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Joseph M. Farley Nuclear Plant - Units 1 and 2
Response to Request for Additional Information Related to License Amendment Request
For Measurement Uncertainty Recapture Power Uprate

Ladies and Gentlemen:

By letter dated October 30, 2019, 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 for a measurement uncertainty recapture power uprate (MUR-PU). This MUR-PU LAR would increase FNP's authorized core power from 2775 megawatts thermal (MWt) to 2821 MWt (ML19308A761).

By email dated March 23, 2020 (ML20084G527), the U.S. Nuclear Regulatory Commission (NRC) 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.

In accordance with 10 CFR 50.91, SNC is notifying the state of Alabama of this license amendment RAI response by transmitting a copy of this letter to the designated state official.

If you have any questions, please contact Jamie Coleman at 205.992.6611.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on April 22, 2020.

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

CAG/was/efb/scm

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
Alabama - State Health Officer for the Department of Public Health
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ENCLOSURE

SNC Response to Request for Additional Information (RAI)

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.

SNC Response:

(a) The different treatment of the order of angular quadrature sets introduces additional uncertainty in the neutron fluence evaluation for extended beltline materials in addition to the 30% preliminary uncertainty determined for the Farley MUR evaluation.

(b) To address this question, a sensitivity study has been performed for the Farley MUR fluence evaluation using the same parameters used in the additional benchmarking analysis in Attachment 3, e.g., a P_3 Legendre expansion and a S_{20} order of treatment of angular quadrature sets. An uncertainty of 15.2% has been introduced by this sensitivity study. When combined with the 30% analytical uncertainty determined for all the other uncertainty contributors by the rule of the square root of the sum of the squares, the total uncertainty of the fluence analysis is 33.6%. This 3.6% increase in total uncertainty is not considered significant because, as described in Attachment 3 to the MUR-PU application, the extended beltline materials are not limiting, with significant margin before becoming limiting materials. In addition, for the lower shell to lower vessel head circumferential weld, which has a calculated fast neutron fluence more than a factor of 5

lower than the prescribed threshold of 1E+17 n/cm^2, this 3.6% increase in the total uncertainty is also insignificant. Therefore, the arguments made in Attachment 3 for the justification of using RAPTOR-M3G for the Farley MUR still apply.

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.

SNC Response:

This response differentiates between Unit 1 and Unit 2, where information is different between the units. Unit 2 will be implemented first followed by Unit 1 during the next scheduled refueling outage for each unit. This information is based on the current engineering and design information for the measurement uncertainty recapture power uprate (MUR-PU).

The modernization of the high-pressure turbine (HPT) includes a complete replacement of the original blading with a flow path design featuring advanced blade profiles and elimination of the partial-arc admission for improved performance. The inlet valves are to be operated in single valve mode (full arc admission) where all of the valves open and close together. During low steam flow conditions during startup, the valves operate in a staggered start sequence up to 40% Flow Demand (Governor Valves #2 and #3 open initially followed by #1 and #4) when all valves are in single valve operation. New components include the four guide blade carriers and the new Siemens BB296FG design rotor and internals (including rotor, blade rings, and blading), no-through bore, double flow, fully integral HP rotor; application of erosion resistant alloys for guide blade

carriers, solid stationary and rotating blade seals, and new upgraded HPT bearings. The original nozzle chamber will be reused and reconfigured for full-arc admission. The nozzle blocks are removed and replaced with flow guide segments. This modernization of the HPT retains the existing outer casing, which will require minimal internal machining to fit the new components. The original HPT uses a double-flow blade path with a control stage and six drum stages on each side. The modernized HPT features a symmetric double-flow blade path with eight stages on each side and features integrally shrouded rotating and stationary reaction blading. The reaction blade path design utilizes advanced blade profiles and increased stage count to improve overall efficiency. Additionally, the Westinghouse OVATION turbine control system is being modified to provide different control signal inputs into the turbine governor valves, such that the governor valves control steam admission consistent with plant operation at the uprated condition.

In accordance with heat balance diagrams provided for Units 1 and 2 for the HPT upgrade and MUR uprate conditions, the low-pressure turbine (LPT) inlet temperatures increase only slightly from their baseline conditions (+0.3 deg F on Unit 2 and +1.5 deg F on Unit 1). The temperature changes to steam conditions reflected in the heat balance diagrams were evaluated with respect to the current missile analysis. The LPT inlet steam temperatures are utilized in the missile analysis probability calculation, as they influence the LPT rotor disk metal temperatures. However, the LPT inlet temperatures for both Farley units after the MUR still fall within those assumed in the original finite element analysis (FEA) for the 13.9 m2 LPTs (and utilized for the original missile analysis). Siemens engineering evaluations of the LPTs for pre- and post-uprate show that pressure and steam admission values similarly increase slightly, and these uprated values continue to be within the design conditions, so the missile analysis remains valid. LPT inlet pressures similarly are only slightly affected (+3.2 psi on Unit 2 and +3.0 psi on Unit 1). Steam flow to the LPT's #1 and #2 is increased as follows: For Unit 2, 139,104 lbm/hr total (+1.6%), and +69,552 lbm/hr (+1.6%) for each individually. For Unit 1, 124,637 lbm/hr total (+1.4%), and approximately +62,318.5 lbm/hr (+1.4%) for each individually. The LPT inlet moisture content values for LPT's #1 and #2 on both units are unchanged, at 0% for both pre- and post-uprate conditions.

The changes to the steam conditions in the LPTs due to the HPT upgrade and MUR uprate are small and remain within the 13.9m2 design conditions and the bounding assumptions used in the missile analysis. Therefore, the missile analysis results as currently documented in report CT-27475 Revision 3, Missile Report, will not be affected by the MUR uprate.

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.

SNC Response:

1. Environmental Conditions Outside Containment and the MSVR

   Maximum Radiation Environments Outside Containment:
   The post-accident radiation environments currently presented in FNP FSAR Table 3.11-1 for outside containment locations are based on the Loss-of-Coolant Accident, reflect a core power level of 2,831 MWt (which encompasses MUR-PU operation), and are based on a core isotopic inventory, fuel cycle length and enrichment that either bounds or remains applicable to the MUR-PU. In summary, the accident radiation environments currently utilized for electrical equipment qualification remain valid for operation at MUR-PU conditions

   Maximum Pressure and Temperatures outside Containment
   The initial fluid operating conditions (i.e., pressure and temperature) in a system control the mass and energy (M&E) releases following a line break, which determine the compartment temperature, pressurization, flooding and environmental effects in the affected area. To assess the impact of MUR-PU on the current design basis M&E releases, the system design basis fluid initial conditions reported in the FNP FSAR Appendix 3K, for each system were compared with the corresponding initial fluid conditions for each system following the MUR-PU.

   With the exception of the main feedwater system (and by association, the portion of the auxiliary feedwater system from the junction with the main feedwater line to the first isolation valve), the current design basis system initial conditions remain unchanged or bound the MUR-PU system conditions.

   The elevated feedwater / auxiliary feedwater temperature from 440°F to 446°F may increase the break effluent enthalpy and associated energy release. However, as stated in FSAR Appendix 3K.4.1.2.3, the areas in the auxiliary building affected by a rupture in the main feedwater system and the auxiliary feedwater system are the same as the main steam system; therefore, the compartment pressurization and environmental consequences associated with the referenced feedwater/auxiliary feedwater line breaks are limited to the main steam valve room (MSVR) and the pipe chase.

Because the M&E release rates associated with a feedwater line break are lower than the M&E release rates for a main steam line break (MSLB), the pressures and temperatures in the MSVR and the pipe chase following a MSLB bound those associated with a feedwater line break.

In accordance with the current licensing basis, the main feedwater line break is the limiting high energy line break for MSVR flooding. The minor estimated increase in the feedwater temperature is considered slightly beneficial to the flooding analysis due to the associated density effect and resultant reduction in the volumetric break flow rate following the main feedwater or auxiliary feedwater pipe rupture; consequently, the results of the current flooding analysis documented in FSAR Section 3K.4.1.2.7, remain bounding for the MUR-PU conditions.

Consequently, the current design basis compartment environmental response to postulated high energy line breaks (HELBs) outside the containment remain valid for the MUR-PU.

2. As discussed in the response to Question 1 above, the MUR-PU conditions are bounded by existing evaluations. Therefore, there is no need to assess the impact on qualified nonsafety-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.

SNC Response:

Cameron has performed Unit specific bounding uncertainty analyses for Farley Unit 1 and Unit 2 (Engineering Reports ER-1180NP, Rev. 1 and ER-1181NP, Rev. 1). Copies of these reports were provided in Attachments 5 and 6 of the LAR. The methodology used for calculating uncertainties in these analyses is based on American Society of Mechanical Engineers (ASME) Performance Test Code (PTC) 19.1-1985 and is consistent with Cameron's Topical Report ER-80P, as supplemented by ER-157P.

Uncertainties for the parameters that are not statistically independent are arithmetically summed to produce groups that are independent of each other, which can be statistically combined. Then, all independent parameters/groups that contribute to the power measurement uncertainty are combined using a square root of the sum of the squares (SRSS) approach. Finally, systematic biases are added to the result to determine the overall power measurement uncertainty.

WCAP-12771 is titled "Westinghouse Revised Thermal Design Procedure Instrument Uncertainty Methodology for Alabama Power Farley Nuclear Plant Units 1 and 2 (Model 54F Steam Generators and 2841 NSSS Power." In general, the methodology used in this WCAP to combine uncertainties for an instrument channel is the SRSS (of those groups of components that are statistically independent). The uncertainties that are dependent are combined arithmetically into independent groups, which are then systematically combined (for example, sensor calibration accuracy is considered dependent with sensor measurement and test equipment accuracy). In general, most uncertainties are considered to be random, two-sided distributions. Some uncertainties that are considered non-random (i.e. biases, for example a seismic shift) are treated outside of the radical of the SRSS. The SRSS technique is endorsed by the NRC and has been endorsed by numerous industry standards (i.e., ISA Standard S67.04, Part I, 1994, "Setpoints for Nuclear Safety-Related Instrumentation.")

The Reactor Power Uncertainty calculation is presently performed using the SRSS methodology noted above. The plant performs a primary/secondary side heat balance once every 24 hours when power is above 15% RTP. Assuming that the primary and secondary sides are in equilibrium, the core power is determined by summing the thermal output of the steam generators, correcting the total secondary power for steam generator blowdown (if not secured), subtracting the RCP heat addition, adding the primary side system losses, and dividing by the core rated Btu/hr at full power. The calorimetric power measurement uses multiple instrument channels (including feedwater temperature, feedwater pressure, feedwater flow, steamline pressure, steam generator blowdown flow). Feedwater flow can be measured by either a venturi

and a differential pressure transmitter, or an ultrasonic flow measurement (such as the Cameron Leading Edge Flow Meter (LEFM)). The “current” feedwater instrumentation would be the venturi and differential pressure transmitter, whereas the MUR-PU proposes using the Cameron LEFM.

The methodology for calculating uncertainties is consistent between the current feedwater instrumentation and the LEFM. The SRSS (which is the methodology used in WCAP-12771 [ML17355A516]) is the current setpoint methodology used for Farley. The SRSS methodology is also used for the LEFM.

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.

SNC Response:

Increased BHP for the condensate pumps was analyzed to support the MUR-PU. The MUR-PU BHP for the condensate pumps at full power conditions was determined as 2,909 hp (2,169 kw) for Unit 1 and 2,937 hp (2,190 kw) for Unit 2. The Electrical Transient Analyzer Program (ETAP) load analyzing model does not have the actual condensate pump power load model before and after MUR power uprate. In ETAP, the power loads for all condensate pumps were conservatively assumed as 3000 hp for each pump. Therefore, the MUR power uprate increased condensate pump loads (U1: 2,909 hp & U2: 2,937 hp) are bounded by the 3,000 hp ETAP calculation load model. The load evaluation of the Unit Auxiliary Transformers (UATs) and Startup Auxiliary Transformers (SATs) for 100% power and 102% power, for Unit 1 and for Unit 2 (ETAP models) do not have any difference. As a result, the condensate pump increased BHP of 25 HP per condensate pump will not adversely affect the UATs and SATs analyzed power capability.

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.

SNC Response:

The Farley MUR uprate evaluation and review of the IST program was based on the fifth ten-year interval IST Program Plan which complies with the requirements of the ASME OM Code 2004 Edition through the 2006 Addenda. The fifth ten-year interval began on December 1, 2017 and concludes on November 30, 2027. A copy of the IST Program was transmitted to the NRC via letter NL-17-2005, dated March 11, 2019 (ML19070A247).

The inservice examination and testing of snubbers is in alignment with the plant ISI and IST programs. The IST program for snubbers uses the same ASME OM code discussed above. The ISI program for snubbers uses ASME Section XI Code 2007 Edition through 2008 Addenda which is the ISI program code for the fifth ten-year interval.

EMIB RAI No. 2

NRC staff could [not] 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.

SNC Response:

The only safety-related snubbers impacted by the MUR are in the main steam system for Farley Unit 1 and Unit 2. The snubbers have load increases associated with increased turbine trip fluid transient loads related to the increased flow rates in the main steam piping due to MUR. There are no safety related snubbers installed inside containment on main steam piping on either Unit, and there are a total of eight safety related snubbers installed on main steam piping inside the main steam valve rooms. There are five safety related snubbers installed on main steam piping on Unit 1 and three safety related snubbers installed on main steam piping on Unit 2. All eight snubbers were evaluated and determined to have adequate margin and are acceptable for post-MUR conditions. See the table below:

Farley Unit 1 & 2 Safety Related Snubber Evaluation Summary for MUR-PU						
Support No	NMP-ES-057-001 Listing	Node	Existing Turbine Trip Load (lbs)	Increase in Turbine Trip Load (lbs)	Existing Turbine Trip Load + Increase in Turbine Trip Load (lbs)	Qualified/ Allowable Load (lbs)
Unit 1						
MS-132	MS-R132B	255	20886	418	21304	93000
MS-127	MS-R127B	401	35044	701	35745	93000
MS-121	MS-R121B	743	27105	542	27647	92000
MS-R111	MS-R111	827	8249	165	8414	111000
MS-R109	MS-R109	892	9237	185	9422	111000
Unit 2						
2MS-R503	2MS-R503A	72	24887	498	25385	50000
2MS-R510	2MS-R510A	230	41478	830	42308	106000
2MS-R516	2MS-R516B	392	26329	527	26856	37500

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).

SNC Response:

Based on the guidance in NRC Regulatory Issue Summary 2002-03, the potential adverse impact from flow-induced vibration (FIV) due to the MUR-PU was evaluated for the applicable safety-related components in the plant. Specifically, Section IV.1.A.ii.b of Attachment 4 of the LAR summarized the evaluation which addressed any potential impact of the MUR-PU on FIV within the reactor vessel (RV) internals components. It concluded the MUR-PU does not adversely impact the FIV with regards to the RV internals because the MUR-PU operating parameters are only negligibly affected.

Similarly, section IV.1.A.vi.c of Attachment 4 of the LAR summarized the evaluation which addressed potential impact of the MUR-PU on FIV in the steam generators (SGs), most notably the SG tubes. It also concluded the MUR-PU does not have an adverse impact on the FIV in the SGs.

NEI 08-10, "Roadmap for Power Uprate Program Development and Implementation," while not endorsed by the NRC, does provide a high-level roadmap for power uprate program development and implementation. This document represents a compilation of power uprate lessons learned and was developed to improve the overall execution of power uprates. The term power uprates as used in this report refers to Extended Power Uprate (EPU), Stretch Power Uprate (SPU), and Measurement Uncertainty Recapture (MUR) Power Uprate (PU). Section 4.9 of the NEI guidance first discusses flow induced vibration relative to reactor vessel internals while referencing Regulatory Guide 1.20, "Comprehensive Vibration Assessment Program for Reactor Internals during Preoperational and Initial Startup Testing," and NRC RS-001, "Review Standard for Extended Power Uprates."

Section 4.9 then goes on to mention that there are a number of areas in the Balance of Plant (BOP) side where evaluation or analysis should be performed on the impact of vibration changes due to a power uprate including the Main Steam Supply System, the Moisture Separator Reheater tubes, the Feedwater Piping Systems, the Feedwater Heater tubes, and the Turbine Generator Building structural analysis. Due to the smaller power increase associated with this MUR-PU, the impact on the BOP side will be less than that from a SPU, such as was done for Farley Units 1 and 2 (ML012140259).

For the MUR-PU, Moisture Separator Reheater (MSR) tubes and Feedwater Heater (FWH) tubes were reviewed for vibration and remain adequate for operation at the MUR-PU conditions. The volumetric cycle steam flow increase for the MSRs is less than 0.2% compared to the current conditions and will have an insignificant impact on tube vibration. For the FWH tubes, the tube cross flow velocities are increasing, but remain well below the critical flow velocities.

For the BOP piping, the implementation of MUR-PU will result in slightly higher flow rates for piping systems within the main power cycle. Piping systems experiencing these higher flow rates need to be reviewed for potential flow-induced vibration issues. The piping vibration reviews and piping vibration monitoring plan are included as part of the start-up testing program related to the overall implementation of the MUR-PU. The piping vibration monitoring plan includes performance of Pre-MUR baseline walkdowns of affected piping to establish a baseline of existing piping vibrations / displacements. The plan also includes post-MUR walkdowns performed at two Post-MUR power levels, the first at the pre-MUR 100% power level to determine whether piping vibrations / displacements have increased from their Pre-MUR levels, and the second at 100 percent of Post-MUR power. Areas of the piping exhibiting a slight peak to peak displacement, less than 1/8", are considered acceptable. Piping exhibiting a noticeable peak to peak displacement, typically equal to or greater than 1/8", is further evaluated to ensure acceptability at the MUR conditions.

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.

SNC Response:

The revised CHECWORKS™ models result in some components' wear rates increasing due to the MUR-PU. Predicted wear rate values which are increasing due to MUR-PU are summarized in the table below. These increased wear rates may alter decisions as to the selection of locations for examination and may result in inspections being required more frequently to identify and correct potential areas of increased degradation.

In accordance with existing FAC procedures, the selection of components for examination includes the results of lines analyzed using the CHECWORKS™ model. Predicted changes in wear rates will be evaluated along with minimum wall thickness requirements to determine the remaining service life of susceptible piping and components. The results of inspections, which may be required more frequently after implementation of MUR-PU, will continue to determine if plans should be made for repair or replacement of affected components.

Description	Number of Analyzed Component Sections	Average Change in Wear Rate (%)	Average Change in Wear Rate (mils/yr)	Average Post Uprate Wear Rate (mils/yr)
Unit 1				
Heater Drain Connection to SGFP	71	7.5%	0.066	0.947
Condensate Flow LP FWH 4 to LP FWH 5	52	14.8%	0.174	1.345
Condensate Flow LP FWH 5 to Heater Drain Connection	35	29.8%	0.294	1.279
HP Turbine Exhaust to MSRs	174	7.9%	0.144	1.962
Extraction Steam to FWH 1	148	9.4%	0.595	6.938
Extraction Steam to FWH 2	82	18.8%	0.330	2.085
Feedwater from SGFP to HP FWH 6	85	6.5%	0.094	1.546
Heater Drain Tank to Heater Drain Pump	38	0.7%	0.006	0.909
Heater Drains LP FWH 5 to HDT	38	0.8%	0.005	0.652
Heater Drains HP FWH 6 to HDT	50	2.3%	0.014	0.619
MSR 2ND Stage Drain Tank to HP FWH 6	123	24.7%	0.156	0.788
Drains from MSR to MSR Shell Drain Tank	67	11.1%	0.047	0.471
Drains MSR Shell Drain Tank to Heater Drain Tank	85	51.9%	0.316	0.925
Unit 2				
Extraction Steam to FWH 2	76	39.4%	0.439	1.556
MSR 1ST Stage Drain Tank to Heater Drain Tank	222	0.4%	0.005	1.080
Drains from MSR to MSR Shell Drain Tank	164	9.7%	0.070	0.798
Heater Drains HP FWH6 to Heater Drain Tank	52	1.4%	0.012	0.865
Extraction Steam to LP FWH 5	66	11.2%	0.342	3.388
Extraction Steam to HP FWH 6	5	9.3%	0.187	2.194
Extraction Steam to HP FWH 6	124	8.9%	0.068	0.834
HP Turbine Exhaust to MSRs	152	9.5%	0.605	6.962
Heater Drains LP FWH 4 to LP FWH 3	33	2.5%	0.022	0.903

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.

SNC Response:

As discussed in Section VII.6.B of Attachment 4 to the LAR, Farley complies with the intent of ANSI N101.2. The dose criteria within ANSI N101.2 is not power level dependent and does not change with the implementation of the MUR. The post-accident containment pressures and temperatures were not affected by the MUR as discussed in Section II.1.D.iii.28 of Attachment 4 of the LAR. Therefore, the protective coatings program at the MUR-PU conditions is unaffected and remains valid after implementation of MUR-PU.

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.

SNC Response:

The limits for chloride, fluoride, and oxygen specified in UFSAR Table 5.2-22 fall within the range for Action Level 1 per the EPRI Primary Chemistry Guidelines (Reference 1). Plant operation is permitted to continue during an Action Level 1 event. Values that are above the Table 5.2-22 limits (> 0.15 ppm or 150 ppb chloride, > 0.15 ppm or 150 ppb fluoride, > 0.1 ppm or 100 ppb oxygen) would be Action Level 2 and an indication that corrective action is needed to prevent significant damage. Exceeding the Action Level 2 limits for 24 hours would require an orderly plant shutdown and cooldown to < 250°F as quickly as is safely possible. The Table 5.2-22 limits account for startup/shutdown conditions and are consistent with the limits provided in the EPRI Guidelines prior to exceeding 250°F.

No significant changes in the bulk chemistry of the primary side are expected due to the MUR power uprate, because it is expected that the bulk chemistry will continue to be controlled after the uprate by plant procedures that are consistent with EPRI Guidelines and site strategic plans. In addition, the uprate temperatures for Farley are in the same range as other plants which control bulk chemistry based on the same industry guidelines. EPRI Guidelines recognize the difference in design and operating characteristics of nuclear plants and prescribe that each plant generate strategic water chemistry plans for both the primary and secondary side.

The Farley Primary Water Chemistry Strategic Plan (Reference 2) incorporates the current EPRI Primary Water Chemistry Guidelines. Descriptions of EPRI Action Levels 1, 2, and 3 are included, as well as the Action Level limits and actions required for chloride, fluoride, and oxygen during power operation.

The planned MUR power uprate is not expected to have any impact on the strategies or goals in site strategic plans. The ability to maintain the reactor coolant chemistry within the UFSAR chemistry specifications and EPRI Guidelines limits will also not be affected. Therefore, SG tube integrity will continue to be able to be maintained after the MUR power uprate.

References:

1. *Pressurized Water Reactor Primary Water Chemistry Guidelines: Volume 1, Revision 7*. EPRI, Palo Alto, CA: 2014. 3002000505.
2. Southern Nuclear Document NMP-CH-100-GL01, Version 2.0, “Farley Primary Water Chemistry Strategic Plan,” May 3, 2018.

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.

SNC Response:

1. The 24 ksi endurance limit is from ASME Section III, Appendix I Figure I-9.2.2. "Design Fatigue Curves for Austenitic Steels, Nickel-Chromium-Iron Alloy, Nickel-Iron-Chromium Alloy, and Nickel-Copper Alloy, …," with tabulated data provided in Table I-9.2.2 for the curves provided. The value is from Curve A with the endurance limit for 1E11 cycles reported to be 23.7 ksi (rounded up to 24 ksi). The stress levels for the tubes at the vibration levels calculated have been shown to be well below this limit. The 1989 Edition, No Addenda is the ASME Code version to which the Farley Model 54F replacement steam generators were designed.

2. The parameters provided in Table IV-1 of Attachment 4 of the LAR only include the parameters at 0% SG tube plugging for simplicity. The complete set of thermal design parameters which were considered in all the NSSS analyses, including the SG tube degradation, were provided in Table 3.2-1 of the LAR Enclosure. That table listed parameters at 0%, 15% and 20% SG tube plugging. For the SG tube integrity evaluation, the thermal design parameters associated with the higher SG tube plugging level were utilized because they are the bounding condition.

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.

SNC Response:

The purpose of the proposed change was to facilitate the RCS flow changes assumed in the MUR analysis. Specifically, while the RCS thermal design flow (TDF) value is not changing for the MUR, the minimum RCS total flow rate values currently specified in TS LCO 3.4.1 will be changing as discussed in the LAR Technical Evaluation. In accordance with NRC approval of WCAP-14483-A, "Generic Methodology for Expanded Core Operating Limits Report" and TSTF-339-A, Revision 2, "Relocate TS Parameters to COLR," the minimum limit for RCS flow (TDF) is proposed for inclusion in the LCO.

The justification for deleting the two flow measurement methods specified in the LCO while replacing them with the minimum limit for RCS flow is contained within WCAP-14483-A and the associated NRC Safety Evaluation Report (SER). This change is also consistent with LCO 3.4.1 in NUREG-1431.

The reason why there is a flow value in this LCO as well as a reference to the COLR is explained below. The SER for WCAP-14483-A agreed with the Westinghouse Owners Group request to:

1. Revise TS 3.4.1 of NUREG-1431, RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits, to relocate the pressurizer pressure, RCS average temperature (T-avg), and RCS total flow rate values to the COLR. The minimum limit for total flow based on that used in the reference safety analysis will be retained in the TS.

The reasoning behind the minimum limit remaining in the TS was discussed in the SER:

> *Although some plants operate with lower steam generator tube plugging levels and thus higher RCS flow rates than those assumed in the safety analyses, a change in RCS flow is an indication of a physical change to the plant which should be reviewed by the NRC staff. Because of this, the staff recommended that if RCS flow rate were to be relocated to the COLR, the minimum limit for RCS total flow based on a staff approved analysis (e. g., maximum tube plugging) should be retained in the TS to assure that a lower flow rate than reviewed by the staff would not be used.*

On page 2 of the SER, the following statement is made regarding the DNB parameter limits:

> *The DNB parameter limits are also based on initial conditions assumed for accidents in which precluding DNB is not a criterion.*

Because the lowest flow rate used in any of the safety analyses is the thermal design flow (TDF), the value for the 'minimum limit for total flow' which complies with the SER requirement is the TDF value of 258,000 gpm, which is proposed for inclusion in LCO 3.4.1.

Draft changes to the TS Bases are provided below for information only. Specifically, the first paragraph under the TS LCO section for TS 3.4.1 is intended to be changed as shown in the bold text below:

> This LCO specifies limits on the monitored process variables - pressurizer pressure, RCS average temperature, and RCS total flow rate - to ensure the core operates within the limits assumed in the safety analyses. **These variables are contained in the COLR to provide operating and analysis flexibility from cycle to cycle. However, the minimum RCS flow, based on the maximum analyzed steam generator tube plugging, is retained in the TS LCO.** Operating within these limits will result in meeting the DNB design criterion in the event of a DNB limited transient.