36005, dated September 16, 1988.

2. Letter from J. D. Woodard to USNRC, "Joseph M. Farley Nuclear Plant 10 CFR 50.46 Annual ECCS Evaluation Model Changes Report for 1991," March 25, 1992.

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

4. 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.

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

6. "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.