

Dominion Nuclear Connecticut, Inc.
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PROPRIETARY INFORMATION – WITHHOLD UNDER 10 CFR 2.390

September 1, 2015

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

Serial No. 15-411
NLOS/WDC R0
Docket No. 50-336
License No. DPR-65

DOMINION NUCLEAR CONNECTICUT, INC.
MILLSTONE POWER STATION UNIT 2
PROPOSED LICENSE AMENDMENT REQUEST
SMALL BREAK LOSS OF COOLANT ACCIDENT REANALYSIS

Pursuant to 10 CFR 50.90, Dominion Nuclear Connecticut, Inc. (DNC) requests an amendment, in the form of changes to the Technical Specifications (TS) to Facility Operating License Number DPR-65 for Millstone Power Station Unit 2 (MPS2). The proposed change will add Supplement 1 to the Framatome-ANP (AREVA) topical reports for Pressurized Water Reactor (PWR) Small Break Loss of Coolant Accident (SBLOCA) Evaluation Model and Generic Mechanical Design Criteria for PWR Fuel Designs to TS 6.9.1.8.b, "Core Operating Limits Report," which lists the analytical methods used to determine the core operating limits. Attachment 1 provides the description and assessment of the proposed change. Attachment 2 provides the marked-up TS pages to reflect the proposed TS changes.

In response to a request for additional information dated November 1, 2012, and acknowledged as acceptable by the NRC in a letter dated July 18, 2013, DNC stated the intent to submit a reanalysis of the MPS2 Final Safety Analysis Report (FSAR) Chapter 14 SBLOCA within a year of NRC approval of Supplement 1 to the AREVA topical reports. In a letter dated September 1, 2015, the NRC issued a safety evaluation for Supplement 1 of the AREVA PWR SBLOCA evaluation model.

Attachment 3 provides a report containing a reanalysis of the MPS2 FSAR Chapter 14 SBLOCA using the Supplement 1 methods. This report contains proprietary AREVA information. The non-proprietary version of this report is provided in Attachment 4. It is respectfully requested that Attachment 3 be withheld from public disclosure in accordance with 10 CFR 2.390. The AREVA application for withholding and affidavit is provided in Attachment 5.

As concluded in Attachment 1, the proposed amendment does not involve a significant hazards consideration pursuant to the provisions of 10 CFR 50.92. The proposed change has been reviewed and approved by the Facility Safety Review Committee.

Attachment 3 contains proprietary information that is being withheld from public disclosure under 10 CFR 2.390. Upon separation of Attachment 3, this letter is decontrolled.

ADD
NRR

DNC requests approval of the proposed amendment by September 1, 2016 to support the core design for the MPS2 Cycle 25 core reload in the spring of 2017. DNC will implement the revised TS within 60 days of NRC approval of the proposed amendment.

If you have any questions regarding this request, please contact Wanda Craft at (804) 273-4687.

Sincerely,



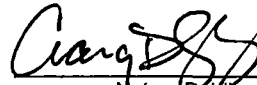
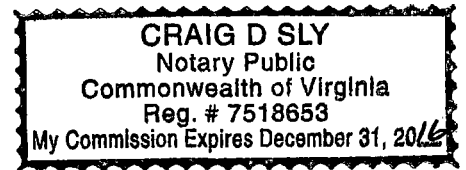
Mark D. Sartain
Vice President – Nuclear Engineering

COMMONWEALTH OF VIRGINIA)
)
COUNTY OF HENRICO)

The foregoing document was acknowledged before me, in and for the County and Commonwealth aforesaid, today by Mark D. Sartain, who is Vice President – Nuclear Engineering of Dominion 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 1ST day of September, 2015.

My Commission Expires: 12/31/16


Notary Public

Attachments:

1. Discussion of Technical Specification Change
2. Marked-up Technical Specification Pages
3. ANP-3315P, Revision 0, Millstone Unit 2 M5[®] Upgrade, Small Break LOCA Analysis Licensing Report (Proprietary)
4. ANP-3315NP, Revision 0, Millstone Unit 2 M5[®] Upgrade, Small Break LOCA Analysis Licensing Report (Non-Proprietary)
5. AREVA Application for Withholding and Affidavit

Commitments made in this letter: None

cc: U.S. Nuclear Regulatory Commission
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Attachment 1

Discussion of Technical Specification Change

**DOMINION NUCLEAR CONNECTICUT, INC.
MILLSTONE POWER STATION UNIT 2**

1.0 Summary Description

Pursuant to 10 CFR 50.90, Dominion Nuclear Connecticut, Inc. (DNC) requests an amendment, in the form of changes to the Technical Specifications (TS), to Facility Operating License Number DPR-65 for Millstone Power Station Unit 2 (MPS2). The proposed changes will add Supplement 1 to the Framatome-ANP (AREVA) topical reports for Pressurized Water Reactor (PWR) Small Break Loss of Coolant Accident (SBLOCA) Evaluation Model, S-RELAP5 Based (EMF-2328(P)(A) – Reference 2) and Generic Mechanical Design Criteria for PWR Fuel Designs (EMF-92-116(P)(A) – Reference 5) to TS 6.9.1.8.b, “Core Operating Limits Report,” which lists the analytical methods used to determine core operating limits. The mark-up to TS 6.9.1.8.b reflecting the proposed changes are provided in Attachment 2.

Supplement 1 to EMF-2328(P)(A), “PWR Small Break LOCA Evaluation Model, S-RELAP5 Based,” is used in the MPS2 SBLOCA reanalysis related to the use of M5 fuel rod cladding, which DNC plans to introduce beginning with the fresh fuel for MPS2 Cycle 25 in the spring of 2017. Supplement 1 to EMF-92-116(P)(A), “Generic Mechanical Design Criteria for PWR Fuel Designs,” describes correction factors to be applied to the RODEX2 predictions to account for fuel pellet thermal conductivity degradation with increasing fuel burnup. Fuel conditions from RODEX2 are used in the SBLOCA S-RELAP5 analysis. Attachment 3 provides a report documenting the reanalysis of the MPS2 Final Safety Analysis Report (FSAR) Section 14.6.5.2 SBLOCA utilizing these updated methods.

2.0 Detailed Description of Proposed Technical Specification Change

TS 6.9.1.8.b, “Core Operating Limits Report,” currently includes 17 documents that define the methods used to determine the core operating limits for MPS2. The proposed revision to this list of documents would add Supplement 1 to Document 5 and Document 16 of TS 6.9.1.8.b. The proposed changes are as follows (deleted text is struck through and added text is italicized and bolded):

Document 5

- 5) EMF-2328(P)(A) **and Supplement 1**, “PWR Small Break LOCA Evaluation Model S-RELAP5 Based,” ~~Framatome-ANP~~

Document 16

- 16) EMF-92-116(P)(A) **and Supplement 1**, ~~Revision 0~~, “Generic Mechanical Design Criteria for PWR Fuel Designs,” ~~Siemens Power Corporation~~.

The specific company name is proposed to be deleted from both documents for clarity as the company name has changed between the issuance of the original documents and Supplement 1 for both documents. The mark-up to TS 6.9.1.8.b reflecting the proposed changes to Documents 5 and 16 are shown in Attachment 2.

3.0 Discussion

In response to a request for additional information dated November 1, 2012 (Reference 1), DNC planned to submit a reanalysis of the MPS2 FSAR Chapter 14 SBLOCA within a year of NRC approval of Supplement 1 to the AREVA topical reports. In a letter dated July 18, 2013 (Reference 3), the NRC acknowledged that DNC stated the intent to submit a reanalysis of the SBLOCA event and concluded the proposed schedule for reanalysis was acceptable and the reanalysis requirement of 10 CFR 50.46 was satisfied. In a letter dated September 1, 2015 (Reference 4), the NRC issued a safety evaluation for Supplement 1 of the AREVA PWR SBLOCA evaluation model.

The MPS2 SBLOCA reanalysis described in Attachment 3 was performed for the AREVA Standard CE14 HTP fuel assembly, which includes M5 fuel rod cladding. DNC plans to introduce this fuel product containing M5 fuel rod cladding beginning with the fresh fuel for MPS2 Cycle 25 in the spring of 2017. An exemption from 10 CFR 50.46 and Appendix K to allow the use of the M5 alloy for fuel rod cladding was approved by the NRC for MPS2 in a letter dated May 12, 2015 (Reference 6). The MPS2 SBLOCA reanalysis uses Supplement 1 to EMF-2328(P)(A), Revision 0, PWR SBLOCA evaluation model.

In addition to the MPS2 SBLOCA reanalysis using the M5 fuel rod cladding, an analysis of the current MPS2 AREVA fuel product with Zirc-4 cladding was performed. A number of break sizes surrounding the limiting break size were reanalyzed with only a change to the current MPS2 fuel product. The limiting 3.78 inch break size remained unchanged from that demonstrated in the MPS2 SBLOCA reanalysis using the M5 fuel product. The peak cladding temperature (PCT) for this limiting break size with the current MPS2 fuel product is 1711°F, or +4°F higher than the 1707°F PCT determined in the reanalysis for the M5 fuel rod cladding product. The analyses demonstrate a negligible difference in PCT between the current MPS2 fuel product and the future fuel product with M5 cladding. Concurrent with implementation of the MPS2 SBLOCA reanalysis, the +4°F PCT for the Zirc-4 fuel will be treated as a penalty (an estimated effect for a change to the evaluation model) under the provisions of 10 CFR 50.46. This penalty will be added to the analysis result provided in MPS2 SBLOCA reanalysis using the M5 fuel product until the current fuel product with Zirc-4 cladding is no longer limiting.

4.0 Technical Evaluation

Attachment 3 provides a description of the reanalysis of the MPS2 SBLOCA including a detailed description of the analysis and the analytical results. As reported in Section 2.0 of Attachment 3, the SBLOCA reanalysis meets the Emergency Core Cooling System (ECCS) performance acceptance criteria in 10 CFR 50.46. Included in the description of the analysis is a discussion of compliance with the limitations contained in the issued NRC safety evaluation associated with the EMF-2328(P)(A) Revision 0 methodology. In the safety evaluation for Supplement 1 of EMF-2328(P)(A) issued September 1, 2015 (Reference 4), the NRC: (1) specifies a break diameter increment resolution, (2) requires that a submittal identifies the critical break size at and below which only one loop seal clears of liquid, and (3) requires that the largest break size that depressurizes

just above the safety injection tank actuation pressure must be included in the break spectrum evaluation. The MPS2 SBLOCA reanalysis complies with the safety evaluation limitations.

This proposed amendment adds Supplement 1 of the AREVA topical report on Generic Mechanical Design Criteria for PWR Fuel Designs, EMF-92-116, Revision 0, to TS 6.9.1.8.b. Supplement 1 describes correction factors to be applied to the RODEX2 predictions to account for fuel pellet thermal conductivity degradation with increasing fuel burnup. Fuel conditions from RODEX2 are used in the SBLOCA S-RELAP5 analysis. Therefore, this change is connected to the addition of Supplement 1 to EMF-2328(P)(A), Revision 0. The NRC issued a safety evaluation for Supplement 1 to EMF-92-116(P)(A), Revision 0 (Reference 5). This safety evaluation does not specify any limitations or restrictions.

In addition to providing fuel conditions for the input to the SBLOCA analysis, the RODEX2 code is used in evaluating the following mechanical fuel design criteria:

- Cladding Collapse
- Cladding Strain
- Fuel Rod Fatigue
- Fuel Densification and Swelling
- Cladding Oxidation, Hydriding, and Crud Buildup
- Fuel Rod Internal Pressure
- Fuel Centerline Melt

Supplement 1 to EMF-92-116(P)(A), Revision 0, also describes the impact that thermal conductivity degradation has on each of these criteria. These criteria are assessed on a cycle-specific basis under 10 CFR 50.59 as part of the normal core reload process.

5.0 Regulatory Evaluation

5.1 Applicable Regulatory Requirements and Criteria

The SBLOCA reanalysis included in Attachment 3 satisfies the requirements of 10 CFR 50.46 and 10 CFR 50 Appendix K. An exemption from 10 CFR 50.46 and Appendix K to allow the use of the M5 alloy for fuel cladding was approved for MPS2 by the NRC in a letter dated May 12, 2015 (Reference 6). The proposed change does not require relief from any other regulatory requirements and does not affect conformance with a General Design Criterion differently than described in the MPS2 FSAR.

5.2 No Significant Hazards Consideration

The proposed changes to TS 6.9.1.8.b add Supplement 1 to AREVA methodology reports EMF-2328(P)(A) and EMF-92-116(P)(A) currently contained in the list of approved methodologies for determining core operating limits at MPS2. Both of these

supplements have been approved by the NRC. In addition, a reanalysis of the MPS2 SBLOCA using Supplement 1 to the AREVA methodology reports is provided for review and approval.

DNC has evaluated whether or not a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of Amendment," as discussed below:

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No.

The proposed changes to TS 6.9.1.8.b permit the use of the recent supplements to the appropriate methodologies to analyze accidents to ensure that the plant continues to meet applicable design criteria and safety analysis acceptance criteria. The proposed changes to the list of NRC-approved methodologies listed in TS 6.9.1.8.b has no impact on plant operation and configuration. The list of methodologies in TS 6.9.1.8.b does not impact either the initiation of an accident or the mitigation of its consequences.

The revised SBLOCA analysis demonstrates MPS2 continues to satisfy the 10 CFR 50.46 ECCS performance acceptance criteria using an NRC-approved evaluation model.

Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No.

The proposed changes have no impact on any plant configuration or system performance. There is no change to the design function or operation of the plant. The proposed changes will not create the possibility of a new or different accident due to credible new failure mechanisms, malfunctions, or accident initiators not previously considered. There is no change to the parameters within which the plant is normally operated, and thus, the possibility of a new or different type of accident is not created.

Therefore, the proposed change does not create the possibility of a new or different kind of accident from those previously evaluated within the FSAR.

3. Does the proposed change involve a significant reduction in a margin of safety?

Response: No.

The proposed changes have no impact on any plant configuration or system performance. Approved methodologies will be used to ensure that the plant continues to meet applicable design criteria and safety analysis acceptance criteria. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above information, DNC concludes that the proposed license amendment presents no significant hazards consideration under the criteria set forth in 10 CFR 50.92(c), and, accordingly, a finding of no significant hazards consideration is justified.

5.3 Precedents

Due to the recent issuance of these supplements, no precedent exists. In addition, no precedent exists for a reanalysis of the MPS2 SBLOCA using these recently issued supplements.

5.4 Conclusion

Based on the considerations discussed above, there is reasonable assurance that (1) the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the requested license amendment will not be inimical to the common defense and security or to the health and safety of the public.

6.0 Environmental Considerations

DNC has reviewed the proposed license amendment for environmental considerations. The proposed license amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion from an environmental assessment as set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

7.0 References

1. J. Alan Price (DNC) letter to USNRC, "Dominion Nuclear Connecticut, Inc., Millstone Power Station Unit 2, Response to Request for Additional Information Regarding the

30-Day Report for Emergency Core Cooling System Model Changes (TAC No. ME 7881)," November 1, 2012, (Proprietary Information).

2. EMF-2328(P)(A), Revision 0, Supplement 1(P), Revision 0, "PWR Small Break LOCA Evaluation Model, S-RELAP5 Based," March 2012.
3. James Kim (USNRC) letter to Dominion Nuclear Connecticut, Inc., "Millstone Power Station, Unit No. 2 – Closure Evaluation for 30-Day Report for Emergency Core Cooling System Model Changes Pursuant to the Requirements of 10 CFR 50.46 (TAC No. ME7881)," July 18, 2013.
4. "Final Safety Evaluation by the Office of Nuclear Reactor Regulation Topical Report EMF-2328(P)(A), Revision 0, Supplement 1, Revision 0, 'PWR [Pressurized Water Reactor] Small Break LOCA [Loss-of-Coolant Accident] Evaluation Model, S-RELAP5 Based', Project No. 728 (TAC No. ME8227)," September 1, 2015.
5. EMF-92-116(P)(A), Revision 0, Supplement 1(P)(A), Revision 0, "Generic Mechanical Design Criteria for PWR Fuel Designs," February 2015.
6. Richard Guzman (USNRC) letter to Dominion Nuclear Connecticut, Inc., "Millstone Power Station, Unit No. 2 – Exemption from 10 CFR 50.46 and Appendix K to Allow Use of M5™ Alloy for Fuel Cladding (TAC No. MF3917)," May 12, 2015.

Attachment 2

Marked-up Technical Specification Pages

2 pages after this cover page

**DOMINION NUCLEAR CONNECTICUT, INC.
MILLSTONE POWER STATION UNIT 2**

May 20, 2015

ADMINISTRATIVE CONTROLS

CORE OPERATING LIMITS REPORT

6.9.1.8 a. Core operating limits shall be established and documented in the CORE OPERATING LIMITS REPORT before each reload cycle or any remaining part of a reload cycle.

- 3/4.1.1.1 SHUTDOWN MARGIN (SDM)
- 3/4.1.1.4 Moderator Temperature Coefficient
- 3/4.1.3.6 Regulating CEA Insertion Limits
- 3/4.2.1 Linear Heat Rate
- 3/4.2.3 TOTAL UNRODDED INTEGRATED RADIAL PEAKING FACTOR - F_r^T
- 3/4.2.6 DNB Margin

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

- 1) EMF-96-029(P)(A) Volumes 1 and 2, "Reactor Analysis System for PWRs Volume 1 - Methodology Description, Volume 2 - Benchmarking Results," Siemens Power Corporation.
- 2) ANF-84-73 Appendix B (P)(A), "Advanced Nuclear Fuels Methodology for Pressurized Water Reactors: Analysis of Chapter 15 Events," Advanced Nuclear Fuels.
- 3) XN-NF-82-21(P)(A), "Application of Exxon Nuclear Company PWR Thermal Margin Methodology to Mixed Core Configurations," Exxon Nuclear Company.
- 4) XN-75-32(P)(A) Supplements 1 through 4, "Computational Procedure for Evaluating Fuel Rod Bowing," Exxon Nuclear Company.
- 5) EMF-2328(P)(A), "PWR Small Break LOCA Evaluation Model S-RELAP5 Based," ^{and Supplement 1} ~~Framatome ANP~~
- 6) EMF-2087(P)(A), "SEM/PWR-98: ECCS Evaluation Model for PWR LBLOCA Applications," Siemens Power Corporation.
- 7) XN-NF-78-44(NP)(A), "A Generic Analysis of the Control Rod Ejection Transient for Pressurized Water Reactors," Exxon Nuclear Company.

May 20, 2015

ADMINISTRATIVE CONTROLS

CORE OPERATING LIMITS REPORT (CONT.)

- 8) XN-NF-621(P)(A), "Exxon Nuclear DNB Correlation for PWR Fuel Designs," Exxon Nuclear Company.
 - 9) XN-NF-82-06(P)(A), and Supplements 2, 4 and 5, "Qualification of Exxon Nuclear Fuel for Extended Burnup," Exxon Nuclear Company.
 - 10) ANF-88-133(P)(A) and Supplement 1, "Qualification of Advanced Nuclear Fuels PWR Design Methodology for Rod Burnups of 62 GWd/MTU," Advanced Nuclear Fuels Corporation.
 - 11) XN-NF-85-92(P)(A), "Exxon Nuclear Uranium Dioxide/Gadolinia Irradiation Examination and Thermal Conductivity Results," Exxon Nuclear Company.
 - 12) ANF-89-151(P)(A), "ANF-RELAP Methodology for Pressurized Water Reactors: Analysis of Non-LOCA Chapter 15 Events," Advanced Nuclear Fuels Corporation.
 - 13) EMF-1961 (P)(A), "Statistical Setpoint/Transient Methodology for Combustion Engineering Type Reactors," Siemens Power Corporation.
 - 14) EMF-2310(P)(A), "SRP Chapter 15 Non-LOCA Methodology for Pressurized Water Reactors," Framatome ANP.
 - 15) EMF-92-153(P)(A) and Supplement 1, "HTP: Departure from Nucleate Boiling Correlation for High Thermal Performance Fuel," Siemens Power Corporation.
 - 16) EMF-92-116(P)(A) ^{and Supplement 1} ~~Revision 0, "Generic Mechanical Design Criteria for PWR Fuel Designs," Siemens Power Corporation.~~
 - 17) BAW-10240(P)(A) Revision 0, "Incorporation of M5™ Properties in Framatome ANP Approved Methods," May 2004.
- c. The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as SHUTDOWN MARGIN, and transient and accident analysis limits) of the safety analysis are met.
- d. The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements thereto, shall be provided upon issuance, for each reload cycle, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.