

July 2, 1999

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

SUBJECT: JOSEPH M. FARLEY NUCLEAR PLANT, UNITS 1 AND 2 RE: REQUEST FOR
ADDITIONAL INFORMATION — STEAM GENERATOR REPLACEMENT
TECHNICAL SPECIFICATIONS CHANGE REQUEST (TAC NOS. MA4393 AND
MA4394)

Dear Mr. Morey:

Your December 1, 1998, and April 21, 1999, letters requested an amendment to the Farley Nuclear Plant, Units 1 and 2 Technical Specifications (TS). The amendment would address TS changes associated with replacing the existing Westinghouse Model 51 steam generators with Westinghouse Model 54F generators. We need additional information, as discussed in the Enclosure, in order to complete our review of your request. Please send us this information within 30 days of the date of this letter. I discussed this request with Mark Ajluni on June 29, 1999, and we mutually established this response date. Please call me at (301) 415-1423 if you need to revise the date.

Sincerely,

Original signed by:

L. Mark Padovan, Project Manager, Section 1
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

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Docket Nos. 50-348 and 50-364

Enclosure: Request for Additional Information

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UNITED STATES
NUCLEAR REGULATORY COMMISSION

WASHINGTON, D.C. 20555-0001

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Sincerely,

A handwritten signature in black ink, appearing to read "L. Mark Padovan".

L. Mark Padovan, Project Manager, Section 1
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-348 and 50-364

Enclosure: Request for Additional Information

cc w/encl: See next page

Joseph M. Farley Nuclear Plant

cc:

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Resident Inspector
U.S. Nuclear Regulatory Commission
7388 N. State Highway 95
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REQUEST FOR ADDITIONAL INFORMATION

STEAM GENERATOR REPLACEMENT TECHNICAL SPECIFICATIONS CHANGE REQUEST

JOSEPH M. FARLEY NUCLEAR PLANT, UNITS 1 AND 2

1. By letter dated February 11, 1999, WCAP-14882-P, "RETRAN-02 Modeling and Qualification Westinghouse Pressurized Water Reactor Non-LOCA [loss-of-coolant-accident] Safety Analysis," was accepted for referencing in licensing applications to the extent specified and under the limitations delineated in the report and in the associated NRC Safety Evaluation. Please address each of the conditions delineated in the report and in the conclusion section of the NRC's Safety Evaluation for WCAP-14882-P.
2. Please provide an electronic copy of the input deck used in the RETRAN-02 analyses of non-LOCA transients performed in support of the steam generator (SG) replacement.
3. We understand that departure from nucleate boiling ratio (DNBR) was evaluated in RETRAN using a partial derivative method as discussed in WCAP-14882-P. Provide values for the partial derivatives used and justify that these values are conservative for DNBR analysis of Farley.
4. In Section 2.1.2.1, "LOCA Forces", provide a description how Leak Before Break was applied to generate the LOCA forces.
5. In Section 2.1.2.2.1, "Method of Analysis:"
 - a. Provide verification that the damping used in the time-history seismic analysis was based on that specified in Regulatory Guide 1.61.
 - b. Indicate if the seismic analysis of the Reactor Coolant Loop model was performed with all 15 steam generator snubbers removed.
6. In Section 2.1.2.2.4, "RCL Supports," and Table 2.1-4:
 - a. Provide the basis and the values for the Faulted Condition allowable load or stress in compression for the SG columns, the Reactor Coolant Pump (RCP) columns and the RCP tie-rods.
 - b. Provide the largest compressive load acting on SG columns, the RCP columns and the RCP tie-rods.
 - c. For the Reactor Vessel Support Structure, provide the limiting load or stress for the support structure under Faulted Condition compressive loads, in accordance with American Society of Mechanical Engineers (ASME) Section III, Subsection NF and Appendix F.

Enclosure

7. In Section 2.1.2.2.5, "RCL Equipment Nozzle Load Evaluation," provide the comparison of the RCL primary equipment nozzle loads to the umbrella allowable loads given in the equipment design specification.
8. What values did you use in the dose analyses for the reactor coolant system mass and volume?
9. What values did you use in the dose analyses for the mass and/or volume initially in the steam generators?
10. Did you address SG tube uncovering in the locked rotor accident?
11. Regarding your SG replacement containment analyses model, please indicate the key input parameters and assumptions that are different from the parameters and assumptions used in your SG uprate containment analyses model.
12. In your SG replacement containment analyses model, the peak LOCA pressure increased slightly from 43.0 psig to 43.8 psig. However, peak main steam line break pressure decreased from 52.4 psig to 52.0 psig, and peak containment temperature decreased from 383 degrees F to 367 degrees F. Please discuss the reasons for the above changes in pressure and temperature.