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10CFR50.46

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

Gentlemen:

Joseph M. Farley Nuclear Plant - Units 1 and 2
10CFR50.46 Annual ECCS Evaluation Model Changes Report for 1993

Provisions in 10CFR50.46 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 1993.

The provisions of 10CFR50.46 also require that a significant change or error in the Evaluation Model be reported to the NRC within 30 days with a proposed schedule for providing a re-analysis or description of other action taken to show compliance with 10CFR50.46 requirements. A significant change or error as defined by 10CFR50.46 is one which results in a calculated peak fuel cladding temperature different by more than 50°F from the temperature calculated for the limiting transient using the last acceptable model or is an accumulation of changes and errors such that the sum of the absolute magnitudes of the respective temperature changes is greater than 50°F. Accordingly, it was determined that on February 21, 1994, a significant error had occurred in the small-break loss of coolant accident (LOCA) analysis results applicable to the Farley VANTAGE-5 analysis. The absolute magnitude of the significant error for the Farley VANTAGE-5 fuel small-break LOCA analysis results has been determined to be 185°F for Unit 1 (up flow configuration) and 166°F for Unit 2 (down flow configuration) and thus reportable within 30 days as required by 10CFR50.46. However, due to the proximity of the annual reporting date, the significant error has been incorporated into the attached annual report.

By letter dated March 11, 1994, Southern Nuclear proposed a re-analysis schedule for the significant PCT errors (safety injection in the broken loop/improved condensation model) identified in our letter dated October 29, 1993. Since both significant model errors (i.e., the significant errors reported on October 29, 1993, and those changes reported herein) affect the NOTRUMP small-break LOCA Evaluation Model, and since the minimum margin to the 10CFR50.46 limit for the two units is at least 265°F, the following re-analysis schedule is proposed. Southern Nuclear proposes to capture a new model at the next licensing action requiring re-analysis of the small-break LOCA and for which NOTRUMP would be used as the Evaluation Model following resolution of the "safety injection in the broken loop/improved condensation model" issue.

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It has been determined that compliance with the requirements of 10CFR50.46 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.

Respectfully submitted,

SOUTHERN NUCLEAR OPERATING COMPANY

Dave Morey

DNM:AA/gps:5046REPO.DOC

Attachment

cc: Mr. S. D. Ebneter
Mr. B. L. Siegel
Mr. T. M. Ross

ATTACHMENT

JOSEPH M. FARLEY NUCLEAR PLANT
10CFR50.46 ECCS EVALUATION MODEL
1993 ANNUAL REPORT

I. BACKGROUND

Provisions in 10CFR50.46 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. 10CFR50.46 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 re-analysis or taking other action as may be needed to show compliance with 10CFR50.46 requirements. 10CFR50.46 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 an accumulation 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 1992. In Reference 2, a significant error report on the small-break LOCA Evaluation Model results for Farley Units 1 and 2 was submitted to the NRC.

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 1993. This report also includes the accumulation of PCT assessments due to the errors and changes in the small-break LOCA Evaluation Model that have been determined to be significant. These significant errors and changes are in addition to those submitted to the NRC on October 29, 1993, as discussed above (Reference 2). The 1993 annual report has been prepared in accordance with the Westinghouse Methodology for Implementation of 10CFR50.46 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 (Reference 4).

II. LARGE-BREAK LOCA

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 obtained for the Farley VANTAGE-5 fuel analysis (References 1 and 4).

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 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%
Up flow Configuration	Down flow Configuration

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

For Farley Units 1 and 2, the limiting size break analysis-of-record for the VANTAGE-5 fuel analysis is a double-ended guillotine rupture of the cold leg piping with a discharge coefficient of CD = 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 analysis-of-record 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 in Table 1 according to the format of WCAP-13451 (Reference 3) and are listed separately because they are not explicitly modeled in the ECCS analysis.

II.B 1993 10CFR50.46 LOCA MODEL ASSESSMENTS

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

II.B.1 Prior LOCA Model Assessments

The structural metal heat modeling correction (-25°F) as shown in Table 1 was reported in the LOCA Model Assessments 1992 Annual Report in Reference 1.

II.B.2 LUCIFER Error Corrections

The LUCIFER code is used to generate the component databases, from raw input data, to be used in the small and large-break LOCA analyses. Errors were found in the VESCAL subroutine of the LUCIFER code. These errors were in the geometric and mass calculations of the vessel and steam generator portions of the data. All LOCA analyses using the LUCIFER code outputs are affected by these error corrections. The errors were corrected in a manner to maintain the consistency of the LUCIFER code.

As shown in Table 1, representative plant calculations indicate a PCT effect of -6°F for Farley Units 1 and 2.

II.B.3 Double-Disk Gate Valve Pressure Equalization

An evaluation of a potential issue concerning use of double-disk gate valves in the ECCS as hot leg isolation valves has been completed. Use of these double-disk gate valves may involve an inner disk pressure equalization line that could set up a leak path into the hot leg during cold leg injection following a LOCA. This condition could lead to inadequate cold leg injection resulting in an increase in PCT.

The design characteristic of a double-disk gate valve provides isolation by the downstream disk sealing against the valve seat. The mechanical seating force and the hydraulic force from the upstream pressure (SI pump) act to provide force to the valve seal surfaces. The double-disk gate valve design results in a volume of fluid which is enclosed between the disks when the valve is closed. As the fluid volume heats up, pressure greater than system pressure may develop and may cause the disks to bind against the seats to the extent that the valves cannot be opened. To avoid this, many double-disk gate valves have been modified to include a pressure equalization line or a small hole in one of the disks to relieve the pressure between the disks. Based on generic leakage calculations, it was determined that the double-disk gate valves modified to eliminate concerns for thermal binding could leak as much as 30 gpm per valve. This leakage into the RCS hot legs (unaccounted for in the ECCS analysis) will increase steam binding during reflood and result in an increase in the calculated peak cladding temperature.

Plant Farley Units 1 and 2 are not affected by this issue since the ECCS hot leg isolation valves are not double-disk gate valves.

II.B.4 Large-Break LOCA Fuel Rod Model Errors

Minor errors in the rod heatup code used in the large break LOCA analyses were corrected. These errors concerned conditions which exist during periods of pellet/clad contact and the internal bookkeeping logic associated with clad thinning.

Representative plant calculations have shown that these corrections have a negligible effect on PCT for near beginning-of-life (BOL) fuel rod conditions (i.e., 2000 MWD/MTU). These effects become prevalent as burnup increases, but are insignificant until pellet/clad contact is predicted for steady-state operating conditions (typically > 8000 MWD/MTU). These corrections therefore result in a negligible PCT impact for large-break LOCA licensing basis PCTs which are calculated with near BOL conditions. This impact is being reported generically as 0°F for Farley Units 1 and 2.

II.B.5 High Temperature Fuel Rod Burst Model

A model for calculating the prediction of zircaloy cladding burst behavior above the previous limit of 1742°F was implemented. This model was described to the NRC in Reference 7.

This issue does not have any impact on the Farley Units 1 and 2 since burst occurs at a PCT below 1742°F in LBLOCA analysis.

II.B.6 Revised Burst Strain Limit Model

A revised burst strain limit model which limits strains is being implemented into the rod heatup codes used in both large-break and small-break LOCAs. This model is identical to that previously approved for use for Appendix K analysis of Upper Plenum Injection plants with WCOBRA/TRAC, as described in WCAP-10924-P-A, Rev. 1, Vol. 1, Add.4, "Westinghouse Large-Break LOCA Best-Estimate Methodology: Volume 1: Model Description and Validation, Addendum 4: Model Revisions," 1991.

The estimated effect on large-break LOCA PCTs ranges from negligible to a moderate, unquantified benefit which will be inherent in calculations once this model is implemented. This model will be implemented in both the large-break and small-break LOCA Evaluation Models during 1994. An estimate of 0°F has been assessed for Farley Units 1 and 2.

II.B.7 Large-Break LOCA Rod Internal Pressure (RIP) Issues

An evaluation of a potential issue concerning the impact of increased beginning-of-life rod internal pressure (RIP) uncertainties on LOCA analyses has recently been completed. Historically, beginning-of-life fuel pressure and temperatures uncertainties were based upon end-of-life considerations. These RIP uncertainties were found to be potentially non-conservative. During the evaluation of this issue, a second issue related to the applicability of generic IFBA fuel analyses to updated LOCA Evaluation Models was also identified and combined with this issue since the underlying mechanisms were the same.

The technical evaluation of this issue concluded that both the RIP uncertainty and the current IFBA designs with 200 psig initial fill pressure fuel typically will result in a maximum + 15°F PCT variation. Consequently, RIP manufacturing uncertainties and 200 psig initial fill pressure IFBA fuel do not have significant effects on the large-break LOCA analyses. Also, based on these results, it was concluded that only nominal RIP (with an upper bound bias) should be used in the LOCA analyses for fuel designs with an initial cold fill pressure > 200 psig. This is consistent with past LOCA analysis.

This conclusion was based upon the implementation of an extended burst and blockage correlation for burst temperatures above 1742°F and a more realistic minimum burnup assumption at hot full power conditions. In addition, any new reloads which would utilize low (< 200 psig) initial fill pressure fuel would be specifically analyzed.

The resolution of this issue for Plant Farley Units 1 and 2 has resulted in eliminating a previously applied temporary PCT penalty while the issue was being investigated.

II.C 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.

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 10CFR50.46 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 VANTAGE-5 fuel 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 10CFR50.46.

III. SMALL-BREAK LOCA

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 obtained for the Farley VANTAGE-5 fuel analysis (Reference 2). 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
Up flow Configuration	Down flow 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 1805°F and 1783°F, respectively. 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. However, the above penalties are listed in Table 2 according to the format of WCAP-13451 (Reference 3) and are listed separately because they are not explicitly modeled in the ECCS analysis.

III.B 1993 10CFR50.46 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 Assessments

In Reference 2, information was submitted to the NRC concerning a significant error/change in the small-break LOCA Evaluation Model PCT results for Farley Units 1 and 2. The significant error/change report included SI in Broken Loop (+150°F), Improved Condensation Model (-150 °F),and Drift Flux Flow Regime (-13°F). The sum of the absolute magnitude of the errors and changes was 313°F and therefore constituted a significant error or change as defined by 10CFR50.46. However, as stated in Reference 2, the net effect of the errors and changes is a 13°F reduction in small-break LOCA PCT for both Units 1 and 2.

III.B.2 Small-Break LOCA Limiting Time in Life

An evaluation of a potential issue with regard to burst/blockage modeling in the Westinghouse small-break LOCA Evaluation Model was recently completed. This potential issue involved a number of synergistic effects, all related to the manner in which the small-break model accounts for the swelling and burst of fuel rods, modeling of the rod burst strain, and resulting effects on clad temperature and oxidation from the metal/water reaction models and channel blockage.

Fuel rod burst during the course of a small-break LOCA analysis was found to potentially result in a significant temperature excursion above the clad temperature transient for a non-burst case. Since the methodology for small-break LOCA analyses had been to perform the analyses at a near beginning-of-life (BOL) condition, where rod internal pressures are relatively low, most analyses did not result in the occurrence of rod burst, and therefore may not have reflected the most limiting time in life PCT. In order to evaluate the effects of this phenomenon, Westinghouse has developed an analytical model which allows the prediction of rod burst PCT effects based upon the existing analysis-of-record.

The impact of this issue on small-break LOCA PCT results was determined to be 52°F for Unit 1 and 35°F for Unit 2, as shown in Table 2.

III.B.3 Charging/Safety Injection Systems Issues

An evaluation of a potential safety issue regarding issues related to the design and use of the miniflow line for the charging/safety injection pumps was recently completed. One issue involves the operation of the centrifugal charging pump (CCP) miniflow line during accident conditions. A CCP runout condition may occur if the CCP injection lines were balanced with the CCP miniflow path closed and credit was taken for operator action to isolate the miniflow line during the accident. Also, the existence of this condition may impact the ECCS flows assumed in plant-specific small-break LOCA analyses. The other issue involves miniflow orifices that are used for the CCP pumps. Two different orifice types have been supplied: 60 or 70 gpm orifice at a differential head of 6000 feet. Additional confirmation testing indicates that the orifice plates will allow a higher than design flow rate through the orifice at the design differential head. As a result, a discrepancy may exist between the installed miniflow line capacity and the ECCS analysis assumptions. The discrepancy would occur if the ECCS analysis assumed that the miniflow line resistance was based on the orifice allowing design flow at the design head as opposed to the higher as tested flow and head. Consequently, the miniflow path may permit more flow than previously determined, which may reduce SI flow during injection.

The specific PCT penalty assessment for Farley Units 1 and 2 is 107°F as shown in Table 2.

III.B.4 Hot Assembly Average Rod Burst Effects

The rod heatup code used in small-break LOCA calculations contains a model to calculate the amount of clad strain that accompanies rod burst. However, the methodology which has historically been used does not apply this burst strain model to the hot assembly average rod. This was done so as to minimize the rod gap and therefore maximize the heat transferred to the fluid channel, which in turn would maximize the hot rod temperature. However, due to mechanisms governing the zirc-water temperature excursion (which is the subject of the SBLOCA limiting time-in-life penalty for the hot rod), modeling of clad burst strain for the hot assembly average rod can result in a penalty for the hot rod by increasing the channel enthalpy at the time of PCT. Therefore, the methodology has been revised such that burst strain will also be modeled on the hot assembly average rod.

Representative plant calculations have shown that this change introduces an approximate 10% increase in the small-break LOCA limiting time-in-life penalty on the hot rod. This results in a 5 °F penalty in Unit 1 and a 4°F penalty in Unit 2. However, the penalty for each unit is being offset in Table 2 by the Revised Burst Strain Limit Model described below. These models will be implemented concurrently in the small-break Evaluation Model rod heatup code in 1994.

III.B.5 Revised Burst Strain Limit Model

See the discussion of this revised model in Section II.B.6.

In small-break LOCA, representative plant calculations indicate that the magnitude of the benefit is conservatively estimated to be exactly offsetting to the penalty introduced by the Hot Assembly Average Rod Burst issue documented above (Table 2). The corresponding PCT assessment for Unit 1 is -5°F and for Unit 2 is -4°F. This model will be implemented in both large-break and small-break Evaluation Models during 1994.

III.B.6 LUCIFER Error Corrections

See the discussion of this error in Section II.B.2.

Representative plant calculations indicate a PCT effect of -16°F for both Units 1 and 2 as shown in Table 2.

III.B.7 ISHIIA Drift Flux Error

An error was discovered both in WCAP-10079-P-A and the relevant coding in NOTRUMP subroutine ISHIIA which led to an incorrect calculation of the drift flux in NOTRUMP when a laminar film annular flow was predicted. The affected equation in WCAP-10079-P-A is Equation G-74 wherein a factor of "g", the gravitational constant, was inadvertently omitted from both the documentation and the equivalent coding. The correction of this error returned NOTRUMP to consistency with the ultimate reference for the affected correlation.

Representative plant analyses were used to estimate a generic PCT effect of 0°F for both Units 1 and 2.

III.B.8 NOTRUMP Point Kinetics Error

An error was discovered in the coding used in the NOTRUMP user external subroutine VOLHEAT. The coding did not correctly perform the calculation described by Equation 3-12-28 of WCAP-10054-P-A. This calculation is only used during the time when the point kinetics option is used to determine the core power before reactor trip. Therefore, any analysis which used the more conservative assumption of constant core power until reactor trip time is not affected by this error. The correction of this error returned NOTRUMP to consistency with WCAP-10054-P-A.

Representative plant analyses were used to estimate a generic PCT effect of 0°F for both Units 1 and 2.

III.B.9 Core Node Initialization Error

An error was discovered in how the properties of CORE NODE components were initialized for non-existent regions in the adjoining FLUID NODE. In particular, this led to artificially high core temperatures during the time step when the core mixture level crossed a node boundary, conservatively causing slightly more core mixture level depression than appropriate during this time step. Correction of this error allows for a smoother mixture level uncovery transient during node crossings.

The nature of this error led to an estimated generic PCT effect of 0°F for both Units 1 and 2.

III.B.10 NOTRUMP Heat Link Pointer Error

An error was discovered in how NOTRUMP initialized certain heat link pointer variables at the start of a calculation. Correction of this error returned NOTRUMP to consistency with the original intent of this section of coding.

Representative plant analyses were used to estimate a generic PCT effect of 0°F for both Units 1 and 2.

III.B.11 Fuel Rod Mode Errors in Small-Break LOCA

A number of minor programming errors were corrected in the fuel rod heatup code used in the small-break LOCA analyses. These corrections were related to:

1. Individual rod plenum temperatures;
2. Individual rod stack lengths;
3. Clad thinning logic;
4. Pellet/clad contact logic;
5. Corrected gamma redistribution;
6. Including ZrO2 thickness at t=O initialization; and
7. Numerics and convergence criteria of initialization.

This change is judged to have a negligible effect on PCT, and, on a generic basis, the estimated effect of 0°F is reported for both Units 1 and 2.

III.B.12 Double-Disk Gate Valve Pressure Equalization

See the discussion of this error in Section II.B.3.

An assessment of this issue on small-break LOCA Evaluation Model PCT results showed a nominal benefit, which is being reported generically as a 0°F impact for both Units 1 and 2.

III.C 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.

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.

It is noted in Table 2 that the sum of the absolute magnitude of the new changes and errors in the small-break Evaluation Model since the last significant changes and errors reported in Reference 2 are also considered to be significant and reportable within 30 days under 10CFR50.46 (i.e., greater than 50°F). The absolute magnitude of the significant errors is 185° F for Unit 1 (up flow) and 166°F for Unit 2 (down flow). These values do not include the significant errors reported to the NRC earlier in Reference 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 10CFR50.46 would be maintained for both Units 1 and 2. A significant error report on the Farley Units 1 and 2 small-break LOCA ECCS analysis results is required to be submitted to the NRC within 30 days under 10CFR50.46. However, this annual report is deemed to fulfill the reporting requirement.

IV. REFERENCES

1. Letter from J. D. Woodard to USNRC, "Joseph M. Farley Nuclear Plant 10CFR50.46 Annual ECCS Evaluation Model Changes Report for 1992," March 16, 1993.

2. Letter from D. N. Morey to USNRC, "Joseph M. Farley Nuclear Plant Peak Clad Temperature (PCT) Calculation," October 29, 1993.

3. WCAP-13451, "Westinghouse Methodology for Implementation of 10CFR50.46 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 (Proprietary), 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.

7. "Extension of NUREG-0630 Fuel Rod Burst Strain and Assembly Blockage Models to High Fuel Rod Burst Temperature," letter ET-NRC-92-3746, N. J. Liparulo (W) to R. C. Jones (NRC), September 16, 1992.

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*	1966	2074
2. Containment Mini-Purge Auto Isolation	3	3
3. Tavg Temperature Uncertainty	8	8
4. Combined SSE and LOCA Events	6	6
5. Transition Core Penalty	50	50
Total Analysis-of-Record PCT* =	2033	2141
B. 1993 10CFR50.46 MODEL ASSESSMENTS		
1. Prior Assessments**	-25	-25
2. LUCIFER Error Corrections	- 6	- 6
C. 10 CFR 50.59 PLANT MODIFICATIONS		
None	0	0
D. TOTAL RESULTANT LARGE-BREAK LOCA PCT	2002	2110

* The PCT values are rounded up to the next highest integer number to avoid reporting in decimal points. In the VANTAGE-5 fuel analysis NRC approval in Reference 5, these values were presented in decimal points.

** Structural Metal Heat Modeling Correction and its associated -25°F PCT assessment for Units 1 and 2 were reported in Reference 2.

TABLE 2

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

A. ANALYSIS-OF-RECORD (VANTAGE-5)	Unit 1, °F	Unit 2, °F
1. ECCS Analysis*	1785	1763
2. Tavg Temperature Uncertainty	20	20
Total Analysis-of-Record PCT =	1805	1783
B. 1993 10CFR50.46 MODEL ASSESSMENTS		
1. Prior Reported Assessments		
a. Effect of SI in Broken Loop*	150	150
b. Effect of Improved Condensation Model*	-150	-150
c. Drift Flux Flow Regime Errors*	-13	-13
2. Burst and Blockage/Time In Life**	52	35
3. Charging/SI Miniflow Assumptions**	107	107
4. Average Rod Burst Strain**	5	4
5. Fuel Rod Burst Strain Limit**	-5	-4
6. LUCIFER Error Corrections**	-16	-16
C. 10 CFR 50.59 PLANT MODIFICATIONS		
None	0	0
D. TOTAL RESULTANT SMALL-BREAK LOCA PCT	1935	1896

* A significant change/error report under 10CFR50.46 was submitted to the NRC in Reference 2.

** The sum of the absolute magnitude of the new changes and errors in the small-break Evaluation Model for Farley Units 1 and 2 since the last significant changes and errors reported in Reference 2 is also considered to be a significant error and thus reportable within 30 days under 10CFR50.46 (i.e., 185°F for Unit 1 and 166°F for Unit 2).