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

March 23, 1994

Dockets 50-348
and 50-364

Mr. D. N. Morey, Vice President
Southern Nuclear Operating Co., Inc.
Post Office Box 1295
Birmingham, Alabama 35201-1295

Dear Mr. Morey:

SUBJECT: RESPONSE TO THE STAFF'S REQUEST FOR ADDITIONAL INFORMATION ON NUREG-0737 ITEM II.D.1 - JOSEPH M. FARLEY NUCLEAR PLANT, UNITS 1 AND 2 (TAC NOS. M84666 AND M84667)

The staff's Safety Evaluation of Item II.D.1 of NUREG-0737, "Completion of Review of Item II.D.1, NUREG-0737, Safety and Relief Valve Testing for Joseph M. Farley Nuclear Power Plant, Units 1 and 2," dated December 16, 1986, identified a concern with the Southern Nuclear Operating Company's (SNC's) evaluation of the piping downstream of the pressurizer power-operated relief valve (PORV) and safety valves for the loop seal discharge transient. Specifically, the Safety Evaluation noted that certain sections of the pipe would be overstressed during the discharge transient, and that SNC had not committed to modifying the piping to conform with the design code requirements. The staff reviewed your August 7, 1992, response and determined that it was not adequate to close this issue. A request for additional information was transmitted to SNC on May 24, 1993, and your response to this request was provide on October 12, 1993.

The staff's review of the October 12, 1993, response has identified several technical deficiencies which are described in the enclosure. While most licensees have either eliminated the cold loop seal or modified the discharge piping to take the transient loads, SNC has attempted to justify the existing design using a non-linear analysis method which is not normally used for piping system analysis. In addition, you have not followed the load combination guidelines developed by the Electric Power Research Institute (EPRI), and accepted by the staff, for the resolution of this issue. Further, the method you used was not technically appropriate for combining responses from the non-linear analysis. As a result, the staff has determined that your proposed approach is not acceptable.

It is requested that you review the deficiencies identified by the staff in the enclosed Request for Additional Information, and, within 45 days of receipt of this letter, provide a schedule for either the re-evaluation of this piping which addresses the staff's concerns, commits to the elimination of the cold loop seal, or commits to modification of the discharge piping to withstand the loads.

The reporting and/or recordkeeping requirements contained in this letter affect fewer than ten respondents; therefore, OMB clearance is not required under P.L. 96-511.

If you have any questions please call me at (301) 504-1463.

Sincerely,

FOR

Byron L. Siegel, Senior Project Manager
Project Directorate II-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

cc w/enclosure: See next page

Enclosure:
Request for Additional Information

Mr. D. N. Morey
Southern Nuclear Operating
Company, Inc.

Joseph M. Farley Nuclear Plant

cc:

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REQUEST FOR ADDITIONAL INFORMATION
REGARDING THE TMI ACTION ITEM II.D.1 OF NUREG-0737
JOSEPH M. FARLEY NUCLEAR PLANT, UNITS 1 AND 2

1. The August 7, 1992, submittal identified that the computer program WECAN was used to perform an elastic/plastic analysis of the pressurizer safety valve piping system. Since an elastic/plastic analysis is not normally performed for piping, the staff requested the licensee to provide a description of the benchmark procedure for this computer program which included the comparisons made between the results of this program and actual test data. The staff also requested that the description of these comparisons include the relevance of the benchmark data to the analysis of the safety valve discharge piping analysis. The licensee's response, dated October 12, 1993, in part, was that benchmarking WECAN against the specific safety valve discharge test is not required because the thermal hydraulic forcing function code was benchmarked against the Electric Power Research Institute (EPRI) test data. The staff has determined that benchmarking of the thermal hydraulic code does not provide any insight regarding the adequacy of the WECAN code to predict the piping response. The purpose of the request was to enable the staff to assess whether the WECAN elastic/plastic analysis of the piping system produces conservative results when predicting piping component strains and pipe support loads for the safety valve discharge event. Therefore, the licensee's October 12, 1993, response was not adequate. The staff again requests that the licensee provide a more complete description of the benchmark procedure for the WECAN computer program, including the comparisons made between the results of this program and actual test data, and that the description of these comparisons include the relevance of the benchmark data to the analysis of the safety valve discharge piping analysis.

2. The staff initially asked the licensee to explain how the emergency load limits specified in Tables 3 and 4 of its August 7, 1992, submittal were met. The licensee's October 12, 1993, response was that the non-nuclear safety (NNS) piping downstream of the valves does not meet the criteria for the emergency condition specified in Table 4. The licensee argues that, from an overall safety standpoint, the analysis contains many conservative assumptions. However, a safety valve discharge event is normally considered a plant upset condition which would result in the use of more conservative design limits than the limits specified in Table 4. The licensee's response is not considered adequate. The staff again requests that the licensee provide a more complete explanation of how the emergency load limits specified in Tables 3 and 4 of its August 7, 1992, submittal were met.

3. The August 7, 1992, submittal, referenced the American Society of Mechanical Engineers (ASME) Code Case N-47 as the basis for the allowable strains. Since this Code Case has not been endorsed in Regulatory Guide 1.84, the staff requested the licensee to explain how the criteria used in the elastic/plastic analysis meets the applicable code criteria that has been endorsed by the regulations or other staff guidance. The licensee's response was that only a few locations in the NNS piping do not meet the emergency condition allowable stress. In

addition, the licensee cited the results of several piping system and component tests as evidence of the conservatism of piping system behavior under dynamic loads. The industry has used these test results, as well as other similar tests, to argue that the criteria for seismic loads is too conservative. However, testing of piping systems has also been performed by EPRI for water hammer type loads. As a result of these tests, EPRI concluded that piping systems can collapse under slug type waterhammer loads. The industry has not used this test data to argue that the criteria for slug type waterhammer loads is overly conservative. The loop seal discharge causes a similar type of slug flow loading condition. The relevant test data demonstrates that a piping system collapse due to slug flow conditions is a credible concern. The licensee's response is not considered adequate. The staff again requests that the licensee provide a more complete explanation of how the criteria meets the applicable code criteria that has been endorsed by the regulations or other staff guidance.

4. The August 7, 1992, submittal, stated that the valve thrust loads and the safe shutdown earthquake (SSE) loads were combined by the square-root-of-the-sum-of-the-squares (SRSS) method. The staff questioned this procedure because there is no technical basis for combining responses based on an elastic analysis with the results of an elastic/plastic analysis. The licensee's justification, in part, was that the magnitude of the SSE was small in comparison to the valve thrust loads. However, because the piping analysis was already inelastic for the valve discharge event, the addition of SSE loads could lead to a significant change in the results for both the piping and the supports. There is no technically defensible method to combine responses from inelastic analyses. The licensee's response to this issue is not adequate. The staff requests that the licensee provide a technically defensible method of combining the loads from the different events.

5. The August 7, 1992, submittal, identified a factor of safety of 1.3 that was used in lieu of 4.0 for concrete expansion anchor bolts. The licensee's response is that the lower factors of safety were used for the NNS piping. The staff does not consider the factor of safety proposed by the licensee for the NNS pipe support anchor bolts to provide an adequate margin of safety. The staff requests that the licensee use a factor of safety that has been endorsed by the staff for concrete anchor bolts.

The reporting and/or recordkeeping requirements contained in this letter affect fewer than ten respondents; therefore, OMB clearance is not required under P.L. 96-511.

If you have any questions please call me at (301) 504-1463.

Sincerely,

ORIGINAL SIGNED BY C. E. CARPENTER FOR:

Byron L. Siegel, Senior Project Manager
Project Directorate II-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

cc w/enclosure: See next page

Enclosure:
Request for Additional Information

DISTRIBUTION

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cc: Farley Service List

OFFICE	LA:PD21:NRR	PE:PD21:NRR	PM:PD21:NRR	D:PD21:NRR
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