From: John Lamb
Sent: Wednesday, November 30, 2022 11:42 AM
To: Sparkman, Wesley A.
Cc: Chamberlain, Amy Christine
Subject: For Your Action - RAI - Farley - TS 3.4.10 LAR, PSV Setpoint (L-2022-LLA-0098)

Importance: High

Wes,

By letter dated June 30, 2022 (Agencywide Documents and Access Management System (ADAMS) Accession No. ML22181B145), Southern Nuclear Operating Company (SNC, the licensee) submitted a licensee amendment request (LAR) to propose modifications to the Joseph M. Farley Nuclear Plant (Farley), Units 1 and 2, Technical Specifications (TS) 3.4.10, “Pressurizer Safety Valves,” regarding the Limiting Condition for Operation (LCO) for pressurizer safety valves (PSVs). The proposed changes would decrease the low side setpoint tolerance value from -1 percent to -2.5 percent for the PSVs.

After reviewing the LAR, the NRC staff requests response to the request for additional information (RAI) given below.

On October 31, 2022, the NRC staff provided a draft RAI question to SNC to make sure that the RAI is understandable, the regulatory basis is clear, to ensure there is no proprietary information, and to determine if the information was previously docketed. On November 29, 2022, a clarifying call was held and SNC stated that it would provide the RAI response 45 days from the date of this email.

If you have any questions, you can contact me at 301-415-3100.

Sincerely,
John

<u>REQUEST FOR ADDITIONAL INFORMATION (RAI)</u>

By letter dated June 30, 2022 (Agencywide Documents and Access Management System (ADAMS) Accession No. ML22181B145), Southern Nuclear Operating Company (SNC, the licensee) submitted a licensee amendment request (LAR) to propose modifications to the Joseph M. Farley Nuclear Plant (Farley), Units 1 and 2, Technical Specifications (TS) 3.4.10, “Pressurizer Safety Valves,” regarding the Limiting Condition for Operation (LCO) for pressurizer safety valves (PSVs). The proposed changes would decrease the low side setpoint tolerance value from -1 percent to -2.5 percent for the PSVs.

After reviewing the LAR, the NRC staff requests response to the request for additional information (RAI) given below.

<u>RAI QUESTION #1</u>

<u>Regulatory Requirements</u>

The regulation Title 10 of the *Code of Federal Regulation* (10 CFR), part 50, Section 36(c)(1) requires that plant TS will include safety limits, limiting safety system settings, and limiting control settings. The regulation 10 CFR 50.36(c)(2)(ii)(C) specifies that a LCO be established for a structure, system, or component (SSC) that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure or presents a challenge to the integrity of a fission product barrier. Farley's PSVs provide, in conjunction with the reactor protection system, overpressure protection for the reactor coolant system.

Licensee LAR Discussion

In Section 2.2, "Reason for the Proposed Change," the licensee stated:

> "This change is proposed to reduce an unnecessarily restrictive LCO. FNP [Farley Nuclear Plant] has PSVs manufactured by Crosby. There have been six instances since 2015 where the PSVs were tested and found outside the ±1% tolerance limits. All out of tolerance test results were outside the low end of the setpoint tolerance (-1%). These as-found results have
> prompted the generation of licensee event reports (LERs) in accordance with 10 CFR 50.73(a)(2)(i)(B), "Any operation or condition prohibited by the plant's Technical Specifications..." The as-found results did not exceed -3% of the pressure setpoint. Based on the lift pressure meeting the Inservice Test (IST) program requirements, no IST scope expansion testing was needed. Since the as-found result was lower than the allowed value in the TS, the condition did not have an adverse impact on its overpressurization function. This is within the safety analysis assumptions that are credited for PSVs, and the plant remained bounded by the accident analyses in the Final Safety Analysis Report (FSAR). Setpoint drift was determined to be the cause of the PSVs lifting low out of tolerance. The PSVs are performing within the design analysis assumptions. Therefore, generating a LER for a PSV that is performing satisfactorily
> within the design analysis assumptions becomes an unnecessary burden for both the licensee and the NRC."

RAI #1

In addition to the LAR discussion of the advantages of the proposed low side setpoint tolerance value of -2.5 percent for the Farley PSVs, the licensee is requested to specify whether the evaluations following the six instances of out-of-tolerance test results for the Farley PSVs indicated any material condition concerns with these valves or the performance of their design capabilities. As referenced in the Farley LAR, this information was provided in the LAR submitted by Exelon for the Byron and Braidwood PSV tolerances on June 27, 2003 (ADAMS ML03181034).

RAI QUESTION #2

Regulatory Basis

The regulation 10 CFR 50, Appendix A, GDC 15, "Reactor Coolant System Design," states that the reactor coolant system and associated auxiliary, control, and protection systems are designed with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences.

RAI #2

In Section 3.0 of the enclosure to the letter dated June 30, 2022, under heading "Margin Between High Pressure Reactor Trip and PSVs," refer to the following statement:

"The calculated uncertainty associated with the Pressurizer Pressure – High RTS [reactor trip system] function was determined to be +28.8 psi [pounds per square inch]. The setpoint uncertainty was determined using a Farley-specific setpoint report utilizing a statistical methodology. The Westinghouse setpoint uncertainty calculation methodology was selected because it was used at many Westinghouse PWRs [pressurized-water reactors], conformed to industry practices such as ISA [International Society of Automation] Standard S67.04, 1987, "Setpoints for Nuclear Safety-Related Instrumentation Used in Nuclear Power Plants," and was previously approved by the NRC."

Provide responses to the following:

(a) Briefly describe the analysis for calculating the +28.8 psi uncertainty (including the statistical combination method of uncertainty components) in the nominal high pressurizer pressure reactor trip setting of 2385 psi gauge (psig) for the Pressurizer Pressure – High Reactor Trip System (RTS) function.

(b) Regulatory Guide (RG) 1.105, Revision 4 (ADAMS Accession No. ML20330A239), endorses American National Standards Institute (ANSI)/International Society of Automation (ISA) Standard 67.04.01-2018, "Setpoints for Nuclear Safety-Related Instrumentation," for uncertainty calculation. However, according to the above statement, ISA Standard S67.04, 1987 is used for this calculation. Confirm that the +28.8 psi uncertainty in the Pressurizer Pressure - High RTS function trip setpoint would be obtained by using the NRC endorsed ISA Standard 67.04.01-2018. If not, what would be the uncertainty based on the 2018 ISA standard?

(c) Based on the +28.8 psi uncertainty, the RTS highest trip pressure would be (2385 + 28.8) = 2413.8 psig, which is less by 9.2 psig from the proposed 2423 psig PSV lower limit. If, in response to item (b) above, based on the ISA Standard 67.04.01-2018, the RTS high trip pressure exceeds the proposed PSV lower limit of 2423 psig, the PSV may actuate prior to the RTS trip. In this scenario, provide an evaluation of the impact on the analysis of events/accidents that result on the overpressurization of the reactor coolant system.

(d) Define the acronym "SAL" given in the last sentence of the first paragraph under heading "Margin Between High Pressure Reactor Trip and PSVs," Does it stand for "Analytical Limit"? Explain the meaning of SAL?

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