L. M. Stinson (Mike)
Vice President

Southern Nuclear
Operating Company, Inc.
40 Inverness Center Parkway
Post Office Box 1295
Birmingham, Alabama 35201

Tel 205.992.5181
Fax 205.992.0341

SOUTHERN COMPANY
Energy to Serve Your World℠

October 6, 2005

NL-05-0068

Docket Nos.: 50-348
50-364

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D. C. 20555-0001

Joseph M. Farley Nuclear Plant Units 1 and 2
Technical Specification Amendment Request to Incorporate
Best Estimate LOCA Analysis Using ASTRUM

In accordance with the provisions of 10 CFR 50.90, Southern Nuclear Operating Company (SNC) is proposing a change to the Joseph M. Farley Nuclear Plant (FNP) Unit 1 and Unit 2 Technical Specifications (TS). This change is to support a revision to the Best Estimate Loss of Coolant Accident (LOCA) for FNP. The NRC recently approved a new Westinghouse Best Estimate LOCA (BELOCA) methodology, ASTRUM. ASTRUM (Automated Statistical Treatment of Uncertainty Method) was submitted in WCAP-16009-P. The NRC issued a Safety Evaluation Report in a letter dated November 5, 2004. Westinghouse issued WCAP-16009-P-A in January 2005. SNC has completed the analysis for FNP and the enclosed proposed amendment is to incorporate a reference to WCAP-16009-P-A in TS section 5.6.5 Core Operating Limits Report (COLR).

Enclosure 1 provides the basis for the proposed change, including an evaluation determining that the proposed change involves no significant hazards consideration as defined in 10 CFR 50.92 and an evaluation that determines this change satisfies the criteria of 10 CFR 51.22 for categorical exclusion from the requirements for an environmental assessment. Marked-up TS page is provided in Enclosures 2, and clean-typed pages are provided in Enclosure 3.

SNC requests approval of the proposed license amendments by October 14, 2006. The proposed changes would be implemented within 60 days of issuance of the amendment.

Upon approval of this proposed license amendment, the results presented in this letter will become the large break LOCA analysis of record for FNP Units 1 and 2.

(Affirmation and signature are provided on the following page.)

Mr. L. M. Stinson states he is a Vice President of Southern Nuclear Operating Company, is authorized to execute this oath on behalf of Southern Nuclear Operating Company and to the best of his knowledge and belief, the facts set forth in this letter are true.

This letter contains no NRC commitments. If you have any questions, please advise.

Respectfully submitted,

SOUTHERN NUCLEAR OPERATING COMPANY

L. M. Stinson

L. M. Stinson

Sworn to and subscribed before me this 6 *day of* October*, 2005.*

Deborah B. Sugamer
Notary Public

My commission expires: NOTARY PUBLIC STATE OF ALABAMA AT LARGE
MY COMMISSION EXPIRES: June 10, 2008
BONDED THRU NOTARY PUBLIC UNDERWRITERS

LMS/CHM

Enclosures: 1. Basis for Proposed Change
2. Marked-Up Technical Specifications Page
3. Clean Typed Technical Specifications Pages

cc: Southern Nuclear Operating Company
Mr. J. T. Gasser, Executive Vice President
Mr. J. R. Johnson, General Manager – Plant Farley
RTYPE: CFA04.054; LC# 14206

U. S. Nuclear Regulatory Commission
Dr. W. D. Travers, Regional Administrator
Mr. R. E. Martin, NRR Project Manager – Farley
Mr. C. A. Patterson, Senior Resident Inspector – Farley

Alabama Department of Public Health
Dr. D. E. Williamson, State Health Officer

Joseph M. Farley Nuclear Plant Units 1 and 2
Technical Specification Amendment Request to Incorporate
Best Estimate LOCA Analysis Using ASTRUM

Enclosure 1

Basis for Proposed Change

Joseph M. Farley Nuclear Plant Units 1 and 2
Technical Specification Amendment Request to Incorporate
Best Estimate LOCA Analysis Using ASTRUM

Enclosure 1

Basis for Proposed Change

1.0 Description

In accordance with the provisions of 10 CFR 50.90, Southern Nuclear Operating Company (SNC) is proposing a change to the Joseph M. Farley Nuclear Plant (FNP) Unit 1 and Unit 2 Technical Specifications (TS). This change is to support a revision to the Best Estimate Loss of Coolant Accident (LOCA) for FNP. The NRC recently approved a new Westinghouse Best Estimate LOCA (BELOCA) methodology, ASTRUM. ASTRUM (Automated Statistical Treatment of Uncertainty Method) was submitted in WCAP-16009-P. The NRC issued a Safety Evaluation Report in a letter dated November 5, 2004. Westinghouse issued WCAP-16009-P-A in January 2005. SNC has completed the analysis for FNP and the enclosed proposed amendment is to incorporate a reference to WCAP-16009-P-A in TS section 5.6.5 Core Operating Limits Report (COLR).

2.0 Proposed Change

The current FNP Technical Specification section 5.6.5, Core Operating Limits Report (COLR), contains references to the analytical methods used to determine the core operating limits as follows:

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

1. WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology," July 1985 (W Proprietary).

   (Methodology for LCOs 3.1.1 - SHUTDOWN MARGIN, 3.1.3 - Moderator Temperature Coefficient, 3.1.5 - Shutdown Bank Insertion Limit, 3.1.6 - Control Bank Insertion Limits, 3.2.3 - Axial Flux Difference, 3.2.1 - Heat Flux Hot Channel Factor, 3.2.2 - Nuclear Enthalpy Rise Hot Channel Factor and 3.9.1 - Boron Concentration)

2. WCAP-10216-P-A, Rev.1A, "Relaxation of Constant Axial Offset Control / F_Q Surveillance Technical Specification," February 1994 (W Proprietary).

   (Methodology for LCOs 3.2.3 - Axial Flux Difference and 3.2.1 - Heat Flux Hot Channel Factor.)

3a. WCAP-12945-P-A, Volume 1, Revision 2, and Volumes 2 through 5, Revision 1, "Code Qualification Document for Best Estimate LOCA Analysis," March 1998 (W Proprietary).

3b. WCAP-12610-P-A, "Vantage+ Fuel Assembly Reference Core Report," April 1995 (W Proprietary).

(Methodology for LCO 3.2.1 - Heat Flux Hot Channel Factor and LCO 3.4.1-RCS Pressure, Temperature and Flow Departure from Nucleate Boiling Limits.)

4. WCAP-8745-P-A, "Design Bases for the Thermal Overpower ΔT and Thermal Overtemperature ΔT Trip Functions," September 1986 (Westinghouse Proprietary)

(Methodology for Overpower ΔT and Thermal Overtemperature ΔT Trip Functions)

5. WCAP-14750-P-A Revision 1, "RCS Flow Verification Using Elbow Taps at Westinghouse 3-Loop PWRs. (Westinghouse Proprietary)

(Methodology for minimum RCS flow determination using the elbow tap measurement.)

6. WCAP-11596-P-A, "Qualification of the Phoenix-P/ANC Nuclear Design System for Pressurized Water Reactor Cores," June 1988

(Methodology for LCO 3.9.1 - Boron Concentration.)

7. WCAP-11397-P-A "Revised Thermal Design Procedure," April 1989

(Methodology for LCO 2.1.1-Reactor Core Safety Limits, LCO 3.4.1-RCS Pressure, Temperature and Flow Departure from Nucleate Boiling Limits.)

This proposed amendment will add the following item:

3c. WCAP-16009-P-A, "Realistic Large Break LOCA Evaluation Methodology Using Automated Statistical Treatment of Uncertainty Method (ASTRUM)," M.E. Nissley, et al., January 2005 (Proprietary).

3.0 Background

Farley Units 1 and 2 are currently operating under the WCOBRA/TRAC best estimate methodology approved by the NRC in 1998. This is reflected in the current COLR Reference in TS 5.6.5.b.3a. The NRC recently approved a new Westinghouse BELOCA methodology, ASTRUM. ASTRUM was submitted in WCAP-16009-P (ref . 3). The NRC issued a Safety Evaluation Report in a letter dated November 5, 2004 (ref. 4). Westinghouse issued WCAP-16009-P-A in January 2005 (ref. 5).

Westinghouse recently underwent a program to revise the statistical approach used to develop the Peak Cladding Temperature (PCT) and oxidation results at the 95th percentile. This method is still based on the Code Qualification Document (CQD) methodology (ref. 2) and follows the steps in the Code Scaling Applicability and Uncertainty (CSAU) methodology. However, the uncertainty analysis (element 3 in CSAU) is replaced by a technique based on order statistics. The ASTRUM methodology replaces the response surface technique with a statistical sampling method where the uncertainty parameters are simultaneously sampled for each case.

4.0 Technical Analysis

Westinghouse reanalyzed the FNP BELOCA using the new approved ASTRUM methodology. The reanalysis was performed and met all the NRC Safety Evaluation Report conditions and limitations identified in NRC letter dated November 5, 2004 (ref. 4).

The WCOBRA/TRAC models for Farley Units 1 and 2 were originally developed for the power uprate which was approved by the NRC in 1998 (ref. 1). Two BELOCA models were utilized in the original analysis of record (AOR), mainly because Unit 1 had an upflow barrel/baffle (B/B) configuration, whereas Unit 2 had a downflow B/B configuration. A parametric study was performed at that time to determine the limiting unit. Unit 2 was determined to be the limiting unit at that time. Therefore, the Unit 2 model was utilized for the subsequent steps of the original application of the best estimate large break LOCA evaluation model.

Subsequent to the original analysis, FNP performed a Replacement Steam Generators (RSG) project for both units, and Unit 2 has been converted to an upflow B/B configuration. These changes were incorporated into the ASTRUM analysis. Moreover, investigations revealed that the remaining differences in the vessels were small enough to justify the use of a single WCOBRA/TRAC geometric model for both Units 1 and 2.

Table 1 lists the major plant parameter assumptions used in the BE LOCA analysis for FNP and Table 2 summarizes the results of the ASTRUM analysis.

The ASTRUM methodology requires the execution of 124 transients to determine a bounding estimate of the 95th percentile of the Peak Clad Temperature (PCT), Local Maximum Oxidation (LMO), and Core Wide Oxidation (CWO) with 95%

confidence level. These parameters are needed to satisfy the 10 CFR 50.46 criteria with regard to PCT, LMO, and CWO. From these 124 calculations, run 104 proved to be the limiting PCT transient, run 051 the limiting LMO transient, and run 014 the limiting CWO transient.

The scatter plot presented on Figure 2 shows the influence of the effective break area on the final PCT. The effective break area is calculated by multiplying the discharge coefficient (CD) with the sample value of the break area, normalized to the cold-leg cross sectional area. Figure 2 is provided to illustrate that the break area is a significant contributor to the variation in PCT.

Figures 3, 4 and 5 are presented to show the limiting cladding transient for each criterion. Figure 3 shows the predicted clad temperature transient at the PCT limiting elevation for run 104. Figure 4 presents the clad temperature transient predicted at the LMO elevation for run 051. Figure 5 shows the PCT trace for the CWO limiting transient (run 014).

Based on the results as presented in Table 2, it is concluded that the FNP Units 1 and 2 continue to maintain a margin of safety to the limits prescribed by 10 CFR 50.46.

5.0 Regulatory Analysis

5.1 10 CFR 50.46 Evaluation

In accordance with 10 CFR 50.46, the conclusions of the best estimate large break LOCA analysis show that there is a high level probability the following criteria are met.

1. The calculated maximum fuel element cladding temperature (i.e., peak cladding temperature (PCT)) will not exceed 2,200°F.

2. The calculated total oxidation of the cladding (i.e., maximum cladding oxidation) will nowhere exceed 0.17 times the total cladding thickness before oxidation.

3 The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam (i.e., maximum hydrogen generation) will not exceed 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.

4. The calculated changes in core geometry are such that the core remains amenable to cooling.

5. After successful initial operation of the Emergency Core Cooling System (ECCS), the core temperature will be maintained at an acceptably low value and decay heat will be removed for the

extended period of time required by the long-lived radioactivity remaining in the core.

5.2 No Significant Hazards Consideration

In accordance with the provisions of 10 CFR 50.90, Southern Nuclear Operating Company (SNC) is proposing a change to the Joseph M. Farley Nuclear Plant (FNP) Unit 1 and Unit 2 Technical Specifications (TS). This change is to support a revision to the Best Estimate Loss of Coolant Accident (LOCA) for FNP. The NRC recently approved a new Westinghouse Best Estimate LOCA (BELOCA) methodology, ASTRUM. ASTRUM (Automated Statistical Treatment of Uncertainty Method) was submitted in WCAP-16009-P. The NRC issued a Safety Evaluation Report in a letter dated November 5, 2004. Westinghouse issued WCAP-16009-P-A in January 2005. SNC has completed the analysis for FNP and the enclosed proposed amendment is to incorporate a reference to WCAP-16009-P-A in TS section 5.6.5 Core Operating Limits Report (COLR).

SNC has evaluated whether or not a significant hazards consideration is involved with the proposed change by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

   No physical plant changes are being made as a result of using the Westinghouse Best Estimate Large Break LOCA (BELOCA) analysis methodology. The proposed TS changes simply involve updating the references in TS 5.6.5.b, Core Operating Limits Report (COLR), to reference the Westinghouse BELOCA analysis methodology. The plant conditions assumed in the analysis are bounded by the design conditions for all equipment in the plant; therefore, there will be no increase in the probability of a LOCA. The consequences of a LOCA are not being increased, since the analysis has shown that the Emergency Core Cooling System (ECCS) is designed such that its calculated cooling performance conforms to the criteria contained in 10 CFR 50.46, "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors." No other accident consequence is potentially affected by this change.

   All systems will continue to be operated in accordance with current design requirements under the new analysis, therefore no new components or system interactions have been identified that could lead to an increase in the probability of any accident previously evaluated in the Updated Final Safety Analysis Report (UFSAR). No changes were required to the Reactor Protection System (RPS) or Engineering Safety Features (ESF) setpoints because of the new analysis methodology.

Therefore, it is concluded that this change does not significantly increase the probability or consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

   There are no physical changes being made to the plant as a result of using the Westinghouse Best Estimate Large Break LOCA analysis methodology. No new modes of plant operation are being introduced. The configuration, operation and accident response of the structures or components are unchanged by utilization of the new analysis methodology. Analyses of transient events have confirmed that no transient event results in a new sequence of events that could lead to a new accident scenario. The parameters assumed in the analysis are within the design limits of existing plant equipment.

   In addition, employing the Westinghouse Best Estimate Large Break LOCA analysis methodology does not create any new failure modes that could lead to a different kind of accident. The design of all systems remains unchanged and no new equipment or systems have been installed which could potentially introduce new failure modes or accident sequences. No changes have been made to any RPS or ESF actuation setpoints.

   Based on this review, it is concluded that no new accident scenarios, failure mechanisms or limiting single failures are introduced as a result of the proposed changes.

   Therefore, the proposed TS changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

   It has been shown that the analytic technique used in the Westinghouse Best Estimate Large Break LOCA analysis methodology realistically describes the expected behavior of the reactor system during a postulated LOCA. Uncertainties have been accounted for as required by 10 CFR 50.46. A sufficient number of LOCAs with different break sizes, different locations, and other variations in properties have been considered to provide assurance that the most severe postulated LOCAs have been evaluated. The analysis has demonstrated that all acceptance criteria contained in 10 CFR 50.46 paragraph b continue to be satisfied.

   Therefore, it is concluded that this change does not involve a significant reduction in the margin of safety.

Based on the above, SNC concludes that the proposed change presents no significant hazards considerations under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

6.0 Environmental Consideration

SNC has reviewed the proposed change pursuant to 10 CFR 50.92 and determined that it does not involve a significant hazards consideration. In addition, there is no significant change in the types or significant increase in the amounts of any effluents that may be released offsite and there is no significant increase in individual or cumulative occupational radiation exposure. Consequently, the proposed TS change has no significant effect on the human environment and satisfies the criteria of 10 CFR 51.22 for categorical exclusion from the requirements for an environmental assessment.

7.0 References

1. Safety Evaluation by the Office of Nuclear Reactor Regulation Related to Amendment No. 137 to Facility Operating License No. NPF-2 and Amendment No. 129 to Facility Operating License No. NPF-8 Southern Nuclear Operating Company, Inc., Et. Al., Joseph M. Farley Nuclear Plant, Units 1 and 2, Docket Nos. 50-348 and 50-364, April 29,1998.

2. Bajorek, S. M., et. al., 1998, "Code Qualification Document for Best Estimate LOCA Analysis," WCAP-12945-P-A, Volume 1, Revision 2 and Volumes 2 through 5, Revision 1, and WCAP-14747 (Non-Proprietary).

3. Nissley, M. E., et. al., 2003, "Realistic Large Break LOCA Evaluation Methodology Using the Automated Statistical Treatment of Uncertainty Method (ASTRUM)," WCAP-16009-P.

4. Letter from H. N. Berkow (NRC) to J. A. Gresham (W) dated November 5, 2004, RE: Final Safety Evaluation for WCAP-16009-P, Revision 0, "Realistic Large Break LOCA Evaluation Methodology Using Automated Statistical Treatment of Uncertainty Method (ASTRUM)," TAC No. MB9483.

5. Nissley, M. E., et. al., January 2005, "Realistic Large Break LOCA Evaluation Methodology Using Automated Statistical Treatment of Uncertainty Method (ASTRUM)," WCAP-16009-P-A

Table 1

Major Plant Parameter Assumptions Used in the BE LOCA Analysis

Parameter	Value
Plant Physical Description	
SG Tube Plugging	≤ 10 %
Plant Initial Operating Conditions	
Reactor Power	≤ 102 % of 2,775 MWt
Peaking Factors	$F_Q \leq 2.5$ $F_{\Delta H} \leq 1.7$
Axial Power Distribution	See Figure 1
Fluid Conditions	
T_{AVG}	567.2 ± 6 °F ≤ T_{AVG} ≤ 577.2 ± 6 °F
Pressurizer Pressure	2,200 psia ≤ P_{RCS} ≤ 2,300 psia
Reactor Coolant Flow	≥ 86,000 gpm/loop
Accumulator Temperature	90 °F ≤ T_{ACC} ≤ 120 °F
Accumulator Pressure	600 psia ≤ P_{ACC} ≤ 680 psia
Accumulator Water Volume	965 ft^3 ≤ V_{ACC} ≤ 995 ft^3
Accident Boundary Conditions	
Single Failure Assumptions	Loss of one ECCS train
Safety Injection Flow	Minimum
Safety Injection Temperature	70 °F ≤ T_{SI} ≤ 100 °F
Safety Injection Initiation Delay Time	≤ 12 sec (with offsite power) ≤ 27 sec (without offsite power)
Containment Pressure	Bounded

Table 2

Best Estimate Large Break LOCA Results

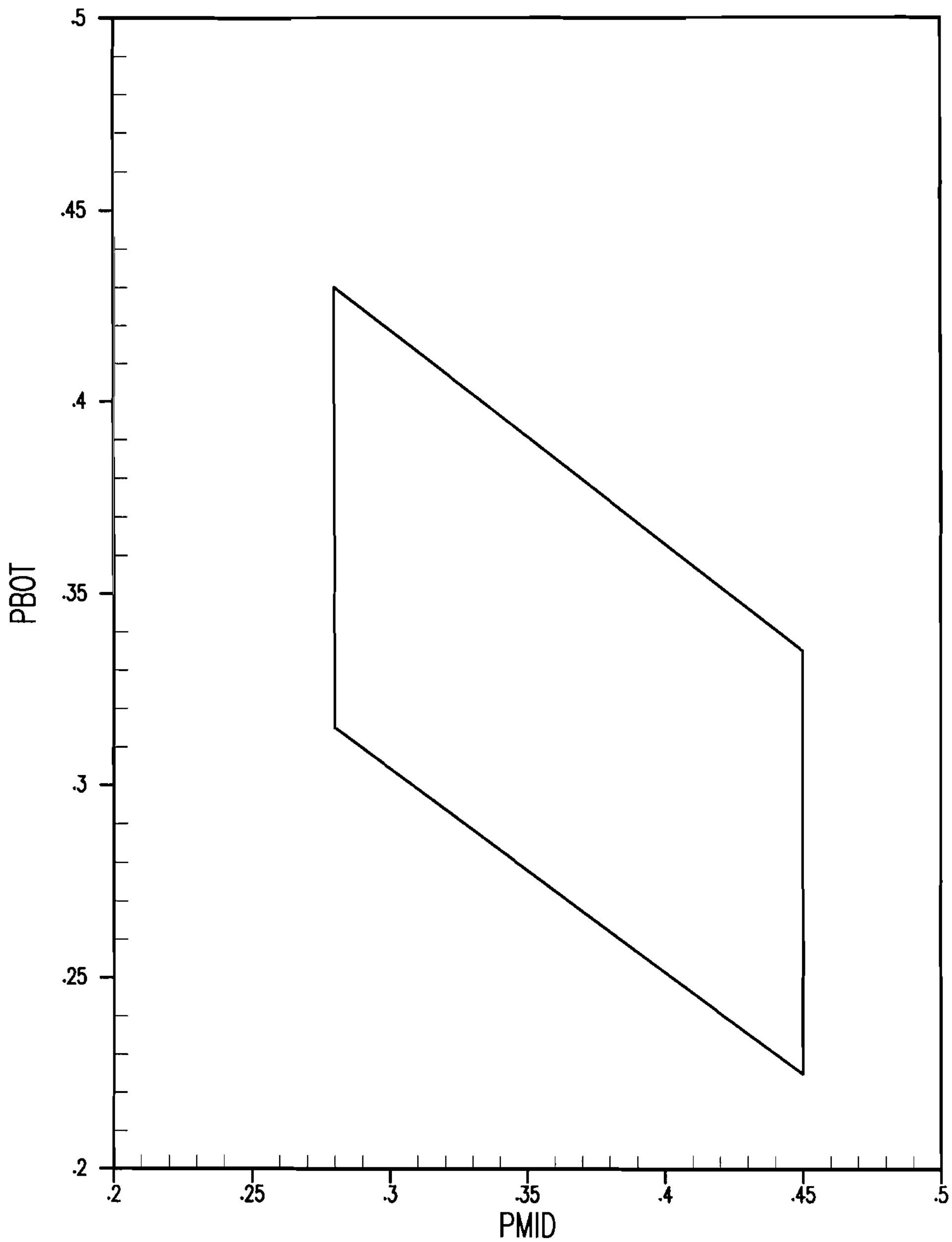
10 CFR 50.46 Requirement	Value	Criteria
95/95 PCT (°F)	1,836*	2,200
95/95 LMO (%)	2.9	17.0
95/95 CWO (%)	0.22	1.00
Coolable Geometry	Core remains coolable	
Long Term Cooling	Core remains cool in long term	

PCT - Peak Clad Temperature
LMO - Local Maximum Oxidation
CWO - Core Wide Oxidation

* Separate from the ASTRUM methodology, an evaluation was performed to assess the ECCS performance during the quarterly Residual Heat Removal (RHR) surveillance testing. The assessment concluded that a 25 °F PCT penalty applies to the licensing basis PCT during the testing period. This value will be tracked as a temporary PCT and will apply only during the period of the test.

Figure 1

Farley BELOCA Analysis Axial Power Shape Operating Space Envelope



PBOT: integrated power fraction in the lower third of the core PMID: integrated power fraction in the middle third of the core

Figure 2

Farley BELOCA Analysis PCT v. Effective Break Area Scatter Plot

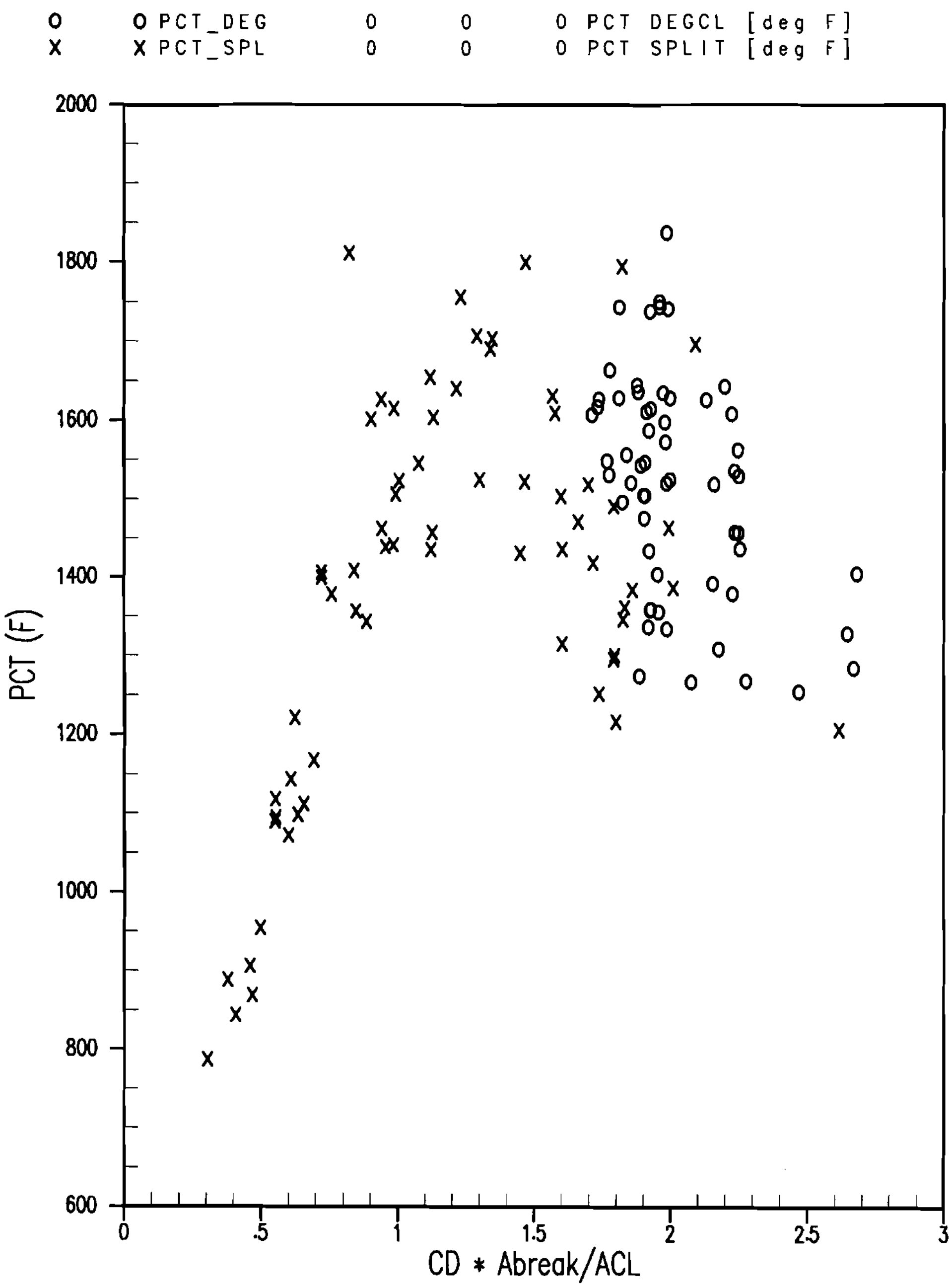


Figure 3

Farley Units 1/2 BELOCA Analysis Clad Temperature Transient at the Limiting Elevation for the Limiting PCT Case (Run 104)

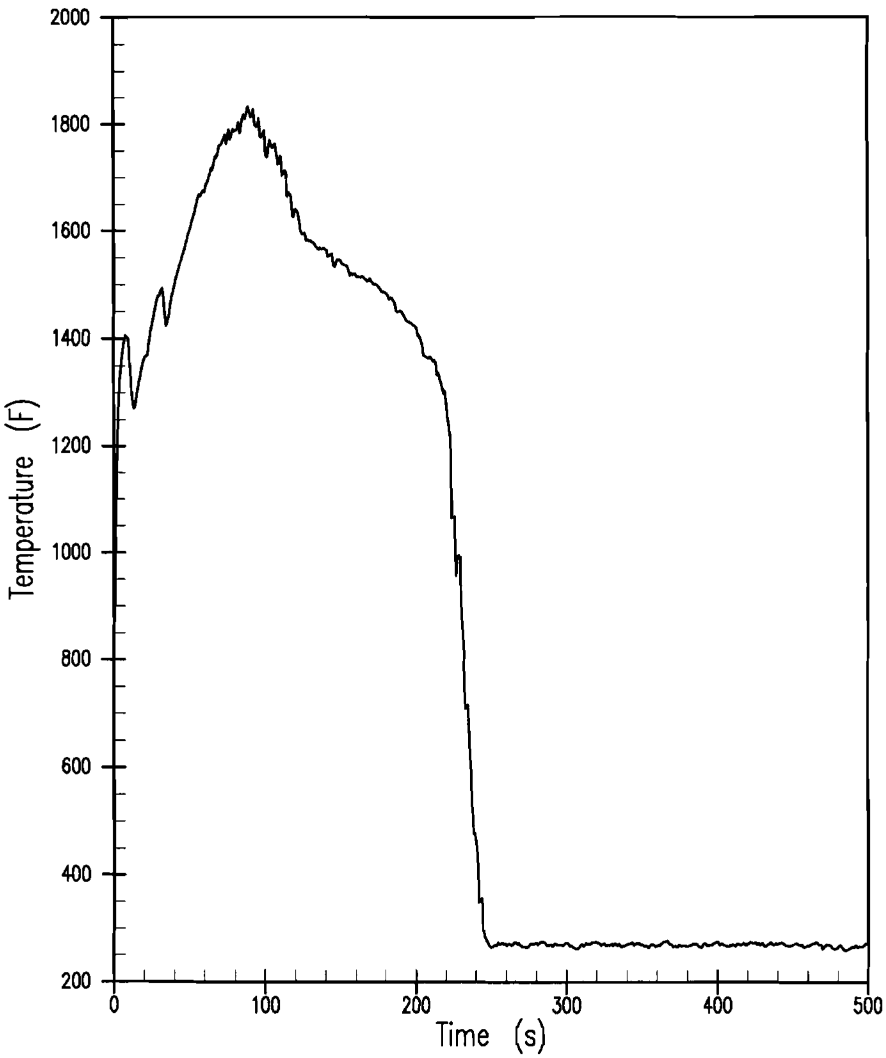


Figure 4

Farley Units 1/2 BELOCA Analysis Clad Temperature Transient at the Limiting Elevation for the Limiting LMO Case (Run 051)

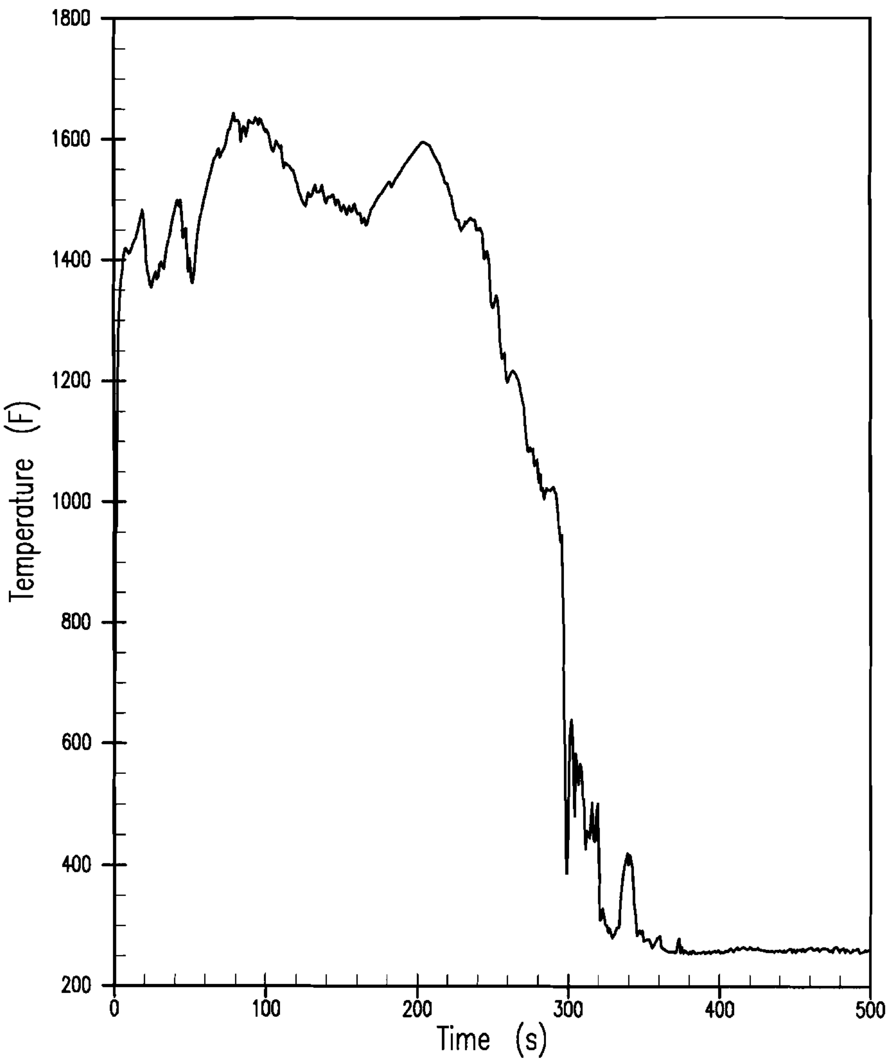
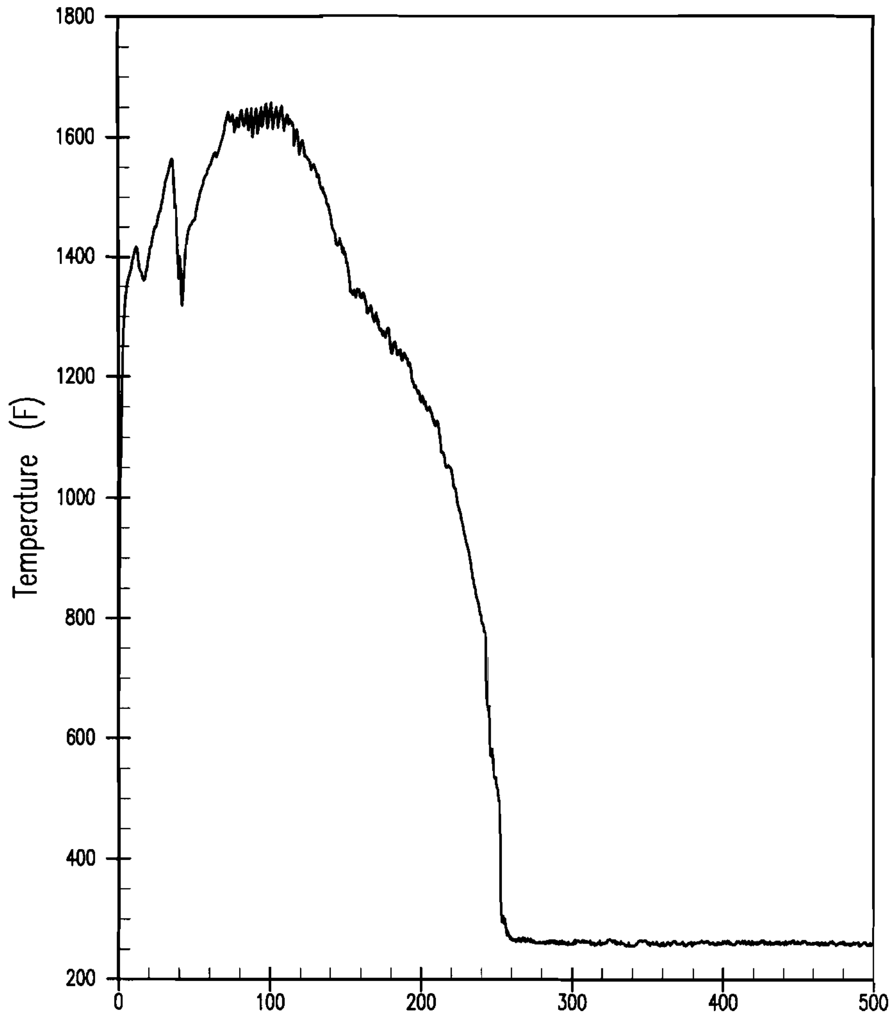


Figure 5

Farley Units 1/2 BELOCA Analysis PCT Transient for the Limiting CWO Case (Run 014)



Joseph M. Farley Nuclear Plant Units 1 and 2
Technical Specification Amendment Request to Incorporate
Best Estimate LOCA Analysis Using ASTRUM

Enclosure 2

Marked-Up Technical Specifications Page

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

3a. WCAP-12945-P-A, Volume 1, Revision 2, and Volumes 2 through 5, Revision 1, "Code Qualification Document for Best Estimate LOCA Analysis," March 1998 (W Proprietary).

3b. WCAP-12610-P-A, "Vantage+ Fuel Assembly Reference Core Report," April 1995 (W Proprietary).

(Methodology for LCO 3.2.1 - Heat Flux Hot Channel Factor and LCO 3.4.1-RCS Pressure, Temperature and Flow Departure from Nucleate Boiling Limits.)

3c. WCAP-16009-P-A, "Realistic Large Break LOCA Evaluation Methodology Using Automated Statistical Treatment of Uncertainty Method (ASTRUM)," M.E. Nissley, et al., January 2005 (Proprietary).

4. WCAP-8745-P-A, "Design Bases for the Thermal Overpower ΔT and Thermal Overtemperature ΔT Trip Functions," September 1986 (Westinghouse Proprietary)

(Methodology for Overpower ΔT and Thermal Overtemperature ΔT Trip Functions)

5. WCAP-14750-P-A Revision 1, "RCS Flow Verification Using Elbow Taps at Westinghouse 3-Loop PWRs. (Westinghouse Proprietary)

(Methodology for minimum RCS flow determination using the elbow tap measurement.)

6. WCAP-11596-P-A, "Qualification of the Phoenix-P/ANC Nuclear Design System for Pressurized Water Reactor Cores," June 1988

(Methodology for LCO 3.9.1 - Boron Concentration.)

7. WCAP-11397-P-A "Revised Thermal Design Procedure," April 1989

(Methodology for LCO 2.1.1-Reactor Core Safety Limits, LCO 3.4.1-RCS Pressure, Temperature and Flow Departure from Nucleate Boiling Limits.)

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

(continued)

Joseph M. Farley Nuclear Plant Units 1 and 2
Technical Specification Amendment Request to Incorporate
Best Estimate LOCA Analysis Using ASTRUM

Enclosure 3

Clean Typed Technical Specifications Pages

Affected Pages

5.6-4
5.6-5
5.6-6

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

3a. WCAP-12945-P-A, Volume 1, Revision 2, and Volumes 2 through 5, Revision 1, "Code Qualification Document for Best Estimate LOCA Analysis," March 1998 (W Proprietary).

3b. WCAP-12610-P-A, "Vantage+ Fuel Assembly Reference Core Report," April 1995 (W Proprietary).

(Methodology for LCO 3.2.1 - Heat Flux Hot Channel Factor and LCO 3.4.1-RCS Pressure, Temperature and Flow Departure from Nucleate Boiling Limits.)

3c. WCAP-16009-P-A, "Realistic Large Break LOCA Evaluation Methodology Using Automated Statistical Treatment of Uncertainty Method (ASTRUM)" M.E. Nissley, et al., January 2005 (Proprietary).

4. WCAP-8745-P-A, "Design Bases for the Thermal Overpower ΔT and Thermal Overtemperature ΔT Trip Functions," September 1986 (Westinghouse Proprietary)

(Methodology for Overpower ΔT and Thermal Overtemperature ΔT Trip Functions)

5. WCAP-14750-P-A Revision 1, "RCS Flow Verification Using Elbow Taps at Westinghouse 3-Loop PWRs. (Westinghouse Proprietary)

(Methodology for minimum RCS flow determination using the elbow tap measurement.)

6. WCAP-11596-P-A, "Qualification of the Phoenix-P/ANC Nuclear Design System for Pressurized Water Reactor Cores," June 1988

(Methodology for LCO 3.9.1 - Boron Concentration.)

7. WCAP-11397-P-A "Revised Thermal Design Procedure," April 1989

(Methodology for LCO 2.1.1-Reactor Core Safety Limits, LCO 3.4.1-RCS Pressure, Temperature and Flow Departure from Nucleate Boiling Limits.)

(continued)

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

5.6.6 Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

a. The reactor coolant system pressure and temperature limits, including heatup and cooldown rates, shall be established and documented in the PTLR for LCO 3.4.3.

b. The analytical methods used to determine the RCS pressure and temperature limits shall be those previously reviewed and approved by the NRC, specifically those described in the NRC letters dated March 31, 1998 and April 3, 1998.

c. The PTLR shall be provided to the NRC upon issuance for each reactor fluence period and for any revision or supplement thereto.

5.6.7 EDG Failure Report

If an individual emergency diesel generator (EDG) experiences four or more valid failures in the last 25 demands, these failures shall be reported within 30 days. Reports on EDG failures shall include a description of the failures, underlying causes, and corrective actions taken per the Emergency Diesel Generator Reliability Monitoring Program.

5.6.8 PAM Report

When a report is required by Condition B or G of LCO 3.3.3, "Post Accident Monitoring (PAM) Instrumentation," a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status.

(continued)

5.6 Reporting Requirements

5.6.9 Tendon Surveillance Report

Any abnormal degradation of the containment structure detected during the tests required by the Pre-stressed Concrete Containment Tendon Surveillance Program shall be reported to the NRC within 30 days. The report shall include a description of the tendon condition, the condition of the concrete (especially at tendon anchorages), the inspection procedures, the tolerances on cracking, and the corrective action taken.

5.6.10 Steam Generator (SG) Tube Inspection Report

A report shall be submitted within 180 days after the initial entry into MODE 4 following completion of an inspection performed in accordance with the Specification 5.5.9, Steam Generator (SG) Program. The report shall include:

a. The scope of inspections performed on each SG,

b. Active degradation mechanisms found,

c. Nondestructive examination techniques utilized for each degradation mechanism,

d. Location, orientation (if linear), and measured sizes (if available) of service induced indications,

e. Number of tubes plugged during the inspection outage for each active degradation mechanism,

f. Total number and percentage of tubes plugged to date, and

g. The results of condition monitoring, including the results of tube pulls and in-situ testing.

5.6.11 Alternate AC (AAC) Source Out of Service Report

The NRC shall be notified if the AAC source is out of service for greater than 10 days.