UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

February 15, 2018

Mr. George A. Lippard, III
Vice President, Nuclear Operations
South Carolina Electric & Gas Company
Virgil C. Summer Nuclear Station
Post Office Box 88, Mail Code 800
Jenkinsville, SC 29065

SUBJECT: VIRGIL C. SUMMER NUCLEAR STATION, UNIT NO. 1 – REQUEST FOR ADDITIONAL INFORMATION RE: RELIEF REQUEST (RR-4-13), USE OF RISK-INFORMED PROCESS AS AN ALTERNATIVE FOR THE SELECTION OF CLASS 1 AND CLASS 2 PIPING WELDS (EPID NO. L-2017-LLR-0133)

Dear Mr. Lippard:

By letter dated October 30, 2017, South Carolina Electric & Gas Company (SCE&G, the licensee) submitted an alternative request for Virgil C. Summer Nuclear Plant, Unit 1. The proposed alternative requests the use of a risk-informed process as an alternative for the selection of Class 1 and Class 2 piping welds in lieu of the inspection and examination requirements specified by the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (BPV), Section XI, "Rules for inservice inspection of nuclear power plant components," Tables IWB-2500-1 and IWC-2500-1.

The U.S. Nuclear Regulatory Commission (NRC) staff has reviewed the licensee's submittal and determined that additional information is needed to continue its review. As discussed with Ms. Dalick, please respond within 45 days of the date of this letter. Please note that the NRC staff's review is continuing, and further requests for information may be developed.

If you have any questions, please contact me at 301-415-1009 or Shawn.Williams@nrc.gov.

Sincerely,

Shawn Williams

Shawn A. Williams, Senior Project Manager
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-395

Enclosure:
Request for Additional Information

cc w/enclosure: Distribution via Listserv

REQUEST FOR ADDITIONAL INFORMATION

RELIEF REQUEST (RR-4-13) FOR USE OF A RISK-INFORMED PROCESS AS AN

ALTERNATIVE FOR THE SELECTION OF CLASS 1 AND CLASS 2 PIPING WELDS

SOUTH CAROLINA ELECTRIC & GAS COMPANY

VIRGIL C. SUMMER NUCLEAR STATION, UNIT NO. 1

DOCKET NO. 50-395

By letter dated October 30, 2017 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML17303B183), South Carolina Electric & Gas Company (SCE&G, the licensee) submitted a relief request (RR) for Virgil C. Summer Nuclear Plant, Unit 1 (VCSNS). The licensee proposed the use of Risk-Informed Inservice Inspection (RI-ISI) Program as an alternative to the selection of Class 1 and Class 2 piping welds in lieu of the inspection and examination requirements specified by the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (BPV) Code, Section XI, "Rules for inservice inspection of nuclear power plant components," Tables IWB-2500-1 and IWC-2500-1.

The U.S. Nuclear Regulatory Commission (NRC) staff has determined that the following requests for additional information (RAI) are required to complete its review.

RAI No. 1

Regarding the risk metrics provided in the RR, please clarify the following:

a. On page 7 of the RR, the licensee states:

> The revised program represents an overall reduction of plant risk of -9.83E-09 for CDF [core damage frequency] and -4.08E-09 for LERF [large early release frequency].

   Please clarify that the negative value of the CDF and LERF risk metrics represent risk reductions and not "negative reductions," in which a reduction of a negative value as provided in the license amendment request (LAR) could imply an increase in risk.

b. On page 8 of the RR in the table "VCSNS Risk Impact Results", the change in LERF for each system is approximately 40% of the corresponding change in CDF, which is consistent with the overall change in CDF and LERF as provided on page 7 of the LAR. However, the changes in LERF for the Emergency Feedwater (EF) and Feedwater (FW) systems have the same values as their corresponding changes in CDF, which does not appear to be consistent with the rest of the systems, where the change in LERF is always less in magnitude than the corresponding change in CDF, and the overall results. Please clarify why the changes in CDF and LERF for the EF and FW systems are equal and not consistent with the other results.

RAI No. 2

According to Regulatory Issue Summary 2007-06 (ADAMS Accession No. ML070650428), the NRC staff expects that licensees fully address all scope elements with Revision 2 of Regulatory Guide (RG) 1.200 (ADAMS Accession No. ML090410014) by the end of its implementation period (i.e., one year after the issuance of Revision 2 of RG 1.200). Revision 2 of RG 1.200 endorses, with exceptions and clarifications, the combined American Society of Mechanical Engineers (ASME)/American Nuclear Society (ANS) Probabilistic Risk Assessment (PRA) standard (ASME/ANS RA-Sa-2009).

On page 3 of Attachment 1 of the LAR, the licensee states:

> Independent PRA peer reviews were conducted under the auspices of the Pressurized Water Reactor Owners Group (PWROG) following the Industry PRA Peer Review process in 2002 and 2016.

The licensee further explains a PRA model update was completed in 2016 and a full scope peer review was performed. Please confirm the following items in regard to the 2016 full scope peer review:

a. Please confirm the full scope peer review was reviewed against the 2009 ASME/ANS PRA standard, as endorsed by RG 1.200, Revision 2. If not, identify any gaps between the peer review and the guidance in RG 1.200, Revision 2.

b. For the disposition of SR IE-A5/A6 in Table 1 of the LAR, the licensee states:

> The initiating event list in the VCSNS PRA was based on a review of other risk assessments, plant operating history, and plant design. This included a review of support systems."

The guidance in RG 1.200, Revision 2 specifies the systematic evaluation of each system, including support systems, needs to be performed "down to the subsystem or train level where necessary." Please confirm if the review of the support systems was provided down to this level, and provide justification if it was not.

RAI No. 3

In Table 1 of Attachment 1 of the RR, for the disposition of SR IE-C1, the licensee cites performance of a sensitivity study to address the Finding, in which a change in consequence from MEDIUM to HIGH was due to a near factor of four increase in medium loss-of-coolant accident (LOCA) frequency. The disposition further questions the basis for this increase, stating the difference may be based on binning or expert elicitation, and, therefore, concluding that the data used in the current PRA model are "sufficient for medium LOCA." Please explain the reasons for the increase in frequency other than how the data, which may include recent updates, have been processed. Please provide justification for taking exception to the factor of four increase in the medium LOCA frequencies and/or include an evaluation of the effect from the sensitivity study of the factor of four increase in the medium LOCA frequencies on the risk metrics applicable to this application.

RAI No. 4

In Table 1 of Attachment 1 of the LAR, for the disposition of SR SY-A4, the licensee states:

> Walkdowns of recent system modifications have been done in support of Fire PRA human reliability.

The licensee's National Fire Protection Association (NFPA) 805 "Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants" LAR (ADAMS Accession No. ML14287A289) was submitted on November 15, 2011, and the Amendments issued on February 11, 2015 (ADAMS Accession No. ML14287A289). Based on the walkdowns, referenced in Table 1 of the LAR, walkdowns were performed six or more years ago. Please provide justification that these walkdowns are adequate to be representative of the as-built, as-operated status of the plant for the version of the PRA used in this application.

RAI No. 5

In Table 1 of Attachment 1 of the RR, for the disposition of SR HR-G7, identify the joint Human Error Probability (HEP) floors that were used. Please confirm that none were < 1E-6 for internal events. If any were <1E-6 for internal events, please provide the basis and the results of a sensitivity evaluation using 1E-6, and include any effects on the "small" impact statement under the "Impact" column in Table 1.

RAI No. 6

In Table 1 of Attachment 1 of the LAR, for the disposition of SR IFEV-A7, the licensee states:

> Limited on-line maintenance makes human induced flooding less significant and it should not affect the failure probability or consequence for any piping welds.

Although on-line maintenance is limited, it is not eliminated and, therefore, there may be risk from human induced flooding. Please justify with a bounding quantification the statements that human induced flooding should not affect the risk, and the likely impact on RI-ISI risk is small and will not add significance.

RAI No. 7

In Table 1 of Attachment 1 of the RR, for the disposition of SR QU-D2 / AS-A5, the licensee cites conservatism in the PRA model and that the risk impact is expected to be small. On the basis that conservatism may underestimate the risk increase or overestimate the risk reduction of the application, provide the following information:

a. When citing conservatism in the PRA model, please confirm that calculation of the differential risk for this application is also conservative (i.e., the risk estimated for the before versus after condition uses the same assumptions, etc., except for the change to any basic event values affected by the application, ensuring that the before value is not overestimated such that subtracting it from the after value could underestimate the risk increase).

b. Please provide quantitative justification with a bounding evaluation, crediting the available recovery options, that the expected impact is small, as stated in the "Impact" column.

<u>RAI No. 8</u>

VCSNS Unit 1 is currently already in its Fourth 10-Year ISI interval, which started on January 1, 2014, and is scheduled to end on December 31, 2023. The licensee implemented its regular ISI program for ASME Code Class 1 and Class 2 piping welds for the first period of the Fourth 10-Year ISI interval. Relief Request RR-4-13 states that the licensee will use its RI-ISI Program for the balance of the Fourth 10-Year Interval (i.e., second and third periods), and "prorate" for examinations it already performed in the first period.

In its submittal, the licensee did not provide any specific information on the weld examinations already performed for the first period, or how these examinations will be "prorated" for the Fourth 10-Year Interval. Additionally, it is not clear if all risk significant examinations that would have been completed during the first period of a RI-ISI Program at VCSNS Unit 1, were performed.

In its submittal, the licensee provided summary tables that include the total weld population in the scope of the VCSNS Unit 1 proposed RI-ISI Program. Please provide the ASME Code classifications (i.e., ASME Class 1 or 2) of the piping welds in the tables. Additionally, confirm that risk significant examinations that should have been performed during the first period of the RI-ISI program were performed during the first period as part of the regular ASME ISI, or will be performed as part of the RI-ISI Program consistent with the requirements of Table IWB-2411-1.

According to ASME Code Case N-578-1, "Risk-Informed Requirements for Class 1, 2, or 3 Piping, Method B, Paragraph -2420 (a), "Successive Inspections," the sequence of piping examinations established during the first inspection interval using the risk-informed process shall be repeated during each successive inspection. Thus, please confirm that the sequence of the piping examinations for periods 2 and 3 of the Fourth 10-year RI-ISI interval will be consistent with the sequence of examinations for 2 and 3 of the Third 10-year RI-ISI interval.

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ADAMS Accession No. ML18023B069 ***via email**

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