From:	Williams, Shawn
Sent:	Monday, March 23, 2020 3:25 PM
To:	Coleman, Jamie Marquess
Cc:	Sparkman, Wesley A.; Bates, Ernest F.
Subject:	Joseph M. Farley Nuclear Plant, Units 1 and 2 - Request for Additional Information RE: Measurement Uncertainty Recapture Power Uprate (EPID L-2019-LLS-0002)
Attachments:	March 23, 2020, Farley MUR RAIs .docx

Dear Ms. Coleman,

By letter dated October 30, 2019, as supplemented on November 25, 2019, the Southern Nuclear Operating Company, Inc., submitted a license amendment request for the Joseph M. Farley Nuclear Plant, Units 1 and 2. The proposed amendment would revise the Renewed Operating Licenses and Technical Specifications to allow for a measurement uncertainty recovery power uprate (MUR-PU) from 2775 megawatts thermal (MWt) to 2821 MWt.

The U.S. Nuclear Regulatory Commission (NRC) staff has determined that additional information is needed as discussed in the Enclosure. A clarification call to ensure mutual understanding was conducted on March 12, 2020. Please respond within 30 days of the date of this e-mail. 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 via e-mail at Shawn.Williams@nrc.gov.

Sincerely,

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

Docket Nos. 50-348 and 50-364

Enclosure:
Request for Additional Information

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REQUEST FOR ADDITIONAL INFORMATION

JOSEPH M. FARLEY NUCLEAR PLANT, UNITS 1 AND 2

MEASUREMENT UNCERTAINTY RECAPTURE POWER UPRATE

LICENSE AMENDMENT REQUEST

SOUTHERN NUCLEAR OPERATING COMPANY, INC.

DOCKET NOS. 50-348 AND 50-364

By letter dated October 30, 2019 (Agencywide Documents Access and Management System (ADAMS) Package Accession No. ML19308A761), as supplemented on November 25, 2019 (ADAMS Accession No. ML19331A099), the Southern Nuclear Operating Company, Inc., submitted a license amendment request (LAR) for the Joseph M. Farley Nuclear Plant (FNP), Units 1 and 2. The proposed amendment would revise the Renewed Operating Licenses and Technical Specifications to allow for a measurement uncertainty recovery power uprate (MUR-PU) from 2775 megawatts thermal (MWt) to 2821 MWt.

The U.S. Nuclear Regulatory Commission (NRC) staff has determined that the following additional information is needed.

SFNB RAI No. 1:

Guidance provided in NRC Regulatory Guide (RG) 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," (ADAMS Accession No. ML010890301) states:

> The uncertainty of the fluence must be 20% (1 σ) or less when the fluence is used to determine RT_{PTS} and RT_{NDT} for complying with 10 CFR 50.61 and Revision 2 of Regulatory Guide 1.99, "Radiation Embrittlement of Reactor Vessel Materials," respectively. It should be recognized that this 20% uncertainty value has been included in the margin term for the RT_{PTS}.

While the analytic uncertainty estimate provided in WCAP-18124-NP-A, "Fluence Determination with RAPTOR-M3G and FERRET," confirms that the analytic uncertainty is estimated to be below 20% for the core-adjacent beltline region, the licensee furnished additional information estimating the uncertainty for the extended beltline region. The estimated uncertainty in the extended beltline, while not explicitly limited to 20%, must still be taken into consideration when confirming that the extended beltline materials do not become limiting with respect to RT_{PTS} and RT_{NDT}.

In section IV.1.C.ii of Attachment 4 to the LAR dated October 30, 2019, it is stated that the anisotropic scattering for the neutron fluence evaluation was modeled with a P_3 Legendre expansion and the angular discretization was modeled with an S_{12} order of angular quadrature.

In the additional benchmarking analysis for neutron transport calculations in Attachment 3 to the same application, it is stated that the anisotropic scattering was modeled with a P_3 Legendre expansion and the angular discretization was modeled with an S_{20} order of angular quadrature.

Explain (a) whether the different treatment of the order of angular quadrature introduces additional uncertainty in the neutron fluence evaluation for extended beltline materials, and (b) whether and how this potential contributor to the uncertainty was taken into account in the uncertainty estimation. If the different angular quadrature does not affect the uncertainty, or was not taken into account, provide a justification.

NPHP RAI No. 1

Section VI.1.A.ii of Attachment 4, "Summary of RIS 2002-03 Requested Information for Farley Nuclear Plant License Amendment Request," to your letter dated October 30, 2019, stated that the "high pressure blading design (specifically the first and second rotating blade rows on each end) will require modernization for MUR conditions. Additionally, the existing Units 1 and 2 high pressure turbines require modernization to increase valve wide-open steam flow capacity, and to recover throttle flow margin to support the MUR-PU." The licensee evaluation also stated that the high-pressure replacement does not require an update to the turbine missile analysis, since the turbine missile analysis is only for the low-pressure turbine, and the low-pressure turbine was not modified.

However, since the components for the turbine are being modified (including modernization to increase valve wide-open steam flow capacity, and to recover throttle flow margin) the effects of these modification can change the steam flow, pressure, temperature and moisture content of the steam including those seen by the low-pressure turbine. Any effects to the low-pressure turbine, such as increased moisture content, temperature or flow can affect the degradation mechanism and stresses on the low-pressure turbine rotor and impact the turbine missile analysis.

Identify what components are being modified for "the modernization to increase valve wide-open steam flow capacity, and to recover throttle flow margin," and what affects do these modifications have on the steam flow, pressure, temperature and moisture content when it is diverted to the low-pressure turbine. Address how these changes have been evaluated for any potential impact to the low-pressure turbine missile analysis and whether it has been revised accordingly.

EENB RAI No. 1

Regulatory Criteria:

10 CFR 50.49 (e)(1) requires that the time-dependent temperature and pressure at the location of the electric equipment important to safety must be established for the most severe design basis accident during and following which this equipment is required to remain functional.

10 CFR 50.49(b)(2) requires qualification of nonsafety-related electric equipment whose failure under postulated environmental conditions could prevent satisfactory accomplishment of safety functions specified in subparagraphs (b)(1) (i) (A) through (C) of paragraph (b)(1) of 10 CFR 50.49 by the safety-related equipment.

Background:

Section V.1.C of Attachment 4 of the LAR, “Environmental qualification of electrical equipment,” states:

- The equipment and components of the equipment qualification program will continue to operate satisfactorily and perform their intended functions at the uprated conditions to satisfy the requirements outlined in 10 CFR 50.49, and the safety-related electrical equipment is qualified to survive the environment at its specific location during normal operation and during an accident.

- The equipment qualification program equipment will accommodate MUR-PU conditions without exceeding electrical equipment qualification margins for the parameters of temperature, pressure, radiation, and similar parameters, as defined by IEEE Standard 323-1974.

This section further stated that “[t]he evaluations determine that there is no impact on the existing analyses or changes to equipment qualification areas and, therefore, the existing analyses remain bounding and the MUR-PU will not affect equipment in the equipment qualification program for EQ.”

The NRC staff reviewed the above sections and noted that according to Table 11.1-1, “FSAR Accidents, Transients, and Other Analyses,” the following accidents/transients previously analyzed in FSAR remain bounded as a result of MUR-PU: major reactor coolant system pipe rupture (LOCA), containment analyses, flooding, high energy line break outside containment, and major secondary system pipe rupture. The NRC staff also noted that according to the evaluation performed in Section II.1, for flooding (item 30), and main steam line break in the Main Steam Valve Room (MSVR) (item 33), current analyses remain valid and unaffected by MUR-PU. Also, according to Section VI.1.B.iii of the Appendix 4 to the LAR the current minimum sump level PH value of 7.21 remains applicable for the MUR-PU.

Issue:

In the LAR, the licensee noted that they have evaluated the impact of the proposed MUR-PU on the Environmental Qualification (EQ) of equipment. SNC asserted that the results of their evaluations showed that electrical equipment that is required to be environmentally qualified per 10 CFR 50.49 will remain qualified (i.e., bounded by the existing EQ). However, the licensee did not provide enough detail for the NRC staff to confirm whether the existing accident analyses for all areas of the plant were performed at 102% rated thermal power (RTP) versus being limited to inside containment and the MSVR (e.g., temperature/pressure profiles, radiation dose calculations, etc.).

It is also unclear as to whether the licensee considered the impact of the proposed change on qualified nonsafety-related equipment (10 CFR 50.49(b)(2)) whose failure under postulated environmental conditions could prevent satisfactory accomplishments of safety functions by the safety-related equipment.

Request:

1. If the accident analyses performed at 102% RTP were limited to inside containment and the MSVR, provide an evaluation that shows that the environmental qualification remains bounded for electric equipment located in areas of the plant that will experience parameter changes (i.e., increase in temperature, pressure, radiation, humidity, chemical spray, etc.) due to the proposed MUR-PU.

2. Assess the impact of the proposed MUR-PU on qualified nonsafety-related equipment (10 CFR 50.49(b)(2)) whose failure in postulated environmental conditions could prevent satisfactory accomplishments of safety functions by the safety-related equipment.

EICB RAI No. 1

Regulation 10 CFR 50, Appendix K, allows licensees to use an assumed power level lower than 1.02 times the licensed power level, provided the proposed alternative value has been demonstrated to account for uncertainties due to power level instrumentation error.

Regulatory Issue Summary (RIS) 2002-03, "Guidance on the Content of Measurement Uncertainty Recapture Power Update Applications," addresses scope and detail of the information that should be provided to the NRC for reviewing measurement uncertainty recapture power uprate applications. To aid licensees in optimizing their measurement uncertainty recapture power uprate applications, the NRC staff developed the guidance in Attachment 1 to the RIS and has been established as a method to meet the requirements of 10 CFR 50, Appendix K.

RIS 2002-03, Attachment 1, Section I.1.D, provides the following detail:

> The dispositions of the criteria that the NRC staff stated should be addressed (i.e., the criteria included in the staff's approval of the technique) when implementing the feedwater flow measurement technique.

The NRC staff had previously reviewed and approved the use of the Cameron LEFM Check and CheckPlus system, as described in Cameron Engineering Report ER-80P and ER-157P respectively. In their approval and documented in the respective safety evaluation, the following criterion is provided:

> Confirm that the methodology used to calculate the uncertainty of the LEFM in comparison to the current feedwater instrumentation is based on accepted plant setpoint methodology (with regard to the development of instrument uncertainty). If an alternative approach is used, the application should be justified and applied to both venturi and ultrasonic flow measurement instrumentation for comparison.

The application did not provide sufficient details for the NRC staff to adequately address this criterion to satisfy 10 CFR 50 Appendix K requirements.

a) Provide a description of the methodology used to calculate the uncertainty of the LEFM, the methodology used for the current feedwater instrumentation and a comparison of the two methodologies.

b) Provide a description of the current plant setpoint methodology.

c) The LAR also states, “[t]he core thermal power uncertainty calculation…is performed in accordance with WCAP-12771.” This WCAP was not included with the application. Provide a description of this methodology from the WCAP.

EEOB RAI No. 1

The regulatory requirements related to electric power system are contained in Title 10 of the Code of Federal Regulations (10 CFR), Appendix A, Criterion 17, "Electric power systems.”

The subject LAR states in Attachment 4, Section V, “Electrical Equipment Design”, subsection titled, “Unit Auxiliary Transformers,” that the only 4160-VAC loads affected by the uprate are Condensate Pumps 1A, 1B, and 1C for Unit 1, and Condensate Pumps 2A, 2B, and 2C for Unit 2. The brake horsepower (BHP) of the condensate pump for the MUR-PU condition will increase by 25 HP but remains within 3000-hp, 1.15 Service Factor rating of the motor.”

Please provide the percent loading of the Unit Auxiliary Transformers and Startup Auxiliary Transformers before and after the 75 HP increase to each unit due to the three Condensate Pumps.

EMIB RAI No. 1

The Farley LAR states that inservice testing (IST) program does not require revision as a result of Farley MUR power uprate. In a previously submitted Farley document for the fifth 10-year IST interval (ADAMS Accession No. ML1703D324), states that “Code of Record” is ASME OM Code 2004 Edition with 2006 Addenda for pumps and valves. Please confirm that these ASME OM Code editions are also used for Farley MUR uprate evaluation and review. In addition, please provide the applicable Code used the inservice examination and testing of snubbers.

EMIB RAI No. 2

NRC staff could locate information in the LAR regarding the evaluation of safety-related snubbers (similar to pumps and valves). Please describe the snubber evaluation and its results. If an evaluation was not performed, justify that the existing evaluation of the snubbers is bounding for the uprated power.

EMIB RAI No. 3

Please explain how the adverse effects from flow-induced vibration of safety-related components was evaluated in the Farley MUR LAR, as discussed in Section 4.9, “Flow Induced Vibration Analysis,” of NEI 08-10, “Roadmap for Power Uprate Program Development and Implementation” (ADAMS Accession No. ML092540581).

NCSG RAI No. 1:

The guidance in RIS 2002-03, “Guidance on the Content of Measurement Uncertainty Recapture Power Uprate Applications,” recommends that a licensee provide information for its flow-accelerated corrosion (FAC) program as part of its LAR. The NRC staff’s acceptance criteria for FAC related reviews are based on maintaining the minimum acceptable wall thickness for components susceptible to FAC.

Section IV.1.E.iii, “Flow-Accelerated Corrosion program,” of the LAR stated that the wear rates for some components will increase due to the proposed MUR power uprate at Farley. The licensee stated that the wear rates at the proposed MUR power uprate conditions were modeled in CHECWORKS™. However, the LAR does not provide information regarding what components are impacted, and by how much the maximum wear rate for an individual component will increase due to the proposed MUR power uprate.

In order to obtain reasonable assurance that components within the lines modeled in CHECWORKS™ will not experience significant degradation at the MUR power uprate conditions, the NRC staff requests predicted wear rate values for individual susceptible components in these lines that will experience the greatest increase in wear rate due to the MUR power uprate conditions. Additionally, if any of these components are expected to have significantly increased wear rates, describe how the current FAC program will manage this degradation.

NCSG RAI No. 2:

NUREG-0800, “Standard Review Plan,” Section 6.1.2, “Protective Coating Systems (Paints) – Organic Materials,” Revision 3, provides the NRC staff guidance to ensure coating systems used inside containment are evaluated to determine suitability for design basis accident (DBA) conditions. This guidance directs the reviewer to verify coating monitoring and maintenance procedures are capable of ensuring that coatings will not fail and become a debris source for the emergency core cooling system. This guidance also instructs the reviewer to determine the suitability of the protective coatings in the (DBA environment when exposed to high temperatures, pressures, and radiation dose.

Section VII.6.B, “Containment Coatings Program,” of the LAR discusses the current licensing basis for the Farley containment coatings program, as well as, the DBA qualifications of the coatings in containment. However, the NRC staff requires clarification on the DBA qualifications for coatings in containment.

For the NRC staff to verify that the qualifications of containment coatings are still bounding of the proposed MUR DBA conditions, provide a comparison of DBA conditions (e.g. temperature, pressure, dose) to the qualification condition for the containment coatings.

NCSG RAI No. 3:

The bases for Farley Technical Specification 3.4.17, “SG Tube Integrity,” state that the content of the Farley Steam Generator Program is Nuclear Energy Institute (NEI) 97-06, “Steam Generator Program Guidelines,” and its referenced EPRI Guidelines. These referenced EPRI Guidelines includes the EPRI “Pressurized Water Reactor [PWR] Primary Water Chemistry Guidelines.” The EPRI PWR Primary Water Chemistry Guidelines contain limits on specific impurities for primary water chemistry and associated actions if these impurity limits are not met.

In order to ensure the integrity of the steam generator (SG) tubes can be maintained at MUR power uprate conditions, the NRC staff reviewed the primary water chemistry program. UFSAR Table 5.2-22, "Reactor Coolant Water Chemistry Specification," provides a maximum concentration of chlorides and fluorides of 0.15 parts per million (ppm) and states that the concentration of oxygen will be maintained below 0.1 ppm. These values are greater than EPRI Primary Water Chemistry Guidelines Rev 7 action level 1 limits for primary water chemistry parameters and may contribute to degradation of SG tubes. Provide the justification for why operations at the MUR power uprate conditions will be able to maintain SG tube integrity with the primary water chemistry limits described in the Farley UFSAR.

NCSG RAI No. 4:

Section 5.4.2.1, "Steam Generator Materials and Design," of NUREG-0800, "Standard Review Plan," provides the NRC staff guidance to review SG designs with respect to potential degradation of the SG tubes. The NRC staff review is focused on maintaining reasonable assurance of SG tube integrity as well as compliance with relevant General Design Criteria (GDC) such as GDCs 14, "Reactor Coolant Pressure Boundary," and 31, "Fracture Prevention of Reactor Coolant Pressure Boundary." This includes an evaluation of potential degradation mechanisms that may cause SG tube wear or fatigue of the SG tubes.

In the Farley LAR certain potential degradation mechanisms are discussed (e.g. fluidelastic instability, turbulence, vortex shedding, tube wear, and fatigue). The licensee concludes that these mechanisms will not be impacted by the proposed MUR power uprate as the stresses induced by these mechanisms will remain below the ASME Code fatigue endurance limit of 24 kilopounds per square inch (ksi).

The source of the ASME Code fatigue endurance limit is not clear to the NRC staff. Additionally, in Section IV.1, "Mechanical/Structural/Material Component Integrity and Design," the licensee stated that the table containing critical thermal design parameters does not assume any SG tube plugging.

Therefore, the NRC staff requests the following information:

1. Clarify the basis for using the "endurance limit of approximately 24 ksi" that is mentioned in Section IV.1.A.vi.c, "SG Tube Wear and FIV [Flow-Induced Vibration] Evaluation."

2. It appears that the thermal design parameters in the LAR do not consider any SG tube plugging. SG tube plugging may have impacts on potential SG tube degradation. For the potential SG tube degradation mechanisms described in the LAR, state whether SG tube plugging was considered in these evaluations. If not, provide a justification for why SG tube plugging does not need to be considered in these evaluations.

STSB RAI No. 1

One of the proposed TS changes on TS page 3.4.1-1 involves deletion of the following bolded text from LCO 3.4.1, "RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits":

RCS DNB parameters for pressurizer pressure, RCS average temperature, and RCS total flow rate shall be within the limits specified in the COLR. The minimum RCS total flow rate shall be

greater than 263,400 GPM when using the precision heat balance method, greater than 264,200 GPM when using the elbow tap method, and greater than the limit specified in the COLR.

The licensee's technical justification on Page E-9 provides the following explanation for the proposed change:

> The MUR-PU DNBR calculations that use the statistical treatment of measurement uncertainties are based on a minimum measured flow of 273,900 GPM compared to the value of 263,400 GPM used in many of the current DNB analyses of record (AORs). The higher core flow is consistent with the value in the COLRs tor the current operating cycles in the Farley units for those DNB events that are limiting below the first mixing vane grid. The DNB analyses which do not use the statistical treatment of measurement uncertainties continue to use the TDF of 258,000 gpm.

Please provide justification for the deletion of the two methods specified in the LCO above. Also, per 10 CFR 50.36 (a)(1), please provide a summary statement of the bases or the proposed TS bases pages accordingly.