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March 24, 2016

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

Serial No. 16-070
NLOS/WDC R0
Docket No. 50-336
License No. DPR-65

DOMINION NUCLEAR CONNECTICUT, INC.
MILLSTONE POWER STATION UNIT 2
RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION FOR PROPOSED LICENSE AMENDMENT REQUEST REGARDING SMALL BREAK LOSS OF COOLANT ACCIDENT REANALYSIS (CAC NO. MF6700)

By letter dated September 1, 2015, Dominion Nuclear Connecticut, Inc. (DNC) submitted a license amendment request (LAR) for Millstone Power Station Unit 2 (MPS2). The proposed amendment would 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. In an email dated February 16, 2016, the Nuclear Regulatory Commission (NRC) transmitted a request for additional information (RAI) to DNC related to the LAR. DNC agreed to respond to the RAI by March 28, 2016.

The attachment to this letter provides DNC's response to the NRC's RAI.

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

Sincerely,

Mark D. Sartain
Vice President – Nuclear Engineering

Vicki L. Hull
NOTARY PUBLIC
Commonwealth of Virginia
Reg. # 140542
My Commission Expires May 31, 2018

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 24TH day of MArch, 2016.

My Commission Expires: 5-31-18.

Vicki L. Hull
Notary Public

A001
NRR

Attachment:
Response to Request for Additional Information for License Amendment Request Regarding Small Break Loss of Coolant Accident Reanalysis

Commitments made in this letter: None

cc: U.S. Nuclear Regulatory Commission
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ATTACHMENT

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION FOR LICENSE AMENDMENT REQUEST REGARDING SMALL BREAK LOSS OF COOLANT ACCIDENT REANALYSIS

DOMINION NUCLEAR CONNECTICUT, INC.
MILLSTONE POWER STATION UNIT 2

By letter dated September 1, 2015, Dominion Nuclear Connecticut, Inc. (DNC) submitted a license amendment request (LAR) for Millstone Power Station Unit 2 (MPS2). The proposed amendment would 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 Technical Specificaton (TS) 6.9.1.8.b, "Core Operating Limits Report," which lists the analytical methods used to determine the core operating limits. In an email dated February 16, 2016, the Nuclear Regulatory Commission (NRC) transmitted a request for additional information (RAI) to DNC related to the LAR.

RAI – 1 (SRXB)

Please identify what core operating limit report (COLR) parameters are governed by the methods described in EMF-2328(P)(A) and Supplement 1, and EMF-92-116(P)(A) and Supplement 1.

DNC Response

The methodology of EMF-2328(P)(A) including Supplement 1 is used to establish limits for the COLR parameters listed below to demonstrate acceptable small break loss of coolant accident results.

TS 3.1.3.6 – Regulating CEA Insertion Limits
TS 3.2.1 – Linear Heat Rate
TS 3.2.3 – Total Unrodded Integrated Radial Peaking Factor - F^T_r

The methodology of EMF-92-116(P)(A) including Supplement 1 is used to establish limits for the COLR parameters listed below to demonstrate the fuel rod design criteria are met.

TS 3.1.3.6 – Regulating CEA Insertion Limits
TS 3.2.1 – Linear Heat Rate
TS 3.2.3 – Total Unrodded Integrated Radial Peaking Factor - F^T_r

RAI – 2 (SRXB)

Core Nodalization: The number of nodes in the core presented in Figure 3-3, "S-RELAP5 SBLOCA Reactor Vessel Nodalization," of Attachment 3 of the license amendment request (LAR) appears to be inconsistent with that described in Supplement 1 to EMF-2328. Please confirm that the core nodalization is consistent with the updated methodology.

DNC Response

Figure 3-3, "S-RELAP5 SBLOCA Reactor Vessel Nodalization," of Attachment 3 of the LAR is a general representation of the vessel model for a Combustion Engineering plant. The number of axial nodes in the active core will vary depending on the active core length for each plant. The active core for MPS2 was nodalized as specified in Section 9.2 of Supplement 1 of EMF-2328(P)(A).

RAI – 3 (SRXB)

Reactivity Feedback: As discussed in Supplement 1 to EMF-2328, when Technical Specifications (TSs) allow a positive moderator temperature coefficient (MTC) at full power, the maximum plausible value will be incorporated in order to allow an increase in power prior to reactor SCRAM. Per the recent Cycle 24 COLR, there appears to be a positive MTC when thermal power is >70%. Please confirm that the maximum plausible MTC value was incorporated into the analysis.

DNC Response

In accordance with Supplement 1 of EMF-2328(P)(A), the maximum plausible value of the moderator temperature coefficient (MTC) at full power was used as a basis for the moderator reactivity feedback used in the MPS2 SBLOCA analysis in the LAR submittal.

RAI – 4 (SRXB)

For the most part, the system parameters and initial conditions in Table 3-1 of Attachment 3 of the LAR are unchanged from those provided in Table 14.6.5.2-3 in the Millstone Power Station, Unit 2 (MPS2) Final Safety Analysis Report (FSAR). Please discuss why the safety injection tank (SIT) Fluid Temperature, SG Secondary Pressure, and the AFW Temperature have changed in this analysis compared to those values in the Table 14.6.5.2-3 in the current FSAR. Additionally, please discuss why the uncertainty in the Radial Peaking Factor has increased compared to the uncertainty discussed in Chapter 14.6.5.2 in the FSAR.

DNC Response

The differences between the reported values in Table 14.6.5.2-3 of MPS2 FSAR and Attachment 3 of the submitted LAR are addressed below. Both the FSAR and LAR analyses were performed under the same SBLOCA methodology, EMF-2328(P)(A). However, after the SBLOCA analysis reported in the FSAR was performed, the methodology guidelines on the bias of some of the input parameters were revaluated in part to bound future plant reloads or to increase analysis conservatism.

- SIT Fluid Temperature – The value used in the FSAR analysis is from the SBLOCA methodology used prior to Supplement 1 of EMF-2328(P)(A). The value used in the LAR analysis follows the guidelines provided in Section 6.0 of Supplement 1 of EMF-2328(P)(A).

- Steam Generator (SG) Secondary Pressure – The FSAR reported value is the steady state pressure calculated by S-RELAP5. The LAR reported value is the target pressure used in the steady state analysis to benchmark SG plant conditions with no tube plugging. The steady state pressure calculated by S-RELAP5 in the LAR SBLOCA analysis, with 500 tubes pugged, is 870 psia. The difference between the initial steam generator pressure of the LAR analysis and the current FSAR analysis is insignificant to the results of the analysis.

- Auxiliary Feedwater Temperature (AFW) – The FSAR analysis used a maximum value of 100°F. The LAR analysis used a nominal value of 70°F. The AFW temperature is a second order effect and the use of a nominal temperature is suggested in the current SBLOCA guidelines of the EMF-2328(P)(A) methodology. Parameters that are not of first order significance are not biased.

- The differences in radial peaking factor uncertainties between the FSAR and LAR reported values are in the rodded augmentation factor used by AREVA, which accounts for the effects of rod insertion on F^T_r. This applies only to CE plants with an unrodded F^T_r TS limit. The higher augmentation value in the LAR analysis is the current value recommended by AREVA to bound future plant reloads. This value will be verified in future cycles of the plant.

<u>RAI – 5</u> (SRXB)

Figure 3-4, "Axial Power Distribution Comparison." Please identify the legend titles for the solid and dashed lines since the staff had difficulty reading them due to the clarity of the text. Additionally, has the axial power distribution changed from the distribution used for the analysis in the current FSAR?

<u>DNC Response</u>

The legends for Figure 3-4 are as follows:

Solid line= Step Axial Adjusted to FQ/FΔH Limit
Dashed line = User Input Step Axial

The adjusted axial shape used in the S-RELAP5 SBLOCA calculations for the LAR and FSAR are slightly different because of differences in fuel cycle designs, but peak at the same elevation. The same guidelines have been applied in the processing of the power distribution to satisfy TS limits.

<u>RAI – 6</u> (SRXB)

Auxiliary Feedwater (AFW) Flow: From comparison between Figure 4-13 (Steam Generator Auxiliary Feedwater Mass Flow Rates - 3.78-inch Break) of the LAR and Figure 14.6.5.2-13 of the current FSAR, it appears that there are significant modeling differences for the AFW flow. Please discuss the differences between the two modeling techniques and provide justification for the changes.

<u>DNC Response</u>

The differences between the two modeling techniques are in the way the control logic of the AFW was set and the flow delivery option used. In both the FSAR and the LAR analyses, the AFW is modeled with a simplified control system that injects after SG low level and delay times are met. However, in the LAR analysis, the time delay was reset on every SG low level demand reached through the transient, which explains the flow discontinuity observed in Figure 4-13 of the LAR. In the FSAR analysis, the time delay is only set on the first actuation of AFW. The other difference is in the flow delivery option used. The LAR analysis delivers a constant minimum AFW flow rate and the FSAR analysis delivers variable flow rate as a function of SG pressure. As a result, the FSAR analysis produces higher flows than the LAR throughout the transient. An assessment has been performed which confirms that the overall lower flow rate in the LAR analysis bounds that produced in the FSAR analysis. Further, the assessment also concludes that the AFW flow rate used in the LAR break spectrum analysis bounds the minimum AFW flow requirements for the plant.

<u>RAI – 7</u> (SRXB)

As this is the first plant-specific application of this NRC-approved Small Break Loss of Coolant Accident (SBLOCA) method and for the staff to gain a better understanding of the SBLOCA phenomena for the transition break (3.79-inch diameter break) case, please provide the following six plots for the transition break that were provided for the limiting break (3.78-inch diameter break):

Figure 4-3: Primary and Secondary System Pressure
Figure 4-6: Loop Seal Void Fraction
Figure 4-9: Inner and Outer Core Collapsed Liquid Level
Figure 4-19: Integrated Break Flow and ECCS Flow
Figure 4-21: Hot Assembly Mixture Level
Figure 4-22: Peak Cladding Temperature at PCT Location

DNC Response

Figures 1 through 6 provide the requested plots for the 3.79-inch transition break.

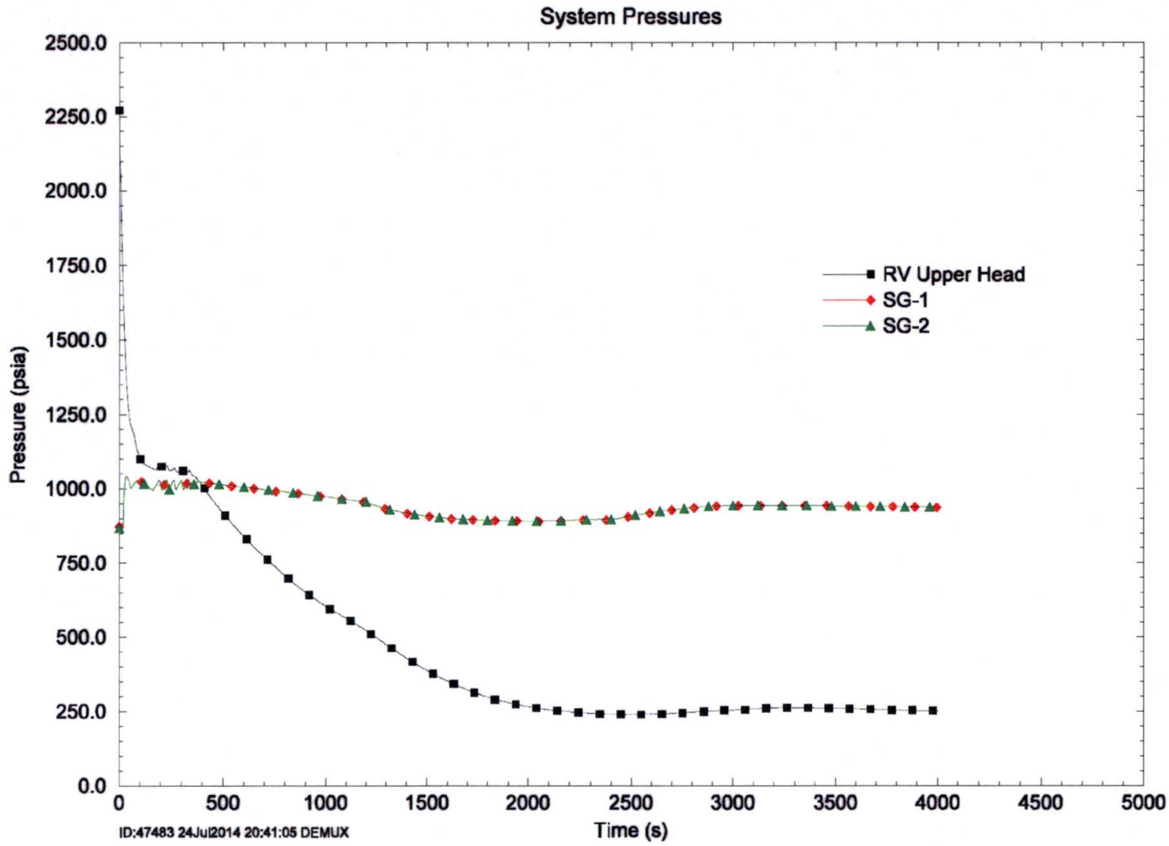

Figure 1: Primary and Secondary System Pressure – 3.79-inch Break

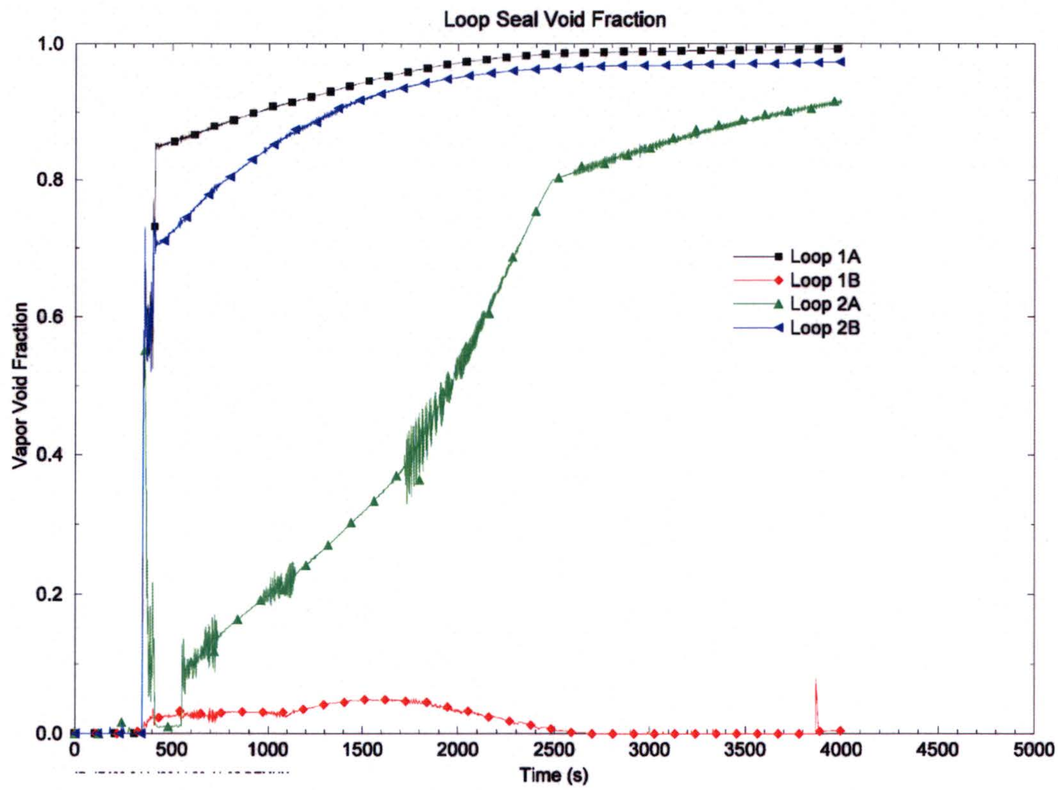

Figure 2: Loop Seal Void Fraction – 3.79-inch Break

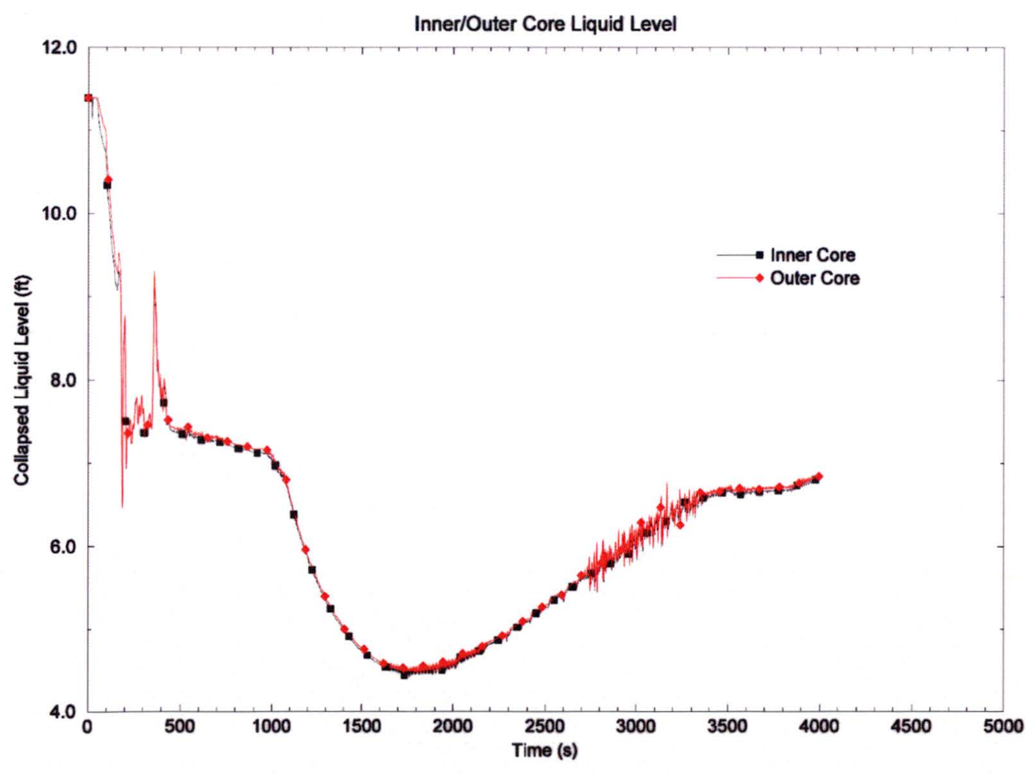

Figure 3: Inner and Outer Core Collapsed Liquid Level – 3.79-inch Break

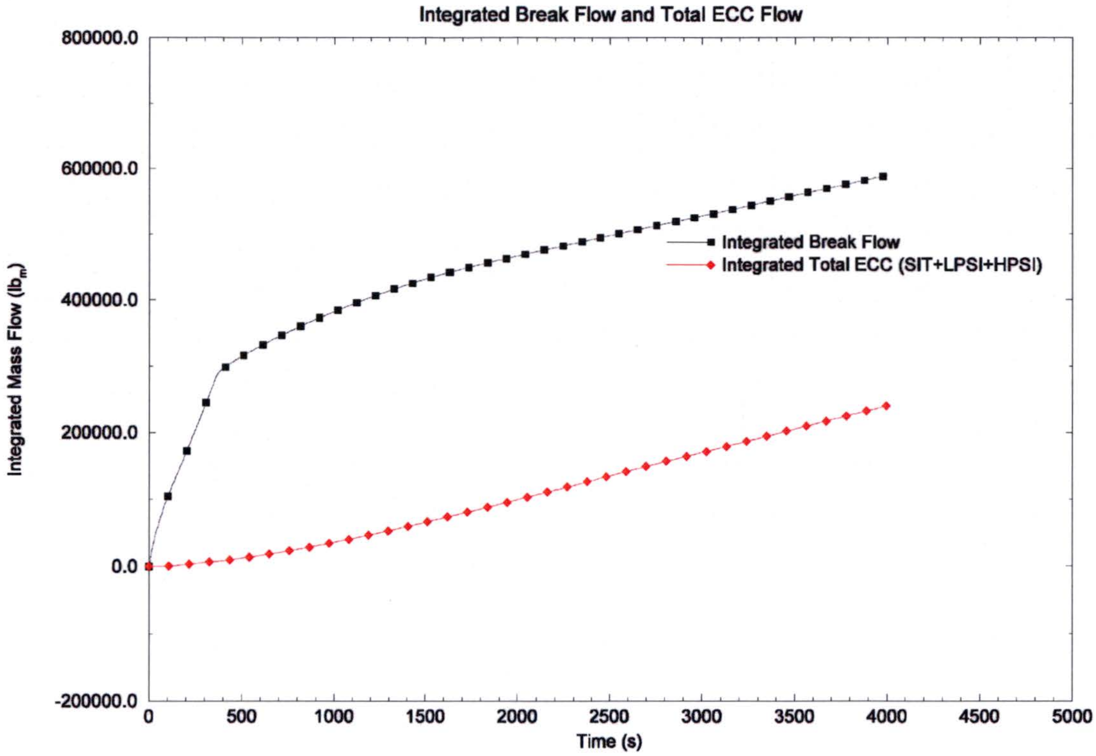

Figure 4: Integrated Break Flow and ECCS Flow – 3.79-inch Break

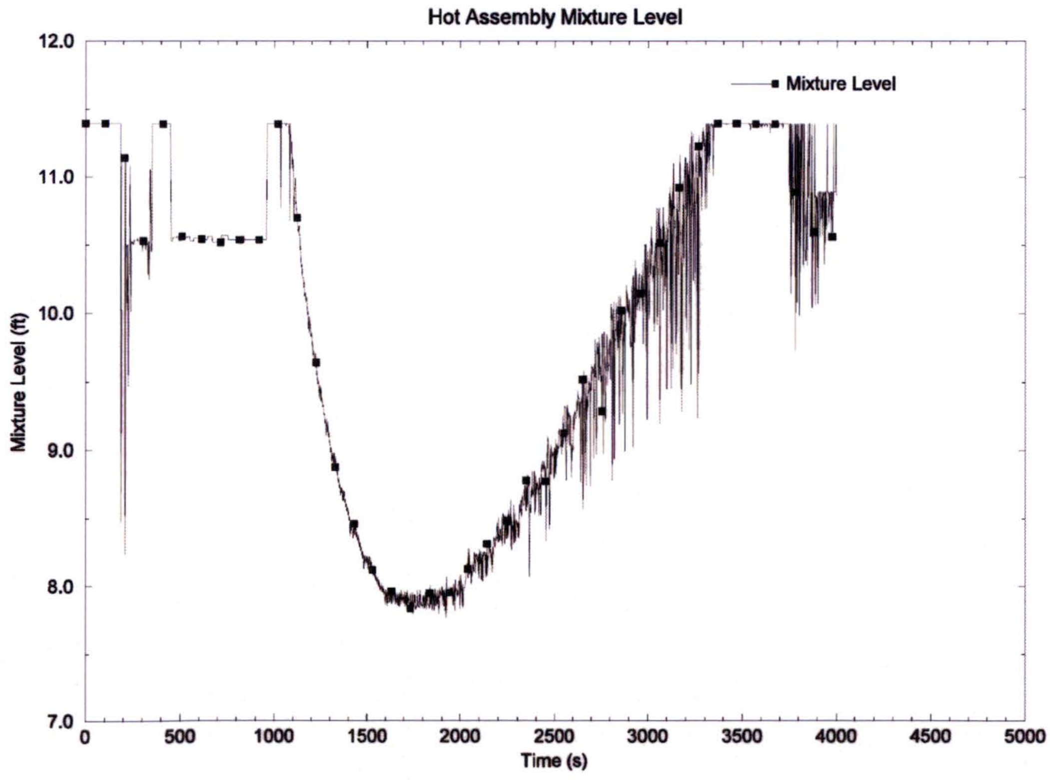

Figure 5: Hot Assembly Mixture Level – 3.79-inch Break

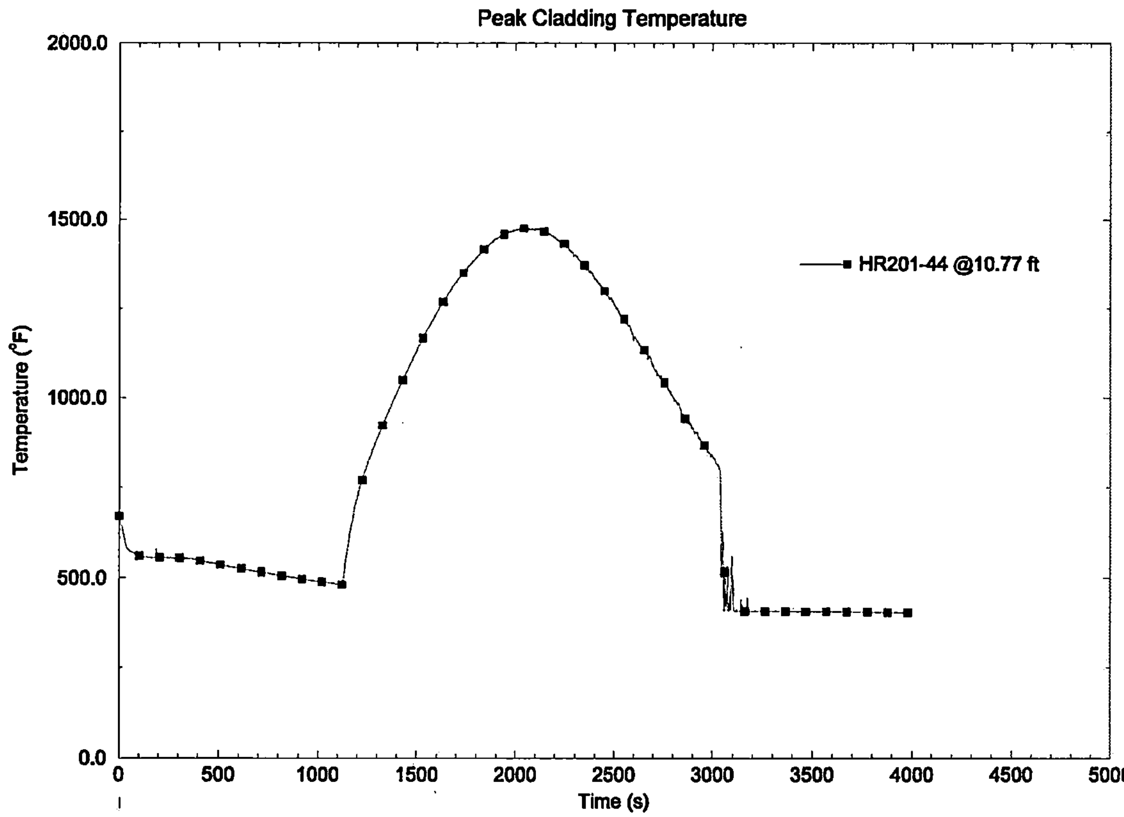

Figure 6: Peak Cladding Temperature at PCT Location – 3.79-inch Break

RAI – 8 (SRXB)

As this is the first plant-specific application of this NRC-approved SBLOCA method and for the staff to gain a better understanding of the Reactor Coolant Pump (RCP) Trip Sensitivity Studies, please provide additional discussion of the following:

a. *Describe the process that was used to determine the limiting break for these cold and hot leg break RCP trip sensitivity studies and please provide the breaks that were analyzed along with their associated peak cladding temperature (PCT) results.*

b. *Were the intact loop seals biased to promote the broken loop clearing, as was in the base break spectrum analysis? If the sensitivity studies did bias the intact loop seals, how was the transition break determined for the cold and hot leg break RCP trip sensitivity studies?*

c. *For the case assuming the relaxation in Appendix K assumptions, what cold and hot leg breaks were rerun to find out the maximum delay time for the operator action?*

d. *The results of the RCP trip sensitivity studies for the cold leg and hot leg breaks conclude that there is at least 2 minutes for the operators to trip the RCPs after the loss of subcooling margin in the cold leg pump suction and there would be additional time if there was some relaxation in the Appendix K requirements. Please elaborate on how the plant's Emergency Operating Procedures (EOPs) are supported by these conclusions.*

DNC Response

Response to Parts a. and b.

The intent of the delayed RCP trip study is to support the EOP RCP trip criteria and demonstrate that 10 CFR 50.46 limits are not exceeded when the trip is delayed by the supported manual trip time. The delayed RCP trip study follows the guidelines provided in Section 5.0 of EMF-2328(P)(A), Supplement 1. For this study, the same break spectrum, biasing of loop seals and transition break size as in the base break spectrum, were applied to both the cold and hot leg break sets. The listing of the break cases and corresponding PCT for both cold and hot leg break sets is provided in Table 1.

Table 1: Summary of Results for the RCP Trip Sensitivity Study

Break Diameter (in)	PCT from Cold Breaks (°F)	PCT from Hot Breaks (°F)
2.00	1075	957
3.00	1372	1158
3.60	1613	1255
3.70	1453	1283
3.75	1354	1299
3.76	1445	1302
3.78	1439	1299
3.785	1448	1280
3.79	1414	1308
4.02	**1644**	1365
4.40	1575	1458
4.60	1532	1516
4.80	1427	1566
5.00	1454	**1575**
5.30	1332	1530
5.50	1366	1484
6.00	1170	1352

Break Diameter (in)	PCT from Cold Breaks (°F)	PCT from Hot Breaks (°F)
7.00	828	1219
8.00	681	1083
9.00	681	970
9.49	681	947

c. For the study assuming the relaxation in Appendix K assumptions, break sizes of 3-inch, 4-inch and 5-inch were analyzed. These cases cover the limiting Appendix K results and are on both sides of the transition break size (3.79 inch).

d. Immediately following a reactor trip, MPS2 operators enter the Standard Post Trip Actions EOP. The EOP directs the operator to check the reactivity control, vital auxiliaries, RCS inventory control safety functions, and then the RCS pressure control safety function. If the pressurizer pressure has dropped to less than the TS trip setpoint for the safety injection actuation signal and a safety injection actuation signal has initiated, the EOP directs the operator to ensure one RCP in each loop is stopped. If pressurizer pressure continues to drop to less than the minimum RCP net positive suction head (NPSH) limit, the EOP directs the operator to stop all RCPs. The RCP NPSH limit corresponds to a pressurizer pressure as a function of RCS cold leg temperature. As an example, at an RCS cold leg temperature of 550°F, pressurizer pressure must be greater than 1550 psia to satisfy the EOP RCP NPSH curve. The saturation pressure of water at a temperature of 550°F is 1045 psia. Following a SBLOCA, the time to reach saturation conditions in the RCS cold leg takes longer than the time to lose NPSH to the RCPs. Therefore, the SBLOCA analysis, which bases the start time for the time-critical operator action to manually trip the RCPs on loss of RCS cold leg subcooling, is conservative. Following approval of the SBLOCA reanalysis contained in the LAR, the EOP time-critical operator action associated with manually tripping the RCPs will be updated to 10 minutes following loss of RCP NPSH. This is consistent with the SBLOCA reanalysis contained in Section 4.3 of Attachments 3 and 4 of the LAR.

<u>RAI – 9</u> (SRXB)

Please provide additional information regarding the evaluation of the Zirc-4 cladding:

a. Please identify which break sizes were rerun and provide their associated PCTs. Also, please provide the PCT vs. time plot for the limiting case.

b. *Did the evaluation include the different RODEX2 input for Zirc-4 and were the correction factors applied for thermal conductivity degradation (TCD)?*

c. *The application states that the PCT penalty will be applied until Zirc-4 is no longer limiting. How will this transition be identified and controlled?*

<u>DNC Response</u>

a. The break sizes and associated PCTs for the Zirc-4 cladding evaluation are listed in Table 2. The PCT vs. time plot for the limiting case is shown in Figure 7.

Table 2: Summary of Results for the Zr-4 SBLOCA Break Spectrum

Break Diameter (in)	3.76	3.78	3.785	3.79	3.90	4.00
PCT (°F)	1646	1711	1682	1541	1586	1677

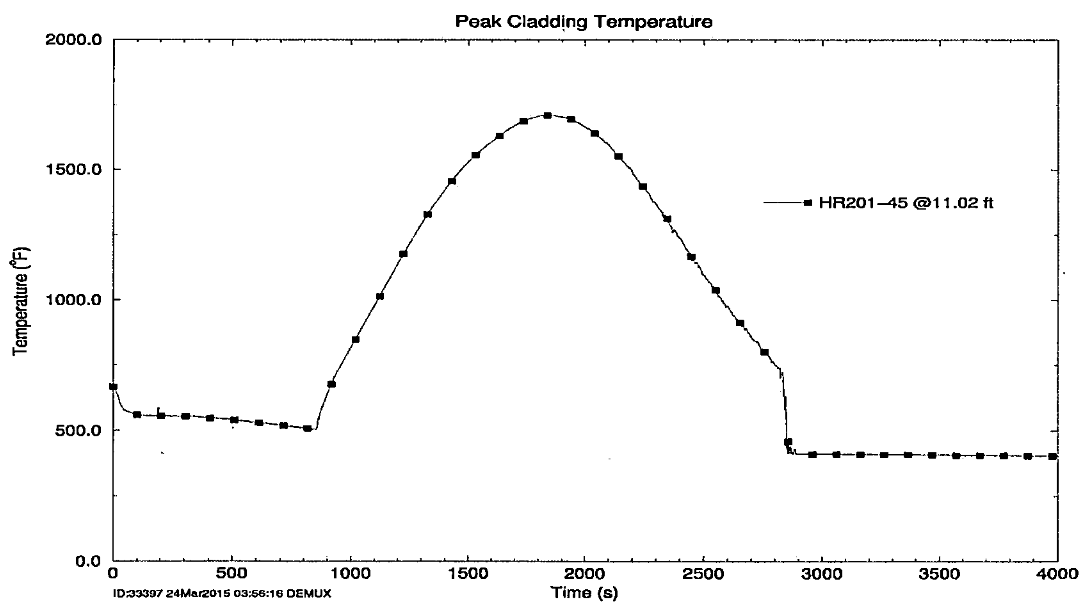

Figure 7: Peak Cladding Temperature at PCT Location – 3.78-inch Break, Zirc-4 Cladding

b. The applicability of TCD corrections to the RODEX2-2A results of the SBLOCA analysis are discussed in the response to RAI-10. The discussion is also applicable to the Zirc-4 cladding SBLOCA evaluation.

c. The cycle design process used by DNC and AREVA will evaluate the need to apply the PCT penalty for Zirc-4 cladding for each reload core. Removal of the PCT penalty when Zirc-4 is no longer limiting will be done in accordance with DNC's 10 CFR 50.46 program.

<u>RAI – 10</u> (SRXB)

Attachment 3 of the LAR does not include discussion regarding the implementation of Supplement 1 to EMF-92-116. Please confirm that the TCD correction factors were applied to the RODEX2 results for input into the SBLOCA analysis.

<u>DNC Response</u>

TCD correction factors developed for fuel temperature predictions in Supplement 1 to EMF-92-116 (P)(A) were not applied in the SBLOCA analysis. The application of TCD fuel temperature correction factors would only affect the fuel stored energy early in the SBLOCA transient. For the SBLOCA analysis, RODEX2-2A is used to determine the initial core and hot pin stored energy. Small breaks evolve through a pump coastdown and natural circulation phase, to a loop draining phase, followed by a boil-off and refill phase. During the coastdown phase of the SBLOCA transient, a single or two-phase forced circulation exists within the RCS, which acts to remove the initial energy of the fuel and deposit it in the steam generators or the containment, thus, preventing a cladding temperature excursion. In either case, the energy content of the fuel has been reduced to that required to transport decay heat out of the fuel by the end of the coastdown phase. The peak cladding temperatures, which occur later in the transient, depend on decay heat versus heat transfer and have no relationship to the initial stored energy within the fuel. Since SBLOCA results are dominated by decay heat, TCD adjustments for the prediction of fuel temperature will not impact SBLOCA results. Therefore, TCD adjustments are not necessary for SBLOCA evaluations. This position is consistent with that described in the response to SNPB RAI-5 on the St. Lucie Unit 2 Fuel Transition (ADAMS Accession Numbers ML15279A222 and ML15279A227).

TCD can also affect the prediction of the rod internal pressure, which affects the limiting fuel power histories selected for the SBLOCA analysis. The power histories used as input into the SBLOCA analysis are based on end of life rod internal pressure calculations, which follow the TCD corrections recommended in Supplement 1 to EMF-92-116 (P)(A).

The use of RODEX2-2A results for input into the SBLOCA analysis is consistent with the method described in EMF-2328, Rev.0, Supplement 1(P)(A).

<u>RAI – 11</u> (SRXB)

The most recent MPS2 SBLOCA 50.46 PCT rackup sheet (ADAMS Accession Number: ML15188A346) contains a number of ECCS model assessments. Was this SBLOCA analysis completed with a version of S-RELAP5 that contains modeling updates that incorporated these assessments? Were any of the modeling updates different than the corrections described in their associated 50.46 reports?

<u>DNC Response</u>

The SBLOCA analysis presented in the LAR submittal was completed using the most current version of S-RELAP5, which takes into account all known reportable errors. Regarding the modeling updates, the analysis is fully compliant with the method described in EMF-2328, Revision 0(P)(A) and Supplement 1(P)(A).