Docket File

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

February 26, 1996

Mr. D. N. Morey
Vice President, Farley Project
Southern Nuclear Operating Company, Inc.
P. O. Box 1295
Birmingham, AL 35201-1295

SUBJECT: STAFF REVIEW OF THE INDIVIDUAL PLANT EXAMINATION (IPE) SUBMITTAL FOR INTERNAL EVENTS AND FLOODS FOR THE JOSEPH M. FARLEY NUCLEAR PLANTS, UNITS 1 AND 2 (TAC NOS. M74408 AND M74409)

Dear Mr. Morey:

By letter dated June 14, 1993, and supplemented November 9, 1994, you responded to Generic Letter (GL) 88-20, "Individual Plant Examinations for Severe Accident Vulnerabilities," and Supplements 1, 2 and 3, thereto. With the assistance of our contractors, we have completed our review of the IPE submittal for internal events and internal flooding. The evaluation package consists of:

- The Staff Evaluation Report (SER) (Enclosure 1)
- The contractor's Technical Evaluation Reports (TERs) for the front-end, back-end, and human reliability analysis reviews (Enclosures 2, 3, and 4)
- A Summary of the IPE Submittal on Internal Events (Enclosure 5)

The Farley IPE did not identify any severe accident vulnerabilities. We noted that as a result of the IPE, you implemented several procedural enhancements which were reflected in your core damage frequency (CDF) estimate. We also noted that you implemented a number of other procedural enhancements and some significant modifications for which you did not take credit in the IPE or were completed after the freeze date for the IPE model or after the IPE was submitted. Your sensitivity studies indicate that these improvements will reduce the estimated CDF of 1.3E-4/reactor-year by more than 20%.

Based on our review of the Farley IPE submittal and associated documentation, we conclude that you have fully met the intent of Generic Letter 88-20. We commend the procedural and hardware enhancement you have made to improve the ability of the operators and the plants to respond to severe accidents.

Generic Letter 88-20 suggested that licensees could use their IPE submittals to address, among other safety issues, Unresolved Safety Issue (USI) A-45, "Shutdown Decay Heat Removal Requirements" and USI A-17, "Systems Interactions in Nuclear Power Plants." As discussed in the front-end TER, these two issues are adequately resolved for the Joseph M. Farley Nuclear Plants, Units 1 and 2.

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If you have any questions regarding this staff SER, please call me at (301) 415-1463.

Sincerely,

Original signed by:

Byron Siegel, Senior Project Manager
Project Directorate II-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket Nos. 50-348 and
50-364

Enclosures:
1. Staff Evaluation Report
2. TER (Front-End)
3. TER (Back-End)
4. TER (Human Reliability Analysis)
5. Summary of IPE Submittal

cc w/encl 1: See next page

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Sincerely,

Byron L Siegel

Byron Siegel, Senior Project Manager
Project Directorate II-2
Division of Reactor Projects - I/II
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Mr. D. N. Morey
Southern Nuclear Operating
Company, Inc.

Joseph M. Farley Nuclear Plant

cc:
Mr. R. D. Hill, Jr.
General Manager -
Southern Nuclear Operating Company
Post Office Box 470
Ashford, Alabama 36312

Mr. Mark Ajluni, Licensing Manager
Southern Nuclear Operating Company
Post Office Box 1295
Birmingham, Alabama 35201-1295

Mr. M. Stanford Blanton
Balch and Bingham Law Firm
Post Office Box 306
1710 Sixth Avenue North
Birmingham, Alabama 35201

Mr. J. D. Woodard
Executive Vice President
Southern Nuclear Operating Company
Post Office Box 1295
Birmingham, Alabama 35201

State Health Officer
Alabama Department of Public Health
434 Monroe Street
Montgomery, Alabama 36130-1701

Chairman
Houston County Commission
Post Office Box 6406
Dothan, Alabama 36302

Regional Administrator, Region II
U.S. Nuclear Regulatory Commission
101 Marietta Street, NW., Suite 2900
Atlanta, Georgia 30323

Resident Inspector
U.S. Nuclear Regulatory Commission
7388 N. State Highway 95
Columbia, Alabama 36319

ENCLOSURE 1

FARLEY NUCLEAR POWER PLANT

INDIVIDUAL PLANT EXAMINATION

STAFF EVALUATION REPORT

I. INTRODUCTION

On June 14, 1993, Southern Nuclear Operating Company submitted the Farley Nuclear Plant (FNP) Individual Plant Examination (IPE) in response to Generic Letter (GL) 88-20 and associated supplements. On August 12, 1994, the staff sent a request for additional information to the licensee. The licensee responded in a letter dated November 9, 1994.

A "Step 1" review of the FNP IPE submittal was performed and involved the efforts of Science & Engineering Associates, Inc., Scientech, Inc., and Concord Associates in the front-end, back-end, and human reliability analysis (HRA), respectively. The Step 1 review focused on whether the licensee's method was capable of identifying vulnerabilities. Therefore, the review considered (1) the completeness of the information and (2) the reasonableness of the results given the FNP design, operation, and history. A more detailed "Step 2" review was not performed for this IPE submittal. A summary of contractors' findings is provided below. Details of the contractors' findings are in the attached technical evaluation reports (Enclosures 2, 3, and 4) of this staff evaluation report (SER). A summary of the IPE submittal on Internal Events is contained in Enclosure 5.

In accordance with GL 88-20, FNP proposed to resolve Unresolved Safety Issue (USI) A-45, "Shutdown Decay Heat Removal Requirements," and USI A-17, "Systems Interactions in Nuclear Power Plants." No other specific USIs or generic safety issues were proposed for resolution as part of the FNP IPE.

II. EVALUATION

FNP is a Westinghouse 3-loop PWR with a large dry containment. The FNP IPE has estimated a core damage frequency (CDF) of 1.3E-4/reactor-year from internally initiated events, including the contribution from internal floods. The FNP CDF compares reasonably with that of other Westinghouse 3-loop PWR plants. Reactor coolant pump (RCP) seal loss of coolant accident (LOCA) contributes 47%, loss of heat sink 25%, LOCAs 19%, and station blackout 9%. The important system/equipment contributors to the estimated CDF that appear in the top sequences are service water (SW), 4160 V AC buses, component cooling water (CCW), emergency core cooling system (ECCS) recirculation, and ECCS injection. The licensee's Level 1 analysis appeared to have examined the significant initiating events and dominant accident sequences.

Based on the licensee's IPE process used to search for decay heat removal (DHR) vulnerabilities, and review of FNP plant-specific features, the staff finds the licensee's DHR evaluation consistent with the intent of the USI A-45 (Decay Heat Removal Reliability) resolution.

The licensee performed a HRA to document and quantify potential failures in human-system interactions and to quantify human-initiated recovery of failure events. The licensee identified the following operator actions as important in the estimate of the CDF: failure to restore SW and/or CCW within 20 minutes, failure to trip the RCPs upon loss of on-service CCW water train, failure to establish steam generator (SG) feed to 2 of 3 SGs, failure to establish feed and bleed, and failure to establish containment spray recirculation. The staff concluded, however, that there were limitations in the HRA approach used by the licensee. First, the staff concluded that human errors related to calibration of equipment were not appropriately treated in the HRA. Although it is unlikely that the omission of calibration errors critically impacts the licensee's overall conclusions from the IPE, the licensee may have missed the opportunity to identify potential contributors to plant safety.

Second, many of the post-initiator human actions were quantified using the Westinghouse application of the Techniques for Human Error Rate Prediction (THERP) methodology. The NRC has reviewed this application of THERP and have identified numerous inherent weaknesses. These weaknesses include: the limited and incorrect consideration of an operator's need to diagnose before performing an action (especially actions beyond the emergency operating procedures (EOPs)); the limited consideration of the time available to an operator to perform an action vs the time needed; and the limited incorporation of plant-specific factors. In addition to these inherent limitations in the methodology used by the licensee, the staff concluded that plant-specific factors and experiences were not adequately factored in estimating human error probabilities related to routine human actions, especially with calibration, and recovery of equipment. Further, the staff notes that the licensee took credit for recovery actions that are not part of the EOPs or abnormal operating procedures (AOPs) (NUREG-1335 specifically requested the licensees not to take credit for actions that are not proceduralized). Regardless of these limitations, however, it appears that the licensee in their systemic examination gained an understanding of the quantitative impact of human performance on core damage and radioactive material release frequencies such that a potential vulnerability was not overlooked.

The licensee's back-end phenomenological uncertainties were addressed qualitatively in the FNP IPE submittal and were based on technical position papers generated by Fauske and Associates. The licensee's back-end analysis appeared to have considered important severe accident phenomena. Among the FNP conditional containment failure probabilities; early containment failure is 0%; late containment failures (within 48 hours) is 3% with LOCA being the primary contributor, and bypass is 0.4% with steam generator tube rupture (SGTR) being the primary contributor. The containment remains intact about 96% of the time. Early radiological releases are dominated by SGTR sequences and late releases are dominated by LOCA sequences. The licensee's response to containment performance improvement program recommendations is consistent with the intent of GL 88-20 and associated Supplement 3.

Some insights and unique plant safety features identified at FNP are:

1. RCP seal LOCA is important because of the use of charging pumps for high head ECCS injection which uses CCW for pump cooling.

2. Diesel driven firewater can be used for cooling the charging pumps.

3. There is no automatic alignment of ECCS from injection to recirculation.

4. Three of five diesel generators (DGs) are swing DGs for powering either unit.

5. The reactor cavity and instrument tunnel would provide an effective barrier to debris dispersal from the cavity following a high-pressure vessel blowdown.

6. The containment design would not facilitate flooding of the reactor cavity, and therefore, would reduce core concrete interaction.

7. The containment remains intact after core damage (>48 hours) to allow airborne fission products to be removed via the natural deposition process.

The licensee used the guidance in NUMARC 91-04 to screen for plant-specific vulnerabilities. In summary, the licensee used 1E-4/year or greater than 50% of CDF for a core damage sequence, and 1E-5/year or greater than 20% of CDF for a containment bypass sequence. Based on these guidelines, the licensee did not identify any severe accident vulnerabilities. Plant improvements, however, were identified. These improvements, listed below, were under consideration for implementation:

(1) Align charging pump suction to the reactor water storage tank and isolating RCP seal return flow upon a loss of cooling to the on-service CCW train.

(2) Align plant fire protection water to provide charging pump cooling if CCW cooling is lost and cannot be recovered.

(3) Align the swing CCW pump to the standby train mechanically while it is powered from the opposite train to maintain seal injection flow on a loss of SW in the on-service train combined with failure of the standby CCW pump.

(4) Operate one SW pump by reducing SW system loads to maintain CCW cooling.

(5) Realign the ECCS to the cold leg recirculation alignment following failure to establish hot leg recirculation.

(6) Verify that major loads on the engineered safety feature buses have been shed prior to aligning a backup diesel to the bus on a single unit loss of offsite power.

(7) Replace the current RCP seal O-rings with new high temperature O-rings during the next scheduled seal maintenance on each RCP.

III. CONCLUSION

Based on the above findings, the staff notes that: (1) the licensee's IPE is complete with regards to the information requested by GL 88-20 (and associated guidance NUREG-1335), and (2) the IPE results are reasonable given the FNP design, operation, and history. As a result, the staff concludes that the licensee's IPE process is capable of identifying the most likely severe accidents and severe accident vulnerabilities, and therefore, that the FNP IPE has met the intent of GL 88-20.

It should be noted, that the staff's review primarily focused on the licensee's ability to examine FNP for severe accident vulnerabilities. Although certain aspects of the IPE were explored in more detail than others, the review is not intended to validate the accuracy of the licensee's detailed findings (or quantification estimates) that stemmed from the examination. Therefore, this SER does not constitute NRC approval or endorsement of any IPE material for purposes other than those associated with meeting the intent of GL 88-20. However, because the licensee intends to continue to use and maintain its probabilistic safety assessment (PSA), the staff encourages the licensee to improve the Farley IPE/PSA in order to make it a valuable tool for other applications. Without the improvements the staff believes that the Farley IPE/PSA will be limited in regards to future regulatory uses.

ENCLOSURE 2

FARLEY NUCLEAR POWER PLANT

INDIVIDUAL PLANT EXAMINATION

TECHNICAL EVALUATION REPORT

(FRONT-END)