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

July 1, 1997

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION RELATED TO POWER UPRATE SUBMITTAL - JOSEPH M. FARLEY NUCLEAR PLANT, UNITS 1 AND 2 (TAC NOS. M98120 AND M98121)

Dear Mr. Morey:

By letter dated February 14, 1997, you submitted a request to amend the Facility Operating Licenses and Technical Specifications (TS) for Farley Nuclear Plant (Farley), Units 1 and 2, to allow for an increase in the licensed thermal power from 2652 MWt to 2775 MWt. Included with your submittal was Westinghouse Nonproprietary Class 3 Report WCAP-14723, "Farley Nuclear Plant Units 1 and 2, Power Uprate Project NSSS Licensing Report," dated January 1997, and applicable changes with the respective Farley Units 1 and 2 TSs.

The staff has performed a preliminary review of your submittal and determined that additional information is required. The enclosure identifies the requested additional information needed.

In order to maintain a timely review schedule and meet your requested target date for completion, it is requested that the information be provided within 30 days of receipt of this letter. If you require any clarification regarding this request, please call me at (301) 415-2426.

Sincerely,

ORIGINAL SIGNED BY:

Jacob I. Zimmerman, Project Manager
Project Directorate II-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket Nos. 50-348 and 50-364

Enclosure: Request for Additional Information

cc w/encl: See next page

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

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Division of Reactor Projects - I/II
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cc:

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

REQUEST FOR ADDITIONAL INFORMATION
REGARDING THE SOUTHERN NUCLEAR OPERATING COMPANY, INC.
REQUEST TO AMEND THE OPERATING LICENSES
FOR THE JOSEPH M. FARLEY NUCLEAR PLANT, UNITS 1 AND 2
THERMAL POWER UPRATE REQUEST

General Questions regarding WCAP-14723

1. Please provide a discussion of the adequacy of the primary and secondary overpressure protection given the relative relieving capacity has gone down (relative to rated power). Include a Standard Review Plan Section 5.2.2 analysis.

2. The submittal indicates that the large break loss-of-coolant accident evaluation model is being changed and that selected other new/improved methods will be used. Please give a description of all the other new or improved methods used to support this license amendment and indicate whether they have received staff approval.

3. The submittal mentions the boron injection tank (BIT) in a few different locations. Please indicate the current function of the BIT and/or if there are plans to remove the tank.

4. The specification for the charging pump discharge pressure is being reduced. Are there any circumstances where injection flow would be necessary or beneficial for make-up or boration (i.e., perhaps an ATWS event) at the pressurizer code safety valve relief pressure that would no longer be available with the new specification?

5. Please provide a description of the transition from fuel with zircaloy cladding to Zirlo cladding. The submittal references both the topical reports for the Vantage 5 and the Vantage+ fuel designs. What fuel design will be used and referenced and describe how any transition core effects will be evaluated. Please provide references for any NRC approvals related to the use of Zirlo cladding at Farley.

6. Please describe and justify the flow streaming effects that would permit a -1°F bias to the temperature measurements (page 7-10 of topical).

7. The analysis T_{ave} window used is 567.2 - 577.2 °F; however, the allowable window in the technical specifications is larger. Describe why the analysis window is not used in the technical specifications.

8. The submittal indicates that using the lowest reactor coolant system (RCS) flow is always used in the analysis. In some analysis, like the main steamline break, higher flow can be more limiting. Please describe how RCS flow is modeled when higher flow is limiting.

9. Please provide an evaluation of your ability to shut the plant down considering all the changes to the main steam pressure, steam flow, RCS flow, residual heat removal flow, and component cooling water temperatures. On page 4-14 of the topical report, a 50 °F/hr cooldown

Enclosure

rate is assumed. Please evaluate the ability to a achieve this cooldown rate for affected scenarios in the Farley Licensing Basis (i.e., single train cooldown, natural circulation cooldown, etc.).

10. Page 4-20 indicates that the analysis of a partial load rejection caused an oscillating plant response. Please provide greater detail regarding the calculated results and any associated effects. Include details regarding the magnitude and length of time that the oscillations occurred.

11. No methodologies are presented for many evaluations performed in the topical report Chapter 5. Please reference the methodologies used in Chapter 5 of the topical report calculations (i.e., rod drop times, core bypass flows, and flow induced vibration).

12. Please provide a reference for the NRC approval of the use of the Westinghouse revised thermal design procedure at Farley and discuss how the transition core effects will be addressed with this thermal design approach.

13. Is Southern Nuclear requesting staff approval of the moderator temperature coefficient limit curve presented in Chapter 7 of the submittal (Figure 7.2-1) or merely showing the currently approved limit curve?

14. Chapter 7.3 presents a number of fuel rod design acceptance limits. For each, please describe where the limit is derived or referenced and if the limit has been accepted by the NRC generically or for Farley specifically.

15. Please verify that the fluence value used to support the technical specification pressure/temperature limit curves (effective through 16 and 14 effective full power years for Units 1 and 2, respectively) will not be exceeded at the higher full power limit.

Questions Regarding Compliance with 10 CFR Part 50, Appendix G, and 10 CFR Part 50, Appendix H

1. Provide the projected maximum end-of-life (EOL) fluences at the inner diameter of the Joseph M. Farley Nuclear Plant (Farley) reactor pressure vessels (RPVs) based on the new uprated power conditions and the revised adjusted reference temperature values for the Farley Units 1 and 2 RPV beltline materials.

2. Provide an assessment of how the proposed power uprate will affect the current pressure-temperature (P-T) limit curves in the Farley Unit 1 and

Unit 2 technical specifications.[1] If the uprated power conditions will change (increase) the adjusted reference temperatures for the most limiting beltline materials in the Farley RPVs, new P-T limit curves should be submitted based on the new uprated conditions and fluences.

3. Provide an assessment of how the proposed thermal uprate will affect the EOL upper-shelf energies for the Farley Units 1 and 2 RPV beltline materials. Include appropriate calculations and figures based on the guidelines of Regulatory Guide 1.99, Rev. 2, "Radiation Embrittlement of Reactor Vessel Material," dated May 1988.

4. Will the revised neutron fluences as a result of the uprated conditions affect the surveillance capsule withdrawal schedule for the Farley Units 1 and 2 RPVs?

5. The staff is providing copies of the Pressurized Thermal Shock (PTS) Summary Files and the Upper Shelf Energy (USE) Summary Files for the Farley Units 1 and 2 RPV beltline materials, as obtained from the NRC Reactor Vessel Integrity Database (RVID), Version 2.0.[2] Update the Summary Files to the extent possible based on the most current data for the Farley RPVs, and using the uprated fluence values for the plants. The updated Summary Files may be used to assist you in your responses to Items 1. — 3. listed above.

Questions Regarding Steam Generator Integrity

1. Summarize the results of the assessment that evaluates the effect of the power uprate on (1) the minimum wall thickness of steam generator tubes, (2) the number of steam generator tubes susceptible to anti-vibration bar wear, and (3) susceptibility of the steam generator tubing to various forms of degradation mechanisms.

2. It is not clear to the staff whether the Southern Nuclear Operating Company, Inc. (SNC), has assessed the structural integrity of the Farley steam generator tubing under uprated power conditions in accordance with Regulatory Guide 1.121 methodology. Clarify and provide the basis for your conclusions.

3. Clarify whether SNC has considered performing any additional surveillance methods to monitor for changes in steam generator degradation as a result of the uprated power conditions. Provide the basis for your conclusions.

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[1] The current P-T Limit Curves in the Farley licensing Bases are TS Figures 3.4-2 and 3.4-3 for Farley Unit 1 (based on License Amendment No. 71, dated June 23, 1987) and TS Figures 3.4-2 and 3.4-3 for Farley Unit 2 (based on License Amendment No. 81, dated December 31, 1990).

[2] The NRC has not yet distributed Version 2.0 of the RVID to the industry; however, the data in the PTS Summary Files and USE Summary Files for Farley Units 1 and 2 should not be appreciably different from the data listed in the corresponding Summary Files from RVID Version 1.1.

4. Section 5.7.1 discussed the structural evaluation of steam generator internals. Provide a list of components that were evaluated and results of the evaluation.

5. Section 5.7.3 discussed the fatigue evaluation of U-bends from a fluid vibration viewpoint. It is not clear to the staff whether SNC has evaluated the small radius (rows 1 and 2) U-bends for degradation from stress corrosion cracking. Clarify and provide the basis for your conclusions.

6. Section 5.7.4.2 stated that the power uprate will not significantly affect outside diameter stress corrosion cracking (ODSCC). Clarify which regions of the steam generator tubes were assessed with respect to ODSCC, including whether the power uprate would affect ODSCC at tube support plates. Provide the basis for your conclusions.

7. SNC has implemented voltage-based alternate tube repair criteria in the Technical Specifications for Farley Units 1 and 2. Discuss whether the uprated power conditions would affect the structural and leakage analyses that are recommended in Generic Letter 95-05. Provide the basis for your conclusions.

8. Section 5.7.5 stated that an analysis was performed to revise the F* criteria in the Farley Unit 2 Technical Specifications to bound the best estimate steam generator outlet pressure at 2785 MWt. It is not clear to the staff whether SNC will submit for staff review a license amendment to revise the F* criteria specified in the Farley Unit 2 Technical Specifications. Please clarify.

Questions Regarding Attachment 6, Section 2.15 - Safety-Related Electrical Equipment Qualification

1. Provide the list of required and qualified radiological doses of the individual safety-related electrical equipment before and after power uprate. In your submittal, it is stated that "for safety-related electrical equipment with uprate doses not bounded by the original design basis, radiological doses at uprate conditions were compared against the dose threshold limits used for the individual components or equipment." We believe that the doses should be bounded by the test report values, not by the dose threshold limits. Explain the differences and why your method is acceptable.

2. Furnish composite loss-of-coolant accident/main steamline break containment temperature profiles before and after power uprate case on the same plot that extends to 30 days. Identify where the composite temperature power uprate profiles are not enveloped by the design basis profile.

3. Explain why the (power) uprated temperature that exceeds the existing design basis profile by a few degrees (i.e., 5°F) toward the end of the composite temperature profiles (greater than 30,000 seconds) is acceptable by having enough margin between 70 seconds and 10,000 seconds.

Should the end of the composite temperature profiles be longer or shorter than 30,000 seconds (8.3 hours)?

Questions Regarding Attachment 6, Section 2.20 - Miscellaneous Electrical Reviews

1. Provide the impact of the load, voltage, and short circuit values for power uprate conditions at all levels of the station auxiliary electrical distribution system (i.e., the onsite power system, the main generator, and its step-up transformer).

2. Provide the result of an analysis which was used to conclude that: (1) the bounding steady-state voltages and motor starting voltages remain within acceptable limits, (2) emergency diesel generator loadings are within the design ratings, and (3) there are no impacts on relay trip set points for loss of voltage or degraded grid voltage protective scheme due to power uprate.

3. State what would be the negative impact on the stability of the units by increasing Farley generation to 920 mWe per unit.

4. Clarify the statement, "There is a slight decrease in the margin of stability for limited faults during valley load conditions. Normal system growth offsets the slight decrease in margin of stability within 3 to 5 years." Please elaborate on how the generation increase due to its power uprate will decrease the stability margin, but the stability will improve later on when the system load grows.

Questions Regarding Attachment 6, Section 2 - Balance of Plant Program Description

1. The increase in the probability of turbine overspeed and associated turbine missile production due to plant operations at the proposed uprated power level have not been addressed. Please demonstrate that plant operations at the proposed uprated power level will not increase the probability of turbine overspeed and associated turbine missile production.

2. With regard to spent fuel pool (SFP) decay heat loads and cooling, provide the following information:

    a. The heat load and corresponding peak calculated SFP temperature for each case analyzed.

    b. Is full core offload a general practice for routine refueling? If it is, how many trains of the SFP cooling system will be available/operable prior to refueling operation?

Questions Regarding Attachment 5, Section 6 - NSSS Accident Analyses

1. In order to evaluate the impact of future plant changes, equipment problems, or other issues for the power uprate, please provide the doses for the control room operator, EAB, and LPZ for the five accidents listed

in Attachment 6. Please demonstrate that the doses for the control room operators comply with the regulatory criteria for control room doses given in 10 CFR Part 50, Appendix A, General Design Criterion 19.

2. For all of the accidents listed in Attachment 6, provide the assumptions along with the calculational methodology to support the dose analysis results, i.e., the modeling, assumptions, input data, and results of the dose analysis for each postulated accident should be provided. What power level was used for the accidents listed in Attachment 6. What is the core radionuclide inventory based on. What meteorological data are the X/Q calculations based on.

3. For the Control Rod Ejection Accident, explain why releases from the secondary side were not included in your evaluation.

Additional Questions

1. In regard to Sections 5.1.1 and 5.2.3 of Reference 2, provide the maximum calculated stress and cumulative fatigue usage factor (CUF) at the critical locations of the reactor pressure vessel (RPV) and internal components (such as RPV nozzles, core plates, core barrel, baffle/barrel, and fuel assembly, etc.). Also, provide the allowable Code limits, and the Code and Code edition used in the evaluation for the power uprate. If different from the Code of record, provide the necessary justification.

2. In regard to Section 5.2.2.2 of Reference 2, provide an assessment of flow-induced vibration of the reactor internal components due to power uprate.

3. In reference to Section 5.4 of Reference 2, provide an evaluation of the control rod drive mechanism with regard to the stress and fatigue usage as a result of the power uprate. Also, provide the allowable Code limits for the critical components evaluated, and the Code and Code edition used for the evaluation. If different from the Code of record, justify and reconcile the differences.

4. In reference to Section 5.5 of Reference 2, provide the methodology and assumptions used for evaluating the reactor coolant piping systems for the power uprate. Also, provide the calculated maximum stress, critical locations, allowable stress limits, and the Code and Code edition used in the evaluation for the power uprate. If different from the Code of record, justify and reconcile the differences.

5. Discuss the analytical methodology and assumptions used in evaluating pipe supports, nozzles, penetrations, guides, valves, pumps, heat exchangers, and anchors at the power uprate conditions. Were the analytical computer codes used in the evaluation different from those used in the original design-basis analysis? If so, identify the new codes and provide justification for using the new codes and state how the codes were qualified for such applications.

6. In regard to Section 5.6 of Reference 2, provide a comparison of the design parameters and transients for the reactor coolant pump (RCP) against the power uprate condition. Also, provide the maximum-calculated stress and CUF for the RCP, the allowable Code limits, and the Code and Code edition used in the evaluation for the power uprate. If different from the Code of record, provide a justification.

7. In regard to Section 5.7.1 of Reference 2, provide a comparison of the design parameters and transients for the Farley steam generators (SGs) Model 51 against the power uprate condition. Also, provide the maximum calculated stress and CUF for the SGs vessel shell and nozzles, the allowable Code limits, and the Code and Code edition used in the evaluation for the power uprate. If different from the Code of record, provide a justification.

8. In reference to Section 5.7.3 of Reference 2, provide a detailed evaluation of the flow-induced vibration of the steam generator U-bend tubes due to power uprate regarding the analysis methodology, vibration level, computer codes used in the analysis and the calculated cross flow velocity. Explain why the tube repair would not be required for at least 13.7 years at the proposed power uprate.

9. In regard to Section 5.8 of Reference 2, provide a comparison of the design parameters and transients for the pressurizer against the power uprate condition. Also, provide the maximum calculated stress and CUF at the critical locations (such as surge nozzle, skirt support, spray nozzle, safety and relief nozzle, upper head/upper shell and instrument nozzle) of the pressurizer, the allowable Code limits, and the Code and Code edition used in the evaluation for the power uprate. If different from the Code of record, provide a justification.

10. In reference to Sections 5.1 and 5.2 of Reference 2, provide the methodology, assumptions, and loading combinations used for evaluating the reactor vessel and internal components with regard to the stress and CUF for the power uprate. Were the analytical computer codes used in the evaluation different from those used in the original design-basis analysis? If so, identify the new codes used and provide justification for using the new codes and state how the codes were qualified for such applications.

11. Discuss how the calculated CUFs for the reactor vessel and piping components compared to the CUFs resulting from the actual loading cycles based on the data recorded during plant operation.

12. Discuss the operability of safety-related mechanical components (i.e., valves and pumps) affected by the power uprate to ensure that the performance specifications and technical specification requirements (e.g., flow rate, close and open times) will be met for the proposed power uprate. Confirm that safety-related motor-operated valves (MOVs) will be capable of performing their intended function(s) following the

power uprate including such affected parameters as fluid flow, temperature, pressure and differential pressure, and ambient temperature conditions. Identify mechanical components for which operability at the uprated power level could not be confirmed.

13. In reference to Reference 3, list the balance-of-plant (BOP) piping systems that were evaluated for the power uprate. Discuss the methodology and assumptions used for evaluating BOP piping, components, and pipe supports, nozzles, penetrations, guides, valves, pumps, heat exchangers, and anchorage for pipe supports. Were the analytical computer codes used in the evaluation different from those used in the original design-basis analysis? If so, identify the new codes and provide justification for using the new codes and state how the codes were qualified for such applications.

14. Provide the calculated maximum stresses for the critical BOP piping systems, the allowable limits, the Code of record and Code edition used for the power uprate conditions. If different from the Code of record, justify and reconcile the differences.

15. Discuss the potential for flow-induced vibration in the balance of plant heat exchangers following the power uprate.

REFERENCES

1. Letter, Southern Company to the NRC, "Joseph M. Farley Nuclear Plant Units 1 and 2, License Nos. DPR-58 and DPR-74, Proposed Facility Operating Licenses and Technical Specification Change Request for Power Uprating," dated February 14, 1997, with attachments.

2. Westinghouse Electric Corporation, WCAP-147239, "Joseph M. Farley Nuclear Plant Units 1 and 2 Power Uprate Project NSSS Licensing Report," dated January 1997 (Attachment V to Reference 2).

3. "Farley Nuclear Plant Units 1 and 2, Power Uprate Project BOP Licensing Report" (Attachment 6 to Reference 1).