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Dominion Energy®

June 2, 2021

U.S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, DC 20555

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| NRA/SS | R0 |
| Docket No. | 50-423 |
| License No. | NPF-49 |

**DOMINION ENERGY NUCLEAR CONNECTICUT, INC.**
**MILLSTONE POWER STATION UNIT 3**
**RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION REGARDING LICENSE AMENDMENT REQUEST FOR MEASUREMENT UNCERTAINTY RECAPTURE POWER UPRATE**

By letter dated November 19, 2020 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML20324A702), Dominion Energy Nuclear Connecticut, Inc. (DENC), submitted a license amendment request (LAR) to the Nuclear Regulatory Commission (NRC) for Millstone Power Station, Unit No. 3 (MPS3). The proposed license amendment would increase the rated thermal power (RTP) level from 3,650 megawatts thermal (MWt) to 3,709 MWt in the MPS3 operating license and in Technical Specification (TS) 1.27, an increase in RTP of approximately 1.6%. The proposed increase is referred to as a measurement uncertainty recapture (MUR) power uprate and is based on utilizing an installed Cameron Technology US LLC (currently known as Sensia, formerly known as Caldon) Leading Edge Flow Meter CheckPlus system as an ultrasonic flow meter located in each of the four main feedwater lines supplying the steam generators to improve plant calorimetric heat balance measurement accuracy. The proposed changes would also involve an editorial correction to TS 2.1.1.1 and revision to TS 3.7.1.1, Action Statement "a" and TS Table 3.7-1, "Operable MSSVs Versus Maximum Allowable Power" to update the maximum allowable power levels corresponding to the number of operable main steam safety valves per steam generator.

In an email dated April 8, 2021, the NRC issued a draft request for additional information (RAI) related to the proposed LAR. On April 20, 2021, the NRC staff conducted a conference call with DENC staff to clarify the request. In an email dated April 22, 2021, the NRC transmitted the final version of the RAI (ADAMS Accession No. ML21112A308). DENC agreed to respond to the RAI within 45 days of issuance, or no later than June 7, 2021.

Attachment 1 provides DENC's response to the RAI. Attachment 2 provides a supplement to clarify that the methodology selected for DENC's planned Loss of Normal Feedwater (LONF) reanalysis has changed from what is presented in MUR LAR, Attachment 4, Sections II.1.10 and III.4-1.1. Instead, the LONF event revision will be performed using an alternate, NRC-approved, methodology.

If you have any questions or require additional information, please contact Shayan Sinha at (804) 273-4687.

Sincerely,

Gerald T. Bischof
Senior Vice President – Nuclear Operations & Fleet Performance

COMMONWEALTH OF VIRGINIA )
)
COUNTY OF HENRICO )

The foregoing document was acknowledged before me, in and for the County and Commonwealth aforesaid, today by Mr. Gerald T. Bischof, who is Senior Vice President – Nuclear Operations and Fleet Performance of Dominion Energy Nuclear Connecticut, Inc. He has affirmed before me that he is duly authorized to execute and file the foregoing document in behalf of that company, and that the statements in the document are true to the best of his knowledge and belief.

Acknowledged before me this 2ND day of June, 2021.

My Commission Expires: 12/31/24.

Notary Public

CRAIG D SLY
Notary Public
Commonwealth of Virginia
Reg. # 7518653
My Commission Expires December 31, 2024

Attachments:

1. Response to Request for Additional Information Regarding License Amendment Request for Measurement Uncertainty Recapture Power Uprate
2. Supplement to Clarify Methodology for Revised Loss of Normal Feedwater Reanalysis

Commitments made in this letter: None

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# ATTACHMENT 1

## Response to Request for Additional Information Regarding License Amendment Request for Measurement Uncertainty Recapture Power Uprate

MILLSTONE POWER STATION UNIT 3
DOMINION ENERGY NUCLEAR CONNECTICUT, INC.

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This attachment provides DENC's response to the RAI.

***<u>Regulatory Basis: Vessels and Internals (NVIB) RAIs</u>***

*The regulation at 10 CFR 50.55a requires that the reactor pressure vessel (RPV) be constructed, designed and analyzed in accordance with the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section III.*

*The regulation at 10 CFR 50.61 requires pressurized thermal shock (PTS) evaluations to ensure that adequate fracture toughness exists for RPV beltline materials in pressurized water reactors (PWRs) to protect against failure during a PTS event. Fracture resistance of RPV beltline materials during PTS events is evaluated by calculating the nil-ductility temperature (RTNDT) for PTS (identified as RTPTS). Section 50.61(b)(1) requires that PWR licensees have projected values of RTPTS accepted by the NRC for each RPV beltline material. Section 50.61(c)(2) requires that RTPTS calculations for RPV beltline materials incorporate credible RPV surveillance material test data that are reported as part of the RPV materials surveillance program required by 10 CFR Part 50, Appendix H.*

**NVIB-RAI-1**

*Section IV.1.A.ii.e, "Mechanical Evaluation," in Attachment 4 of the LAR states that the MUR power uprate design conditions do not affect the current design bases for seismic and loss-of-coolant accident (LOCA) loads. The licensee further stated that the stress levels caused by the flow induced vibration on the core barrel assembly and upper internals are low and remain well below the material high-cycle fatigue endurance limit. Summarize the mechanical evaluation that demonstrates the core barrel assembly and upper internals are not affected by the MUR power uprate design conditions, including a discussion on the acceptance criteria, resulting stresses and fatigue endurance limits.*

**DENC Response to NVIB-RAI-1**

The goal of analyzing flow-induced vibration (FIV) on the core barrel assembly and upper internals is to assess its impact on structural integrity of the components contained within the MPS3 reactor vessel. Extensive evaluations were previously completed in support of the MPS3 stretch power uprate (SPU) LAR, which involved a Rated Thermal Power (RTP) increase of 7%. The MPS3 MUR power uprate LAR, which is currently under NRC review, further increases the RTP level by approximately 1.6%. The MUR power increase considers changing the operating power plus uncertainty but remains bounded by the SPU operating power plus uncertainty which was previously analyzed. Based on this understanding, a comparison to the evaluations completed for the SPU increase was relied upon for justification of the MUR increase.

The methodology followed in the SPU evaluation in SPU LAR Attachment 5 (Reference 1-1), Section 2.2.3 involved scaling the structural response of the FIV according to analytical and experimental formulations. Specifically, by relating parameters such as flow rate (Mechanical Design Flow or "MDF"), vessel inlet temperature, and vessel outlet temperature. The change in input parameters is shown to be negligible (<1%), so the FIV evaluation completed for the SPU program was considered to be applicable to the MUR program, as these parameters are the only change noted in this evaluation between the two programs.

The SPU evaluation included the reactor internals components that are considered to be limiting with regards to FIV. These components consist of the core barrel in the lower internals assembly and the guide tubes in the upper internals assembly. For other reactor internal components such as the lower radial restraints, upper core plate alignment pins, lower support plate, and the lower support columns, the vibratory response is extremely small. The calculated stresses were obtained by scaling previously generated FIV stresses to what would be expected after SPU implementation. For conservatism, the values were scaled based on the hot functional flow rates, which are typically about 4% higher than the MDF rate.

Furthermore, the seismic and LOCA analyses for the MPS3 SPU program are considered to be applicable to the MUR program. The seismic inputs and plant model parameters from the SPU program are also applicable to the MUR power uprate. Additionally, the

conservative assumptions in the LOCA force calculations bound the changes found in the MUR program. Therefore, existing SPU parameters were determined to be applicable to the MUR conditions.

The ASME Code (1998 Edition with 2000 Addenda, Section III, Division 1 and Section II, Part D) combined with measured data, forms the basis for the acceptance criteria for mechanically induced stresses/strains produced by FIV. Although the particular components analyzed do not constitute a pressure boundary, the acceptability of the components impacted by the uprate relating to alternating stresses for high-cycle fatigue was assessed. The alternating stresses were calculated and compared according to the ASME Code rules on high-cycle fatigue or measured experimental data on strains limits in the MPS3 SPU LAR.

**NVIB-RAI-2**

*Section IV.1.A.ii.f, "Structural Evaluation," in Attachment 4 of the LAR states that evaluations were performed to demonstrate that the structural integrity of reactor internal components is not adversely affected by the MUR power uprate design conditions. The NRC staff requests the licensee to (a) summarize the structural evaluation that demonstrates the reactor internal components are not adversely affected by the MUR power uprate design conditions, including a discussion on the acceptance criteria and resulting stresses and (b) discuss whether there are any cracks in any of the RPV internal components. If there are any cracks, discuss whether an evaluation has been performed and how this evaluation demonstrates sufficient structural integrity of the degraded reactor internal component under the MUR power uprate conditions.*

**DENC Response to NVIB-RAI-2**

**DENC Response to NVIB-RAI-2, part (a)**

Extensive evaluations were previously completed in support of the MPS3 SPU LAR, which involved an RTP increase of 7%. The MPS3 MUR power uprate LAR, which is currently under NRC review, further increases the RTP level by approximately 1.6%. The MUR power increase considers changing the operating power plus uncertainty but remains bounded by the SPU operating power plus uncertainty which was previously analyzed. Based on this understanding, a comparison to the evaluations completed for the SPU increase was relied upon for justification of the MUR increase.

The inputs for all components remained unchanged, thus no new calculations were performed, and the results from the SPU calculations were documented in the SPU LAR. Five areas were evaluated, including the following: Control Rod Insertability Evaluation, Baffle Bolt Evaluation, FIV Evaluation, Critical Reactor Internal Components Structural Evaluation, and an Upper and Lower Core Plate Evaluation. The FIV evaluation is addressed in NVIB-RAI-1, the Upper and Lower Core Plate Evaluation is addressed in

NVIB-RAI-3, and the Baffle Bolt Evaluation is addressed in NVIB-RAI-4. Further discussion on the remaining two evaluations is included below.

*Control Rod Insertability Evaluation*
This evaluation determined the maximum mechanical load acting on the guide tubes. This load was then compared to the allowable load on the guide tubes. The methodology assessed the predicted lateral load/displacement against drop tests that applied lateral loads/deflections to the guide tube to confirm control rod insertion. These tests considered the effects of both temporary (elastic) and permanent (plastic) deformation.

The applicable loadings that were involved in this evaluation include mass flow and acoustic loads, system loads, and safe shutdown earthquake (SSE) loads. These three loads, as previously generated for the SPU program, were deemed to be applicable for use in the MUR program. This applicability was confirmed based on either bounding or unchanging inputs to the specified loading conditions.

MPS3 uses a 17x17, 96-inch style guide tube. The allowable loads and acceptance criteria for this style of guide tube and control rod insertion were shown to be met for the SPU analysis documented in SPU LAR Attachment 5 (Reference 1-1), Section 2.2.3 (which bound MUR conditions).

*Critical Reactor Internal Components Structural Evaluation*
The purpose of this evaluation was to assess the impact of the MUR power uprate program on the MPS3 reactor internal components including the lower core support plate atypical region, lower support columns, and core barrel nozzle weldments. The following methodology was previously used for the SPU program evaluation in SPU LAR Attachment 5, Section 2.2.3:

1. Determine the changes in load conditions due to the power uprate. Generally, the power uprate would change the loads on the internal components during normal and upset conditions due to Heat Generation Rates (HGR) from gamma heating and thermal fluid transients.
2. Compare the MPS3 core support component design configurations to other design configurations that have been used for detailed stress analyses, which were performed under similar loading conditions. Any differences in design dimensions need to be reconciled in the component stress evaluation. If there are any major design differences, component stresses need to be determined independently.
3. Compare thermal loadings (thermal transients and HGRs) used in core support component analysis, which were performed under similar loading conditions to the thermal loadings for MPS3 uprate conditions.
4. Determine if the thermal loadings of similar components in detailed stress analyses, which were performed under similar loading conditions bound the MPS3 thermal loadings.

The primary inputs to these evaluations include heat generation rates and design transients. Both of these inputs are deemed to be applicable to both the SPU and MUR programs, due to conservatisms included in the SPU analysis.

No plant-specific ASME Code stress report was written for MPS3. The reactor internal components were analyzed to meet the intent of the ASME Code, Section III criteria. Therefore, based on the previous evaluations and current practices, the guidelines in Subsection NG of the ASME Code were used for this evaluation. In conclusion, the allowable loads and acceptance criteria for the internal components analyzed were shown to be met for the SPU analysis (which bound MUR conditions).

**DENC Response to NVIB-RAI-2, part (b)**

As part of the 10-year visual examinations, DENC has not identified cracking in any MPS3 RPV internal components.

**NVIB-RAI-3**

*Section IV.1.A.ii.g, "Upper and Lower Core Plate Structural Analysis," in Attachment 4 of the LAR states that thermal design transients, heat generation rates, and operating conditions affect thermal loads on the upper and lower core plates. The licensee stated that for the MUR power uprate, current analysis of record (AOR) thermal design transients and heat generation rates remain applicable because the MUR power uprate operating conditions are bounded by the operating conditions in the current AOR. The licensee further stated that the maximum primary plus secondary stress intensity of the upper and lower core plate and cumulative usage factor remain acceptable. Summarize how the existing structural analysis for the upper and lower core plates is still applicable under the MUR power uprate conditions, including a discussion on the acceptance criteria and resulting stresses.*

**DENC Response to NVIB-RAI-3**

Extensive evaluations were previously completed in support of the MPS3 SPU LAR, which involved an RTP increase of 7%. The MPS3 MUR power uprate LAR, which is currently under NRC review, further increases the RTP level by approximately 1.6%. The MUR power increase considers changing the operating power plus uncertainty but remains bounded by the SPU operating power plus uncertainty which was previously analyzed. Based on this understanding, a comparison to the evaluations completed for the SPU increase was relied upon for justification of the MUR increase.

The purpose of the Upper and Lower Core Plate Evaluation was to assess the impact on structural integrity of MPS3 reactor internals lower core plate (LCP) and upper core plate (UCP) with regard to the proposed MUR power uprate program. The methodology for qualifying the structural integrity of the MPS3 UCP and LCP was investigated to determine the driving inputs which produce the stresses on the upper and lower core plates. It was determined that the LCP and UCP geometry and material, core plate

supports, HGRs and loads, fluid-thermal loads, thermal design transients (including number of cycles) and mechanical loads are all used in the stress calculation for the LCP and UCP. For the MUR power uprate, it was either determined that the inputs listed have not changed, that the change observed is negligible, or that the observed change is bounded by the AOR.

The original LCP and UCP evaluation documented effects on the SPU program for MPS3 in SPU LAR Attachment 5 (Reference 1-1), Section 2.2.3. This evaluation made use of analyses completed for a similar plant during a replacement steam generator (RSG) program. Therefore, the structural qualification and fatigue evaluation is the original design basis of the MPS3 reactor internals. Based on similarities in design and thermal loading, the evaluation completed for the LCP and UCP at the similar plant was used in the current analysis for the MPS3 SPU program.

The allowable stresses are obtained from the ASME Code Requirements (1974 edition, Division 1, Section III, Subsection NG) which is consistent with the original evaluation. For this evaluation the allowable stress, as prescribed by the 1974 ASME Boiler Code, were used as the acceptance criteria.

**<u>NVIB-RAI-4</u>**

*Section IV.1.A.ii.h, "Baffle-Barrel Region Evaluations," in Attachment 4 of the LAR states that the baffle bolts are subjected to primary loads consisting of deadweight, hydraulic pressure differentials, LOCA and seismic loads, and secondary loads consisting of preload and thermal loads resulting from reactor coolant system (RCS) temperatures and gamma heating rates. The licensee stated that it evaluated the baffle former bolt maximum displacement at the MUR power uprate design conditions. The licensee concluded that the existing thermal and structural analysis of the baffle-barrel region results remain bounding for the MUR power uprate design conditions. The NRC staff requests the licensee to summarize (a) how the existing thermal and structural analysis of the baffle barrel region remain bounding under the MUR power uprate design conditions, including a discussion on the baffle former bolt maximum displacement and stresses under the MUR power uprate design conditions and (b) the inspection history and results of former baffle bolts and plates.*

**<u>DENC Response to NVIB-RAI-4</u>**

**<u>DENC Response to NVIB-RAI-4, part (a)</u>**

Extensive evaluations were previously completed in support of the MPS3 SPU LAR, which involved an RTP increase of 7%. The MPS3 MUR power uprate LAR, which is currently under NRC review, further increases the RTP level by approximately 1.6%. The MUR power increase considers changing the operating power plus uncertainty but remains bounded by the SPU operating power plus uncertainty which was previously

analyzed. Based on this understanding, a comparison to the evaluations completed for the SPU increase was relied upon for justification of the MUR increase.

The baffle bolt evaluation was used to assess the structural acceptability of the baffle-former bolts for the MPS3 MUR power uprate program. The methodology used in the SPU power uprate evaluation in SPU LAR Attachment 5 (Reference 1-1), Section 2.2.3 determined the cumulative fatigue damage resulting from the thermal loading. This cumulative fatigue damage factor was then compared to the allowable factor. Calculation of the damage factor was completed using the following methodology:

1. The geometry of the MPS3 baffle-former bolts was reviewed and compared to the bolt geometry used to determine a design fatigue curve. This fatigue curve was developed from test data used for qualification of the standard four-loop, upflow, baffle-former bolts. The generation of this fatigue curve included the margins specified in the ASME Boiler and Pressure Vessel Code, Section III, Division 1, 1998 Edition, for using test results to qualify a component. It was assumed that the fatigue curve had the same "shape" as the design fatigue curve for stainless steel. This is consistent with the procedure provided in the ASME Code, Appendix II, for performing fatigue tests using accelerated loadings.
2. Baffle-barrel temperatures were determined for normal and upset service conditions. This analysis addressed heating rates associated with the historic and projected fuel loading patterns.
3. Displacements of the baffle plates were determined at the bolt locations, using the previous fatigue curve and the temperatures determined in Steps 1 and 2. Using linear scaling, displacements were also determined for other applicable load conditions.
4. Fatigue usage was determined using Miner's Rule.

The inputs to this evaluation included RCS temperature and flow parameters for MUR and SPU conditions, the four-loop, upflow, baffle-bolt design fatigue curve, design transients, HGRs, and baffle-barrel temperatures. An evaluation of the aforementioned inputs was completed to determine applicability of the SPU parameters for the MUR program. All of the listed inputs were deemed to have not changed and are applicable to the MUR program for MPS3. Therefore, the results from the SPU program are also applicable to the MUR power uprate program.

The conclusion reached is that an acceptable damage factor was calculated for the SPU program, and this remains applicable to the MUR power uprate program. Therefore, the baffle-former-barrel configuration is acceptable for the effects of the MUR, and the calculated usage is well below the allowable fatigue usage limit of 1.0. Note that this analysis considered a full baffle-former bolting pattern. No Acceptable Bolting Pattern Analysis (ABPA) methods were applied in the SPU or the MUR evaluations.

**DENC Response to NVIB-RAI-4, part (b)**

The MPS3 reactor vessel core support barrel former baffle plates and baffle bolting have been subjected to one VT-3 examination during each 10-year ISI interval. The first interval exam was performed during the 3R05 outage (June 1995), the second interval exam was performed during the 3R11 outage (April 2007), and the third interval exam was performed during the 3R17 outage (April 2016). No relevant indications were identified on the core support barrel baffle plates or baffle bolting during any of these exams.

**NVIB-RAI-5**

*Section IV.1.A.iii.a, "Bottom Mounted Instrumentation (BMI)," in Attachment 4 of the LAR discusses the stress analysis of the BMI guide tubes. The licensee stated that the range of vessel core inlet temperatures for the MUR power uprate is 536.7 degrees Fahrenheit (°F) to 555.8°F, which is lower than the RPV core inlet temperature in the existing analysis. These temperatures are bounded by the BMI guide tube design temperature of 560°F. Clarify whether the existing stress analysis of the BMI guide tubes is based on the design temperature of 560°F.*

**DENC Response to NVIB-RAI-5**

The BMI guide tubes stress analysis is based on the design temperature of 560°F. This temperature is bounding of both the pre-MUR and post-MUR power uprate core inlet temperatures.

**NVIB-RAI-6**

*Section IV.1.C.i, "Pressurized Thermal Shock (PTS) Calculations," in Attachment 4 of the LAR states that the limiting reference temperature for PTS (RTPTS) value of 130°F applies to Lower Shell Plate B9820-2. The licensee stated that this is a change from the AOR that had a limiting RTPTS value of 133°F pertaining to Intermediate Shell Plate B9805-1. The NRC staff requests the licensee to (a) clarify the cause of the changes in the RTPTS value from 133°F to 130°F and the limiting beltline material, and (b) discuss whether the RTPTS value of 130°F was derived based on the MUR power uprate conditions.*

**DENC Response to NVIB-RAI-6**

**DENC Response to NVIB-RAI-6, part (a)**

Regulatory Background

Per 10 CFR 50.61, "Fracture toughness requirements for protection against pressurized thermal shock events [Reference 1-3]," the use of results from the plant-specific surveillance program may result in an $RT_{PTS}$ value that is higher or lower than the value calculated without plant-specific surveillance data. As described in MPS3 FSAR Section 5.3, the methodology described in Regulatory Guide (RG) 1.99, Revision 2 [Reference 1-6] is periodically updated to incorporate the effects of irradiation exposure using in reference leg temperature calculations. RG 1.99 includes positions that determine Chemistry Factor (CF) without the use of surveillance data (Position 1.1) or with the use of surveillance data (Position 2.1).

MPS3 Analysis of Record $RT_{PTS}$ Background

The AOR for pressurized thermal shock (PTS) for MPS3 is contained in Table 2.1.3-4 of the SPU LAR, Attachment 5 [Reference 1-1]. The limiting $RT_{PTS}$ value (consistent with the SPU LAR) is shown in Table 5.2-7 of the MPS3 FSAR [Reference 1-2].

As described Section 2.1.3.2.4 in the SPU LAR, Attachment 5:

> *The limiting material is Intermediate Shell Plate B9805-1, with the more limiting $RT_{PTS}$ value occurring for calculations using the RG 1.99, Rev. 2 Position 1.1 Chemistry Factor, as opposed to the Position 2.1 Chemistry Factor calculated from credible surveillance data. The most limiting $RT_{PTS}$ value at 54 EFPY [effective full power years] for Plate B9805-1 is 133°F.*

Thus, the AOR limiting $RT_{PTS}$ value is considered to be 133°F, which corresponds to Intermediate Shell Plate B9805-1 evaluated with a Position 1.1 CF (i.e., calculated without the use of surveillance data). However, using the credible surveillance plate data per Position 2.1, an $RT_{PTS}$ value of 111°F was calculated for Intermediate Shell Plate B9805-1 in the SPU analysis (see Table 2.1.3-4 of Reference 1-1).

Since the surveillance data is credible and the Position 2.1 calculation results in a lower calculated $RT_{PTS}$ value, the Position 2.1 results could have been used in lieu of the Position 1.1 results for the SPU analysis per 10 CFR 50.61. However, instead of taking credit for the credible surveillance data, the SPU LAR Attachment 5 conservatively reported the Position 1.1 CF result of 133°F for Intermediate Shell Plate B9805-1, instead of the 111°F value from using the Position 2.1 CF.

If the SPU report had elected to take credit for the credible surveillance data for Intermediate Shell Plate B9805-1, this material would no longer have been the most

limiting material. Instead, the limiting $RT_{PTS}$ value would have been 130°F corresponding to Lower Shell Plate B9820-2 (see Table 2.1.3-4 of Reference 1-1).

MPS3 MUR $RT_{PTS}$ Evaluation

The $RT_{PTS}$ calculations supporting the MUR [Reference 1-4] are shown in Table 1-1. This analysis is based on fluence values derived from MUR power uprate conditions. Consideration of the MUR increased the peak 54 EFPY fluence value from the SPU analysis from a value of 2.70 x $10^{19}$ n/cm$^2$ (SPU) to 2.72 x $10^{19}$ n/cm$^2$ (MUR). Since the $RT_{PTS}$ values are reported as whole numbers, this slight increase in fluence did not change the highest calculated $RT_{PTS}$ values. Specifically, the $RT_{PTS}$ values corresponding to Intermediate Shell Plate B9805-1 using Position 1.1 (133°F), Intermediate Shell Plate B9805-1 using Position 2.1 (111°F), and Lower Shell Plate B9820-2 (130°F) are unchanged from the fluence increase. It is also noted that the Position 2.1 CF for Intermediate Shell Plate B9805-1 for both the SPU and MUR utilize the same surveillance data, available in WCAP-16629-NP [Reference 1-5]. The credibility of the data is also assessed in WCAP-16629-NP.

The MUR LAR $RT_{PTS}$ analysis takes credit for the available credible surveillance data for Intermediate Shell Plate B9805-1, and thus the $RT_{PTS}$ value corresponding to this material is 111°F. The 133°F result based on the Position 1.1 CF is shown for information only. As a result, the limiting $RT_{PTS}$ value is 130°F, which corresponds to Lower Shell Plate B9820-2.

The change in the limiting $RT_{PTS}$ value and material is the result of decisions made with the implementation of 10 CFR 50.61 with respect to surveillance data, and is not a result of changes in fluence projections, surveillance data available since the SPU LAR, and/or material properties. The change in the limiting $RT_{PTS}$ value from 133°F to 130°F and the limiting beltline material complies with 10 CFR 50.61 and the Positions of Regulatory Guide 1.99, Revision 2.

**DENC Response to NVIB-RAI-6, part (b)**

As described in the response to NVIB-RAI-6, part (a), the $RT_{PTS}$ value of 130°F considers the fluence value derived from MUR power uprate conditions. The value of 130°F was unchanged from the analysis of record based on the fluence increase resulting from the MUR power uprate conditions.

**Table 1-1**
**MUR $RT_{PTS}$ Calculations for the MPS3 Beltline and Extended Beltline Materials at 54 EFPY[(a)]**

| Reactor Vessel Location | Reactor Vessel Material | R.G. 1.99, Rev. 2 Position | CF[(b)] (°F) | Fluence[(c)] (n/cm², E > 1.0 MeV) | FF[(c)] | Initial $RT_{NDT}$ [(b)] (°F) | $\Delta RT_{NDT}$ (°F) | $\sigma_U$ (°F) | $\sigma_\Delta$ (°F) | Margin (°F) | $RT_{PTS}$ (°F) |
|---|---|---|---|---|---|---|---|---|---|---|---|
| Beltline | Intermediate Shell Plate B9805-1 | 1.1 | 31.0 | $2.72 \times 10^{19}$ | 1.2671 | 60 | 39.3 | 0 | 17 | 34.0 | 133 |
| | | 2.1 | 26.7 | $2.72 \times 10^{19}$ | 1.2671 | 60 | 33.8 | 0 | 8.5[(d)] | 17.0 | 111 |
| | Intermediate Shell Plate B9805-2 | 1.1 | 31.0 | $2.72 \times 10^{19}$ | 1.2671 | 10 | 39.3 | 0 | 17 | 34.0 | 83 |
| | Intermediate Shell Plate B9805-3 | 1.1 | 31.0 | $2.72 \times 10^{19}$ | 1.2671 | 0 | 39.3 | 0 | 17 | 34.0 | 73 |
| | Lower Shell Plate B9820-1 | 1.1 | 51.0 | $2.72 \times 10^{19}$ | 1.2671 | 10 | 64.6 | 0 | 17 | 34.0 | 109 |
| | Lower Shell Plate B9820-2 | 1.1 | 44.0 | $2.72 \times 10^{19}$ | 1.2671 | 40 | 55.8 | 0 | 17 | 34.0 | 130[(f)] |
| | Lower Shell Plate B9820-3 | 1.1 | 37.0 | $2.72 \times 10^{19}$ | 1.2671 | 20 | 46.9 | 0 | 17 | 34.0 | 101 |
| | Intermediate Shell Longitudinal Weld Seams 101-124 A,B,C | 1.1 | 31.8 | $2.72 \times 10^{19}$ | 1.2671 | -50 | 40.3 | 0 | 20.1[(e)] | 40.3 | 31 |
| | | 2.1 | 6.7 | $2.72 \times 10^{19}$ | 1.2671 | -50 | 8.5 | 0 | 4.2[(e)] | 8.5 | -33 |
| | Intermediate to Lower Shell Girth Weld Seam 101-171 | 1.1 | 31.8 | $2.72 \times 10^{19}$ | 1.2671 | -50 | 40.3 | 0 | 20.1[(e)] | 40.3 | 31 |
| | | 2.1 | 6.7 | $2.72 \times 10^{19}$ | 1.2671 | -50 | 8.5 | 0 | 4.2[(e)] | 8.5 | -33 |
| | Lower Shell Longitudinal Weld Seams 101-142 A,B,C | 1.1 | 31.8 | $2.72 \times 10^{19}$ | 1.2671 | -50 | 40.3 | 0 | 20.1[(e)] | 40.3 | 31 |
| | | 2.1 | 6.7 | $2.72 \times 10^{19}$ | 1.2671 | -50 | 8.5 | 0 | 4.2[(e)] | 8.5 | -33 |
| Extended Beltline | Nozzle Shell Plate B9804-1 | 1.1 | 31.0 | $0.0814 \times 10^{19}$ | 0.3770 | 40 | 11.7 | 0 | 5.8[(e)] | 11.7 | 63 |
| | Nozzle Shell Plate B9804-2 | 1.1 | 51.0 | $0.0814 \times 10^{19}$ | 0.3770 | 20 | 19.2 | 0 | 9.6[(e)] | 19.2 | 58 |
| | Nozzle Shell Plate B9804-3 | 1.1 | 31.0 | $0.0814 \times 10^{19}$ | 0.3770 | 0 | 11.7 | 0 | 5.8[(e)] | 11.7 | 23 |
| | Inlet Nozzle B9806-3 | 1.1 | 58.0 | $0.0814 \times 10^{19}$ | 0.3770 | 10 | 21.9 | 0 | 10.9[(e)] | 21.9 | 54 |
| | Inlet Nozzle B9806-4 | 1.1 | 58.0 | $0.0814 \times 10^{19}$ | 0.3770 | 0 | 21.9 | 0 | 10.9[(e)] | 21.9 | 44 |
| | Inlet Nozzle R5-3 | 1.1 | 44.0 | $0.0814 \times 10^{19}$ | 0.3770 | -10 | 16.6 | 0 | 8.3[(e)] | 16.6 | 23 |
| | Inlet Nozzle R5-4 | 1.1 | 51.0 | $0.0814 \times 10^{19}$ | 0.3770 | 0 | 19.2 | 0 | 9.6[(e)] | 19.2 | 38 |
| | Nozzle Shell Longitudinal Weld 101-122A | 1.1 | 39.8 | $0.0814 \times 10^{19}$ | 0.3770 | -10 | 15.0 | 0 | 7.5[(e)] | 15.0 | 20 |
| | Nozzle Shell Longitudinal Weld 101-122B, 101-122C | 1.1 | 39.8 | $0.0814 \times 10^{19}$ | 0.3770 | -50 | 15.0 | 0 | 7.5[(e)] | 15.0 | -20 |

| Reactor Vessel Location | Reactor Vessel Material | R.G. 1.99, Rev. 2 Position | CF(b) (°F) | Fluence(c) ($n/cm^2$, E > 1.0 MeV) | FF(c) | Initial $RT_{NDT}$ (b) (°F) | $\Delta RT_{NDT}$ (°F) | $\sigma_U$ (°F) | $\sigma_\Delta$ (°F) | Margin (°F) | $RT_{PTS}$ (°F) |
|---|---|---|---|---|---|---|---|---|---|---|---|
| | Nozzle Shell to Intermediate Shell Girth Weld 103-121 | 1.1 | 41.0 | $0.0814 \times 10^{19}$ | 0.3770 | -40 | 15.5 | 0 | 7.7(e) | 15.5 | -9 |
| | Inlet Nozzle Weld 105-121A | 1.1 | 45.3 | $0.0814 \times 10^{19}$ | 0.3770 | -60 | 17.1 | 0 | 8.5(e) | 17.1 | -26 |
| | Inlet Nozzle Weld 105-121B | 1.1 | 75.4 | $0.0814 \times 10^{19}$ | 0.3770 | -50 | 28.4 | 0 | 14.2(e) | 28.4 | 7 |
| | Inlet Nozzle Weld 105-121C | 1.1 | 75.4 | $0.0814 \times 10^{19}$ | 0.3770 | -50 | 28.4 | 0 | 14.2(e) | 28.4 | 7 |
| | Inlet Nozzle Weld 105-121D | 1.1 | 75.4 | $0.0814 \times 10^{19}$ | 0.3770 | -50 | 28.4 | 0 | 14.2(e) | 28.4 | 7 |

Notes:

(a) The 10 CFR 50.61 [Reference 1-3] methodology was utilized in the calculation of the $RT_{PTS}$ values.

(b) Values are consistent with those utilized in the analysis of record, the SPU evaluation [Reference 1-1].

(c) Maximum fluence values for the beltline ($2.72 \times 10^{19}$ $n/cm^2$) and extended beltline ($8.14 \times 10^{17}$ $n/cm^2$) considering MUR conditions are conservatively used for every material in the region. FF = fluence factor as calculated with the 10 CFR 50.61 [Reference 1-3] methodology.

(d) A reduced $\sigma_\Delta$ term is used since the surveillance data is deemed credible per WCAP-16629-NP [Reference 1-5].

(e) Per 10 CFR 50.61 [Reference 1-3], $\sigma_\Delta$ need not exceed 0.5 * $\Delta RT_{NDT}$. Therefore, the $\sigma_\Delta$ has been reduced.

(f) The limiting $RT_{PTS}$ value for MPS3 is 130°F, which corresponds to Lower Shell Plate B9820-2. Note that Intermediate Shell Plate B9805-1 resulted in a higher $RT_{PTS}$ value of 133°F when the surveillance data was not used; however, the $RT_{PTS}$ value for Intermediate Shell Plate B9805-1 is 111°F when the credible surveillance data is used. Thus, taking credit for the credible Intermediate Shell Plate B9805-1 surveillance data, the limiting material for MPS3 is Lower Shell Plate B9820-2.

***Regulatory Basis: Nuclear Systems Performance (SNSB) RAIs***

*Appendix A to 10 CFR 50 establishes minimum criteria (General Design Criteria or GDC) for the safe operation of light water reactors. GDC 10, "Reactor Design", requires that the reactor core and associated coolant, control, and protection systems be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences. Fuel design limits are challenged by transients described in the MPS3 Updated Final Safety Analysis Report (UFSAR) sections 15.1.3, 15.3.1, 15.3.2, 15.4.3, and 15.6.1.*

*GDC 15, "Reactor coolant system design", requires that reactor coolant system and associated auxiliary, control, and protection systems be 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. UFSAR sections 15.2.6, 15.2.7, and 15.3.2 describes analyses of anticipated operational occurrences which could challenge the reactor coolant pressure boundary.*

*GDC 28, "Reactivity limits", requires reactivity control systems to be designed with appropriate limits on potential reactivity increases so the effects of a postulated rod ejection accident can result in neither damage to the reactor coolant pressure boundary nor result in sufficient disturbance to impair the core cooling capability. The transient described in UFSAR section 15.4.8 helps to demonstrate that this criterion is met.*

*GDC 31, "Fracture prevention of reactor coolant pressure boundary", requires that the reactor pressure boundary be designed with sufficient margin to ensure that the probability of rapidly propagating failure is minimized under postulated accident conditions. UFSAR sections and 15.2.8 and 15.3.3 describes analyses of postulated accidents which could challenge the reactor coolant pressure boundary.*

*Regulatory Information Summary (RIS) 2002-03 (ADAMS Accession No. ML013530183) provides guidance to addressees on the scope and detail of information that should be provided to NRC for reviewing MUR power uprate applications. The guidance states that in areas for which existing AOR bound plant operation at the proposed power level, the staff will not conduct a detailed review.*

*Natural circulation cooldown is a portion of the transients described in UFSAR sections 15.2.6, 15.2.7, 15.2.8, and 15.3.2. The current licensing analysis for this event documents compliance with the guidance in Branch Technical Position (BTP) 5-1.*

**SNSB-RAI-7**

*The licensee described the power level assumed when analyzing accidents and transients for MPS3 in Table II-1 of Attachment 4 to the LAR. However, the NRC staff notes that the information in the LAR, Table II-1 is not consistent with the information in Table 15.0-2 of the UFSAR, Revision 33. Specifically, there is a discrepancy for the thermal power level listed for accidents in UFSAR sections 15.1.3, 15.3.1, 15.3.2, 15.4.3, 15.4.8, 15.6.1, and the DNB analysis in UFSAR section 15.3.3. To ensure that various analyses can be accepted without detailed review, please explain the discrepancy between the power listed in Table 15.0-2 of the UFSAR and Table II-1 of Attachment 4 to the LAR for accidents described in UFSAR sections 15.1.3, 15.3.1, 15.3.2, 15.4.3, 15.4.8, 15.6.1, and the DNB analysis in UFSAR section 15.3.3. For any of these cases, if the discrepancy exists because a portion of the analysis was performed assuming a power level that does not bound the uprated power level, please provide a justification or summarize the updated analysis.*

**DENC Response to SNSB-RAI-7**

SNSB-RAI-7 identifies discrepancies between the power level information listed in the MPS3 MUR LAR Attachment 4, Table II-1 and MPS3 FSAR Table 15.0-2. DENC's review of the tables identified two primary differences between the reported powers. First, FSAR Table 15.0-2 inconsistently presents initial power information. Second, FSAR Table 15.0-2 includes power levels reflecting the transient analyses that provide input to the MUR DNB design basis. These transient analyses are based on a 3666 MWt NSSS power.

Regarding Item 1, three general inconsistencies were identified in the FSAR Table 15.0-2 reported powers. The inconsistencies include: (a) lack of use of an explicit identifier for core power, (b) lack of inclusion of the power uncertainty, and (c) lack of inclusion of the conservative MUR power level (3712 MWt) for events with a DNB acceptance criterion. The issue was entered into DENC's corrective action system and is limited to the FSAR table. The MPS3 safety analyses described in FSAR Chapter 15 use the appropriate power levels and are not impacted. MUR LAR Table II-1 has been reviewed and is affirmed to document appropriate and consistent inputs for power level.

With respect to Item 2, select differences between FSAR Table 15.0-2 and LAR Table II-1 reflect the transient analyses that provide statepoint operating conditions into the MUR DNB design basis. Select DNB calculations were performed at a conservative, scaled power level of 101.7% of 3650 MWt (3712 MWt) along with statepoint operating conditions generated by a system transient analysis performed at a NSSS power level of 3666 MWt. The statepoint conditions generated at 3666 MWt NSSS power have been validated as applicable for the MUR power uprate because the minor power level difference has a negligible effect on the local core conditions (temperature, pressure, and flow) relevant for the calculation of DNB. The combination of the system transient operating conditions (based on a NSSS power of 3666 MWt) and the scaled, conservative power level of 3712 MWt results in a conservative calculation of DNB for comparison to the DNB acceptance criteria. The approach taken is presently summarized in FSAR

Section 15.0.3.1. As described in the MUR LAR and FSAR Section 15.0.3.1, all DNB events show acceptable DNB performance. This power scaling is not applicable to the FSAR Section 15.1.3 and 15.4.3 DNB events identified in SNSB-RAI-7.

Table 1-2 outlines the corrections required to consistently report initial power in FSAR Table 15.0-2. Further, Table 1-2 identifies those analyses where the DNB calculation was performed at a conservative, scaled core power of 3712 MWt along with statepoint operating conditions generated at 3666 MWt NSSS power. Each FSAR section identified by the NRC as containing a power discrepancy is included. The FSAR Section 15.1.2 is also included since its associated FSAR Table 15.0-2 entry is impacted by the power reporting inconsistencies, and the transient analysis supporting the DNB calculation was performed at 3666 NSSS power.

For all listed events, the references for NRC approval and/or NRC-approved methods presented in LAR Table II-1 remain accurate and valid.

**Table 1-2: MPS3 FSAR Table 15.0-2 Corrections and Clarifications**

| FSAR Section and Event | Current FSAR Table 15.0-2 Initial Power [MWt] | Corrected FSAR Table 15.0-2 Initial Power [MWt] Note 1 | Basis for Correction or Discrepancy |
|---|---|---|---|
| 15.1.2 - Feedwater System Malfunctions that Result in an Increase in Feedwater Flow | 0 (NSSS) and 3666 (NSSS) | 0 **NSSS** and 3666 **NSSS** **0 Core and 3712 Core (DNB)** | FSAR Table 15.0-2 does not currently identify the conservative, scaled core power assumed in DNB calculations. |
| 15.1.3 - Excessive Increase in Secondary Steam Flow | 3666 (NSSS) | **3712 Core (DNB)** | FSAR Table 15.0-2 does not currently list the conservative core power assumed in DNB calculations. No power scaling was applied as described in MPS3 MUR LAR Attachment 4, Section II.1.3. |
| 15.3.1 - Partial Loss of Forced Reactor Coolant Flow<br><br>15.3.2 - Complete Loss of Forced Reactor Coolant Flow | 3666 (NSSS) 3712 *(not identified as a core power in Table 15.0-2)* | 3666 **NSSS** 3712 **Core (DNB)** | FSAR Table 15.0-2 does not currently include the "core" identifier for power level. A conservative, scaled core power is assumed in DNB calculations. |
| 15.3.3 - Locked Rotor - Rods in DNB | 3666 (NSSS) 3712 *(not identified as a core power in Table 15.0-2)* | 3666 **NSSS** 3712 **Core (DNB)** | FSAR Table 15.0-2 does not currently include the "core" identifier for power level. A conservative, scaled core power is assumed in DNB calculations. |
| 15.4.3 - RCCA Misalignment | 3666 (NSSS) 3712 (Core) | 3712 Core (DNB) | Although FSAR Table 15.0-2 presently lists the conservative core power assumed in DNB calculations, no power scaling was applied as described in MPS3 MUR LAR Attachment 4, Section II.1.17. |
| 15.4.8 - Rod Ejection | 0 (NSSS) and 3650 *(not identified as a core power in Table 15.0-2)* | **0 Core and 3723 Core** | FSAR Table 15.0-2 does not currently include the "core" identifier for power level nor identify that calorimetric uncertainty was applied to the event analysis. |
| 15.6.1 - Inadvertent Opening of a Pressurizer Safety or Relief Valve | 3666 (NSSS) | 3666 **NSSS** **3712 Core (DNB)** | FSAR Table 15.0-2 does not currently include the conservative, scaled core power assumed in DNB calculations. |

Note 1: Power uncertainty is included in the initial power as appropriate for each event.

**SNSB-RAI-8**

*The licensee provided an evaluation for the natural circulation cooling event in support of the MUR power uprate in Section III.3-1 of Attachment 4 to the LAR. In their evaluation, the licensee indicated that the current AOR was performed at 3650 MWt, which does not bound the power level proposed in the MUR power uprate. The licensee has repeated the natural circulation cooling analysis at conditions bounding the MUR power uprate, and confirmed that RCS cooldown and adequate boron mixing can be achieved. However, the licensee did not describe RCS pressure control or depressurization in their evaluation. The initial licensing evaluation, as well as an evaluation performed in support of a stretch power uprate, confirmed that pressure control can be achieved during a natural circulation cooling event. Please confirm that RCS pressure control can be maintained and depressurization can be achieved in a natural circulation cooling event at conditions bounding the MUR uprate.*

**DENC Response to SNSB-RAI-8**

The natural circulation cooldown AOR described in the MPS3 SPU LAR (performed at 3650 MWt) concluded that RCS pressure control could be maintained by the use of pressurizer auxiliary spray and pressurizer power operated relief valves (PORVs), and that the RCS pressure could be reduced low enough to allow Residual Heat Removal (RHR) system initiation in approximately eleven hours (which is well before the required 24 hours). The evaluation of natural circulation cooldown was reevaluated for a rated thermal power of 3723 MWt, which bounds the MUR Power Uprate conditions. This reevaluation determined that the conclusions from the previously performed analyses for 3650 MWt remained valid at MUR conditions, including meeting the required timeframe to initiate RHR. System and component evaluations at the MUR conditions have confirmed that the pressurizer PORVs continue to have the capability to control and reduce RCS pressure when required. Additional evaluations have shown that the pressurizer auxiliary spray can also control and reduce RCS pressure when needed at MUR conditions. Therefore, RCS pressure control can be maintained during a natural circulation cooldown at conditions bounding the MUR uprate and depressurization to a pressure low enough to initiate RHR can be achieved.

***Regulatory Basis: Electrical Engineering (EEEB) RAIs***

*GDC 17, "Electric power systems," of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR 50, states in part, "An onsite electric power system and an offsite electric power system shall be provided to permit functioning of structures, systems, and components important to safety. The safety function for each system (assuming the other system is not functioning) shall be to provide sufficient capacity and capability to assure that (1) specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded as a result of anticipated operational occurrences and (2) the core is cooled and containment integrity and other vital functions are maintained in the event of postulated accidents."*

*RIS 2002-03, "Guidance on the Content of Measurement Uncertainty Recapture Power Uprate Applications," Section V, "Electrical Equipment Design," states in part, "A discussion of the effect of the power uprate on electrical equipment... For equipment that is not bounded by existing analyses of record, a detailed discussion should be included to identify and evaluate the changes related to the power uprate. Specifically, this discussion should address the following items: ...D. grid stability."*

*Office of Nuclear Reactor Regulation office instruction LIC-105, "Managing Regulatory Commitments Made by Licensees to the NRC", Revision 7 (ADAMS Accession No. ML16190A013), Section 4.1, "Creation of Regulatory Commitments," states in part, "Regulatory commitments...do not warrant either legally binding requirements or inclusion in updated final safety analysis reports (UFSARs) or programs subject to a formal regulatory change control mechanism."*

## EEEB-RAI-9

*The NRC staff evaluated the LAR for consistency with RIS 2002-03, GDC 17, and the MPS3 UFSAR. Specifically, as referenced in Attachment 4 of the LAR, Section V, "NRC Regulatory Issue Summary 2002-03 Topic: Electrical Equipment Design," RIS 2002-03 specifies that a discussion of the effect of the power uprate on electrical equipment be included in the LAR, specifically in four areas, one of which is grid stability. A typical grid stability study for a nuclear power plant assesses (1) the impacts of the loss, through a single event, of the largest capacity being supplied to the grid, (2) the removal of the largest load from the grid, (3) the most critical transmission line if unavailable that results in the loss of offsite power, (4) any increased main generator output adverse effects, and (5) confirms adequate reactive power support at the lowest post-contingency 345kV switchyard voltage. The NRC staff has determined that the LAR for MUR implementation does not provide sufficient information about grid stability and the 345 kV switchyard to complete its review.*

*Section 3.0, "Technical Analysis," in Attachment 1 of the LAR states, "In support of meeting the ISO-New England requirements, main generator upgrades will be required to transmit additional megawatts electric (MWe) to the grid at uprate conditions."*

*Section V.1.F.i, "Main Generator," in Attachment 4 of the LAR states, "The main generator requires upgrades in order to accommodate the MUR Power Uprate at the required ISO-New England power factor."*

*Please provide the MPS3 combined main generator (MG) output (in megawatts (MW) and/or mega voltamperes (MVA) segregating the contributions due exclusively to the MG upgrade and MUR implementation (the NRC only regulates the MUR portion).*

**DENC Response to EEEB-RAI-9**

The proposed MUR power uprate would increase reactor thermal power from 3650 MWt to 3709 MWt. This increase in reactor thermal power would yield an increase in main generator output of approximately 21 MWe. The existing generator is nearing its end of life expectancy, and the generator supplier (GE) has issued many Technical Information Letters that would require significant repairs. DENC has chosen to upgrade the generator rather than only conduct these repairs. While the current generator has a nameplate rating of 1354.7 MVA, the upgraded generator will have a nameplate rating of 1500 MVA. The 1500 MVA rating would ensure that MPS3 will meet the requirements for reactive loading, with a power factor of 0.95 at the point of interconnection in accordance with ISO New England "Schedule 22, Large Generator Interconnection Procedures [Reference 1-10]." However, MPS3 will not be able to reach this full generator capacity, because the unit would be limited by the licensed reactor thermal power that is proposed in the MUR LAR. The MPS3 main generator will be operating well below this 1500 MVA value when the unit is at the proposed MUR power level.

In summary, there would be no quantifiable generator output increase solely due to generator upgrades, so the expected increase of approximately 21 MWe would be due to the increase in reactor thermal power from the proposed MUR uprate.

**EEEB-RAI-10**

*Section V.1.D, "Grid Stability," in Attachment 4 of the LAR states that an Interconnection System Impact Study will be performed in accordance with the processes of ISO-NE Schedule 22, Large Generator Interconnection Procedures and will evaluate the impact of the proposed interconnection on the reliability and operation of the New England Transmission System. The licensee also identified the study would consist of a short circuit analysis, a stability analysis, a power flow analysis (including thermal analysis and voltage analysis), a system protection analysis and any other analyses that are deemed necessary by the System Operator (ISO-NE) in consultation with the Interconnecting Transmission Owner (Eversource).*

*Section V.1.G, "Switchyard Interface," in Attachment 4 of the LAR states:*

*The 345 kV switchyard is discussed in FSAR Section 8.1.3. An Interconnection System Impact Study (refer to Section V.1.D) will be performed in accordance with the processes of Reference V.1 (see Regulatory Commitment in Attachment 5). The Interconnection System Impact Study will contain any other analyses that are deemed necessary by the System Operator (ISO-NE) in consultation with the Interconnecting Transmission Owner (Eversource), which includes evaluation of the 345 kV switchyard and distribution system. This analysis of the switchyard will be prepared by Eversource and will ensures the functionality of the switchyard and its associated components that would be affected by the MUR power uprate.*

*In Attachment 5 of the LAR, the following regulatory commitment was submitted by the licensee:*

*(Commitment) DENC will complete an Interconnection System Impact Study, including a grid stability analysis, as described in Attachment 4, Section V.I.D of the MPS3 MUR Power Uprate LAR submittal. (Scheduled Completion Date) The Interconnection System Impact Stability Study will be completed prior to implementation of the MUR Power Uprate for MPS3."*

*Please provide additional details about what explicit actions DENC will take including notifying the NRC if the grid stability study results do not meet DENC's or the preparer's designated standard considering effects on (1) MPS3 grid resiliency, (2) 345 kV switchyard functionality, and (3) GDC 17 requirements for onsite and offsite power systems.*

**DENC Response to EEEB-RAI-10**

The System Operator (ISO-NE) will be performing a system impact study per References 1-7 and 1-8. This study includes:

1. Steady State Analysis: This analysis will be performed to demonstrate compliance with voltage and thermal loading criteria and will identify any system upgrades to satisfy these criteria. The analysis will verify that the power increase will not have any significant adverse impact upon the reliability or operating characteristics of the bulk power system (grid). For steady-state analysis, the maximum output for the MPS3 generator will be used during summer operation.
2. Stability Analysis: This analysis demonstrates that the proposed power increase will not have any significant adverse impact upon the stability, reliability, or operating characteristics of the bulk power systems under the most severe conditions. Studies demonstrating dynamic performance must then simulate conditions that stress the system beyond typical combinations of load level, generation dispatch, and power transfers. For stability analysis, the maximum output for the MPS3 will be used during winter operation.
3. Short Circuit: This analysis will be conducted to demonstrate that the short circuit duties will not exceed equipment capability and will identify any system upgrades required to satisfy the criterion. The base case will include all generating facilities and elective transmission upgrades on the date the study is commenced that: (i) are directly connected to the New England Transmission System, (ii) are interconnected to the affected systems and may have an impact on the interconnection request; and (iii) have a pending higher queued position and may impact on the MPS3 interconnection request.
4. Voltage and Reactive Power Performance of the Bulk Power System: The system impact study will evaluate the compliance of the voltage control capability with the requirements of Reference 1-9.

Upon completion of the study, DENC will confirm the effects of grid resiliency, switchyard functionality, and GDC-17 compliance (which is summarized in MPS3 FSAR Section 3.1.2.17). If the study produces satisfactory results under the MUR uprated conditions, then the proposed changes from the MUR LAR will be implemented. If the study produces unsatisfactory results under the MUR uprated conditions, the MUR LAR will not be implemented, and DENC will inform the NRC of this outcome and describe the expected actions for resolution.

**References:**

1-1. Dominion Energy Letter Serial No. 07-0450, "Dominion Nuclear Connecticut, Inc. Millstone Power Station Unit 3 License Amendment Request Stretch Power Uprate," dated July 13, 2007. Attachment 5: Millstone Power Station Unit 3 Stretch Power Uprate Licensing Report [ADAMS Accession No. ML072000400].
1-2. Millstone Power Station Unit 3 Safety Analysis Report, Revision 33.02, dated March 31, 2021.
1-3. Code of Federal Regulations, 10 CFR 50.61, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events," dated January 4, 2010.
1-4. Dominion Energy Letter Serial No. 20-043, "Dominion Energy Nuclear Connecticut, Inc. Millstone Power Station Unit 3 Proposed License Amendment Request Measurement Uncertainty Recapture Power Uprate," dated November 19, 2020 [ADAMS Accession Number ML20324A703].
1-5. Westinghouse Report WCAP-16629-NP, Revision 0, "Analysis of Capsule W from the Dominion Nuclear Connecticut Millstone Unit 3 Reactor Vessel Radiation Surveillance Program," September 2006.
1-6. U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research, Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," May 1988 [ADAMS Accession Number ML003740284].
1-7. ISO New England Planning Procedure 5-3, Guidelines for Conducting and Evaluating Proposed Plan Application Analysis.
1-8. ISO New England Planning Procedure 5-6, Interconnection Planning Procedure for Generation and Elective Transmission Upgrades.
1-9. ISO New England Operating Procedure No. 14 - Technical Requirements for Generators, Demand Response Resources, Asset Related Demands and Alternative Technology Regulation Resources.
1-10. ISO-New England Schedule 22 – Large Generator Interconnection Procedures.

# ATTACHMENT 2

## Supplement to Clarify Methodology for Revised Loss of Normal Feedwater Reanalysis

**MILLSTONE POWER STATION UNIT 3**
**DOMINION ENERGY NUCLEAR CONNECTICUT, INC.**

Attachment 4, Section III.4-1.1 of the Millstone Power Station Unit 3 (MPS3) Measurement Uncertainty Recapture (MUR) License Amendment Request (LAR) submittal, dated November 19, 2020 (ADAMS Accession No. ML20324A702), discussed the planned reanalysis of select Final Safety Analysis Report (FSAR) Chapter 15 events to support implementation of NRC-approved WCAP-17642-P-A, "Westinghouse Performance Analysis and Design Model (PAD5)," [Reference 2-1]. The discussion of the Loss of Normal Feedwater (LONF) event in Section II.1.10 also described how this event will be reanalyzed to implement WCAP-17642-P-A. The purpose of this supplement is to clarify that the methodology selected for the Dominion Energy Nuclear Connecticut, Inc. (DENC) planned LONF reanalysis has changed from what is presented in the MUR LAR Attachment 4, Sections II.1.10 and III.4-1.1. Instead, the LONF event revision will be performed using an alternate, NRC-approved, methodology.

The current FSAR Chapter 15.2.7 LONF event was performed using the Westinghouse RETRAN transient analysis method (WCAP-14882-P-A) [Reference 2-2]. The LONF revision will be performed using the same RETRAN method as the current LONF analysis. The revised analysis will be confirmed each reload using the Dominion Energy Reload Methodology (VEP-FRD-42-A) [Reference 2-3]. Prior to PAD5 implementation, DENC will perform a review of the LONF revision and submit any items deemed necessary for NRC review per the criteria of 10 CFR 50.59. The LONF revision will reflect the MUR power level.

**References:**

2-1. WCAP-17642-P-A, Revision 1, "Westinghouse Performance Analysis and Design Model (PAD5)," November 2017.
2-2. WCAP-14882-P-A and WCAP-15234-A, "RETRAN-02 Modeling and Qualification for Westinghouse Pressurized Water Reactor NON-LOCA Safety Analyses," April 1999.
2-3. Topical Report VEP-FRD-42-A, Revision 2, Minor Revision 2, "Reload Nuclear Design Methodology," October 2017.