Southern Nuclear

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Docket Nos. 50-348 NL-22-0361
50-364

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D. C. 20555-0001

Joseph M. Farley Nuclear Plant – Units 1 and 2
Response to Request for Additional Information Related to License Amendment Request to Revise Technical Specification 5.5.17, "Containment Leakage Rate Testing Program" to Increase Calculated Peak Containment Pressure

By letter dated December 13, 2021 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML21348A733), Southern Nuclear Operating Company (SNC) submitted a license amendment request (LAR) for the Joseph M. Farley Nuclear Plant, Units 1 and 2 (FNP). The LAR proposes to revise the FNP Technical Specification (TS) 5.5.17, "Containment Leakage Rate Testing Program" to update the peak calculated containment internal pressure for the design basis loss of coolant accident (LOCA) from 43.8 psig to 45 psig.

By email dated May 3, 2022, the U.S. Nuclear Regulatory Commission (NRC) notified SNC that additional information is needed for the staff to perform their review.

The enclosure to this letter provides the SNC response to the NRC Request for Additional Information (RAI).

The conclusions of the No Significant Hazards Consideration and Environmental Consideration contained in the original application have been reviewed and are unaffected by this response.

In accordance with 10 CFR 50.91, a copy of this application, including attachments, is being provided to the designated Alabama Official.

This letter contains no regulatory commitments. If you have any questions, please contact Ryan Joyce at 205.992.6468.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 20th day of June 2022.

Respectfully submitted,

C. A. Gayheart
Director, Regulatory Affairs
Southern Nuclear Operating Company

CAG/was/cbg

Enclosure: SNC Response to Request for Additional Information (RAI)

cc: Regional Administrator, Region II
NRR Project Manager – Farley 1 & 2
Senior Resident Inspector – Farley 1 & 2
Alabama – State Health Officer for the Department of Public Health
RType: CFA04.054

ENCLOSURE

Joseph M. Farley Nuclear Plant – Units 1 and 2

Response to Request for Additional Information Related to License Amendment Request to Revise Technical Specification 5.5.17, "Containment Leakage Rate Testing Program" to Increase Calculated Peak Containment Pressure

SNC Response to Request for Additional Information

REQUEST FOR ADDITIONAL INFORMATION

REGARDING CHANGES TO TECHNICAL SPECIFICATION 5.5.17 TO INCREASE

CALCULATED PEAK CONTAINMENT PRESSURE

EPID: L-2021-LLA-0229

LICENSE AMENDMENT REQUEST

JOSEPH M. FARLEY NUCLEAR PLANT, UNITS 1 AND 2

SOUTHERN NUCLEAR OPERATING COMPANY

DOCKET NOS. 50-348 AND 50-364

By letter dated December 13, 2021 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML21348A733), Southern Nuclear Operating Company (the licensee) submitted a license amendment request (LAR) for the Joseph M. Farley Nuclear Plant, Units 1 and 2 (FNP). The LAR proposes to revise the FNP Technical Specifications (TS) 5.5.17, "Containment Leakage Rate Testing Program" to update the peak calculated containment internal pressure for the design basis loss of coolant accident (LOCA) from 43.8 psig to 45 psig. The design limit for the containment internal pressure is 54 psig.

The U.S. Nuclear Regulatory Commission (NRC) staff has determined that the following requests for additional information (RAI) are needed.

RAI 01:

REGULATORY BASIS:

Appendix A to Title 10, Code of Federal Regulations, Part 50 (10 CFR 50), "General Design Criteria for Nuclear Power Plants" (GDC), Criterion 16, "Containment design," states:

> *Reactor containment and associated systems shall be provided to establish an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment and to assure that the containment design conditions important to safety are not exceeded for as long as postulated accident conditions require.*

RAI:

In Section 3.4 of the enclosure to the LAR, the licensee states that containment pressure analyses to determine the limiting LOCA were performed using Version 8.1 of the GOTHIC computer program. The NRC notes that Version 6.0 of GOTHIC code was used in the analysis of record (AOR) as described in the FNP Final Safety Analysis Report (FSAR), Section 6.2. In Section 3.4 of the enclosure to the LAR, the licensee states that "[t]he change in code version does not result in a numerically significant departure from the previous analysis and does not yield a benefit to peak pressure and temperature results."

The AOR and the revised analysis performed in support of this LAR do not utilize the same version of GOTHIC. Since GOTHIC is not an NRC-approved code but is an accepted code for containment response analysis for FNP, the NRC staff requests that the licensee provide data *to document that a verification analysis was performed to justify the assertion that the change in version does not result in a numerically significant departure from the previous analysis. The NRC staff requests that the licensee provide comparison results between the two versions of the GOTHIC code. For example, the licensee could provide a comparison of curves for the pressure and temperature response of the containment for the Double-Ended Hot Leg (DEHL) and Double-Ended Pump Suction (DEPS) cases documented in Figures 1 through 4 in Section 3.4.1 of the enclosure to the LAR.*

SNC Response:

SNC has performed a thorough benchmark between GOTHIC Versions 6.0 and 8.1 utilizing the most limiting case. The benchmark for LOCA was performed ahead of the Nuclear Safety Advisory Letter (NSAL) re-analysis using the analysis of record (AOR) double ended pump suction leg (minimum Safety Injection (SI)) case. The results are seen below in figures reproduced from the benchmark. The figures below were produced using the GOTHIC compare feature and include two traces for each variable included in the graph – one from the 6.0 run and one from the 8.1 run. In addition to comparing the results between the two versions, all code changes made for each version between 6.0 and 8.1 were evaluated for any effect they may have on the results produced by the containment model.

None of the changes which impacted SNC's containment model yielded a numerically significant difference in the reported results. However, one of the code changes did impact the calculation of sump water temperature in the very beginning of the transient. The AOR did not model any initial liquid in the sump and therefore the 8.1 version model is calculating unreasonably high temperatures in the first half second of the transient. This can be alleviated by initializing the model with a small amount of liquid in the sump. This is discussed in Section 3.4 of the LAR as one of the changes SNC made to the containment model as part of the NSAL analysis.

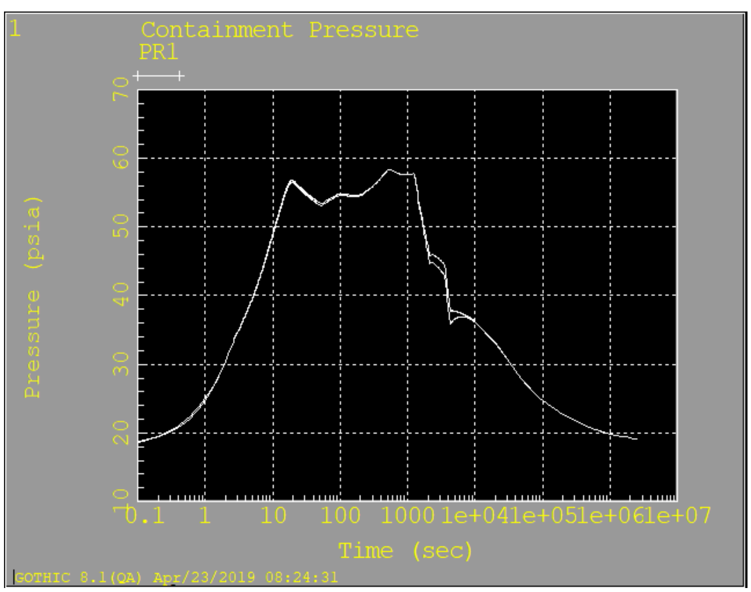

Figure RAI 1-1: Containment Pressure Comparison

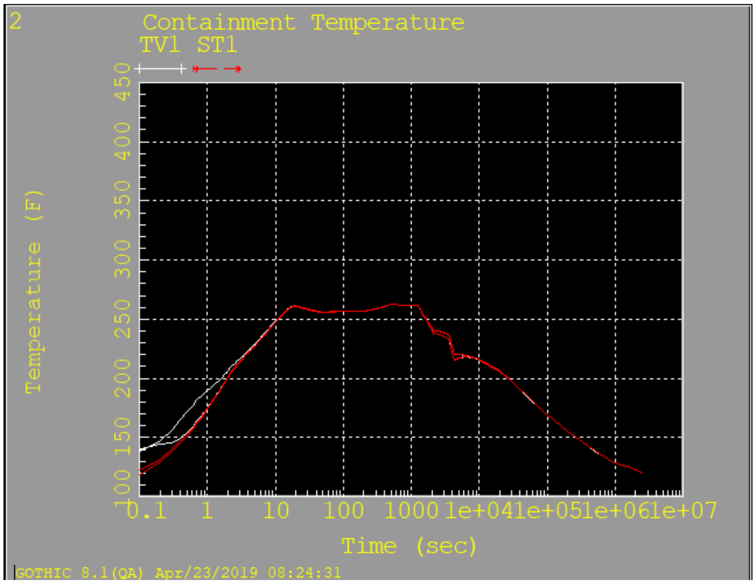

Figure RAI 1-2: Containment Vapor Temperature (white) and Saturation Temperature at the Steam Pressure, Pstm (red)

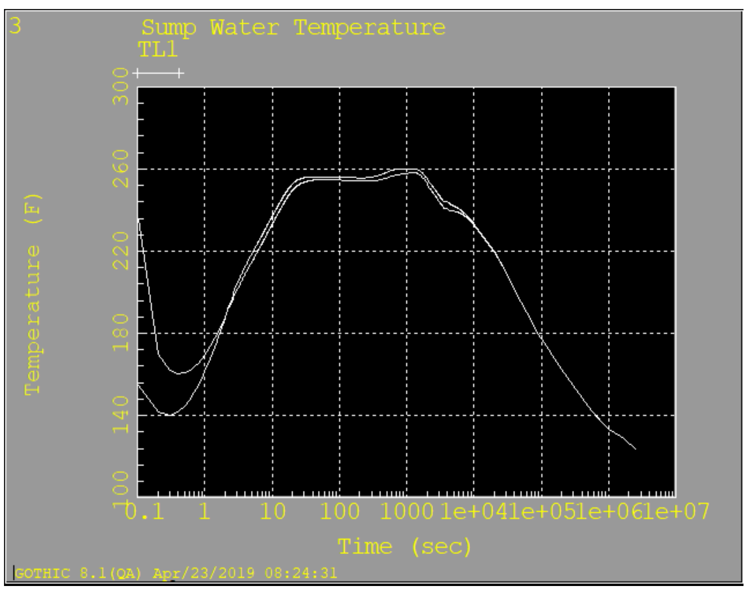

Figure RAI 1-3: Sump Temperature

<u>RAI 02</u>:

REGULATORY BASIS:

Appendix A to 10 CFR 50, GDC, Criterion 16 "Containment design," as stated above.

Appendix A to 10 CFR 50, GDC, Criterion 38 "Containment heat removal" states, in part:

A system to remove heat from the reactor containment shall be provided. The system safety function shall be to reduce rapidly, consistent with the functioning of other associated systems, the containment pressure and temperature following any loss-of-coolant accident and maintain them at acceptably low levels.

RAI:

Section 6.2.1.3.6 of the FNP FSAR states that "[d]uring sump water recirculation, a third pressure peak occurs due to steam evolution from the reactor because of boiloff of the hotter core injection water." The FSAR states that the maximum sump water temperature of 260 degrees Fahrenheit (°F) occurs at 1252 seconds (s). The NRC staff requests the licensee provide the following information to ensure acceptability of the containment heat removal system, consistent with requirements of GDC 38:

(a) Updated sump water temperature profile and the value of the peak water temperature during the LOCA recirculation phase.

SNC Response:

The peak sump temperature for the double ended pump suction (minimum SI) case is 259 °F at 1370 seconds. However, switchover to recirculation occurs at
2,139 seconds, which is after the calculated peak. Below, two graphs of the sump water temperature versus time are presented. The first graph shows the sump water temperature as a log function of time for the entire transient. The second graph is zoomed in to the area of the peak temperature and time of switchover to recirculation as a linear function of time. The highest temperature during recirculation occurs at the beginning of recirculation. This temperature is approximately 253 °F.

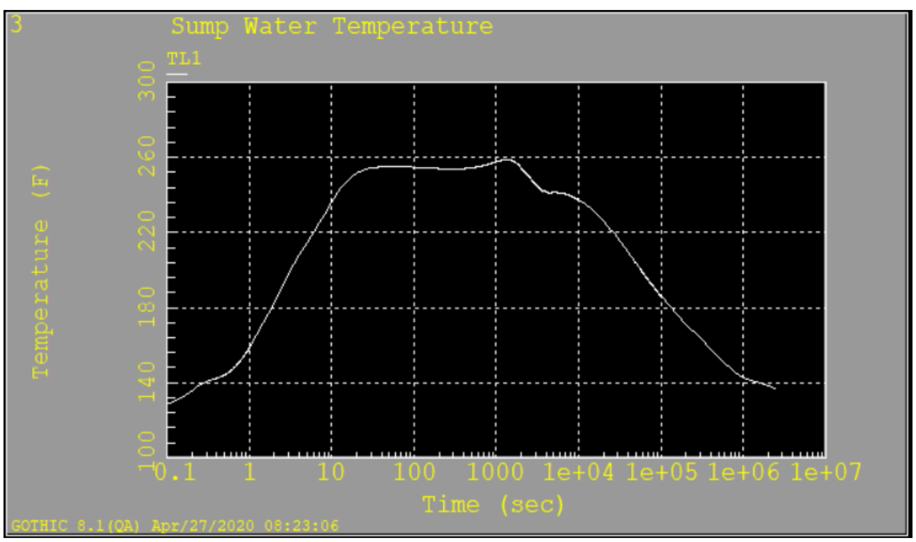

Figure RAI 2-1: Sump Temperature (log function)

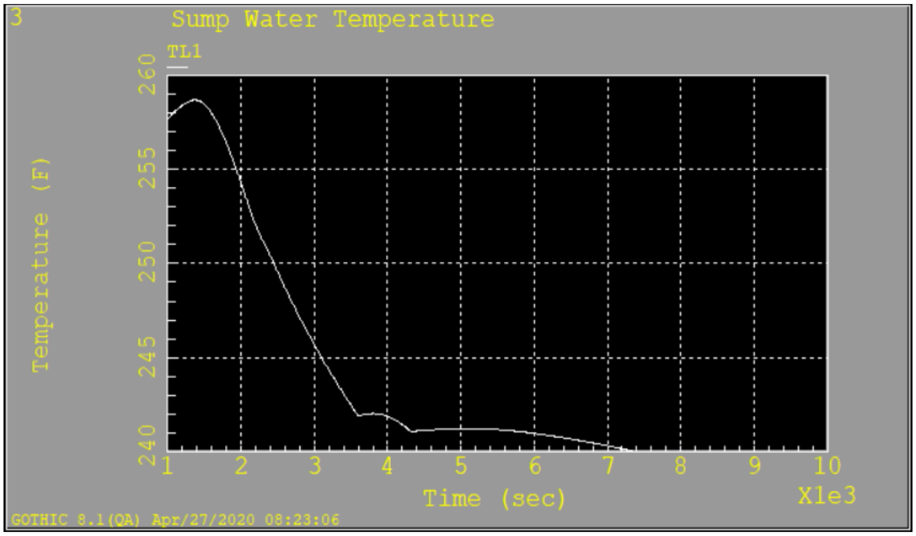

Figure RAI 2-2: Sump Temperature (linear function)

(b) *Minimum net positive suction head (NPSH) available (NPSHA) for the pumps that draw water from the sump during the LOCA recirculation phase.*

SNC Response: See Table RAI 2-1 below.

(c) *In case containment accident pressure (CAP) is used for the minimum NPSHA calculation in (b), confirm that it is less than the vapor pressure at the sump water temperature.*

SNC Response:

SNC does not credit CAP for minimum NPSHA calculations.

(d) *Maximum NPSH required (NPSHR) for the pumps in (b) at their respective applicable flow rates.*

SNC Response: See Table RAI 2-1 below.

(e) *Minimum NPSH margin = NPSHA-NPSHR*

For the items above, please confirm that the analysis reflects the latest sump strainer design installed to meet Generic Safety Issue 191, "Assessment of Debris Accumulation on PWR Sump Performance."

SNC Response:

All NPSH analysis from Table RAI 2-1 uses the latest sump strainer design which was installed to meet GSI-191 performance requirements.

Table RAI 2-1

Pump	**(b) NPSHA (ft)**	**(d) NPSHR (ft)**	**(e) NPSH Margin (NPSHA-NPSHR) (ft)**
U1 “A” Train RHR	20.7	18	2.7
U1 “B” Train RHR	19.6	18	1.6
U1 “A” Train CTMT Spray	20.8	18	2.8
U1 “B” Train CTMT Spray	21.2	18	3.2
U2 “A” Train RHR	19.2	18	1.2
U2 “B” Train RHR	19.3	18	1.3
U2 “A” Train CTMT Spray	20.7	18	2.7
U2 “B” Train CTMT Spray	19.1	18	1.1

Note:

- The minimum NPSH margin occurs in a sump water temperature range of 212-291^{o}F where the minimum values are the same.
- NPSHR is 18 ft for all pumps at flow rates of 4500 gpm for RHR and 3400 gpm for CTMT Spray.