Dominion Nuclear Connecticut, Inc.
Millstone Power Station
Rope Ferry Road
Waterford, CT 06385

Dominion

SEP 26 2001

Docket No. 50-423
B18470

Re: 10 CFR 50.55a(f)(5)(iii)
10 CFR 50.55a(f)(6)(i)

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

Millstone Power Station, Unit No. 3
Response to a Request for Additional Information
Risk-Informed Inservice Inspections (ISI) Program Plan

In a letter dated July 25, 2000,[1] a relief request was submitted to the U.S. Nuclear Regulatory Commission (NRC) to allow implementation of a Risk-Informed Inservice Inspection Program at Millstone Unit No. 3. A conference call was conducted on July 18, 2001, to discuss Millstone Unit No. 3 responses to NRC questions received on June 27, 2001[2]. The purpose of this letter is to transmit the responses to those questions, which are contained in Attachment 1. We request approval of this relief request prior to January 31, 2002. This will allow Millstone Unit No. 3 to implement the Risk Informed ISI program plan during next refueling outage currently scheduled for early September 2002.

There are no regulatory commitments contained within this letter.

If you should have any questions on the above, please contact Mr. Ravi Joshi at (860) 440-2080.

Very truly yours,

DOMINION NUCLEAR CONNECTICUT, INC.

J. Alan Price, Vice President
Nuclear Technical Services - Millstone

Attachment (1)
cc: See next page

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[1] Northeast Nuclear Energy Company letter to U.S. Nuclear Regulatory Commission, "Risk-Informed Inservice Inspection Program Plan, Request for Relief From ASME Section XI," dated July 25, 2000.

[2] U.S. Nuclear Regulatory Commission facsimile to Dominion Nuclear Connecticut, Inc., "Millstone Nuclear Power Station, Unit 3, Facsimile Transmission, Draft Request for Additional Information (RAI) to be Discussed in an Upcoming Conference Call (TAC No. MA9740)," dated June 27, 2001.

A047

cc: H. J. Miller, Region I Administrator
V. Nerses, NRC Senior Project Manager, Millstone Unit No. 3
NRC Senior Resident Inspector, Millstone Unit No. 3

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Attachment 1

Millstone Power Station, Unit No. 3

Response to a Request for Additional Information
Risk-Informed Inservice Inspections (ISI) Program Plan
Supplemental Information

Response to a Request for Additional Information
Risk-Informed Inservice Inspections (ISI) Program Plan
Supplemental Information

In a letter dated July 25, 2000,[(1)] a relief request was submitted to the U.S. Nuclear Regulatory Commission (NRC) to allow implementation of a Risk-Informed Inservice Inspection Program at Millstone Unit No. 3. A conference call was conducted on July 18, 2001, to discuss Millstone Unit No. 3 responses to NRC questions received on June 27, 2001[(2)]. The questions and associated responses are presented below.

Question 1

1) One major step in the WCAP process is the identification of degradation mechanisms and the development of corresponding pipe failure frequencies. The requested Table 1 summarizes the qualitative results of this step by identifying the different degradation mechanisms, combinations of mechanisms, and the prevalence of the different mechanism. The calculated ranges in Table 1 summarize the quantitative results of the analysis. This information will illustrate how the degradation mechanism identification and failure frequency development step in the WCAP methodology was implemented, and provide an overview of the results generated. Please expand the current Table 3.4-1 to include the following information.

a) System	b) Degradation Mechanism/ Combination	c) Failure Probability Range at 40 years with no ISI		d) Number of Susceptible Segments	e) Comments
		Leak	Disabling Leak		

a) System: Each system included in the analysis.

b) Degradation Mechanism/Combination: Segment failure probabilities are characterized in the WCAP method by imposing all degradation mechanisms in a segment (even if they occur at different welds) and the worst case operating conditions at the segment on a "representative" weld, and using the resulting failure probability for the segment. Please identify the dominant degradation mechanisms and combination of degradation mechanisms selected in each system. The reported mechanisms should cover all segments in the system. The table in the current submittal is not clear about which specific degradation

(1) Northeast Nuclear Energy Company letter to U.S. Nuclear Regulatory Commission, "Risk-Informed Inservice Inspection Program Plan, Request for Relief From ASME Section XI," dated July 25, 2000.

(2) U.S. Nuclear Regulatory Commission facsimile to Dominion Nuclear Connecticut, Inc., "Millstone Nuclear Power Station, Unit 3, Facsimile Transmission, Draft Request for Additional Information (RAI) to be Discussed in an Upcoming Conference Call (TAC No. MA9740)," dated June 27, 2001.

mechanisms or combination of mechanisms are included in the leak estimates provided.

c) Failure Probability Range at 40 years with no ISI: For each dominant degradation mechanism and combination of degradation mechanisms, please provide the range of estimates developed for the leak and disabling leak sizes as applicable. The table in the current template provided the range of leak estimates only.

d) Number of Susceptible Segments: Please identify the total number of segments susceptible to each dominant degradation mechanism and combination of degradation mechanisms.

e) Comments: The contents of this column are still being developed. It should provide further explanation and clarifications on the degradation mechanism and results as appropriate. Examples of items to be included are identification of which degradation mechanism are applied to socket welds and if a break calculation was needed to evaluate pipe whip constraints.

Response

The Table information provided below was modified based on our July 18, 2001, telephone conversation with the Staff. It was determined that column d) is not required. The information in the Table has been modified to meet the proposed Table 3.4-1 of the Revision 1, "Example Submittal For Plants that Follow the WOG Methodology (WCAP-14572)".

System (Note 1)	Dominant Potential Degradation Mechanism(s)/ Combination(s)	Range of Cumulative Failure Probability at 40 years with no ISI		Comments
		Small Leak	Disabling Leak (by disabling leak rate) (Note 2)	
RCS	Thermal Fatigue	6.6E-9 - 9.7E-5	LLOCA = 5.8E-8 - 9.2E-6 MLOCA = 8.1E-9 - 3.2E-5 SLOCA = 8.1E-9 - 3.4E-5 SYS = 1.2E-7 - 2.4E-5	Included evaluation of 70 butt welded segments with potential for either thermal fatigue or no mechanism (default classified as thermal fatigue). Potential for high cycle thermal fatigue considered in some small diameter segments.
	Dissimilar Metal Weld	5.6E-5 - 5.9E-5	LLOCA = 3.8E-5 - 3.9E-5 MLOCA = 3.9E-5 - 4.0E-5 SLOCA = 3.9E-5 - 4.1E-5	Included evaluation of 14 segments with dissimilar metal butt welds at vessel safe ends (RPV, SG, & PZR); includes consideration for PWSCC and Thermal Fatigue.

System (Note 1)	Dominant Potential Degradation Mechanism(s)/ Combination(s)	Range of Cumulative Failure Probability at 40 years with no ISI		Comments
		Small Leak	Disabling Leak (by disabling leak rate) (Note 2)	
	Thermal Fatigue, Water Hammer	3.8E-6	MLOCA = 6.4E-6 SLOCA = 7.1E-6	Included evaluation of 2 butt welded segments with a low potential for thermal fatigue and subject to dynamic loading during opening of the PZR PORV's.
	Vibrational Fatigue	5.8E-7	SLOCA = 2.3E-7	Included evaluation of 4 segments with ¾" socket weld test connections to Loop Stop Valves for potential vibration.
	Water Hammer	8.3E-7 - 6.3E-6	SLOCA = 1.2E-7 - 5.4E-6	Included evaluation of three 1" diameter socket welded segments for potential failure during water hammer loads under abnormal conditions.
	Seismic Loads/Impact	3.4E-5	SYS = 2.5E-5	Included evaluation of 1 normally isolated segment for structural loading resulting from movements of the spray line. Default (no mechanism) thermal fatigue.
SIH	Thermal Fatigue	7.0E-7 - 8.0E-6	SYS = 3.7E-7 - 3.9E-6	Included evaluation of 6 butt welded segments. Default (no mechanism) thermal fatigue.
	Vibrational Fatigue	5.3E-6	SYS = 3.0E-6	Included evaluation of 1 butt welded segment and restrictive orifices that may cause moderate vibration during an accident response.
SIL	Thermal Fatigue	1.4E-7 - 8.9E-5	SYS = 1.7E-7 - 6.1E-5	Included evaluation of 6 butt welded segments. Possibility of back leakage from check valve leakage considered.
CHS	Vibrational Fatigue	2.2E-4	SYS = 2.3E-4	Included evaluation of 4 segments having both a socket weld to the RC pump and a butt welded flange. No specific vibration noted, but considered OE from North Anna for a vibration fatigue failure of the same socket weld.
RHS	Thermal Fatigue	8.3E-8	SYS = 5.7E-8 - 5.8E-8	Included evaluation of 2 segments with a pipe to valve butt weld selected on each segment that only operates in Modes 4, 5, 6 and is important to shutdown risk. Potential for thermal fatigue if valve leakage occurs in Modes 1 & 3.

NOTES:(1) RCS - Reactor Coolant System, SIH - High Pressure Safety Injection system, SIL - Low Pressure Safety Injection System, CHS - Chemical Volume & Control System, RHS - Residual Heat Removal System

(2) Disabling leak rate - LLOCA, MLOCA, SLOCA, and SYS (system disabling leak). When no leak rate is shown, this is the system disabling leak rate.

Question 2

Another major step in the WCAP process is assignment of segments into safety significance categories based an integrated decision making process, and the selection of segments for inspection locations. The requested Table 3 summarizes the results of the safety significance categorization process as determined by the quantitative criteria, by the expert panel's deliberation on the medium safety significant segments, and by the expert panel's deliberations based on other considerations. The summarizing information requested in Table 3 will provide an overview of the distribution of the safety significance of the segments based on the quantitative results, and the final distribution based on the integrated decision making. Each segment has four RRWs calculated, a CDF with and without operator action, and a LERF with and without operator action. Please provide the following Table.

System	Number of Segments with Any RRW >1.005	Number of Segments with Any RRW Between 1.005 and 1.001	Number of Segments with Any RRW Between 1.005 and 1.001 Placed in HSS	Number of Segments with All RRW < 1.001 Selected for Inspection

Response

The Table information provided below was modified based on our July 18, 2001, telephone conversation with the Staff. It was determined that an additional column titled "Total Number of Segments Selected for Inspection (High Safety Significant Segments)" needed to be added to the Table. The information in the Table has been modified to meet the proposed Table 3.7-1 of Revision 1, "Example Submittal For Plants that Follow the WOG Methodology (WCAP-14572)."

System (Note 1)	Number of Segments with Any RRW >1.005	Number of Segments with Any RRW Between 1.005 and 1.001	Number of segments with all RRW < 1.001	Number of Segments with Any RRW Between 1.005 and 1.001 Placed in HSS	Number of Segments with All RRW < 1.001 Selected for Inspection	Total Number of Segments Selected for Inspection (High Safety Significant Segments)
RCS	41	26	27	13	2	56
SIH	0	0	7	0	0	0
SIL	0	0	6	0	0	0
CHS	4	0	0	0	0	4
RHS	0	0	2	0	2	2
Total	45	26	42	13	4	62

NOTE: (1) RCS - Reactor Coolant System, SIH - High Pressure Safety Injection System, SIL - Low Pressure Safety Injection System, CHS - Chemical Volume & Control System, RHS - Residual Heat Removal System

Question 3

Another major step in the WCAP process is development of the consequences of segment ruptures. The WCAP methodology requires that a summary of the consequences be developed for each system and provided to the expert panel during their deliberations. Please provide this summary for each system. The summary will illustrate that the appropriate types of consequences (i.e., initiating events, mitigating system failure, and combinations) are included in the evaluation and will provide an overview of the results of the step.

Response

As discussed in our July 28, 2001, telephone conversation, we did not provide the expert panel with a paragraph type summary of the consequences for each system as suggested in the WCAP methodology. We agreed to explain exactly what we did as a substitute for this guidance. The expert panel was provided with a summary table which grouped piping segments by similarities. It was believed that the table provided a better understanding of the system consequences because it broke everything down into segments. This level of detail exceeds the level of information suggested by the Topical Report. The table was used continuously by the expert panel to ensure consistency among the piping segment rankings. It was an easy reference point to use to refer back to segments with similar consequences or RRWs when reviewing the worksheets individually and no indirect consequences affected the Class 1 piping segments. As an example of what was used, a portion of the original summary table is provided as follows:

Group Number	Piping Segments	RRW (CDF) Value	Consequence	Failure Mechanism
1-Hot Leg from Vessel to MOV	RCS-01,RCS-03	1.038	Small, Medium or Large LOCA	Dissimilar Metal Weld
5-Cold Leg from RCP to MOV	RCS-13,14,15,16	1.0057	Small, Medium or Large LOCA	Vibrational Fatigue
8-From CVCS Connector to AV8036A(B,C,D)	RCS-25,26,27,28	1.000	No Impact	Thermal Fatigue
15-Cold Leg 1A (1D) Tee to CVCS Connection CV	RCS-47,48	1.017	Small or Medium LOCA	Thermal Fatigue
17-Cold Leg 1A (1D) Tee to Pressurizer Spray Valve	RCS-51,52	1.0055	Loss of 1 Train of Pressurizer Spray, Small or Medium LOCA	Thermal Fatigue
36-From CV 8367A(B,C,D) to RCP No. 1 Seal	CHS-01,02,03, 04	1.039	Small LOCA	Vibrational Fatigue
37-From CV 8905A (C) to CV8949A (C)	SIH-01,02	1.000	Loss of 1 Train of Hot Leg Recirculation	Thermal Fatigue
38-From Charging Injection RO to CV 8900A (B,C,D)	SIH-03,04,05,06	1.000	Loss of Charging to 1 Loop	Thermal Fatigue
40-From CV 8956A (B,C,D) and CV 8948(B,C,D) to CV 8847A(B,C,D)	SIL-01,02,03,04	1.000	Loss of RWST Inventory Inside Containment, Loss of 1 Acc/RHR/HPSI Injection Line	Thermal Fatigue
42-from MV 8701A (2B) to MV8701C (2C)	RHS-01,02	1.000	No Impact	Thermal Fatigue

Question 4

Please add the statement that the sensitivity study to address uncertainty as described on page 125 (Section 3.6.1) of WCAP-14572, Rev. 1-NP-A was performed and identify how many segments' RRW increased from below 1.001 to greater than or equal to 1.005. If the sensitivity study was not performed, provide a description and justification of any deviation.

Response

The uncertainty analysis as described on page 125 (Section 3.6.1) of WCAP 14572, Rev. 1-NP-A was performed. As a result of the uncertainty analysis, no segments' RRW increased from below 1.001 to greater than or equal to 1.005.

Question 5

Please state that the change in risk calculations were performed according to all the guidelines provided on page 213 (Section 4.4.2) of WCAP-14572, Rev. 1-NP-A or provide a description and justification of any deviation.

Response

The change in risk calculations were performed according to the applicable guidelines provided on page 213 of the WCAP with one small deviation. The proposed program is Class 1 only at Millstone Unit No. 3. Class 1 piping includes the RCS piping and small portions of other systems directly connected to the RCS piping. The justification for this deviation is that all this piping is within the containment and subject to leak detection from equipment such as radiation monitors and sump level indicators. Bullet 3 on page 213 of the WCAP discusses only RCS piping, but does not include portions of other systems connected directly to the RCS piping. For this piping, the failure probability with ISI for those being inspected by NDE and without ISI for those not being inspected was used along with credit for leak detection. Millstone Unit No. 3 has no augmented programs applicable to Class 1 piping and all the other guidelines applicable to augmented programs were not used.

Question 6

The quantitative change in risk results are adequately summarized in the current template tables 3-5 and 3-10. Please state that all four criteria for accepting the final selection of inspection locations provided on pages 214 and 215 in WCAP-14572 Rev. 1-NP-A were applied. If all four criteria were not used, please provide a description and justification of the deviation. If comparison with any of the criteria indicated that "reevaluation" of the selected locations was needed, please identify the criteria that required the reevaluation and summarize the results of the reevaluation. If the results of the reevaluation can be found in the footnotes of Table 5-1, please refer to the footnotes.

Response

All four criteria for accepting the results discussed on pages 214 and 215 in the WCAP were applied. The change in risk evaluation resulted in the identification of 2 piping segments for which examinations are required. When addressing Criterion 1, it was found that two RCS piping segments dominated the slight increase in CDF/LERF. If examinations were included on these two piping segments in the RI-ISI program, the overall risk change would be a slight risk reduction. Criteria 2 and 3 did not result in the addition of examinations. As directed in Criterion 4, the change in risk calculations were revised to credit these two additional examinations. Therefore, this evaluation resulted in the identification of 2 piping segments for which examinations are now required.

Question 7

Briefly describe the qualifications, experience, and training of the users of the SRRA code on the capabilities and limitations of the code.

Response

An engineering team/piping subpanel was established that included expertise from ISI, NDE, materials, welding, and stress analysis with access to expertise from system engineering. The team was trained in the failure probability assessment methodology used in the structural reliability and risk assessment (SRRA) code and the Westinghouse Risk-Informed ISI process as outlined in WCAP-14572, Revision 1-NP-A and the Revision 1-NP-A Supplement 1. Several of the team members have been involved in the Risk-Informed ISI process since its beginnings in the early 1990s and have participated in the development of the process. The team members that actually ran the SRRA code provided significant input into the use of the code for the process as it is currently available for use in the industry today.

Question 8

Intentionally left blank.

Question 9

Please confirm that SRRA code was only used to calculate failure probabilities for the failure modes, materials, degradation mechanisms, input variables, and uncertainties it was programmed to consider as discussed in the WCAP Supplement 1, page 15. For example, the SRRA code should only be applied to standard piping geometry (circular piping geometry with uniform wall thickness). If the code was applied to any non-standard geometry, please describe how the SRRA inputs were developed.

Response

The SRRA code was used to calculate failure probabilities for the failure modes, materials, degradation mechanisms, input variables and uncertainties it was programmed to consider as discussed in the WCAP Supplement 1. All the piping configurations included in the RI-ISI program could be adequately modeled using the SRRA code.

Question 10

Please describe any sensitivity studies performed to support the use of the SRRA code.

Response

No formal sensitivity studies were performed since the program was used within its capabilities, and validation of the program during its development was supported by studies documented in the WCAP Supplement 1. However, preliminary usage of the program included normal exploratory variation of the inputs to determine the impact on the results. For the final analyses, the engineering team/piping subpanel assessed industry and plant experience, plant layout, materials, and operating conditions and identified potential failure mechanisms and causes. Information was gathered from various sources to provide input for the SRRA model. Resulting failure probabilities were compared against postulated damage mechanisms and industry/plant experience for reasonableness. Examples include the expectation of higher failure probabilities for vibratory fatigue and thermal fatigue from thermal cycling, and lower failure probabilities for no active mechanism (default thermal fatigue).

Question 11

Intentionally left blank.

Question 12

Please summarize the system design features and other physical characteristics of the plant as reflected in the risk evaluations that determined the location and the number of locations selected for inspection (this question was suggested for addition by Westinghouse at the May 22, 2001, public meeting).

Response

Millstone Unit No. 3 is a modern 4 Loop Westinghouse PWR plant designed and constructed to ASME Section III (1971 edition with addenda). As such, the piping was designed with greater detail in the definition of plant operating transients and was subject to a more comprehensive analysis of their effect on fatigue life than earlier designs. The detailed stress analysis results and other supporting information was available for development of this Class 1 RI-ISI program.

Because Millstone Unit No. 3 is an ASME Code plant and is of a later design vintage, certain design features simplified the development of this program. For example only 20 B-J socket welds greater than nominal pipe size 1 exist in the Class 1 boundary. This is a very small number compared to older plants and helps eliminate most vibratory fatigue failure concerns. Similarly although much of the SIH piping is small bore, it is butt welded instead of socket welded. In addition, most of the vents and drains connected to the RCS loop piping are constructed with a restrictive orifice to limit leakage if the attached piping were to fail, simplifying the scope of the Class 1 program.

Due to its later vintage, the plant was able to take advantage of earlier lessons learned and could avoid issues such as external chloride induced stress corrosion cracking. Its operating experience has been positive and there have been no Class 1 piping failures other than vibratory fatigue-induced cracks in socket welded instrument taps. There are no known active piping degradation mechanisms although the potential for high cycle thermal fatigue has been considered. Accounting for possible issues with dissimilar metal welds involving Alloy 600, many such welds (14 of 16 on the main reactor coolant loop piping) are included for examination in the proposed program.

As with other plants that have developed a Class 1 RI-ISI program, LOCA initiation was the primary consequence of failure within the Class 1 boundary. This was numerically evident in the risk evaluation results and a primary focus of the Expert Panel. The selections for the RI-ISI program were geared to support prevention of this type of event and at least one location was selected for examination in each high safety significant segment. On the other hand, much of Class 1 drain piping is normally isolated and has little failure consequence resulting in its low safety significant classification.

Question 13

Section 3.4 of your submittal states that, "Generally, the SSRA code was used to estimate where the possible ranges of failure probability would fall. The final probability selected was determined by team members using the relevant information." Page 83 of the Topical states that for Westinghouse Owners Group (WOG) plant application, "(SRRA) tools were used to estimate the failure probabilities for the piping segment." Pages 6 and 7 of the related safety evaluation also state that the failure probability estimate, "is subsequently used to represent the failure probability of the weld." Please explain how your method comports with the approved Topical and the Safety Evaluation. Please also provide an example of the maximum range provided to the expert panel from which to select a value.

Response

There was no selection of failure probabilities from a range of possible values. The engineering team was provided the calculated failure probabilities based on the mechanisms and different results from SRRA runs for "with and without ISI," small leaks, disabling leaks, etc., so that they could determine the reasonableness of the numbers. However, the failure probabilities as calculated by the program were used directly in subsequent risk calculations and were not modified by the engineering team or expert panel. The failure probability estimation process was thus in accordance with the approved Topical Report and its Safety Evaluation Report.

Question 14

In the staff's IPE data base (a data base that includes the results of all the original submitted versions of the IPEs) the MP3 Conditional Core Damage Probability (CCDP) for large Loss of Coolant Accident (LOCA) (>6") is 2.07E-2, Medium LOCA (between 2" and 6") is 1.69E-2, Small LOCA Between 3/8 and 2") is 4.00E-4.

a) What CCDPs and Conditional Large Early Release Probabilities (CLERPs) do you currently have for these LOCA sizes? If the CCDPs and/or CLERPs are location dependent, please provide the different estimates.

Response

LOCA Size	CCDP	CLERP
Large (<6")	1.22E-02	1.76E-05
Medium (between 2" and 6")	1.01E-02	1.96E-05
Small (between 3/8" and 2")	4.94E-03	1.01E-05

The CCDPs and CLERPs are not location dependent.

The IPE CCDP values provided from the staff's IPE database are correct for Medium and Large LOCA; however, the IPE CCDP for Small LOCA is 2.67E-4. When these IPE CCDPs are compared with the current MP3 model CCDPs, the Large LOCA CCDP decreased by a factor of two and the Small LOCA CCDF increased by a factor of 20. In both cases, the modeling of operator actions pertaining to long term decay heat removal were the major reasons for these differences.

The major contributor to core damage due to Large LOCA is the sequence involving operator error to transfer to sump recirculation following successful injection. The human error probability associated with this action decreased from 6E-03 in the IPE to 2E-03 in the current MP3 model. The current human error probability was calculated using a more recent methodology, the HCR/ORE curves from EPRI TR-100259, "An Approach to the Analysis of Operator Actions in Probabilistic Risk Assessment" Final Report, June 1992 and plant specific simulator data. The decrease in this operator action alone resulted in significantly reducing the large LOCA CCDP.

For small LOCA, the IPE modeled two success paths for long term decay heat removal, 1) controlled primary depressurization to conserve RWST inventory and 2) sump recirculation. The operator error probability associated with the controlled primary depressurization was 1.0E-2. The operator error probability associated with sump recirculation was 3.0E-3. These operator actions were considered to be completely independent of one another with an overall human error probability associated with long term decay heat removal of 3.0E-5. This value greatly reduced the importance of the operator action to provide a long term means of decay heat removal within the IPE. This magnitude of human error reduction is prohibited within the current MP3 model. No credit is taken to initiate controlled primary depressurization which includes throttling back safety injection and securing the quench spray pumps to conserve RWST. The rationale for not taking credit for this operator action is due to the uncertainty of whether the timing of the operator actions would be sufficient to conserve RWST inventory to prevent sump recirculation from being required. The operator error probability associated with the transfer to sump recirculation is assigned a value of 6.0E-04 in the current model. So although the current transfer to sump recirculation human error probability decreased from the IPE, the overall operator error probability to provide long term decay heat removal increased within the current model. This, in turn, contributed significantly to the increase in the CCDP for small LOCA.

<u>Question 15</u>

During the review of the WOG Topical and the associated pilot application, it was expected and observed that segments would be distributed throughout the four regions on the WCAP Structural Element Selection Matrix. Your evaluation, however, resulted in only four redundant segments being placed in Region 1.

Question 15(a)

a) Please describe the sequence and timing of events leading to core damage and large early release following the failure of the four segments in Region 1 (rupture of the charging seal injection lines).

Response

This piping segment (assume the same for all four segments) is a length of charging seal injection piping (1.5" to 2" diameter) which connects directly to the reactor coolant pump and given a rupture would result in a loss of RCS inventory out the break. Therefore, a rupture of this line will result in a small LOCA. Failure of this line follows the typical sequence and timing of a small LOCA in the existing PRA model. It is dominated by sequences where injection and decay heat removal are successful but failure of sump recirculation occurs. These sequences result in a late core damage and early release as a result of containment failure shortly after vessel failure due to over pressurization.

Question 15(b)

b) Please provide the four, without ISI, estimated risk measures (the CDF with and without operator action, and the LERF with and without operator action) for these segments.

Response

No operator action was credited and therefore, the CDF without operator action is 2.78E-08/yr and the LERF without operator action is 5.69E-12/yr.

Question 15(c)

c) The submittal states that Westinghouse Owner's Group Peer Review Certification was conducted for the MP3 Probabilistic Risk Assessment (PRA) model in 1999. Please provide any Observation and Fact sheets regarding the Accident Sequence Evaluation subelements supporting the LOCA sequences analysis, and on the sequences used to model the rupture of the charging seal injection lines.

Response

The pertinent Observation and Fact sheets regarding the Accident Sequence evaluation subelements supporting the LOCA sequences analysis, and on the sequences used to model the rupture of the charging seal injection lines are as follows:

FACT/OBSERVATION REGARDING PRA TECHNICAL ELEMENTS

***OBSERVATION (ID: AS-1)* / Element AS / Subelement 17, 18, 22**

The success criteria and associated bases, including the definition of core damage, that were used to develop the event tree logic were originally developed in the PSS. While the SBO Coping Studies used an acceptable definition of core damage (Peak core temperatures > 2200° F), those bases are not always clearly stated in the documentation of the current PSA update, e.g., the event tree calculational files. It is not clear that a consistent definition of core damage was used to develop all the success criteria and operator time windows.

LEVEL OF SIGNIFICANCE

B or C

POSSIBLE RESOLUTION

The MP3 PRA team should consider adopting an industry accepted definition of core damage for future updates, such as core exit temperatures > 1200° F. Preferably the definition should correspond to some observable measurement or quantity that the operators can determine so that the tie in to the HRA time windows is clear and specific. All the event sequence development success criteria and time windows should refer to one consistent core damage definition. If success criteria from the original PSS are continued to be used, their relationship to the adopted core damage definition needs to be understood. This point is emphasized in the ASME PRA standard, Draft 10 and 11 on the Success Criteria Element.

PLANT RESPONSE OR RESOLUTION

This will be performed in the next update.

LEVELS OF SIGNIFICANCE FOR FACTS AND OBSERVATIONS

A.	Extremely important and necessary to address to ensure the technical adequacy of the PRA, the quality of the PRA, or the quality of the PRA update process. (Contingent Item for Grade Assignment.)
B.	Important and necessary to address, but may be deferred until the next PRA update (Contingent Item for Grade Assignment.)
C.	Considered desirable to maintain maximum flexibility in PRA Applications and consistency in the Industry, but not likely to significantly affect results or conclusions.
D.	Editorial or Minor Technical Item, left to the discretion of the host utility.
S.	Superior treatment, exceeding requirements for anticipated applications and exceeding what would be found in most PRAs.

FACT/OBSERVATION REGARDING PRA TECHNICAL ELEMENTS

***OBSERVATION (ID: AS-4)* / Element AS / Subelement 9, 23**

While the 24 hour mission time is generally used, there are examples where it is bypassed. In an earlier version of the SBO event tree, there was a top event "MIT" to capture the functions of mitigating the RCP seal LOCA after electric power recovery was a success. In the most recent update this function was not included, so there seems to be successfully terminated sequences where there is a seal LOCA initiated, AC is restored, and the mission time for LOCA mitigation is truncated at the time of successful recovery. This assumption is optimistic but probably does not impact the CDF calculation in a significant way.

LEVEL OF SIGNIFICANCE

B

POSSIBLE RESOLUTION

It is recommended that for Seal LOCA sequences the mission time for successful mitigation be carried out at least until the leak rate is essentially eliminated via RCS depressurization, or at least 24 hours. Otherwise provide justification why the omission of seal LOCA mitigation does not significantly impact the results. In general, for scenarios in which equipment support functions are recovered, allowing the equipment to be re-started and run, the potential for failure to re-start and failure to run for the entire mission time should be evaluated.

PLANT RESPONSE OR RESOLUTION

This is a completeness issue for SBO sequences and does not impact the MP3 RI-ISI results.

LEVELS OF SIGNIFICANCE FOR FACTS AND OBSERVATIONS

A.	Extremely important and necessary to address to ensure the technical adequacy of the PRA, the quality of the PRA, or the quality of the PRA update process. (Contingent Item for Grade Assignment.)
B.	Important and necessary to address, but may be deferred until the next PRA update (Contingent Item for Grade Assignment.)
C.	Considered desirable to maintain maximum flexibility in PRA Applications and consistency in the Industry, but not likely to significantly affect results or conclusions.
D.	Editorial or Minor Technical Item, left to the discretion of the host utility.
S.	Superior treatment, exceeding requirements for anticipated applications and exceeding what would be found in most PRAs.

FACT/OBSERVATION REGARDING PRA TECHNICAL ELEMENTS

***OBSERVATION (ID: AS-6)* / Element AS / Subelement 6, 14, 17**

There are examples of safety functions and consequential events that either have been omitted, or for which the technical basis for exclusion is not well understood by the PRA team. Examples include:
a) whether the potential for pressurized thermal shock induced failures of the reactor pressure vessel were considered during severe overcooling transients,
b) whether pressure induced STGR during secondary depressurization was considered,
c) whether the potential for consequential bypasses such as letdown isolation and seal return line isolation were considered, and
d) whether the need to isolate the accumulators following injection was considered in development of the event tree logic.

LEVEL OF SIGNIFICANCE

C

POSSIBLE RESOLUTION

In a future update bring the technical basis for treatment of such issues, if available, forward into the current documentation or develop suitable basis. The documentation should include as exhaustive a list as possible of critical safety functions needed to avoid core damage, such as the list used in the EOP critical safety functions, and should discuss how they are addressed in the event tree logic.

PLANT RESPONSE OR RESOLUTION

These examples need to be addressed but there are not likely to significantly affect the RI-ISI results.

LEVELS OF SIGNIFICANCE FOR FACTS AND OBSERVATIONS

A.	Extremely important and necessary to address to ensure the technical adequacy of the PRA, the quality of the PRA, or the quality of the PRA update process. (Contingent Item for Grade Assignment.)
B.	Important and necessary to address, but may be deferred until the next PRA update (Contingent Item for Grade Assignment.)
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FACT/OBSERVATION REGARDING PRA TECHNICAL ELEMENTS

OBSERVATION (ID: AS-12) / **Element AS** / **Subelement 7, 24**

The event sequence pictures in the MP3 event tree analysis notebook show, for small LOCA, an EDG branch, which the notebook explains is a way to filter out contributions from station blackout-related loss of RCP seal cooling during the quantification. However, inspection of the quantification fault tree model showed the expected logic (i.e., no EDG branch), where any SLOCA contributor was "and"ed with SLOCA mitigation logic. Another example is the absence, on the transient event trees, of PORV challenges, which are in fact modeled in the quantification fault tree.

The event sequence illustrations and explanation in the event tree notebook are somewhat confusing relative to what is modeled in the actual CDF model.

LEVEL OF SIGNIFICANCE

B (Although the event tree pictures are not used for quantification, they are presented as documentation that scenarios have been modeled appropriately. Pictures illustrating quantification techniques can be used, but an accurate representation of the actual sequence (either in event tree or fault tree top logic form) should also be presented.)

POSSIBLE RESOLUTION

Consider explaining in either the event tree or quantification notebooks how the actual scenario is defined and the quantification model logic is set up. Also consider including, in the event tree notebook, the quantification fault tree top logic that corresponds to each event tree.

PLANT RESPONSE OR RESOLUTION

The quantification fault tree model provides the expected sequence logic; however, this calls for documentation enhancements within the event trees. There is no impact on the RI-ISI results.

LEVELS OF SIGNIFICANCE FOR FACTS AND OBSERVATIONS

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B.	Important and necessary to address, but may be deferred until the next PRA update (Contingent Item for Grade Assignment.)
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FACT/OBSERVATION REGARDING PRA TECHNICAL ELEMENTS

OBSERVATION (ID: AS-13) **/ Element AS / Subelement 7**

The event tree calculation and the systems notebooks appear to include a relatively large number of conservative assumptions. While each of these viewed singly are reasonable, the peer review team is concerned that the accumulation of so many small conservative assumptions may influence the CDF estimate and may distort the relative risk significance of modeled SSCs. Achievement of the higher grades 3 and 4 in this certification process emphasize the realism of the PSA.

LEVEL OF SIGNIFICANCE

B

POSSIBLE RESOLUTION

To enhance confidence that the PRA can be effective in Grade 3 or 4 applications either avoid these conservative assumptions or justify why they do not, when considered cumulatively, influence the realistic estimation of CDF and LERF.

PLANT RESPONSE OR RESOLUTION

As stated in this observation, there is a concern that the accumulation of so many small conservative assumptions may influence the CDF estimate. Since only a portion of the PRA model (LOCA sequences) is used in the RI-ISI process, this accumulation concern is keep to a minimum. The conservatisms identified within the LOCA trees pertained to the use of a fault exposure factor. These have been applied to the calculation for standby components' unavailability. The peer review noted in Observation and Fact Sheet DA-8 that these are non-standard and probably conservative. The level of significance assigned for DA-8 was "C" which are not likely to significantly affect results. These factors have been removed in a recent PRA update.

LEVELS OF SIGNIFICANCE FOR FACTS AND OBSERVATIONS

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B.	Important and necessary to address, but may be deferred until the next PRA update (Contingent Item for Grade Assignment.)
C.	Considered desirable to maintain maximum flexibility in PRA Applications and consistency in the Industry, but not likely to significantly affect results or conclusions.
D.	Editorial or Minor Technical Item, left to the discretion of the host utility.
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FACT/OBSERVATION REGARDING PRA TECHNICAL ELEMENTS

OBSERVATION (ID: AS-14) **/ Element AS / Subelement 7**

Not all relevant systems are credited. For example, MFW, IA, condensate systems are not included in the model as a backup to the AFW system (By not including these systems the importance of the AFW system may be over-stated (may mask other risk significant contributors).

LEVEL OF SIGNIFICANCE

B. May impact Maintenance Rule importance of the AFW system and its associated components. Could impact risk-informed AOT of the AFW components for on-line maintenance, and possibly lead to unrealistic AOT for other components.

POSSIBLE RESOLUTION

Consider modeling the MFW system.

PLANT RESPONSE OR RESOLUTION

The AFW system is modeled within the Small LOCA event tree. If the contribution due to failure of the AFW system and another credited backup, Bleed and Feed is examined, this sequence contributes a minimal amount to the overall CDF due to small LOCA. In addition the operator action to initiate feed and bleed cooling is highly dependent on other decay heat removal recovery actions, such as recovering MFW, so the additional benefit is lessened.

LEVELS OF SIGNIFICANCE FOR FACTS AND OBSERVATIONS

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Question 16

The submittal states that "at least one" structural element per HSS segment in the reactor coolant loop piping will be inspected. How many segments per loop are HSS and how many are Low Safety Significant (LSS)? How many volumetric inspections will be done in each reactor coolant loop?

Response

The main (or large diameter) reactor coolant loop piping has been divided into 5 segments per loop. All 5 segments in each loop are HSS. None were determined to be LSS. Each HSS segment will receive 1 volumetric exam. Therefore, 5 exams per reactor coolant loop will be performed.

Question 17

What criteria did you use to differentiate between High Failure Importance and Low Failure Importance in Figure 3.7-1 of WCAP-14572, Rev. 1-NP-A? Please include the break size and frequency (or 40-year probability).

Response

The guidance provided in the WCAP under Section 3.7 was used. Within the guidance it shows that a segment would be considered of high failure importance if:

^{P}Large Leak $> 10^{-3} - 10^{-4}$ per 40 year operating life.

For this submittal, the lower failure probability value of 10^{-4} was conservatively used as the threshold for High Failure Importance.

The CHS system segments CHS-01, CHS-02, CHS-03, & CHS-04, RCP pump seal injection lines, were considered to be of High Failure Importance since they are the only ones to exceed 10^{-4} cumulative 40 year large leak failure probability. The lines are 1½" to 2" in diameter. The Break Size/Disabling Leak basis was 20% of the 8 gpm seal injection flow or 1.6 gpm. The 40 year failure probability was calculated as 2.2E-4.

Question 18

The failure probability estimates used to support the statistical analysis are developed specifically to meet the statistical model input parameter definitions. These parameters are different from the failure estimate parameters used in support of the segment ranking and change in risk calculations, and it is expected that the value of the parameter would also be different. If your methodology deviated from the Topical report, please describe and justify your criteria and calculations:

Please confirm that the "Probability of a [unacceptable] Flaw" and "Conditional Probability of Leak/Year/Weld" are calculated for MP3 using the SRRA code as described on page 171 of the Topical report. How many SRR calculations were made to support the statistical analysis? Please confirm that the suggested probability of detection of 0.2 and the "Target Leak Rate/Year/Weld" as provided in Table 3.7-1 were used.

Response

There was no deviation from the methodology of the Topical Report. The "Probability of Flaw" (Perdue model cell C6) and "Conditional Probability of Leak/Yr/Weld" (Perdue model cell C8) are calculated in accordance with page 171 of the Topical Report. The detailed method of deriving the input parameters from SRRA was obtained by direct communication with the pilot plant described in the Topical Report. This same method was used at Millstone. More specifically, the C6 parameter is taken from the results of a SRRA auxiliary routine which is given parameters: end of license (EOL) age of the plant, the current age of the plant, piping OD, thickness, number of flaws/inch, flaw median depth, and standard deviation of depth (latter three items from SRRA intermediate results). The C8 parameter is based on SRRA results for cumulative failure probability at EOL (40 yr) and cumulative failure probability at current age (CA = 15 yr for Millstone), and is calculated as:

$$C8 = \frac{(P_{EOL} - P_{CA})}{(EOL - CA)}$$

The probability of detection (parameter C7) as set at 0.2, the same as was used in the pilot plant. The "Target Leak Rate/yr/weld" (parameter C10) was set using the Table 3.7-1 of the Topical Report. Thus, there were no deviations from the approved Topical Report. The auxiliary SRRA routine was run 13 times, once for each unique combination of pipe diameter and wall thickness. There were no additional SRRA runs required since the required input parameters were available from the existing SRRA runs. There was one Perdue model analysis for each HSS segment.