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Alabama Power
the southern electric system

10 CFR 50.46

Docket Nos. 50-348
50-364

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

Gentlemen:

Joseph M. Farley Nuclear Plant
10 CFR 50.46 Annual ECCS Evaluation Model Changes Report for 1990

The October 17, 1988, revision to 10 CFR 50.46 requires applicants and holders of operating licenses or construction permits to annually notify the Nuclear Regulatory Commission (NRC) of insignificant errors and changes in the ECCS Evaluation Models. Enclosed is Alabama Power Company's report for calendar year 1990 in compliance with this requirement for Joseph M. Farley Nuclear Plant Units 1 and 2.

Attachment A provides information regarding the effect of the ECCS Evaluation Model modifications on the peak cladding temperature (PCT) results. Attachment B provides a summary of the plant change safety evaluations performed through December 31, 1990, that impact PCT under the provisions of 10 CFR 50.59. Please note that the facility change safety evaluations included in Attachment B reflect only those which result in non-zero PCT penalty assessments. This information package constitutes Alabama Power Company's report for 1990 to the NRC as part of annual reporting required by 10 CFR 50.46(a)(3)(ii).

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. This determination is based on the fact that the total large-break and small-break resultant PCTs reported in Attachment B (i.e., including ECCS Evaluation Model modifications and all non-zero PCT penalties associated with the plant change safety evaluations performed under 10 CFR 50.59) are below the PCT limit of 2200°F.

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Alabama Power Company recently became aware of an application error in the calculation of PCT for Farley Nuclear Plant Unit 2. The error was associated with the downflow configuration in Unit 2 when using the 1981 Evaluation Model with BASH, which resulted in a 90°F increase in the large-break LOCA PCT for Unit 2. (The earlier sensitivity studies using the 1978 Evaluation Model had indicated the the Unit 1 upflow configuration was conservative for Unit 2.) This error was significant and thus reportable within 30 days under the provisions of the 10 CFR 50.46 (a)(3). A separate report was recently submitted by Alabama Power Company on February 18, 1991, which provided a detailed discussion of the error and the correction to the large-break LOCA PCT for Unit 2. It should be noted that the Unit 2 results reported in the enclosed annual report do not address this error since the annual report covers 1990 calendar year only.

If there are any questions, please advise.

Respectfully submitted,

ALABAMA POWER COMPANY

J. D. Woodard

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JDW/REM:maf0811

Attachments

cc: Mr. S. D. Ebneter
Mr. S. T. Hoffman
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ATTACHMENT A

EFFECT OF WESTINGHOUSE ECCS EVALUATION MODEL MODIFICATIONS ON THE LOCA ANALYSIS RESULTS

BACKGROUND

The October 17, 1988, revision to 10 CFR 50.46 requires 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. 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 References 2 and 3, information regarding modifications to the Westinghouse large-break and small-break Loss-of-Coolant Accident (LOCA) ECCS Evaluation Models was submitted to the NRC. It should be noted that the information on large-break LOCA provided in Reference 2 is applicable to Farley Unit 1 only, since the Farley Unit 2 large-break LOCA was reanalyzed separately in support of a recently approved license amendment (Reference 4). The license amendment provided changes to the Unit 2 Technical Specifications to allow an average of 15% steam generator tube plugging (SGTP) with a peak of 20% in any one steam generator (from 10% uniform which was in effect for Unit 1 as of December 31, 1990). The amendment also included an approximate 1.5% reduction in the reactor coolant system thermal design flow to 261,600 gpm (from 265,500 gpm).

The following presents an assessment of the effect of the modifications to the Westinghouse ECCS Evaluation Models on the LOCA analysis results. As stated above, the modifications to the Westinghouse ECCS Evaluation Models on the large-break LOCA analysis apply to Unit 1 only, because the current licensing basis analysis for Unit 2 uses a modified version of the methods used for Unit 1.

LARGE-BREAK LOCA

ECCS EVALUATION MODEL

The large-break LOCA analyses for Farley Units 1 and 2 were examined to assess the effect of the applicable modifications to the Westinghouse large-break LOCA ECCS Evaluation Model on PCT results. The large-break LOCA analyses results for Unit 1 were calculated using the 1981 version of the Westinghouse large-break LOCA ECCS Evaluation Model incorporating the BASH analysis technology. For Unit 2, a modified version of the 1981 Evaluation

Model with BASH was used in support of the license amendment discussed above. 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 Standard Fuel Assembly	17x17 Standard Fuel Assembly
F_Q = 2.40	F_Q = 2.32
F-delta-H = 1.62	F-delta-H = 1.62
SGTP* = 10% Uniform	SGTP* = 15% Average/20% Peak
Upflow Configuration Assumed	Upflow Configuration Assumed

* SGTP = Steam Generator Tube Plugging Limit

For Farley Unit 1, the limiting break resulted from the double-ended guillotine rupture of the cold leg piping with a discharge coefficient of C_D = 0.4. The calculated peak cladding temperature was 2013°F.

For Farley Unit 2, the limiting break size previously established was reanalyzed to support the recently approved changes in the Technical Specifications. The analysis-of-record peak clad temperature was 2049°F as obtained by the modified version of the 1981 Evaluation Model with BASH. The analysis-of-record peak clad temperature for Unit 2 also included the combined effects of previous evaluations in order to form a new design basis for Unit 2.

The modifications to the Westinghouse ECCS Evaluation Models discussed in Reference 2 which could affect the large-break LOCA analysis results for Unit 1 are described below.

MODIFICATIONS TO THE BASH ECCS EVALUATION MODEL (Farley Unit 1 Only)

Several improvements were made to the BASH computer code to treat special analysis cases which are related to the tracking of fluid interfaces and which could affect the plant analysis results.

1) A modification to prevent the code from aborting was made to the heat transfer model for the special situation when the quench front region moves to the bottom of the BASH core channel. The quench heat supplied to the fluid node below the bottom of the active fuel was set to zero.

2) A modification to prevent the code from aborting was made to allow negative initial movement of the liquid/two-phase and liquid-vapor interfaces. The coding in these areas was generalized to prevent mass imbalance in the special case where the liquid/two-phase interface reaches the bottom of the BASH core channel.

3) Modifications to prevent the code from aborting were made to increase the dimensions of certain arrays for special applications.

4) A modification was made to write additional variables to the tape of information to be provided to LOCBART.

5) Typographical errors in the coding of some convective heat transfer terms were corrected, but the corrections have no effect on the BASH analysis results since the related terms are always set equal to zero.

6) A modification was made to the BASH coding to reset the cold leg conditions in a conservative manner when the accumulators empty. The BASH model is initialized at the bottom of core recovery with the intact cold legs and lower plenum full of liquid. Flow into the downcomer then equals the accumulator flow. The modification removed most of the intact cold leg water at the accumulator empty time by resetting the intact cold leg conditions to a high quality two-phase mixture.

   In a typical BASH calculation, the downcomer is nearly full when the accumulators empty. The delay time, prior to the intact cold leg water reaching saturation, is sufficient to allow the downcomer to fill from the addition of safety injection fluid before the water in the cold legs reaches saturation. When the intact cold leg water reaches saturation, it merely flows out of the break. The cold leg water, therefore, does not affect the reflood transient.

   However, in a special case where a substantial time was required to fill the downcomer after the accumulators emptied, the fluid in the intact cold legs reached saturation before the downcomer filled, which artificially perturbed the transient response by incorrectly altering the downcomer fluid conditions causing the code to abort.

The Farley Unit 1 LOCA analysis results could be affected by the modifications specified in items 1, 2, 3, 4, 5, and 6 above. While there is no adverse effect on the PCT calculation for the majority of the changes which apply to Farley Unit 1 discussed above, a conservative estimate of 10°F has been assessed and tracked for use in determining the available margin to the limits of 10 CFR 50.46.

MODIFICATIONS TO THE WREFLOOD COMPUTER CODE (Farley Unit 1 Only)

In Reference 2, modifications are reported for the 1981 ECCS Evaluation Model which form the fundamental framework for application of the BASH methodology. The modifications made to the WREFLOOD computer code described for the Westinghouse 1981 ECCS Evaluation Model were carried into the WREFLOOD computer code used for BASH analyses.

In the BASH methodology, the WREFLOOD code is only used to calculate the bottom of core recovery time. Therefore, this modification has no effect on the BASH ECCS Evaluation Model calculations.

MODIFICATION TO THE LOCBART COMPUTER CODE

No modifications 've beer made since those outlined in Reference 5.

CONTAINMENT PURGE LINES OPEN EVALUATION

A safety evaluation of the effect of containment purge lines being open coincident with the large-break LOCA event was performed. An estimate of the large-break LOCA analysis PCT results was projected. The evaluation determined that the large-break LOCA analysis PCT results could be affected by a 4°F increase. This effect was included in the Unit 2 reanalysis.

RESULTANT LARGE-BREAK LOCA PCT

As discussed above, modifications 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 Calculated Result (Analysis-of-Record)	2013°F	2049°F
B.	Modifications to Westinghouse ECCS Evaluation Model	+ 10°F	+ N/A *
C.	Containment Purge Lines Open Evaluation	+ 4°F	*
D.	ECCS Evaluation Model Modifications Resultant PCT	2027°F	2049°F

Therefore, the sum of the absolute magnitude of the PCT assessments introduced as a result of the modifications and errors in the large-break LOCA ECCS Evaluation Model are 14°F for Unit 1 and 0°F for Unit 2.

* The Unit 2 limiting large-break LOCA reanalysis was performed by Westinghouse to support the recently approved SGTP limit of 15% average/20% peak. The reanalysis used the latest version of the 1981 Evaluation Model with BASH; thus, the previous 1981 Evaluation Model's (also with BASH) penalty is not applicable. Also included in the analysis-of-record for Unit 2 was the combined effects of previous safety evaluations.

SMALL-BREAK LOCA

ECCS EVALUATION MODEL

The small-break LOCA analyses for Farley Units 1 and 2 were also examined to assess the effect of the applicable modifications to the Westinghouse ECCS Evaluation Models on PCT results reported in Chapter 15, Section 3 of the FSAR. The small-break LOCA analyses results were calculated using the 1974 small-break LOCA ECCS Evaluation Model incorporating the WFLASH analysis technology. For Farley Units 1 and 2, the limiting size small-break resulted from a six-inch equivalent diameter break in the cold leg. The calculated PCT was 1712°F. The analysis assumed the following information important to the small-break LOCA analyses:

- Core Power = 1 02 X 2652 MWT
- 17x17 Standard Fuel Assembly
- F_Q = 2.32
- F-delta-H = 1.55
- Auxiliary Feedwater Flow = 1050 gpm (Total)

The modifications to the Westinghouse ECCS Evaluation Models discussed in References 2 and 3 which could affect the small-break LOCA analysis results found in Chapter 15, Section 3 in the Farley Units 1 and 2 FSAR are described below.

WFLASH ECCS EVALUATION MODEL CODE

Following the accident at Three Mile Island Unit 2, additional attention was focused on the small-break LOCA, and Westinghouse submitted a report, WCAP-9600 (Reference 6), to the Nuclear Regulatory Commission (NRC) detailing the performance of the Westinghouse small-break LOCA Evaluation Model which utilized the WFLASH computer code. In NUREG-0611 (Reference 7) the NRC staff questioned the validity of certain models in the WFLASH computer code and required licensees to justify continued acceptance of the model. Section II.K.3.30 of NUREG-0737 (Reference 8) clarified the NRC post-TMI requirements regarding small-break LOCA modeling and required licensees to revise their small-break LOCA ECCS models along the guidelines specified in NUREG-0611.

Following the issuance of NUREG-0737, Westinghouse and the Westinghouse Owners Group decided to develop the NOTRUMP (Reference 9) computer code for use in a new small-break LOCA ECCS Evaluation Model (Reference 10). The NRC approved the use of NOTRUMP for small-break LOCA ECCS analyses in May 1985.

Since approval of the NOTRUMP small-break LOCA ECCS Evaluation Model in 1985, the WFLASH computer code has not been maintained as part of the Westinghouse ECCS Evaluation Model computer code.

In section II.K.3.31 of NUREG-0737, the NRC required that each licensee submit a new small-break LOCA analysis using an NRC-approved small-break LOCA Evaluation Model which satisfied the requirements of NUREG-0737 section II.K.3.30. NRC Generic Letter 83-35 (Reference 11) relaxed the requirements of item II.K.3.31 by allowing a more generic response and providing a basis for retention of the existing small-break LOCA analyses. Provided that the previously existing model results were demonstrated to be conservative with respect to the new small-break LOCA model approved under the requirements of NUREG-0737 section II.K.3.30 (NOTRUMP), plant-specific analyses using the new small-break LOCA Evaluation Model would not be required. In WCAP-11145 (Reference 12), Westinghouse and the Westinghouse Owners Group demonstrated that the results obtained from calculations with WFLASH were conservative relative to those obtained with NOTRUMP. Compliance with item II.K.3.31 of NUREG-0737 for Farley has been completed by referencing WCAP-11145 and determined to be acceptable by the NRC.

Westinghouse, therefore, has not been modifying, investigating, or evaluating proposed changes to the WFLASH small-break LOCA ECCS Evaluation Model. Thus, there are no modifications to report.

SBLOCTA-IV COMPUTER CODE

The following modifications to the LOCTA-IV computer code in the small-break LOCA ECCS Evaluation Model have been made:

1) A test was added in the rod-to-steam radiation heat transfer coefficient calculation to preclude the use of the correlation when the wall-to-steam temperature differential dropped below the useful range of the correlation. This limit was derived based upon the physical limitations of the radiation phenomena.

   There is no effect of the modification on reported PCTs since the erroneous use of the correlation forced the calculations into aborted conditions.

2) An update was performed to allow the use of fuel rod performance data from the revised Westinghouse (PAD 3.3) model.

   An evaluation indicated that there is an insignificant effect of the modification on reported PCTs.

3) Modifications supporting a general upgrade of the computer program were implemented as follows:

   a) removal of unused or redundant coding,

   b) better coding organization to increase the efficiency of calculations, and

   c) improvements in user friendliness

      i) through defaulting of some input variables,
      ii) simplification of input,
      iii) input diagnostic checks, and
      iv) clarification of the output.

   Verification analyses calculations demonstrated that there was no effect on the calculated output resulting from these changes.

4) Three modifications improving the consistency between the Westinghouse fuel rod performance data (PAD) and the small-break LOCTA-IV fuel rod models were implemented:

   a) The form of the equation for the density of Uranium-Dioxide was corrected to calculate thermal expansion only in two dimensions, which is consistent with the way in which the fuel rod is modeled in the LOCTA codes.

   b) The correlation for the specific heat of water vapor at temperatures over 1590°F was improved.

   c) An error in the equation for the pellet/clad contact pressure was corrected.

   The Uranium-Dioxide density correction is estimated to have a maximum PCT benefit of less than 2°F, while the contact resistance modification has no PCT effect since it is not used.

SAFETY INJECTION BACK PRESSURE FIX EVALUATION

A safety evaluation of the effect of spilling broken loop safety injection water to containment back pressure instead of to reactor coolant system back pressure was performed for both units. An evaluation of the effect of this modeling change on the small-break LOCA analysis PCT results was performed as documented in section 15.3.1.2.2 of the Farley Units 1 and 2 FSAR. The evaluation determined that the LOCA analysis PCT results could be affected by a 46°F increase. This 46°F increase has been previously reported in an Alabama Power Company to NRC letter dated January 14, 1988.

RESULTANT SMALL-BREAK LOCA PCT

As discussed above, modifications 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 Calculated Result (Analysis-of-Record)	1712°F	1712°F
B.	Modifications to Westinghouse ECCS Evaluation Model	- 2°F	- 2°F
C.	Safety Injection Back Pressure Fix Evaluation	+ 46°F	+ 46°F
D.	ECCS Evaluation Model Modifications Resultant PCT	1756°F	1756°F

Therefore, the sum of the absolute magnitude of the PCT assessments introduced as the result of the modifications and errors in the small-break LOCA ECCS Evaluation Model is 48°F for Units 1 and 2.

CONCLUSIONS

An evaluation of the effect of modifications to the Westinghouse ECCS Evaluation Model, as reported in References 2 and 3, was performed for both the large-break LOCA and small-break LOCA analysis results. When the effects of the ECCS model changes were combined with the current plant analysis results, it was determined that compliance with the requirements of 10 CFR 50.46 would be maintained.

REFERENCES

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

2. NS-NRC-89-3463, "10 CFR 50.46 Annual Notification for 1989 of Modifications in the Westinghouse ECCS Evaluation Model," Letter from W. J. Johnson (Westinghouse) to T. E. Murley (NRC), dated October 5, 1989.

3. NS-NRC-89-3464, "Correction of Errors and Modifications to the NOTRUMP Code in the Westinghouse Small Break LOCA ECCS Evaluation Model Which Are Potentially Significant," Letter from W. J. Johnson (Westinghouse) to T. E. Murley (NRC), dated October 5, 1989.

4. Docket No. 50-364, "Issuance of Amendment No. 79 to Facility Operating License No. NPF-8 Regarding Steam Generator Tube Plugging - Joseph M. Farley Nuclear Plant, Unit 2 (TAC No. 77236)," NRC letter from S. T. Hoffman to W. G. Hairston, III, December 6, 1990.

5. WCAP-10266-P-A, Revision 2 (Proprietary), WCAP-10267-A, Revision 2 (Non-Proprietary), Besspiata, J. J., et al., "1981 Version of the Westinghouse ECCS Evaluation Model Using the BASH Code," March 1987.

6. "Report on Small Break Accidents for Westinghouse Nuclear Steam Supply System," WCAP-9601 (Non-Proprietary), June 1979, WCAP-9600 (Proprietary), June 1979.

7. "Generic Evaluation of Feedwater Transients and Small Break Loss-of-Coolant Accidents in Westinghouse Designed Operating Plants," NUREG-0611, January 1980.

8. "Clarification of TMI Action Plan Requirements," NUREG-0737, November 1980.

9. "NOTRUMP - A Nodal Transient Small Break and General Network Code," WCAP-10079-P-A (Proprietary), WCAP-10080-A (Non-Proprietary), Meyer, P. E., et al., August 1985.

10. "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.

11. "Clarification of TMI Action Plan Item II.K.3.31," NRC Generic Letter 83-35 from D. G. Eisenhut, November 2, 1983.

12. "Westinghouse Small Break ECCS Evaluation Model Generic Study With the NOTRUMP Code," WCAP-11145, Rupprecht, S. D., et al., August 1985.

ATTACHMENT B

EFFECT OF SAFETY EVALUATIONS AGAINST THE LOCA ANALYSIS RESULTS

LARGE-BREAK LOCA

DESCRIPTION OF PLANT MODIFICATIONS

As discussed below, the large-break Loss-of-Coolant Accident (LOCA) analysis results have been supplemented by safety evaluations of plant design changes under 10 CFR 50.59 that have assessed penalties to the fuel peak cladding temperature (PCT).

1) A safety evaluation of the effect of loose parts in the Unit 1 RCS (unrecovered grid strap sections) was performed to determine the effect of this condition on the large-break LOCA analysis PCT results. Completed in 1988, the evaluation determined that the Unit 1 large-break LOCA analysis PCT results could be affected by a 60°F increase.

2) A safety evaluation of the effect of the temperature uncertainties on the large-break LOCA analysis was performed as part of a proposed Technical Specifications change to remove the RTD bypass loops for Unit 1. The temperature uncertainties are associated with the accuracy of the instrumentations, the accuracy of the calibration equipment, and the procedures used for calibrating and reading the instrumentations. Completed in 1990, the evaluation determined that the Unit 1 large-break LOCA analysis PCT results could be affected by a 3°F increase.

RESULTANT LARGE-BREAK LOCA PCT

As discussed above, plant modifications could affect the resultant PCT as follows:

		Unit 1	Unit 2
0.	Resultant PCT from ECCS Evaluation Model Modifications Reported in Attachment A	2027°F	2049°F
1.	10 CFR 50.59 Safety Evaluation for Loose Parts (Grids)	+ 60°F	+ N/A
2.	10 CFR 50.59 Safety Evaluation for RCS Temperature Uncertainties	+ 3°F	+ *
	Total Large-Break Resultant PCT	2090°F	2049°F

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* In the case of Unit 2, the large-break LOCA analysis-of-record which was performed in support of a recently approved SGTP limit of 15% average/20% peak accounted for the effect of RCS temperature uncertainties.

Therefore, the total PCT assessments introduced as a result of plant modifications are 63°F for Unit 1 and 0°F for Unit 2.

SMALL-BREAK LOCA

DESCRIPTION OF PLANT MODIFICATIONS

As described below, the small-break LOCA analysis results have been supplemented by safety evaluations of plant design changes under 10 CFR 50.59 which have assessed penalties to the PCT.

1) A safety evaluation of the effects of a plant design change for upflow conversion (Unit 1 only) was performed. As documented in section 15.3.1.2.2 of the Farley Units 1 and 2 Final Safety Analysis Report (FSAR), the evaluation of the effect of this plant design change on the small-break LOCA analysis PCT results was calculated. The study determined that the Unit 1 small-break LOCA analysis PCT results could be affected by a 117°F increase. This evaluation was completed in 1982.

2) A safety evaluation of the effect of loose parts in the Unit 1 RCS (unrecovered grid strap sections) was performed. An evaluation of the effect of this condition on the small-break LOCA analysis PCT results was performed. The evaluation determined that the Unit 1 small-break LOCA analysis PCT results could be affected by a 32°F increase. This evaluation was completed in 1988.

3) A safety evaluation of the effect of the temperature uncertainties on the small-break LOCA was performed as part of a proposed Technical Specifications change to remove the RTD bypass loops for Unit 1. The temperature uncertainties are associated with the accuracy of the instrumentations, the accuracy of the calibration equipment, and the procedures used for calibrating and reading the instrumentations. The evaluation determined that the small-break LOCA analysis PCT results could be affected by a 2°F increase. Since the instrumentations and procedures are common between the two units, the same penalty applies to Unit 2 also. This evaluation was completed in 1990.

RESULTANT SMALL-BREAK LOCA PCT

As discussed above, plant modifications could affect the resultant PCT as follows:

		Unit 1	Unit 2
0.	Resultant PCT from ECCS Evaluation Model Changes/Errors Reported in Attachment A	1756°F	1756°F
1.	10 CFR 50.59 Safety Evaluation for Upflow Conversion	+ 117°F	N/A
2.	10 CFR 50.59 Safety Evaluation for Loose Parts (Grids)	+ 32°F	N/A
3.	10 CFR 50.59 Safety Evaluation for RTD Temperature Uncertainty	+ 2°F	+ 2°F
	Total Resultant PCT	1907°F	1758°F

Therefore, the resultant PCT assessments introduced as a results of plant modifications is 151°F for Unit 1 and 2°F for Unit 2.

CONCLUSIONS

An evaluation of the effect of modifications to the Westinghouse ECCS Evaluation Model as reported in Attachment A's References 2, 3 and 4 was performed for both the large-break LOCA and small-break LOCA licensing basis analysis results. It was determined that compliance with the requirements of 10 CFR 50.46 would be maintained when plant design changes, performed under 10 CFR 50.59, which could affect the LOCA analysis results were combined with the effect of the ECCS Evaluation Model modifications applicable to Farley Units 1 and 2.