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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 Allow Elimination of the Encapsulation Vessels
Around the First Containment Spray (CS) and Residual Heat Removal (RHR) / Low
Head Safety Injection (LHSI) Recirculation Suction Isolation Valves

Ladies and Gentlemen:

By letter dated July 29, 2021, Southern Nuclear Operating Company (SNC) submitted a license amendment request (LAR) to the Joseph M. Farley Nuclear Plant (FNP) Unit 1 Renewed Facility Operating License (NPF-2), and the Unit 2 Renewed Facility Operating License (NPF-8) to allow elimination of the encapsulation vessels around the first containment spray (CS) and residual heat removal (RHR) / low head safety injection (LHSI) recirculation suction isolation valves.

By email dated November 16, 2021 (ML21210A242), the U.S. Nuclear Regulatory Commission (NRC) staff notified SNC that additional information is needed for the staff to complete their review.

The enclosure to this letter provides the SNC response to the NRC request for additional information (RAI).

This letter contains no NRC 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 December 13, 2021.

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

CAG/was/cbg

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

cc: NRC Regional Administrator
NRC NRR Project Manager – Farley 1&2
NRC Senior Resident Inspector – Farley 1 & 2
SNC Document Control R-Type: CFA04.054

Joseph M. Farley Nuclear Plant - Units 1 and 2
Response to Request for Additional Information Related to License Amendment Request to Allow Elimination of the Encapsulation Vessels Around the First Containment Spray (CS) and Residual Heat Removal (RHR) / Low Head Safety Injection (LHSI) Recirculation Suction Isolation Valves

ENCLOSURE

SNC Response to Request for Additional Information (RAI)

REQUEST FOR ADDITIONAL INFORMATION

REGARDING ELIMINATION OF THE ENCAPSULATION VESSELS AROUND THE FIRST

CONTAINMENT SPRAY AND RESIDUAL HEAT REMOVAL/LOW HEAD SAFETY

INJECTION RECIRCULATION SUCTION ISOLATION VALVES

EPID: L-2021-LLA-0136

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 July 29, 2021 (Agency-wide Documents Access and Management System Accession No. ML21210A242), Southern Nuclear Operating Company (the licensee) submitted a license amendment request for the Joseph M. Farley Nuclear Plant, Units 1 and 2 (FNP). The license amendment request proposes to modify the Updated Final Safety Analysis Report (UFSAR) to allow the removal of the encapsulation vessels around the first Containment Spray and Residual Heat Removal/Low Head Safety Injection recirculation suction isolation valves.

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

RAI 01

In Section 3 “Technical Evaluation” of the proposed amendments, the licensee states that containment sump suction piping design pressure is 80 pound per square inch gauge (psig) and the design temperature is 300 degrees Fahrenheit (°F). The licensee also stated that “the normal operating conditions for the containment sump suction piping are significantly lower as they are isolated with borated water under stagnant conditions.” In its stress analysis to ensure the piping integrity, the licensee considers the containment sump suction piping as moderate-energy piping.

The NRC staff notes that the FNP UFSAR Revision 29, Section 3K.2.1, “Analysis Criteria (General),” states:

> The effects of pipe whip were considered only for those piping systems whose operating pressure and temperature exceed 275 psig and 200°F, respectively [denoted as high-energy piping]. For piping systems whose pressure exceeds 275 psig, but whose temperature does not exceed 200°F, or whose temperature exceeds 200°F, but whose pressure does not exceed 275 psig [denoted as moderate-energy piping], the effects of a critical crack only were considered.

Please clarify (1) the normal operating conditions (operating pressure and operating temperature) for the containment sump suction piping, and (2) whether the definition of the moderate-energy piping used in the proposed amendments is consistent with FNP’s current design and licensing basis.

SNC Response:

(1) The design conditions of 80 psig and 300°F used in the stress analysis calculations are not applicable for establishing the energy level of these piping segments. Instead, the design conditions provided in the LAR were used to bound post-accident conditions and provide the limiting condition inputs to the stress analysis calculations. The determination of the energy level of piping is based on operational plant conditions (normal reactor operation conditions, upset conditions and test conditions). The sump suction piping lines for both the RHR/LHSI and CS systems up to the first isolation valve outside of containment are isolated during operational plant conditions with a standing water leg exposed to containment atmosphere. The operational plant conditions are lower than the design conditions reflected in the LAR. The expected operational plant conditions for these isolated lines would not exceed a pressure of 50 psig and a temperature of 120°F based on exposure to the containment environment and the head of water in the isolated lines.

(2) SNC proposes the use of the definition of moderate-energy per BTP 3-4, Revision 3 to avoid ambiguity in the LAR with respect to the application for the use of the break exclusion criterion. This differs from FNP's current licensing basis. The LAR intent is to limit the BTP 3-4, Revision 3 definition of moderate-energy to just the RHR/LHSI and CS sump suction piping segments up to the first isolation valve outside of containment. See SNC response to RAI 02 for further discussion on the use of this criteria.

RAI 02

As noted in RAI 01 above, the licensee's proposed amendments consider the containment sump suction piping as moderate-energy piping, and, therefore, only considers the effects of leakage cracks were considered. In its stress analysis to ensure the piping integrity, the licensee showed that the maximum stress range does not exceed 0.4 (1.2 Sh + SA) for ASME Code such that leakage cracks need not be postulated for those portions of the containment sump suction piping.

Please clarify whether the criteria used for postulating leakage cracks used in the proposed amendments is consistent with FNP's current design and licensing basis. If not, please specify the version of NRC Branch Technical Position (BTP) 3-4, "Postulated Rupture Locations in Fluid System Piping Inside and Outside Containment," that was used in the stress analysis in the proposed amendments.

SNC Response:

The LAR requests a change to the facility as described in the FSAR to allow application of the Standard Review Plan (SRP) 6.2.4, Revision 3 alternate criterion for meeting General Design Criterion (GDC)-56 for the RHR/LHSI and CS system sump suction piping up to the first isolation valve outside of containment. This alternate criterion requires that the design conforms with SRP 3.6.2, Revision 3 requirements, which in turn requires application of BTP 3-4, Revision 3 moderate-energy break exclusion provisions. The SRP 6.2.4, Revision 3 criterion was developed after FNP was licensed and is therefore not part of the FNP licensing basis. The criterion of maximum stresses less than 0.4(1.8 Sh +SA) required by BTP 3-4 Revision 3 was not used but instead a legacy criterion was used. The legacy criterion of maximum stresses less than 0.4(1.2 Sh +SA) is based on MEB 3-1, Revision 1 provided in SRP 3.6.2, Revision 1. As mentioned in the LAR submittal, this legacy criterion is more conservative than what is required by BTP 3-4, Revision 3 and SRP 3.6.2, Revision 3, and was chosen in order to maintain consistency with the existing stress calculations.

<u>RAI 03</u>

In Section 3 “Technical Evaluation” of the proposed amendments, the licensee states:

> The following criteria, inputs, and assumptions were used in the stress analysis:
> - The Code of Record is ASME Boiler and Pressure Vessel Code, Section III, 1971 Edition, up to and including Addenda through 1971.
> - The piping analysis model utilize the ASME Boiler and Pressure Vessel Code, Section III, 1974 which has been evaluated as acceptable with no adverse impacts to the Code of Record.

Please clarify whether the use of ASME Section III, 1974 Edition for performing piping analysis is the analysis of record for FNP.

<u>SNC Response:</u>

(1) Farley uses software developed by Bechtel Power Corporation. The software used for the stress analysis calculations is ME101 which was developed and is controlled under Bechtel’s nuclear-grade software quality assurance program. This software is the analysis method of record as described in Farley FSAR Section 3L.3.2.2.

(2) The ME101 software requires the applicable code year as an input, but ASME Section III 1971 is not one of the available options in the software. Therefore, ASME Boiler and Pressure Vessel Code, Section III, 1974 was used. A code reconciliation of the applicable differences between the two code years is documented in the Southern Nuclear Piping Stress Analysis Procedure and is summarized as follows.

a) NC-3611.1(b) of the ’71 code addresses limits of calculated stresses due to sustained loads, thermal expansion, additive stresses, and occasional loads. There is no formula given for stress due to weight, sustained loads other than pressure, and occasional loads. A comparison of this paragraph to NC-3652.1 and NC-3652.2 of the ‘74 Code demonstrates that the intent of the ‘71 code is met by the formulas in the 74 code.

b) NC-3672.9 of the '71 Code addresses combining of expansion stresses. There is a formula given for this combination in the '71 Code. Review of Paragraphs NC-3652.3-4 of the '74 Code indicates that a different formula has been used, however, this formula is conservative compared to that of the '71 Code due to the application of a stress intensification factor for both bending and torsional moments

c) Figure NC-3672.9(a)-l of the '71 Code tabulates flexibility and stress intensification factors. Comparison of this Figure to Figures NC-

3673.2(b)-1 thru 5 of the '74 Code reveal minor differences in the factors. However, in each case, the '74 Code is found to be equivalent to or conservative when compared to the corresponding factor of the '71 Code.

It is concluded that using the 1974 code year input is acceptable as it is either equivalent to or conservative with respect to the 1971 edition and addenda.