From: Devlin-Gill, Stephanie
Sent: Tuesday, May 3, 2022 10:35 AM
To: Sparkman, Wesley A.
Subject: RE: RAIs for Farley LAR re. TS 5.5.17 (EPID: L-2021-LLA-0229)
Attachments: Final to SNC RAIs EPID L-2021-LLA-0229.pdf

Wes,

I noticed I stated the incorrect date. For the attached RAIs, the NRC staff anticipates your response to the attached RAIs by COB Monday, June 20, 2022.

stephanie

From: Devlin-Gill, Stephanie
Sent: Tuesday, May 03, 2022 9:42 AM
To: Sparkman, Wesley A. <WASPARKM@southernco.com>
Subject: RAIs for Farley LAR re. TS 5.5.17 (EPID: L-2021-LLA-0229)

Wes,

Attached are the NRC staff's requests for additional information (RAIs) for SNC's license amendment request for the Joseph M. Farley Nuclear Plant, Units 1 and 2 (FNP) (Agency-wide Documents Access and Management System Accession No. ML21348A733), which 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 NRC staff anticipates your response to the attached RAIs by COB ~~Friday, June 17, 2021~~.

stephanie

Stephanie Devlin-Gill (she/her)
Licensing Project Manager
w. 301-415-5301
U.S. NRC, NRR, DORL, LPL2-1

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From: Devlin-Gill, Stephanie

Created By: Stephanie.Devlin-Gill@nrc.gov

Recipients:
"Sparkman, Wesley A." <WASPARKM@southernco.com>
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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.

RAI 02:

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.

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

(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.

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

(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.”