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March 30, 2011

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

Serial No. 11-109
NLOS/WDC R0
Docket No. 50-336
License No. DPR-65

DOMINION NUCLEAR CONNECTICUT, INC.
MILLSTONE POWER STATION UNIT 2
RELIEF REQUEST RR-04-11 IMPLEMENTATION OF A RISK-INFORMED INSERVICE INSPECTION PROGRAM BASED ON ASME CODE CASE N-716

Pursuant to 10 CFR 50.55a(g)(3)(i), Dominion Nuclear Connecticut, Inc. (DNC) requests authorization to implement a risk-informed inservice inspection (RI-ISI) program based on the American Society of Mechanical Engineers (ASME) Code Case N-716, as documented in the attached Millstone Power Station Unit 2 (MPS2) Relief Request RR-04-11 (Attachment 1). RR-04-11 has been developed consistent with the Electric Power Research Institute (EPRI) template. This approach is similar to the many submittals the NRC staff has approved to date: ANO Units 1 and 2 (Safety Evaluation Report (SER) letters dated June 2, 2010 and January 5, 2011, respectively), Calvert Cliffs Units 1 and 2 (SER letter dated November 19, 2009), DC Cook Units 1 and 2 (SER letter dated September 28, 2007), Ginna (SER letter dated October 1, 2010), Grand Gulf (SER letter dated September 21, 2007), Nine Mile Point Unit 1 (SER letter dated March 15, 2010), North Anna Units 1 and 2 (SER letter dated January 21, 2011), River Bend (SER letter dated April 2008), Vogtle Units 1 and 2 (SER letter dated March 3, 2010), and Waterford 3 (SER letter dated April 28, 2008).

In accordance with 10 CFR 50.55a(a)(3)(i), the proposed alternative to the referenced requirements may be approved by the NRC provided an acceptable level of quality and safety are maintained. DNC considers the proposed alternative to meet this requirement.

DNC requests to implement this alternative for the entire fourth 10-year ISI interval for MPS2. The MPS2 fourth 10-year interval began April 1, 2010 and will end March 31, 2020.

DNC requests review and approval of the attached relief request by March 31, 2012 in order to plan completion of first period examinations.

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

Sincerely,

J. A. Price
Vice President – Nuclear Engineering

Attachments: (1)

1. Relief Request RR-04-11 Implementation of a Risk-Informed Inservice Inspection Program Based on ASME Code Case N-716

Commitments made in this letter:

1. None

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ATTACHMENT 1

RELIEF REQUEST RR-04-11
IMPLEMENTATION OF A RISK-INFORMED INSERVICE INSPECTION PROGRAM BASED ON ASME CODE CASE N-716

DOMINION NUCLEAR CONNECTICUT, INC.
MILLSTONE POWER STATION UNIT 2

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MILLSTONE POWER STATION UNIT 2
REQUEST FOR ALTERNATIVE

1. INTRODUCTION

Millstone Power Station Unit 2 (MPS2) is currently in the fourth 10-year inservice inspection (ISI) interval as defined by the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Section XI Code for Inspection Program B [Ref. 5.13]. The MPS2 fourth 10-year ISI interval began April 1, 2010. MPS2 plans to implement a risk-informed/safety-based (RIS_B) ISI program during the entire fourth 10-year interval.

The fourth 10-year ISI interval ASME Section XI code of record for MPS2 is the 2004 Edition with No Addenda for Examination Category B-F, B-J, C-F-1, and C-F-2 Class 1 and 2 piping components.

The objective of this submittal is to request the use of the RIS_B process for the ISI of Class 1 and 2 piping. The RIS_B process used in this submittal is based upon ASME Code Case N-716, *Alternative Piping Classification and Examination Requirements, Section XI Division 1* [Ref. 5.3], which is founded in large part on the RI-ISI process as described in Electric Power Research Institute (EPRI) Topical Report (TR) 112657 Rev. B-A, *Revised Risk-Informed Inservice Inspection Evaluation Procedure* [Ref. 5.1].

1.1 Relation to NRC Regulatory Guides 1.174 and 1.178

As a risk-informed application, this submittal meets the intent and principles of Regulatory Guide (RG) 1.174, *An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis* [Ref. 5.4], and RG 1.178, *An Approach for Plant-Specific Risk-Informed Decisionmaking Inservice Inspection of Piping* [Ref. 5.5]. Additional information is provided in Section 3.4.2 relative to defense-in-depth.

1.2 Probabilistic Risk Assessment Quality

The MPS2 probabilistic risk assessment (PRA) model was originally developed in response to the NRC Generic Letter 88-20 on Individual Plant Examinations (IPE). The IPE was submitted to the NRC in December, 1993. This was accepted by the NRC and used for various applications. The PRA model and supporting documentation was maintained and updated to accurately reflect the current plant configuration and operating practices consistent with its intended application. In November, 2003 DNC submitted a risk-informed inservice inspection (RI-ISI) program for MPS2 to the NRC. This program was developed in accordance with Westinghouse Owners Group WCAP-14572, Revision 1-NP-A, and was supported by the MPS2 PRA model which was released in October 2002. The NRC responded with a safety evaluation report in a letter dated April 1, 2005, stating the proposed RI-ISI program was an acceptable alternative to the requirements of ASME Boiler and Pressure Vessel Code, Section XI for ISI and Class 1 piping.

The MPS2 PRA model continued to be maintained and enhanced to accurately reflect the current plant configuration and operating practices. The model received a formal industry PRA peer review in 2006. Facts and Observations (F&Os) concerning the model were recorded and prioritized to be addressed in the following model update or to be considered as a future model enhancement. In 2006 and 2007, several revisions were made to the MPS2 PRA to address F&Os from the peer review and provide additional model enhancements. A self assessment was performed on this model in October, 2007 to assess the MPS2 PRA model against ASME PRA standard RA-Sb-2005 and Revision 1 of the NRC Regulatory Guide

1.200. This self assessment identified all PRA modeling and documentation supporting requirements (SRs) where the model did not meet Capability Category II of the ASME PRA Standard.

In January 2011, the MPS2 PRA model was updated to meet the Capability Category II supporting requirements for internal events and internal flooding of the ASME/ANS Combined PRA standard ASME/ANS RA-Sa–2009 [Ref. 5.9] and Regulatory Guide 1.200, Rev. 2 [Ref. 5.10]. The model quality was reviewed and gaps where Capability Category II supporting requirements are not met are addressed in Table 1.1 of this attachment [Ref. 5.2, Ref. 5.16]. The model quality was also reviewed to confirm it met Capability Category III for supporting requirements IFSN-A14 and IFSN-A16, which is appropriate for this application based on the use of the internal flooding model. Based on the above, DNC considers the current MPS2 PRA model, which was used to support the RIS_B Program, has an acceptable level of quality to support this application.

2. PROPOSED ALTERNATIVE TO CURRENT INSERVICE INSPECTION PROGRAMS

2.1 ASME Section XI

ASME Section XI Examination Categories B-F, B-J, C-F-1, and C-F-2 currently contain requirements for the nondestructive examination (NDE) of Class 1 and 2 piping components.

The alternative RIS_B Program for piping is described in Code Case N-716. The RIS_B Program will be substituted for the current program for Class 1 and 2 piping (Examination Categories B-F, B-J, C-F-1 and C-F-2) in accordance with 10 CFR 50.55a(a)(3)(i) by alternatively providing an acceptable level of quality and safety. Other non-related portions of the ASME Section XI Code will be unaffected.

2.2 Augmented Programs

The impact of the RIS_B application on the various plant augmented inspection programs listed below were considered. This section documents only those plant augmented inspection programs that address common piping with the RIS_B application scope (e.g., Class 1 and 2 piping).

- An augmented inspection program exists to identify and inspect all Alloy 600/82/182 locations in accordance with ASME Code Case N-722 as required by 10 CFR 50.55a. The RIS_B Program proposed in this submittal does not take credit for these augmented inspections. The analysis for Primary Water Stress Corrosion Cracking (PWSCC) was performed as outlined in Code Case N-716 and selections for inspections were made to meet all the criteria of Code Case N-716. Welds that have been overlaid for mitigation were still considered susceptible to PWSCC. Weld overlays were not credited to reduce the PWSCC population.

- The plant augmented inspection program for flow accelerated corrosion (FAC) per Generic Letter (GL) 89-08, *Erosion/Corrosion-Induced Pipe Wall Thinning*, is relied upon to manage the FAC degradation mechanism but is not otherwise affected or changed by the RIS_B Program.

3. RISK-INFORMED / SAFETY-BASED ISI PROCESS

The process used to develop the RIS_B Program conformed to the methodology described in Code Case N-716 and consisted of the following steps:

- Safety Significance Determination
- Failure Potential Assessment

- Element and NDE Selection
- Risk Impact Assessment
- Implementation Program
- Feedback Loop

These steps are explained in detail in the following steps.

3.1 Safety Significance Determination

The systems assessed in the RIS_B Program are provided in Table 3.1 of this attachment. The piping and instrumentation diagrams and additional plant information, including the existing plant ISI Program, were used to define the piping system boundaries.

Per Code Case N-716 requirements, piping welds are assigned safety-significance categories which are used to determine the inspection techniques. High Safety Significant (HSS) welds are determined in accordance with the requirements below. Low Safety Significant (LSS) welds include all other Class 2, 3, or Non-Class welds.

(1) Class 1 portions of the reactor coolant pressure boundary (RCPB), except as provided in 10 CFR 50.55a(c)(2)(i) and (c)(2)(ii);

(2) Applicable portions of the shutdown cooling pressure boundary function. That is, Class 1 and 2 welds of systems or portions of systems needed to utilize the normal shutdown cooling flow path either:

(a) As part of the RCPB from the reactor pressure vessel (RPV) to the second isolation valve (i.e., farthest from the RPV) capable of remote closure or to the containment penetration, whichever encompasses the larger number of welds; or

(b) Other systems or portions of systems from the RPV to the second isolation valve (i.e., farthest from the RPV) capable of remote closure or to the containment penetration, whichever encompasses the larger number of welds;

(3) That portion of the Class 2 feedwater system [> 4 inch nominal pipe size (NPS)] of pressurized water reactors (PWRs) from the outer containment isolation valve to the steam generator;

(4) Piping within the break exclusion region (> NPS 4 inch) for high-energy piping systems as defined by the Owner. (MPS2 has not defined a break exclusion region and does not take credit for one in the high energy line break analysis.) This may include Class 3 or Non-Class piping; and

(5) Any piping segment whose contribution to core damage frequency (CDF) is greater than 1E-06/yr or 1E-07/yr for large early release frequency (LERF) based upon a plant-specific PRA of pressure boundary failures (e.g., pipe whip, jet impingement, spray, inventory losses). This may include Class 3 or Non-Class piping

3.2 Failure Potential Assessment

Failure potential estimates were generated utilizing industry failure history, plant-specific failure history, and other relevant information. These failure estimates were determined using the guidance provided in EPRI TR-112657 (i.e., the EPRI RI-ISI methodology), with the exception of the deviation discussed below.

Table 3.2 of this attachment summarizes the failure potential assessment by system for each degradation mechanism that was identified as potentially operative.

A deviation to the EPRI RI-ISI methodology has been implemented in the failure potential assessment for MPS2. Table 3-16 of EPRI TR-112657 contains criteria for assessing the potential for thermal stratification, cycling, and striping (TASCS). Key attributes for horizontal or slightly sloped piping greater than NPS 1 inch include:

1. The potential exists for low flow in a pipe section connected to a component allowing mixing of hot and cold fluids; or

2. The potential exists for leakage flow past a valve, including in-leakage, out-leakage and cross-leakage allowing mixing of hot and cold fluids; or

3. The potential exists for convective heating in dead-ended pipe sections connected to a source of hot fluid; or

4. The potential exists for two phase (steam/water) flow; or

5. The potential exists for turbulent penetration into a relatively colder branch pipe connected to header piping containing hot fluid with turbulent flow

   AND

   $\Delta T > 50°F$,

   AND

   Richardson Number > 4 (this value predicts the potential buoyancy of a stratified flow)

These criteria, based on meeting a high cycle fatigue endurance limit with the actual ΔT assumed equal to the greatest potential ΔT for the transient, will identify locations where stratification is likely to occur, but allows for no assessment of severity. As such, many locations will be identified as subject to TASCS where no significant potential for thermal fatigue exists. The critical attribute missing from the existing methodology that would allow consideration of fatigue severity is a criterion that addresses the potential for fluid cycling. The impact of this additional consideration on the existing TASCS susceptibility criteria is presented below.

- **Turbulent Penetration TASCS**

  Turbulent penetration typically occurs in lines connected to piping containing hot flowing fluid. In the case of downward sloping lines that then turn horizontal, significant top-to-bottom cyclic ΔTs can develop in the horizontal sections if the horizontal section is less than about 25 pipe diameters from the reactor coolant piping. Therefore, TASCS is considered for this configuration.

  For upward sloping branch lines connected to the hot fluid source that turn horizontal or in horizontal branch lines, natural convective effects combined with effects of turbulence penetration will keep the line filled with hot water. If there is no potential for in-leakage towards the hot fluid source from the outboard end of the line, this will result in a well-mixed fluid condition where significant top-to-bottom ΔTs will not occur. Therefore, TASCS is not considered for these configurations. Even in fairly long lines, where some heat loss from the outside of the piping will tend to occur and some fluid stratification may be present, there is no significant potential for cycling as has been observed for the in-leakage case. The effect of TASCS will not be significant under these conditions and can be neglected.

- **Low flow TASCS**

  In some situations, the transient startup of a system (e.g., shutdown cooling suction piping) creates the potential for fluid stratification as flow is established. In cases where no cold fluid source exists, the hot flowing fluid will fairly rapidly displace the cold fluid in stagnant lines, while fluid mixing will occur in the piping further removed from the hot source and stratified conditions will exist only briefly as the line fills with hot fluid. As such, since the situation is transient in nature, it can be assumed that the criteria for thermal transients (TT) will govern.

- **Valve leakage TASCS**

  Sometimes a very small leakage flow of hot water can occur outward past a valve into a line that is relatively colder, creating a significant temperature difference. However, since this is generally a "steady-state" phenomenon with no potential for cyclic temperature changes, the effect of TASCS is not significant and can be neglected.

- **Convection Heating TASCS**

  Similarly, there sometimes exists the potential for heat transfer across a valve to an isolated section beyond the valve, resulting in fluid stratification due to natural convection. However, since there is no potential for cyclic temperature changes in this case, the effect of TASCS is not significant and can be neglected.

In summary, these additional considerations for determining the potential for thermal fatigue as a result of the effects of TASCS provide an allowance for considering cycle severity. The above criteria have previously been submitted by EPRI to the NRC [Ref. 5.6] to assist in review of this type of risk-informed submittal. The methodology used in the MPS2 RIS_B application for assessing TASCS potential conforms to these updated criteria.

3.3 Element and NDE Selection

Code Case N-716 and lessons learned from the Grand Gulf [Ref. 5.7] and DC Cook [Ref. 5.8] RIS_B pilot applications provide criteria for identifying the number and location of required examinations. Ten percent of the HSS welds shall be selected for examination as follows:

(1) Examinations shall be prorated equally among systems to the extent practical, and each system shall individually meet the following requirements:

   (a) A minimum of 25% of the population identified as susceptible to each degradation mechanism and degradation mechanism combination shall be selected.
   (b) If the examinations selected above exceed 10% of the total number of HSS welds, the examinations may be reduced by prorating among each degradation mechanism and degradation mechanism combination, to the extent practical, such that at least 10% of the HSS population is inspected.
   (c) If the examinations selected above are not at least 10% of the HSS weld population, additional welds shall be selected so that the total number selected for examination is at least 10%.

(2) At least 10% of the RCPB welds shall be selected.

(3) For the RCPB, at least two-thirds of the examinations shall be located between the inside first isolation valve (IFIV) (i.e., isolation valve closest to the RPV) and the RPV.

(4) A minimum of 10% of the welds in that portion of the RCPB that lies outside containment (OC) (e.g., portions of the main feedwater system in boiling water reactors) shall be selected.

(5) A minimum of 10% of the welds within the break exclusion region (BER) shall be selected. (This does not apply to MPS2 since no BER has been defined, nor is one credited in the high energy line break analysis.)

In contrast to a number of RI-ISI program applications where the percentage of Class 1 piping locations selected for examination has fallen substantially below 10%, Code Case N-716 mandates that 10% be chosen. A brief summary is provided below, and the results of the selections are presented in Table 3.3 of this attachment. Section 4 of EPRI TR-112657 was used as guidance in determining the examination requirements for these locations.

Unit	Class 1 Welds(1)		Class 2 Welds(2)		NC Welds(3)		All Piping Welds(4)	
	Total	Selected	Total	Selected	Total	Selected	Total	Selected
2	553	63	1236	7	10	3	1799	73

Notes
(1) Includes all Category B-F and B-J locations.
(2) Includes all Category C-F-1 and C-F-2 locations.
(3) NC - non-class, Fire Protection.
(4) Regardless of safety significance, Class 1, 2 and 3 piping components will continue to be pressure tested as required by the ASME Section XI Program. VT-2 visual examinations are scheduled in accordance with the station's Pressure Test Program that remains unaffected by the RIS_B Program.

3.3.1 Additional Examinations

If the flaw is original construction or otherwise is acceptable, Code rules do not require any additional inspections. Except for the case of Class 1 Alloy 600/82/182 welds that have been overlaid to mitigate PWSCC concerns, any unacceptable flaw will be evaluated per the requirements of ASME Code Section XI, IWB-3500 and/or IWB-3600. As part of performing evaluation to IWB-3600, the degradation mechanism that is responsible for the flaw will be determined and accounted for in the evaluation. The process for ordinary flaws is to perform the evaluation using ASME Section XI. If the flaw meets the criteria, then it is noted and appropriate successive examinations scheduled. If the nature and type of the flaw is service-induced, then similar systems or trains will be examined.

If the flaw is found unacceptable for continued operation, it will be repaired in accordance with IWA-4000 and/or applicable ASME Section XI Code Cases. The need for extensive root cause analysis beyond that required for IWB-3600 evaluation is dependent on practical considerations (i.e., the practicality of performing additional NDE or removing the flaw for further evaluation during the outage). IWB-3600 requires the evaluation to be submitted to the NRC. Finally, the evaluation will be documented in the corrective action program and the Owner reports required by Section XI.

The evaluation will include whether other elements in the segment or additional segments are subject to the same root cause conditions. Additional examinations will be performed on those elements with the same root cause conditions or degradation mechanisms. The additional examinations will include HSS elements up to a number equivalent to the number of elements required to be inspected during the current outage. If unacceptable flaws or relevant conditions are again found similar to the initial problem, the remaining elements identified as susceptible will be examined during the current outage. No additional examinations need be performed if there are no additional elements identified as being susceptible to the same root cause conditions.

Flaws discovered in welds that have been overlaid to mitigate PWSCC concerns are addressed by MPS2 Relief Request RR-04-07 [Ref. 5.12], currently submitted to the NRC for approval. Additional examinations will be performed in accordance with Relief Request RR-04-07 and any agreed upon actions identified during the approval of Relief Request RR-04-07. DNC anticipates that these welds will be addressed specifically in the next revision of 10CFR50.55a, at which time the MPS2 weld inspection program will be updated to incorporate applicable new regulatory requirements.

3.3.2 Program Relief Request

An attempt has been made to select RIS_B locations for examination such that a minimum of >90% coverage (i.e., Code Case N-460 criteria) is attainable. However, some limitations will not be known until the examination is performed since some locations may be examined for the first time by the specified techniques. In instances where locations at the time of the examination fail to meet the >90% coverage requirement, the process outlined in 10 CFR 50.55a will be followed.

Per footnote 3 of Table 1 of Code Case N-716, when the required examination volume or area cannot be examined due to interference by another component or part geometry, limited examinations shall be evaluated for acceptability. Acceptance of limited examinations or volumes shall not invalidate the results of the change-in-risk evaluation (paragraph 5 of Code Case N-716). The change in risk evaluation of Code Case N-716 is consistent with previous RI-ISI applications and meets RG 1.174 change-in-risk acceptance criteria. Areas with acceptable limited examinations, and their bases, shall be documented.

Consistent with previously approved RI-ISI submittals, MPS2 will calculate coverage and use additional examinations or techniques in the same manner it has for traditional Section XI examinations. Experience has shown this process to be weld-specific (e.g., joint configuration). As such, the effect on risk, if any, will not be known until that time. Relief requests will be submitted per the guidance of 10 CFR 50.55a(g)(5)(iv) within one (1) year after the end of the interval.

3.4 Risk Impact Assessment

The RIS_B Program development was conducted in accordance with RG 1.174 and 1.178 and the requirements of Code Case N-716, and the conservative risk impact assessment performed demonstrates the RIS-B Program does not have an adverse effect on safety.

This evaluation categorized segments as HSS or LSS in accordance with Code Case N-716, and then determined what inspection changes are proposed for each system. The changes include adjusting the number and location of inspections and in many cases improving the effectiveness of the inspection to account for the findings of the RIS_B degradation mechanism assessment. For example, examinations of locations subject to thermal fatigue will be conducted on an expanded volume and will be focused to enhance the probability of detection (POD) during the inspection process.

3.4.1 Quantitative Analysis

MPS2 has conducted a risk impact analysis per the requirements of Section 5 of Code Case N-716. The analysis estimates the net change in risk due to the positive and negative influences of adding and removing locations from the inspection program. This analysis was performed to ensure that the change in risk of implementing the RIS_B Program meets the requirements of RG 1.174 and 1.178. Code Case N-716 criterion requires that the cumulative change in CDF and LERF be less than 1E-07 and 1E-08 per year per system, respectively. Code Case N-716 also requires that the cumulative increase in overall CDF and LERF be less than 1E-06 and 1E-07 per year, respectively, for the implementation of the RIS_B Program.

In accordance with Code Case N-716, bounding estimates for pipe failure frequency, conditional core damage probability (CCDP), and conditional large early release probability (CLERP) were used to simplify the risk impact assessment calculations. Welds susceptible to FAC were assumed to be managed by the plant FAC Augmented Inspection Program. HSS welds susceptible to an identified degradation mechanism besides FAC were assigned the failure frequency of 2E-07/yr. HSS welds that had no identified degradation mechanisms were assigned a failure frequency 1E-08/yr. LSS welds were conservatively assigned a failure frequency of 2E-07/yr. This approach is consistent with Table 3 of Code Case N-716.

With respect to assigning failure potential for LSS piping, the criteria are defined by Table 3 of Code Case N-716. That is, those locations identified as susceptible to FAC (or another mechanism and also susceptible to water hammer) are assigned a high failure potential. Those locations susceptible to thermal fatigue, erosion-cavitation, corrosion or stress corrosion cracking are assigned to a medium failure potential, and those locations that are identified as not susceptible to degradation are assigned a low failure potential.

Bounding CCDP and CLERP values were conservatively used for all welds in the RIS_B Program for the risk impact assessment. Use of these bounding values simplifies the risk impact assessment, and demonstrates that the use of the RIS_B Program does not have an adverse affect on safety. Applying bounding CCDPs and CLERPs to all welds is conservative in all cases where the RIS_B Program does not increase the number of inspections over the ASME Section XI ISI Program. For MPS2, the number of inspections increases in the Chemical Volume and Control System (CVCS). Therefore, using bounding CCDP and CLERP values would over-estimate the reduction in risk due to the increased number of inspections for the CVCS system. To avoid this non-conservatism by crediting too much risk reduction from the additional inspections, the total risk impact of the RIS_B Program is considered without any risk reduction credited from the CVCS system. There is also a segment of Fire Protection (FP) piping included in the program that was not inspected under Section XI. This increased number of inspections would also result in a risk reduction. However, the piping segment is connected by mechanical joints instead of welds. Because the change in risk is evaluated on a per weld basis, no risk reduction from the FP system was credited.

The CCDP and CLERP values used to assess risk impact were estimated based on the potential consequences of pipe break locations under consideration. For CDF, the most limiting case considered was a large break loss of coolant accident (LOCA). The CCDP for a large break LOCA was estimated to be 5.91E-3/yr for MPS2. For LERF, the most limiting case considered was an interfacing systems LOCA (ISLOCA). The CCDP and CLERP for an ISLOCA were estimated using the probability of a check valve to fail open and allow the ISLOCA to take place, since it is assumed core damage and large early release will take place once in the event of an ISLOCA. The resulting CLERP for an ISLOCA was estimated to be 1.97E-5/yr. These bounding values were conservatively applied to all welds in the RIS_B Program for the risk impact assessment.

The likelihood of pressure boundary failure (PBF) is determined by the presence of different degradation mechanisms and the rank is based on the relative failure probability. The basic likelihood of PBF for a piping location with no degradation mechanism present is given as x_o and is expected to have a value less than 1E-08/yr. Piping locations identified as medium failure potential have a likelihood of $20x_o$. These PBF likelihoods are consistent with References 9 and 14 of EPRI TR-112657. In addition, the analysis was performed both with and without taking credit for enhanced inspection effectiveness due to an increased POD from application of the RIS_B approach.

Table 3.4 in this attachment presents a summary of the proposed RIS_B Program for the Fourth 10-year ISI Interval Program versus the MPS2 Third 10-year Interval Program which was based on ASME Section XI 1989 Edition with No Addenda. The presence of FAC was adjusted in the quantitative analysis by excluding their impact on the failure potential rank. The exclusion of the impact of FAC on the failure potential rank and, therefore, in the determination of the change in risk is appropriate because FAC is a degradation mechanism managed by separate, independent plant augmented inspection programs. The RIS_B Program credits and relies upon the plant augmented inspection programs to manage these degradation mechanisms. The plant FAC Program will continue to determine where and when examinations are performed. Hence, since the number of FAC examination locations remains the same "before" and "after" and no delta exists, there is no need to include the impact of FAC in the performance of the risk impact analysis.

As described above, the total risk impact of implementing the RIS_B Program is considered without any credit taken from risk reduction resulting from an increased number of inspections in the CVCS and FP systems, or from an increased POD. The total change in CDF and LERF is 6.34E-08 and 2.11E-10 events per year, respectively. These conservative estimates for the change in risk are still below the limits described in Code Case N-716.

As indicated in the following table, this evaluation has demonstrated that unacceptable risk impacts will not occur from implementation of the RIS_B Program, and satisfies the acceptance criteria of RG 1.174 and Code Case N-716. Negative numbers indicate a decrease in risk associated with a particular system under the RIS_B Program. Only welds susceptible to TT or TASCS include improved POD credit. Systems with no welds identified with TT or TASCS have the same risk impact with and without POD credit.

MPS2 Risk Impact Assessment Summary

System	With POD Credit		Without POD Credit	
	Delta CDF	Delta LERF	Delta CDF	Delta LERF
SDC – Shutdown Cooling	8.87E-11	2.96E-13	8.87E-11	2.96E-13
FW – Main Feedwater	1.77E-10	5.91E-13	1.77E-10	5.91E-13
MS – Main Steam	8.87E-09	2.96E-11	8.87E-09	2.96E-11
SI – Safety Injection	4.65E-08	1.55E-10	5.34E-08	1.78E-10
CVCS – Chemical and Volume Control System	0.00E+00*	0.00E+00*	0.00E+00*	0.00E+00*
RC – Reactor Coolant	-6.12E-09	-2.04E-11	-9.16E-10	-3.05E-12
RBCCW – Reactor Building Closed Cooling Water	5.91E-10	1.97E-12	5.91E-10	1.97E-12
FP – Fire Protection	0.00E+00*	0.00E+00*	0.00E+00*	0.00E+00*
CP – Containment Purge	1.18E-09	3.94E-12	1.18E-09	3.94E-12
CHP – Containment Hydrogen Purge	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Total	**5.13E-08**	**1.71E-10**	**6.34E-08**	**2.11E-10**

*No credit is taken for systems where the number of inspections increased with N-716, as noted above

3.4.2 Defense-in-Depth

The intent of the inspections mandated by ASME Section XI for piping welds is to identify conditions such as flaws or indications that may be precursors to leaks or ruptures in a system's pressure boundary. Currently, the process for selecting inspection locations is based upon structural discontinuity and stress analysis results. As depicted in ASME White Paper 92-01-01 Rev. 1, *Evaluation of Inservice Inspection Requirements for Class 1, Category B-J Pressure Retaining Welds*, this method has been ineffective in identifying leaks or failures. EPRI TR-112657 and Code Case N-716 provide a more robust selection process founded on actual service experience with nuclear plant piping failure data.

This process has two key independent ingredients; one, a determination of each location's susceptibility to degradation and, two, an independent assessment of the consequence of the piping failure. These two ingredients assure defense-in-depth is maintained. By evaluating a location's susceptibility to degradation, the likelihood of finding flaws or indications that may be precursors to leak or ruptures is increased. A generic assessment of high-consequence sites has been determined by Code Case N-716 supplemented by plant-specific evaluations, thereby requiring a minimum threshold of inspection for important piping whose failure would result in a LOCA. Finally, Code Case N-716 requires that any plant-specific piping with a contribution to CDF of greater than 1E-06/yr or 1E-07/yr for LERF be included in the scope of the application.

One pipe run was determined to contribute to a CDF greater than 1E-06/yr (i.e., 2.27E-06/yr CDF value) due to a flooding concern in an electrical room and was designated HSS. This non-class pipe is a portion of the fire protection system that runs through the East DC Switchgear Room. After performing the degradation mechanism analysis, localized corrosion (LC) in the form of micro-biologically induced corrosion (MIC) and pitting (PIT) was determined a concern for pipe failure. The ten mechanical connections on this run of pipe were assigned reference numbers, with MIC and PIT noted as degradation mechanisms. Twenty-five percent (three) of the component areas were selected for examination over the interval.

All locations within the Class 1, 2, and 3 pressure boundaries will continue to be pressure tested in accordance with the Code, regardless of its safety significance.

3.5 Implementation and Monitoring

Upon approval of the RIS_B Program, procedures that comply with the guidelines described in EPRI TR-112657 and Code Case N-716 will be prepared to implement and monitor the program as needed. The new program will be implemented in the fourth 10-year ISI interval. No changes to the Technical Specifications or Updated Final Safety Analysis Report are necessary for program implementation.

The applicable aspects of the ASME Code not affected by this change will be retained, such as inspection methods, acceptance guidelines, pressure testing, corrective measures, documentation requirements, and quality control requirements. Existing ASME Section XI Program implementing procedures will be retained and modified to address the RIS_B process, as appropriate.

The monitoring and corrective action program will contain the following elements:

A. Identify
B. Characterize
C. (1) Evaluate, determine the cause and extent of condition identified
   (2) Evaluate, develop a corrective action plan(s)
D. Decide
E. Implement
F. Monitor
G. Trend

The RIS_B Program is a living program requiring feedback of new relevant information to ensure the appropriate identification of HSS piping locations. As a minimum, this review will be conducted on an

ASME period basis. In addition, significant changes may require more frequent adjustment as directed by NRC Bulletin or GL requirements, or by industry and plant-specific feedback.

For preservice examinations, MPS2 will follow the rules contained in Section 3.0 of Code Case N-716. Welds classified HSS require preservice inspection. The examination volumes, techniques, and procedures shall be in accordance with Table 1 of the Code Case. Welds classified as LSS do not require preservice inspection.

4. PROPOSED ISI PROGRAM PLAN CHANGE

A comparison between the RIS_B Program and the ASME Section XI 1989 Edition with No Addenda Program requirements for B-F, B-J, C-F-1, C-F-2 and in-scope (all HSS) piping at MPS2 is provided in Table 5.1 in this attachment. The numbers reflect the MPS2 ISI Program for the third 10-year ISI interval based on the 1989 Edition Code of Record. During the third 10-year interval, MPS2 performed additional exams on LSS thin walled C-F-1 pipe that would have been exempt from Code requirements: Forty-eight (48) SI LSS welds and four (4) CVCS LSS welds. These numbers are included in Table 5.1 in this attachment and were included in the delta risk calculation comparison as a conservative measure.

MPS2 intends to start implementing the RIS_B Program during the first period of the fourth inspection interval. The fourth 10-year ISI interval will implement 100% of the inspection locations selected for examinations per the RIS_B Program. Examinations shall be performed such that the period percentage requirements of ASME Section XI are met.

5. REFERENCES/DOCUMENTATION

5.1 EPRI TR-112657, *Revised Risk-Informed Inservice Inspection Evaluation Procedure*, Rev. B-A, December 1999

5.2 EPRI TR-1018427, *Nondestructive Evaluation: Probabilistic Risk Assessment Technical Adequacy Guidance for Risk-Informed In-Service Inspection Programs,* December 2008

5.3 ASME Code Case N-716, *Alternative Piping Classification and Examination Requirements, Section XI Division 1,* approved April 19, 2006

5.4 Regulatory Guide 1.174, *An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis,* Rev. 1, November 2002

5.5 Regulatory Guide 1.178, *An Approach for Plant-Specific Risk-Informed Decisionmaking Inservice Inspection of Piping,* Rev. 1, September 2003

5.6 Letters from P. J. O'Regan (EPRI) to Dr. B. Sheron (USNRC) dated February 28, 2001 and March 28, 2001, *"Extension of Risk-Informed Inservice Inspection Methodology"*

5.7 USNRC Safety Evaluation for Grand Gulf Nuclear Station Unit 1, Request for Alternative GG-ISI-002-Implement Risk-Informed ISI Based on ASME Code Case N-716, dated September 21, 2007 [ML072430005]

5.8 USNRC Safety Evaluation for DC Cook Nuclear Plant, Units 1 and 2, Risk-Informed Safety-Based ISI Program for Class 1 and 2 Piping Welds, dated September 28, 2007 [ML072620553]

5.9 ASME/ANS RA-S-2008, "Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications" and its 2009 addendum (ASME/ANS RA-Sa-2009)

5.10 Regulatory Guide 1.200, An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities, Rev. 2, March 2009

5.11 Not used

5.12 MPS2 Relief Request RR-04-07 "Examination Criteria for Weld Overlays", Letter Serial No. 10-323, July 29, 2010

5.13 ASME Boiler and Pressure Vessel Code Section XI, Rules for Inservice Inspection of Nuclear Power Plant Components

Supporting Dominion Documentation

5.14 ETE-ISI-2011-0001 Rev. 1, "Millstone Unit 2 4th Interval Code Case N-716 Risk Informed Submittal", February 2011

5.15 SM-1632 Rev. 1, "Millstone Unit 2 Code Case N-716 Internal Flooding Analysis and Risk Impact Assessment", February 2011

5.16 "MPS2 PRA Model Notebook Appendix A.1 – Internal Events Model Self Assessment", Rev. 2, February 2011

5.17 ER-AA-ISI-RI-100, Dominion Risk Informed Program

5.18 ER-AA-ISI-RI-101, The Dominion Risk Informed Period Update Process

5.19 ER-AA-ISI-101, Dominion Inservice Inspection Program Preparation and Change Control Process

Table 1.1
PRA Quality Gap Analysis: MPS2

SR	Gap	Impact on RI-ISI Program
AS-A10	Better documentation of the operator actions and system impacts is required in the AS.1 notebook.	Not Significant. This is judged to be a documentation consideration only and does not affect the technical adequacy of the PRA model.
AS-A4	All operator actions required for each key safety function modeled should be identified, described, and listed. As part of the 2009 model update, the event trees were revised to address the SRs not met. The AS.1 notebook was also revised to better document the event trees and to address this SR self assessment comment. An operator actions section has been added to each section of the AS.1 notebook that lists the operator actions modeled for each event tree. However, the table of operator actions is left blank for this revision of the AS.1 notebook.	Not Significant. This is judged to be a documentation consideration only and does not affect the technical adequacy of the PRA model.
AS-A7	Operator action to throttle auxiliary feedwater (AFW) after power restoration following a station black out (SBO) is assumed successful. No justification is provided for omitting this sequence.	Not Significant. Only initiating events related to internal flooding or failure of the primary pressure boundary are quantified in this application. This gap relates only to a station blackout scenario event, which does not affect this application.
AS-B2	The modeling of the availability of the steam supply to the turbine-driven auxiliary feedwater (TDAFW) pump correctly accounts for steam generator tube rupture (SGTR), steam line breaks (SLBs) and main feed line breaks (MFLBs). However, the AS.1 notebook does not include a discussion on this change.	Not Significant. This is judged to be a documentation consideration only and does not affect the technical adequacy of the PRA model.
AS-B3	Better documentation of the operator actions and system impacts is required in the AS.1 notebook.	Not Significant. This is judged to be a documentation consideration only and does not affect the technical adequacy of the PRA model.
AS-B5a, [AS-B6]	A discussion of how plant and system configurations are applied in the PRA models is required in the AS.1 notebook.	Not Significant. This is judged to be a documentation consideration only and does not affect the technical adequacy of the PRA model.
AS-C2	Better documentation of the operator actions, system impacts, and initiating event (IE) to event tree (ET) correlations is required in the AS.1 notebook.	Not Significant. This is judged to be a documentation consideration only and does not affect the technical adequacy of the PRA model.
DA-C10	Review of the procedures would only be necessary if the demand and run time data were estimated based on the number of times a procedure is performed. During the 2009 model update, this data was obtained based on real plant data obtained from the station logs and the plant computer. Therefore, review of the procedures is not necessary, but documentation needs to be updated to indicate this.	Not Significant. This is judged to be a documentation consideration only and does not affect the technical adequacy of the PRA model.
DA-C15 (DA-C16)	There were no plant specific loss of offsite power (LOOP) events for MPS2 for the update period. Therefore, no plant-specific recovery times are available, but this is not currently documented in the MPS2 HR.3 notebook.	Not Significant. This is judged to be a documentation consideration only and does not affect the technical adequacy of the PRA model.

Table 1.1 PRA Quality Gap Analysis: MPS2		
SR	**Gap**	**Impact on RI-ISI Program**
HR-G3	The Human Reliability Analysis (HRA) Calculator was used for the quantification of the human error probabilities (HEPs). The Calculator includes the performance shaping factors (PSFs) listed in this SR. However, the documentation of why different PSFs are applicable or selected is still lacking.	Not Significant. This is judged to be a documentation consideration only and does not affect the technical adequacy of the PRA model.
IE-A6, [IE-A8]	Interviews of plant personnel must be conducted to determine if MPS2 initiating events have been overlooked.	Not Significant. Additional initiating events, if identified, would not be considered in the RI-ISI program. The program considers only internal flooding events and consequences of HSS weld failures, which are all accounted for.
IE-C1b, [IE-C3]	Some HEPs are used for both initiating event logic and post-initiator logic. Separate initiating events should be developed. The modeling of recovery actions in other initiator fault trees uses the same HEP for post-initiator failures. This SR will remain as Not Met until separate initiator recovery actions are developed.	Not significant. The limiting conditions are used to develop HEPs where the same action may be performed under different conditions for initiating event logic and post-initiator logic. Current model results are conservative, so meeting this SR would not impact the RI-ISI program.
IFPP-A3	Documentation in notebook IF.1 should include discussion of why multi-unit flood areas (and scenarios) are not relevant for MPS2.	Not Significant. This is judged to be a documentation consideration only and does not affect the technical adequacy of the PRA model.
IFSO-A5	Documentation in the IF.1 and IF.2 notebooks needs to be enhanced to identify the pressures and temperatures of flood sources.	Not Significant. This is judged to be a documentation consideration only and does not affect the technical adequacy of the PRA model.
LE-A1	Containment isolation does require some support systems so it should be in the plant damage state (PDS) tree using bridge trees. This supporting requirement should remain open, but a sensitivity removing valves that require support systems to close or remain closed could show it not to be applicable for a given application.	Not significant. Contributions to LERF from flooding scenarios were very low, so a containment isolation sensitivity would not cause any additional flooding scenarios to be considered HSS. The bounding LERF case used for the change in risk evaluation was an ISLOCA. The ISLOCA CLERP is based on a check valve failure probability instead of model quantification results, and would also not be affected by the sensitivity described.
LE-C2a, [LE-C2]	In the 2009 model, two mitigations were examined. The first, feeding a ruptured steam generator, was not implemented due to uncertainty about whether the allowed time, 22 minutes, is sufficient. The second, depressurizing the RCS, was not modeled in the Level 2 model due to time constraints. Although these need to be resolved before this SR can be closed out, the current model is conservative.	Not significant. Current model results are conservative, so meeting this SR would not impact the RI-ISI program.
LE-C2b, [LE-C3]	In the 2009 model the sequences have not been reviewed for repair possibilities. Since repair has not been credited, the current model is conservative.	Not significant. Current model results are conservative, so meeting this SR would not impact the RI-ISI program.
LE-C5, [LE-C6]	The containment isolation does require some support systems so it should be in the PDS tree using bridge trees. This supporting requirement should remain open, but a sensitivity removing valves that require support systems to close or remain closed could show it not to be applicable for a	Not significant. Contributions to LERF from flooding scenarios were very low, so a containment isolation sensitivity would not cause any additional flooding scenarios to be considered HSS. The bounding LERF case used for the change in risk evaluation was an ISLOCA. The ISLOCA CLERP is based on a check valve

Table 1.1 PRA Quality Gap Analysis: MPS2		
SR	**Gap**	**Impact on RI-ISI Program**
	given application. This SR remains Not Met.	failure probability instead of model quantification results, and would also not be affected by the sensitivity described.
LE-C6, [LE-C7]	The 2009 update includes HEPs for feeding the steam generators and depressurizing the RCS, but the case for depressurizing the steam generators could not be verified to be achievable in the time allowed. The 2009 update also includes a detailed assessment of containment isolation, but HEPs still need to be generated. Current results are conservative for depressurization.	Not significant. Current model results are conservative, so meeting this SR would not impact the RI-ISI program.
LE-C8a, [LE-C9]	In the 2009 model, LE.3 notebook discusses survivability of Containment Air Recirculation (CAR) fans, but there should be justification for other equipment. Although the SR cannot be closed out at this point, there is potential to justify other components.	Not Significant. This is judged to be a documentation consideration only and does not affect the technical adequacy of the PRA model.
LE-C8b, [LE-C10]	In the 2009 model the sequences have not been reviewed for potential LERF reductions. The current model is conservative.	Not significant. Current model results are conservative, so meeting this SR would not impact the RI-ISI program.
LE-C9b, [LE-C12]	Since there are no LERF sequences with containment failure, a review of significant accident progression sequences for possible additional credit to sequences after containment failure makes no difference. But before closing out this SR a statement should be added to LE.2 or QU.2 saying so.	Not Significant. This is judged to be a documentation consideration only and does not affect the technical adequacy of the PRA model.
LE-D1b, [LE-D2]	EQE calculation file 52204-R-002 should be brought under Dominion documentation control, and the applicable portions of the report summarized in the LE notebook. This was not done, but it does not relate to LERF because in all cases LERF is due to either bypass or failure to isolate rather than due to containment failure.	Not Significant. This is judged to be a documentation consideration only and does not affect the technical adequacy of the PRA model.
LE-D6, [LE-D7]	Although containment isolation was reevaluated, there are two weaknesses. First, since isolation requires support systems, the availability of these systems should be looked at. Second, there are some HEPs that are missing. These issues should be resolved before this SR is closed out, but sensitivities could show that LERF results are not significantly affected.	Not significant. Contributions to LERF from flooding scenarios were very low, so a containment isolation sensitivity would not cause any additional flooding scenarios to be considered HSS. The bounding LERF case used for the change in risk evaluation was an ISLOCA. The ISLOCA CLERP is based on a check valve failure probability instead of model quantification results, and would also not be affected by the sensitivity described.
LE-F1a, [LE-F1]	Since the 2009 update does not document the PDS frequencies, the PDS contribution to LERF is not computed. Not meeting this SR has no effect on the overall LERF.	Not Significant. This is judged to be a documentation consideration only and does not affect the technical adequacy of the PRA model.
LE-F1b, [LE-F2]	Expand the cutset descriptions to include a discussion of reasonableness to document that conservative (or non-conservative) assumptions have not skewed the LERF results.	Not Significant. This is judged to be a documentation consideration only and does not affect the technical adequacy of the PRA model.

Table 1.1
PRA Quality Gap Analysis: MPS2

SR	Gap	Impact on RI-ISI Program
LE-G2	The LE notebooks do not document the containment capacity analysis, which references an EQE document that could not be found, or the Level 2 fault tree.	Not Significant. This is judged to be a documentation consideration only and does not affect the technical adequacy of the PRA model.
LE-G3	Since the 2009 update does not document the PDS frequencies, the PDS contribution to LERF is not computed. Not meeting this SR has no effect on the overall LERF.	Not Significant. This is judged to be a documentation consideration only and does not affect the technical adequacy of the PRA model.
LE-G4	Document key assumptions and key sources of uncertainty associated with the LERF analysis, including results and important insights from sensitivity studies.	Not significant. Contributions to LERF from flooding scenarios were very low, so a sensitivity study to examine uncertainty would not cause any additional flooding scenarios to be considered HSS. The bounding LERF case used for the change in risk evaluation was an ISLOCA. The ISLOCA CLERP is based on a check valve failure probability instead of model quantification results, and would also not be affected by the sensitivity described.
LE-G5	Limitations in the LERF analysis that would impact applications have not been identified.	Not Significant. This is judged to be a documentation consideration only and does not affect the technical adequacy of the PRA model. RI-ISI is not considered an application affected by limitations in the LEF model.
QU-B5	Section 3.11 of the SY.3 notebooks documents the breaks in the system fault trees. However, the list of circular logic breaks was not included in the notebook revisions for this model update.	Not Significant. This is judged to be a documentation consideration only and does not affect the technical adequacy of the PRA model.
QU-E3	The parametric uncertainty analysis has yet to be completed. It will be documented in the QU.3 notebook.	Not significant. There were no flooding scenarios that were close to the threshold for inclusion in HSS scope and would potentially become HSS with an uncertainty analysis. Likewise, the change in risk results were not close to the threshold of the acceptance criteria, so inclusion of an uncertainty analysis would not affect the acceptability of implementing the RI-ISI program in as an alternative to the Section XI ISI program.
QU-E4	The Dominion fleet approach for addressing this SR is to assemble the sources of uncertainty and assumptions from each of the notebooks that are characterized as possible sources of uncertainty into a QU.4 notebook. Once this notebook is developed, then it will be referenced and used for specific applications to identify the sources of uncertainty that impact the application. Currently, there is no QU.4 notebook for the MPS2 model.	Not significant. There were no flooding scenarios that were close to the threshold for inclusion in HSS scope and would potentially become HSS with an uncertainty analysis. Likewise, the change in risk results were not close to the threshold of the acceptance criteria, so inclusion of an uncertainty analysis would not affect the acceptability of implementing the RI-ISI program in as an alternative to the Section XI ISI program.
SC-B5	The SC.1 notebook needs to be enhanced to include a specific comparison of MPS2 success criteria to other similar plants and note/explain any significant differences, and to determine whether the references to the Calvert Cliffs Interim Reliability Evaluation Program are still appropriate.	Not Significant. This is judged to be a documentation consideration only and does not affect the technical adequacy of the PRA model.
SY-A19, [SY-A21]	A room cooling matrix has not been included in the notebook to show which areas have room heatup calculations completed.	Not Significant. This is judged to be a documentation consideration only and does not affect the technical adequacy of the PRA model.

Table 1.1 PRA Quality Gap Analysis: MPS2		
SR	**Gap**	**Impact on RI-ISI Program**
SY-A20, [SY-A22]	No new room heatup calculations were performed as part of the latest model update. Components in rooms requiring ventilation are assumed to fail upon loss of ventilation. A room cooling matrix should be added to the SY.1 notebook similar to the other Dominion model notebooks.	Not Significant. The modeling for ventilation is conservative because equipment is assumed to fail without ventilation in all cases where a room heatup calculation was not available. Because current model results are conservative, meeting this SR would not impact the RI-ISI program.
SY-A4	While the individual plant examination (IPE) documentation and conversations with the PRA engineers indicate that these tasks were performed, no documentation exists (walkdown sheets, system engineer interviews) to support this supposition. During the 2009 model update, the system notebooks were updated to include additional information on plant walkdowns and plant personnel interviews. The walkdowns and interviews have not been completed at this time. This SR will remain Not Met until documentation is complete.	Not Significant. This is judged to be a documentation consideration only and does not affect the technical adequacy of the PRA model.
SY-A6	The system notebooks meet this SR as is. However, this SR would be improved with simplified schematics included in each system notebook to illustrate the system boundaries.	Not Significant. This is judged to be a documentation consideration only and does not affect the technical adequacy of the PRA model.
SY-B12, [SY-B11]	The Instrument Air (IA) system does credit air bottles or accumulators for when IA is failed. However, there is no documentation of their capacity.	Not Significant. This is judged to be a documentation consideration only and does not affect the technical adequacy of the PRA model.
SY-B6	Room heatup calculations have been completed for many of the rooms and ventilation failures added as appropriate. However, it needs to be confirmed that all areas where PRA credited equipment is located have been evaluated for room cooling dependencies. Need to document this in a room cooling matrix in SY.1 similar to the other Dominion model notebooks.	Not Significant. This is judged to be a documentation consideration only and does not affect the technical adequacy of the PRA model.
SY-B7	Room heatup calculations have been completed for many of the rooms and ventilation failures added as appropriate. However, it needs to be confirmed that all areas where PRA credited equipment is located have been evaluated for room cooling dependencies. Need to document this in a room cooling matrix in SY.1 similar to the other Dominion model notebooks.	Not Significant. This is judged to be a documentation consideration only and does not affect the technical adequacy of the PRA model.
SY-C2	System schematics in the model notebooks have not been updated.	Not Significant. This is judged to be a documentation consideration only and does not affect the technical adequacy of the PRA model.

Table 3.1
N-716 Safety Significance Determination: MPS2

System Description	Weld Count	N-716 Safety Significance Determination					Safety Significance	
		RCPB	SDC	PWR: FW	BER	>1E-6^CDF >1E-7^LERF	High	Low
CVCS – Chemical and Volume Control System	79	✓					✓	
	51							✓
FW - Main Feedwater	39			✓			✓	
RC - Reactor Coolant	314	✓					✓	
SDC - Shutdown Cooling	20		✓				✓	
SI - Safety Injection *	162	✓	✓				✓	
	926							✓
FP - Fire Protection	10					✓	✓	
CHP - Containment Hydrogen Purge	11							✓
CP - Containment Purge	14							✓
MS - Main Steam	109							✓
RBCCW - Reactor Building Closed Cooling Water	64							✓
SUMMARY RESULTS FOR ALL SYSTEMS	533	✓					✓	
	39			✓			✓	
	42		✓				✓	
	10					✓	✓	
	1175							✓
TOTALS	**1799**						**624**	**1175**

*HSS SI welds: Twenty-two (22) welds in SDC
140 welds in RCPB

Table 3.2
Failure Potential Assessment Summary: MPS2

System(1)(2)	Thermal Fatigue(3)		Stress Corrosion Cracking(4)				Localized Corrosion(5)			Flow Sensitive(6)	
	TASCS	TT	IGSCC	TGSCC	ECSCC	PWSCC	MIC	PIT	CC	E-C	FAC
CVCS		✓									
FW											✓
RC	✓	✓				✓					
SDC						✓					
SI		✓	✓								
FP							✓	✓			

Notes
(1) Systems are described in Table 3.1.
(2) A degradation mechanism assessment was performed only on high safety significant piping segments.
(3) Thermal Fatigue: TASCS Thermal Stratification, Cycling, Striping, TT Thermal Transient
(4) Stress Corrosion Cracking: IGSCC Intergranular, TGSCC Transgranular, ECSCC External Chloride, PWSCC Primary Water
(5) Localized Corrosion: MIC Micro-Biologically-Induced , PIT Pitting, CC Crevice
(6) Flow Sensitive: E-C Erosion Cavitation, FAC Flow-Accelerated Corrosion

Table 3.3
N-716 Element Selections: MPS2

System[1]	Selections	HSS	DMs[2]	RCPB	RCPB^IFIV(3)	RCPB^OC(4)	BER
CVCS	**Required**	8 of 79	TT 10 of 41	8 of 79	8	n/a	n/a
	Made	12	TT 12	12	12	n/a	n/a
FW	**Required**	4 of 39	n/a	n/a	n/a	n/a	n/a
	Made	4	n/a	n/a	n/a	n/a	n/a
RC	**Required**	32 of 314	TT 2 of 6 TASCS 2 of 5 TASCS, TT 4 of 16 PWSCC 2 of 5 TT, PWSCC 1 of 2	32 of 314	23	n/a	n/a
	Made	34	TT 3 TASCS 2 TASCS, TT 4 PWSCC 2 TT, PWSCC 1	34	33	n/a	n/a
SDC	**Required**	2 of 20	PWSCC 1 of 1	2 of 20	2	n/a	n/a
	Made	2	PWSCC 1	2	1	n/a	n/a
SI	**Required**	17 of 162	TT 8 OF 29 IGSCC 3 OF 9	14 OF 140	10	n/a	n/a
	Made	18	TT 8 IGSCC 3	4	3	n/a	n/a
FP	**Required**	1 of 10	MIC, PIT 3 of 10	0 of 0	n/a	n/a	n/a
	Made	3	MIC, PIT 3	0	n/a	n/a	n/a
TOTAL	**Required**	**63 of 624**	**31 of 124**	**56 of 553**	**43**	n/a	n/a
	Made	**73**	**39**	**63**	**54**	n/a	n/a

Notes
(1) Systems are described in Table 3.1.
(2) No more than 10% of HSS piping welds are required to be selected for examination. Degradation Mechanism (DM) selections may be reduced to meet this requirement.
(3) Reactor Coolant Pressure Boundary between first isolation valve and the reactor. For RPCB^IFIV 2/3 requirement is for total of RPCB^IFIV and is not required to be met per system.
(4) Reactor Coolant Pressure Boundary outside containment, typically does not apply to PWRs.

Table 3.4 Risk Impact Analysis Results: MPS2

System[1]	Safety Significance	Failure Potential		Inspections			CDF Impact		LERF Impact	
		DMs	Rank[2]	SXI[3]	RIS_B	Delta	w/ POD	w/o POD	w/ POD	w/o POD
CVCS	High	TT	Medium	0	12	12	-1.28E-08	-4.26E-11	-7.09E-09	-2.36E-11
CVCS	High	None	Low	0	0	0	0.00E+00	0.00E+00	0.00E+00	0.00E+00
CVCS	Low	None	Medium	4	0	-4	2.36E-09	7.88E-12	2.36E-09	7.88E-12
Total							0.00E+00[4]	0.00E+00[4]	0.00E+00[4]	0.00E+00[4]
FW	High	None	Low	10	4	-6	1.77E-10	5.91E-13	1.77E-10	5.91E-13
Total							1.77E-10	5.91E-13	1.77E-10	5.91E-13
RC	High	PWSCC	Medium	4	2	-2	1.18E-09	3.94E-12	1.18E-09	3.94E-12
RC	High	TT	Medium	1	3	2	-2.84E-09	-9.46E-12	-1.18E-09	-3.94E-12
RC	High	TASCS	Medium	0	2	2	-2.13E-09	-7.09E-12	-1.18E-09	-3.94E-12
RC	High	TT,PWSCC	Medium	2	1	-1	5.91E-10	1.97E-12	5.91E-10	1.97E-12
RC	High	TASCS,TT	Medium	3	4	1	-3.19E-09	-1.06E-11	-5.91E-10	-1.97E-12
RC	High	None	Low	31	22	-9	2.66E-10	8.87E-13	2.66E-10	8.87E-13
Total							-6.12E-09	-2.04E-11	-9.16E-10	-3.05E-12
SDC	High	PWSCC	Medium	1	1	0	0.00E+00	0.00E+00	0.00E+00	0.00E+00
SDC	High	None	Low	4	1	-3	8.87E-11	2.96E-13	8.87E-11	2.96E-13
Total							8.87E-11	2.96E-13	8.87E-11	2.96E-13
SI	High	TT	Medium	13	8	-5	-3.90E-09	-1.30E-11	2.96E-09	9.85E-12
SI	High	IGSCC	Medium	2	3	1	-5.91E-10	-1.97E-12	-5.91E-10	-1.97E-12
SI	High	None	Low	33	7	-26	7.68E-10	2.56E-12	7.68E-10	2.56E-12
SI	Low	None	Medium	85	0	-85	5.02E-08	1.67E-10	5.02E-08	1.67E-10
Total							4.65E-08	1.55E-10	5.34E-08	1.78E-10
FP	High	MIC,PIT	Medium	0	3	3	-3.00E-07	-3.69E-11	-3.00E-07	-3.69E-11
Total							0.00E+00[4]	0.00E+00[4]	0.00E+00[4]	0.00E+00[4]
CHP	Low	None	Medium	0	0	0	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Total							0.00E+00	0.00E+00	0.00E+00	0.00E+00
CP	Low	None	Medium	2	0	-2	1.18E-09	3.94E-12	1.18E-09	3.94E-12
Total							1.18E-09	3.94E-12	1.18E-09	3.94E-12
MS	Low	None	Medium	15	0	-15	8.87E-09	2.96E-11	8.87E-09	2.96E-11
Total							8.87E-09	2.96E-11	8.87E-09	2.96E-11
RBCCW	Low	None	Medium	1	0	-1	5.91E-10	1.97E-12	5.91E-10	1.97E-12
Total							5.91E-10	1.97E-12	5.91E-10	1.97E-12
Program Total							5.13E-08	1.71E-10	6.34E-08	2.11E-10

Notes

(1) Systems are described in Tables 3.1.

(2) The failure potential rank for high safety significant (HSS) locations is assigned as "High", "Medium" or "Low" dependent upon potential susceptibly to the various types of degradation mechanisms. [Note: LSS locations were conservatively assumed to be a rank of Medium (i.e., "Assume Medium").]

(3) Only those ASME Section XI Code inspection locations that received a volumetric examination in addition to a surface examination are included in the count. Inspection locations previously subjected to a surface examination only were not considered in accordance with Section 3.7.1 of EPRI TR-112657.

(4) No credit was taken for increased number of inspections in the CVCS and FP Systems.

Table 5.1
Inspection Location Selection Comparison Between ASME Section XI Code and Code Case N-716: MPS2[4]

System[1]	Safety Significance		Failure Potential		Code Category	Weld Count	Section XI		Code Case N-716	
	High	Low	DMs	Rank[2]			Vol/Sur	Sur Only	RIS_B	Other[3]
CVCS	✓		TT	Medium	B-J	39	0	11	10	–
					B-F	2	0	2	2	
CVCS	✓		None	Low	B-J	38	0	8	0	–
CVCS		✓	None	Medium	C-F-1	51	4[6]	0	0	–
FW	✓		None	Low	C-F-2	39	10	0	4	–
										–
RC	✓		PWSCC	Medium	B-F	5	4	1	2	
RC	✓		TT	Medium	B-J	6	1	0	3	–
RC	✓		TASCS	Medium	B-J	5	0	0	2	–
RC	✓		TT, PWSCC	Medium	B-F	2	2	0	1	
RC	✓		TASCS, TT	Medium	B-J	16	3	0	4	–
RC	✓		None	Low	B-J	266	23	49	19	
					B-F	14	8	0	3	
										–
SDC	✓		PWSCC	Medium	B-F	1	1	0	1	–
SDC	✓		None	Low	B-J	19	4	0	1	–
SI	✓		TT	Medium	B-J	25	10		7	–
					B-F	4	3		1	
SI	✓		IGSCC	Medium	B-J	9	2		3	
SI	✓		None	Low	B-J	102	24		4	–
					C-F-1	22	9		3	
SI		✓	None	Medium	C-F-1	926	85[7]	34	0	–

Table 5.1
Inspection Location Selection Comparison Between ASME Section XI Code and Code Case N-716: MPS2(4)

System(1)	Safety Significance		Failure Potential		Code Category	Weld Count	Section XI		Code Case N-716	
	High	Low	DMs	Rank(2)			Vol/Sur	Sur Only	RIS_B	Other(3)
FP	✓		MIC, PIT	Medium	NC	10(5)	0	0	3	–
CHP		✓	None	Medium	C-F-2	11	0	0	0	–
CP		✓	None	Medium	C-F-2	14	2	0	0	–
MS		✓	None	Medium	C-F-2	109	15	3	0	–
RBCCW		✓	None	Medium	C-F-2	64	1	0	0	–

Notes

(1) Systems are described in Table 3.1.

(2) The failure potential rank for HSS locations is assigned as "High", "Medium" or "Low" dependent upon potential susceptibly to the various types of degradation mechanisms. [Note: LSS locations were conservatively assumed to be a rank of Medium (i.e., "Assume Medium").]

(3) The column labeled "Other" is generally used to identify plant Augmented Inspection Program locations credited per Section 4 of Code Case N-716. Code Case N-716 allows the existing plant Augmented Inspection Program for IGSCC (Categories B through G) to be credited toward the 10% requirement. MPS did not take credit for any augmented inspections in meeting the 10% sampling requirement. The "Other" column has been retained in this table solely for uniformity purposes with other RIS_B application template submittals.

(4) The break location was not designated in this application because the risk impact assessment was performed with bounding estimates of weld failure consequences in place of weld specific consequences.

(5) These locations on FP pipe are mechanical connections versus welds.

(6) These four (4) exams were actually not required per Section XI Code requirements. They were exempt based on pipe wall thickness.

(7) Forty-eight (48) of these exams were actually not required per Section XI Code requirements. They were exempt based on pipe wall thickness.