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SOUTHERN COMPANY
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U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555

Joseph M. Farley Nuclear Plant
Technical Specification Change Request
Pressure Temperature Limits Report

Ladies and Gentlemen:

By letter dated July 23, 1997, Southern Nuclear Operating Company (SNC) submitted a Technical Specification (TS) Change Request associated with the relocation of the reactor coolant system (RCS) pressure-temperature (P-T) limits from the TS to the Pressure Temperature Limits Report (PTLR) in accordance with the guidance provided in Generic Letter (GL) 96-03, Relocation of the Pressure Temperature Limit Curves and Low Temperature Overpressure Protection System Limits. By letter dated November 14, 1997, the NRC issued a Request for Information (RAI) requesting clarification of the methodology used associated with the SNC TS amendment request. Furthermore, discussions with the NRC Staff resulted in changes to the methodology documented in WCAP-14040-NP-A, Revision 2, Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Curves, used to develop the Farley PTLR. In addition, the best estimate copper and nickel values determined by the Combustion Engineering Vessel Owners Group in response to GL 92-01, Revision 1, Supplement 1, have been incorporated in the Farley Nuclear plant reactor vessel integrity analysis.

Enclosure 1 of this letter provides the response to the RAI and the methodology used by SNC to generate the P-T limits and setpoints associated with low temperature overpressure protection. Enclosures 2 and 3 provide the revised PTLR for Farley Units 1 and 2 respectively. Enclosure 4 provides the revised technical specification pages associated with this change. Although there were only minor changes to the technical specification pages submitted with the July 23, 1997 submittal, all technical specification pages associated with this technical specification amendment are included for completeness. Enclosure 5 provides a revised safety analysis. Enclosure 6 provides a revised significant hazards evaluation.

A001/1

If you have any questions, please advise.

Respectfully submitted,

SOUTHERN NUCLEAR OPERATING COMPANY

DM Morey

Dave Morey

Sworn to and subscribed before me this 18th day of December 1997

Sherry Ann Mitchell
Notary Public

My Commission Expires: 12/17/2000

REM/maf:PTLR2NRC.DOC

Enclosures:
1. Response to November 14, 1997 NRC Staff Request for Additional Information
2. Revised Unit 1 Pressure Temperature Limits Report
3. Revised Unit 2 Pressure Temperature Limits Report
4. Technical Specification Pages
5. Safety Analysis
6. Significant Hazards Evaluation

cc: Mr. L. A. Reyes, Region II Administrator
Mr. J. I. Zimmerman, NRR Project Manager
Mr. T. M. Ross, Plant Sr. Resident Inspector
Dr. D. E. Williamson, State Department of Public Health

ENCLOSURE 1

ENCLOSURE 1

NRC STAFF REQUEST FOR ADDITIONAL INFORMATION
PRESSURE TEMPERATURE LIMITS REPORT
FARLEY LICENSE AMENDMENT REQUEST

Question:
The proposed license amendments request approval to use a Pressure/Temperature Limits Report (PTLR) in accordance with Generic Letter 96-03, "Relocation of the Pressure Temperature Limit Curves and Low Temperature Overpressure Protection System Limits." Approval of the license amendments will allow the pressure/temperature (PT) limits to be changed without NRC approval. The proposed Technical Specification (TS) Section 6.9.15 references both the approved topical report, WCAP-14040-NP-A, Revision 2, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limits Curves," and the approved NRC safety evaluation report. However, the approved topical (pg. 3-6) states that the methodology is "applicable only when the pressurizer power-operated relief valves are used for the COMS [cold overpressure mitigating system]." Because the residual heat removal (RHR) relief valves are used for cold overpressure protection at Farley, the topical report is not applicable for developing setpoints for the RHR relief valves. The submittal does not state how the low-temperature overpressure protection (LTOP) setpoints are verified as acceptable. Although the LTOP or RHR setpoints are not being changed with this submittal or being moved to the proposed PTLR, each time the PT limits are revised the LTOP system must be reevaluated to ensure the current setpoints are acceptable for meeting the functional requirements of the system and capable of protecting the new PT limits. Because the approved methodology for determining the setpoints is not applicable to Farley, how will you demonstrate that the RHR setpoints are still acceptable relative to the new PT limits?

Because the approved methodology, which is referenced in the TS, is not applicable to RHR setpoints, please update the submittal and the proposed TS and describe in detail how the approved methodology will be applied. If a different methodology is presented to determine the RHR setpoints, the methodology should be described in detail, explaining how calculations will be performed and how to account for uncertainties and setpoint drift. A discussion of the limiting overpressure transients should be included and how controls are in place and will be maintained to assure the RHR relief valves will protect the plant for all low temperature overpressure transients. If a new methodology is used, it should also be referenced in the TS. Please include sample calculations for the first application (for the proposed PT limits) of the methodology. The required elements of a methodology are contained in the generic letter.

Note: The above request was subsequently revised to include a request to address the seven "Requirements for Methodology and PTLR" included in Generic Letter 96-03.

Southern Nuclear Response: The response to the NRC requests are included in the following "Methodology for Determination of Reactor Coolant System Pressure Temperature Limits and Low Temperature Overpressure Protection System."

JOSEPH M. FARLEY NUCLEAR PLANT

METHODOLOGY FOR DETERMINATION OF REACTOR COOLANT SYSTEM PRESSURE TEMPERATURE LIMITS AND LOW TEMPERATURE OVERPRESSURE PROTECTION SYSTEM

The methodology for determining the reactor coolant system pressure temperature limits includes the determination of low temperature overpressure protection setpoints and is best described by addressing the seven "Requirements for Methodology and PTLR" found in Generic Letter 96-03.

1. **Describe the transport calculation methods including computer codes and formulas used to calculate neutron fluence. Provide references.**

   Section 2.2 of WCAP-14040-NP-A, Revision 2, provides the methodology for determining the neutron fluence for the surveillance capsules and the reactor vessel with the exception that, as requested by the NRC, calculated fluence values ($\phi_{Calc.}$) are used in lieu of best-estimate fluence ($\phi_{Best\ Est.}$) described in WCAP-14040-NP-A, Revision 2.

2. **Briefly describe the surveillance program. Licensee transmittal letter should identify by title and number report containing the Reactor Vessel Surveillance Program and surveillance capsule reports. Topical/generic report contains place holder only. Reference Appendix H to 10 CFR 50.**

   The reactor vessel material surveillance program for Farley Nuclear Plant Unit 1 is described in WCAP-8810, Alabama Power Company Joseph M. Farley Nuclear Plant Unit No. 1 Reactor Vessel Radiation Surveillance Program, dated December 1976. To date, four surveillance capsules have been removed from Farley Nuclear Plant Unit 1 as documented in the following test reports submitted to the NRC in accordance with 10 CFR 50, Appendix H:

   - WCAP-14196, Analysis of Capsule W from the Alabama Power Company Farley Unit 1 Reactor Vessel Radiation Surveillance Program, dated February 1995.
   - WCAP-11563, Revision 1, Analysis of Capsule X from the Alabama Power Company Joseph M. Farley Unit 1 Reactor Vessel Radiation Surveillance Program, dated September 1987.
   - WCAP-10474, Analysis of Capsule U from the Alabama Power Company Joseph M. Farley Unit 1 Reactor Vessel Radiation Surveillance Program, dated February 1984.
   - WCAP-9717, Analysis of Capsule Y from the Alabama Power Company Farley Unit No. 1 Reactor Vessel Radiation Surveillance Program, dated June 1980.

   The reactor vessel material surveillance program for Farley Nuclear Plant Unit 2 is described in WCAP-8956, Alabama Power Company Joseph M. Farley Nuclear Plant Unit No. 2

Reactor Vessel Radiation Surveillance Program, dated August 1977. To date, three surveillance capsules have been removed from Farley Nuclear Plant Unit 2 as documented in the following test reports submitted to the NRC in accordance with 10 CFR 50, Appendix H:

- WCAP-12471, Analysis of Capsule X from the Alabama Power Company Joseph M. Farley Unit 2 Reactor Vessel Radiation Surveillance Program, dated December 1989.
- WCAP-11438, Analysis of Capsule W from the Alabama Power Company Joseph M. Farley Unit 2 Reactor Vessel Radiation Surveillance Program, dated April 1987.
- WCAP-10425, Analysis of Capsule U from the Alabama Power Company Joseph M. Farley Unit 2 Reactor Vessel Radiation Surveillance Program, dated October 1983.

To assure continued compliance with the requirements of 10 CFR 50, Appendix H, Surveillance Requirement 4.4.10.1.2 for Farley Nuclear Plant Units 1 and 2 associated with the P-T limits requires that the reactor vessel material irradiation surveillance specimens be removed and examined in accordance with 10 CFR 50, Appendix H.

3. Describe how the LTOP system limits are calculated applying system/thermal hydraulics and fracture mechanics. Reference SRP Section 5.2.2; ASME Code Case N-514; ASME Code, Appendix G; Section XI as applied in accordance with 10 CFR 50.55.

Farley Nuclear Plant utilizes the residual heat removal system relief valves (RHRRVs) for low temperature overpressure protection (LTOP) of the RCS from brittle fracture by assuring that the limits of Appendix G are not exceeded. The RHRRVs are spring loaded, bellows-type valves which have a setpoint of 450 psig and are designed to provide rated flow at 495 psig (i.e., 10% accumulation). In order to assure that the RHRRVs are available to protect the RCS from an LTOP event, Technical Specification (TS) 3.4.10.3 requires that the RHR suction valves be open and the RHRRVs operable with a lift setting less than or equal to 450 psig or that the RCS be depressurized with a vent of greater than or equal to 2.85 square inches at RCS temperatures less than or equal to 310°F.

The design basis transients for the Farley Nuclear Plant LTOP system consist of a heat input transient and a mass input transient with the RCS in a water-solid condition. The worst-case heat input transient assumes the start of a single reactor coolant pump with a temperature differential of 50°F existing between the RCS and any one steam generator. At RCS temperatures less than or equal to 180°F, the worst-case mass input transient is assumed to be the inadvertent start of one high head safety injection (HHSI) pump with a maximum flow rate of 590 gallons per minute based on the maximum number of operable HHSI pumps allowed by TS 3.1.2.3. For RCS temperatures greater than 180°F, the worst-case mass input transient assumes the inadvertent operation of three HHSI pumps with a maximum total flow rate of 1000 gallons per minute at zero backpressure. These three transients discussed above are utilized to determine the RCS pressure for further analysis.

The Farley Nuclear Plant LTOP analysis consists of a determination of RCS pressures resulting from each of the design basis LTOP transients based on the relief capacity of the RHRRVs and the following conservative assumptions:

- Credit is taken for flow through only one RHRRV due to single failure of the other RHRRV;
- No flow through the RHRRVs is credited in the analysis until RCS pressure achieves the 10% accumulation pressure for the RHRRVs of 495 psig;
- Flashing is assumed to occur at the valve discharge;
- No credit is taken for a bubble in the pressurizer; and
- The analysis is performed at isothermal conditions in the RCS and provides protection against the steady-state Appendix G limit.

At RCS temperatures less than or equal to 180°F, the most-limiting design basis transient results in an RCS pressure of 495 psig. The resulting pressure is compared to the proposed Appendix G steady-state limit curve to assure that the resulting RCS pressure of 495 psig does not exceed the allowable RCS pressure. The following table provides an Example of Comparison of Limiting Design Basis Transient (LDBT) to Appendix G Steady State Limit Curve.

Example of Comparison of Limiting Design Basis Transient to Appendix G Steady State Limit Curve for Farley Unit 2

RCS Temperature (°F)	RCS Pressure (LDBT)(psig)	Appendix G Steady State Limit Curve (psig)
70	495	498
180	495	626
181	562.5	629
260	562.5	1070
261	795	1080
310	795	1749

As stated above, the RCS pressure for each of the above temperatures are compared to the proposed steady-state Appendix G curve to assure that the RCS pressure does not exceed the Appendix G allowable pressure for the corresponding temperature. If this criteria is met, the Farley Nuclear Plant LTOP system provides adequate protection for the proposed Appendix G curves. As can be seen from the above comparison, the Farley Nuclear Plant LTOP system provides adequate protection for the Appendix G curves.

If the projected RCS pressure exceeds the Appendix G allowable pressure for the corresponding temperature, changes to the RHRRV characteristics, e.g., capacity, relief setpoint, accumulation, may be required. These modifications may require a change to TS 3.4.10.3.

The Farley Nuclear Plant LTOP enable temperature is the temperature below which the LTOP system is required to be operable in accordance with Section 3.4 of WCAP-14040-NP-A, Revision 2. The LTOP enable temperature is compared to the RCS cold leg temperature of 310°F stated in the applicability statement of TS 3.4.10.3 to assure the RCS overpressure protection systems are available at temperatures below the LTOP enable temperature.

If 310°F is not an acceptable LTOP enable temperature, a change to Technical Specification 3.4.10.3 will be required.

In order to minimize setpoint uncertainties and drift, Farley Nuclear Plant tests the RHRRVs on an accelerated basis from that required by the ASME Code. Bench tests are performed at 18 month intervals on a rotating basis for at least one of the RHRRVs to verify the setpoint in accordance with TS Surveillance Requirement 4.4.10.3.1(c). This frequency is more stringent than that required by the ASME Code for class 2 relief valves.

Additionally, Farley Nuclear Plant surveillance test procedures currently use an RHR relief valve setpoint of 445±5 psig for the setpoint. This approximately 1% tolerance is more stringent than the ASME Code requirement of 3% tolerance. The use of 1% setpoint tolerance for the RHRRV setpoint coupled with the 10% accumulation provide adequate protection against setpoint drift. The increased surveillance test frequency, the reduced RHRRV setpoint and setpoint tolerance, coupled with the analysis assumption that flow does not start until inlet pressure reaches 450 psig + 10% accumulation, i.e., 495 psig, provide assurance that the RHR relief valves will provide adequate protection against the limits of Appendix G.

ASME Code Case N-514 is not used for Farley calculations.

4. Describe the method for calculating the ART using Regulatory Guide 1.99, Revision 2.

Section 2.4 of WCAP-14040-NP-A, Revision 2, provides the methodology for calculating the adjusted reference temperature in accordance with Regulatory Guide 1.99, Revision 2.

5. Describe the application of fracture mechanics in constructing P-T curves based on ASME Code, Appendix G. Section XI, and SRP Section 5.3.2.

Sections 2.5 and 2.6 of WCAP-14040-NP-A, Revision 2, provides the application of fracture mechanics in constructing P-T curves. The resulting P-T limit curves are adjusted to account for the 60 psi ΔP between the reactor vessel beltline and the RHRRVs associated with the operation of three reactor coolant pumps (RCPs) at RCS temperatures greater than or equal to 110°F. At RCS temperatures less than 110°F, the number of operating RCPs is limited to one and the resulting ΔP correction of 25 psig is applied.

6. **Describe how the minimum temperature requirements in Appendix G to 10 CFR 50 are applied to P-T curves.**

   Section 2.7 of WCAP-14040-NP-A, Revision 2, provides the methodology for determination of the minimum temperature requirements in 10 CFR 50, Appendix G. The minimum temperature requirement is adjusted as necessary to assure the RCS pressure resulting from design basis LTOP transients does not exceed the steady state Appendix G limit.

7. **Describe how the data from multiple surveillance capsules are used in the ART calculation.**

   Section 2.4 of WCAP-14040-NP-A, Revision 2, provides the methodology for calculating the adjusted reference temperature with multiple surveillance capsules.

   Describe procedure if measured value exceeds predicted value.

   As stated in Section 2.4 of WCAP-14040-NP-A, Revision 2, if the measured value exceeds the predicted value, a supplement to the PTLR must be provided to demonstrate how the results affect the approved methodology.

WHEN OTHER PLANT DATA ARE USED

1. **Identify the source(s) of data when other plant data are used.**

   Farley Nuclear Plant does not rely on surveillance data from other licensees for its reactor vessel integrity analysis. Therefore, this item is not applicable to Farley Nuclear Plant.

2a. **Identify by title and number the safety evaluation report that approved the use of data for the plant. Justify applicability.**

   Farley Nuclear Plant does not rely on surveillance data from other licensees for its reactor vessel integrity analysis. Therefore, this item is not applicable to Farley Nuclear Plant.

   OR

2b. **Compare licensee data with other plant data for both the radiation environments (e.g., neutron spectrum, irradiation temperature) and the surveillance test results.**

Farley Nuclear Plant does not rely on surveillance data from other licensees for its reactor vessel integrity analysis. Therefore, this item is not applicable to Farley Nuclear Plant.