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U. S. Nuclear Regulatory Commission
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Washington, DC 20555

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Docket No.	50-423
License No.	NPF-49

DOMINION NUCLEAR CONNECTICUT, INC.
MILLSTONE POWER STATION UNIT 3
10 CFR 50.55a RELIEF REQUEST IR-3-14, UPDATE TO THE RISK-INFORMED INSERVICE INSPECTION PROGRAM FOR THE THIRD 10-YEAR INSPECTION INTERVAL

Pursuant to the provisions of 10 CFR 50.55a(a)(3)(i), Dominion Nuclear Connecticut, Inc. (DNC) requests approval for Relief Request IR-3-14 for the update and continued implementation of a Risk-Informed Inservice Inspection (RI-ISI) program for ASME Class 1 piping at Millstone Power Station Unit 3.

The proposed update to the ISI program, for Class 1 piping only, is based on the risk-informed methodology described in Westinghouse Owners Group WCAP-14572, Revision 1-NP-A, "Westinghouse Owners Group Application of Risk-Informed Methods to Piping Inservice Inspection Topical Report." DNC submitted a RI-ISI program using this methodology for the second 10-year interval ISI program in a letter dated July 25, 2000, and supplemented by letters dated November 16, 2000 and September 26, 2001. The request was approved by NRC Safety Evaluation Report (SER) dated March 12, 2002 (TAC No. MA9740). Relief Request IR-3-14 is submitted for the third 10-year interval of the piping inspection program, which began on April 23, 2009, and uses the same methodology previously approved for the second 10-year interval.

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

Sincerely,

Leslie N. Hartz
Vice President – Nuclear Support Services

Attachments:

1. Relief Request IR-3-14, Update to the Risk-Informed ISI Piping for the Third 10-Year Inspection Interval
2. PRA Technical Adequacy for RI-ISI Program

Commitments made in this letter:

1. None

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ATTACHMENT 1

RELIEF REQUEST IR-3-14, UPDATE TO THE RISK-INFORMED INSERVICE INSPECTION PROGRAM FOR THE THIRD 10-YEAR INSPECTION INTERVAL

DOMINION NUCLEAR CONNECTICUT, INC.
MILLSTONE POWER STATION UNIT 3

Proposed Alternative

In Accordance with 10 CFR 50.55a(a)(3)(i)

--Alternative Provides Acceptable Level of Quality and Safety--

1. ASME Code Components Affected

ASME Code Class:	Code Class 1
References:	ASME Section XI, Table IWB-2500-1
Examination Category:	B-F, B-J
Item Number:	N/A
Description:	Risk-Informed ISI Program, Third 10-Year Interval Update
Components:	ASME Class 1 Piping and Vessel Nozzle Safe Ends

2. Applicable Code Edition and Addenda

ASME Section XI, 2004 Edition (no Addenda)

3. Applicable Code Requirement

Pursuant to 10 CFR 50.55a (a)(3)(i), DNC requests to update the Millstone Unit 3 (MPS3) Inservice Inspection (ISI) Program, for Class 1 piping only, continuing the use of a Risk-Informed Inservice Inspection Program (RI-ISI) as an alternative to the current requirements of Class 1 examination Categories B-F and B-J as specified in Table IWB-2500-1 of the 2004 Edition with no Addenda of ASME Section XI.

The proposed revision to the ISI program, for Class 1 piping only, is based on the risk-informed methodology described in Westinghouse Owners Group WCAP-14572, Revision 1-NP-A, "Westinghouse Owners Group Application of Risk-Informed Methods to Piping Inservice Inspection Topical Report." (References 1-2). A similar revision to the second interval 10-year ISI program was submitted by letter dated July 25, 2000, and supplemented by letters dated November 16, 2000 and September 26, 2001. The request was approved by NRC Safety Evaluation Report (SER) dated March 12, 2002 (TAC No. MA9740). This request for an alternative to the current requirements for the third 10-year interval uses the same methodology previously approved for the second 10-year interval.

MPS3 has entered the third 10-year interval as defined by the Code for Inspection Program B, which began on April 23, 2009 and ends on April 22, 2019.

4. Reason for Request

The objective of this submittal is to update and continue the implementation of the RI-ISI program for the third 10-year interval to the ISI program plan, for Class 1 piping only. The risk-informed methodology used in this submittal is described in Westinghouse Owners Group WCAP-14572, Revision 1-NP-A, 'Westinghouse Owners Group Application of Risk-Informed Methods to Piping Inservice Inspection Topical Report", (referred to as 'WCAP-14572, A-version" for the remainder of this document).

5. Proposed Alternative and Basis for Use

ASME Section XI Class 1 Categories B-F and B-J currently contain the requirements for examining (via non-destructive examination (NDE)) Class 1 piping components. This current program submittal is limited to ASME Class 1 piping, including piping currently exempt from requirements. The alternative RI-ISI program for piping is described in WCAP-14572, A-version (References 1-2). DNC proposes to substitute the Class 1 RI-ISI for the ASME Section XI requirements, Category B-F and B-J examination program on piping. Other non-related portions of the ASME Section XI Code will be unaffected. A summary of the proposed alternative inspection program and comparison with the previously approved program is presented in Table 1 of this attachment. The table was prepared consistent with the recommendations of the Nuclear Energy Institute (NEI) document NEI 04-05, "Living Program Guidance to Maintain Risk-Informed Inservice Inspection Programs for Nuclear Plant Piping Systems," (Reference 3).

Basis for Use

The MPS3 ISI program for the examination of Class 1 piping is in accordance with a risk-informed process submitted in a letter dated September 26, 2000. The NRC approved this request on March 12, 2002 (TAC No. MA9740). The authorization to implement the alternative was limited to the second 10-year interval and thus requires a re-submittal for the third 10-year interval. Since 2002, most U.S. nuclear power plants have implemented similar Risk-Informed Inservice Inspection programs, with similar requirements for review and update. As a result, a task force was formed by NEI to formulate consistent guidance for maintaining these programs. The task force included representatives from reactor operating companies, ASME committees, Electric Power Research Institute (EPRI), and Westinghouse. The result of this effort is guidance document NEI 04-05, "Living Program Guidance To Maintain Risk-Informed Inservice Inspection Programs For Nuclear Plant Piping

Systems", published April, 2004. While not specifically approved by the NRC, the NRC staff reviewed the document as it was being developed and provided comments.

In accordance with the guidance provided by NEI 04-05, a periodic evaluation and update was performed in conjunction with the end of the second 10-year ISI inspection interval at MPS3. The updated program resulting from this review is the subject of this relief request.

In accordance with the guidance provided by NEI 04-05, Table 1 is included in this attachment identifying the revised number of High Safety Significant (HSS) segments and selected inspection elements for the updated program. For comparison, it also lists similar information for the previous RI-ISI program. The total number of HSS piping segments increased by six, from 62 to 68. One additional segment was added from the reactor coolant system (RCS) and five were added from the high pressure safety injection (SIH) system.

These segments are added based on the results of the revised risk analysis and the expert panel evaluation. Significant changes to the risk analysis include (1) updates to reliability and initiating event frequencies, (2) Probabilistic Risk Assessment (PRA) model updates to meet PRA standards described in RG-1.200 Rev. 1 (Revision 1 is currently accepted for use and is being used, although Revision 2 will become effective in 2010), (3) consideration of internal flooding and spurious SI, and (4) consideration of human factors (HEP). A more detailed description of PRA updates and quality is provided in Attachment 2. As a result of the changes to the risk analysis, the Large Early Release Frequency (LERF) results for large and medium LOCA are now negligible because the Reactor Coolant System (RCS) depressurizes quickly and has less opportunity to cause a steam generator tube rupture. The reduced LERF results in an overall reranking of segment risk measures that required reconsideration by the expert panel.

Table 1 also shows a net reduction of two in the number of inspection elements (welds and base material volumes) requiring volumetric or surface examinations, from 79 to 77. The significant factors affecting the number of selected welds include the following.

- All 14 large bore A82/A182 nickel alloy welds were selected for volumetric examination. These include eight on the reactor vessel nozzles and six on the pressurizer. The six pressurizer nozzle welds have all been mitigated with a full structural overlay using Primary Water Stress Corrosion Cracking (PWSCC) resistant weld material. Three of these welds are located on Low Safety Significant (LSS) segments. The eight steam generator primary piping nozzles were confirmed to not contain A82/A182 welds.

- A formal review of potential branch line thermal cyclic fatigue was conducted in accordance with the EPRI Materials Reliability Program (MRP) document MRP-146 and its supplement MRP-146S. As a result, some drain lines were determined to be non-susceptible to thermal fatigue, allowing examinations on those lines to be excluded. Some drain lines were retained as susceptible to the thermal fatigue and examination elements for those lines were retained in accordance with the MRP guidance. The net change represents a significant decrease in the number of examination elements on drain lines.

- The six additional HSS segments resulted in adding five welds for volumetric examination and one (reactor head vent line) for visual examination.

The number of selected welds in each HSS segment was confirmed to be adequate by an update to the original Perdue analysis.

All VT-2 examinations in the Risk Informed inspection plan are performed during the system pressure test each refueling outage.

An updated "Change in Risk Evaluation" was performed for the revised RI-ISI inspection program described above, and the risk from the revised program continues to remain lower when compared to the last deterministic ASME Section XI inspection program.

6. Duration of Proposed Alternative

This proposal requests approval to implement a RI- ISI program for the third 10-year Inservice Inspection interval, which started on April 23, 2009, and is scheduled to be completed on April 22, 2019.

7. Precedents

The original request for relief for MPS3 to implement a RI-ISI program in the Second Interval was submitted July 25, 2000 and granted by letter dated March 12, 2002 (ADAMS Accession No. ML020570312).

References

1. WCAP-14572, Revision 1-NP-A, Westinghouse Owners Group Application of Risk-Informed Methods to Piping Inservice Inspection Topical Report", February 1999.

2. WCAP-14572 Revision 1-NP-A, Supplement 1, "Westinghouse Structural Reliability Risk Assessment (SRRA) Model for Piping Risk-Informed Inservice Inspection", February 1999.

3. NEI 04-05, "Living Program Guidance to Maintain Risk-Informed Inservice Inspection Programs for Nuclear Plant Piping Systems", published April 2004, Nuclear Energy Institute, Washington, DC.

4. Materials Reliability Program: Management of Thermal Fatigue in Normally Stagnant Non-Isolable Reactor Coolant System Branch Lines (MRP-146). EPRI, Palo Alto, CA: 2005, 1011955.

5. Materials Reliability Program: Management of Thermal Fatigue in Normally Stagnant Non-Isolable Reactor Coolant System Branch Lines – Supplemental Guidance (MRP-146S). EPRI, Palo Alto, CA: 2009, 1018330.

Table 1
MPS3 STRUCTURAL ELEMENT SELECTION
RESULTS AND COMPARISON TO ASME SECTION XI
1989 EDITION REQUIREMENTS

System	High Safety Significant Segments (No. of HSS in Augmented Program / Total No. of Segments in Aumented Program)	Degradation Mechanism(s)	Safety Class	ASME Code Exam Category	Total Weld Count (Welds requiring Volumetric (Vol) and Surface (Sur))		ASME XI Program Examinations		2rd Interval RI-ISI		3rd Interval RI-ISI[a]	
					Vol & Sur	Sur only	Vol & Sur	Sur only	SES Matrix Region	Number of Exam Locations	SES Matrix Region	Number of Exam Locations
CHS	4 (0 / 0)	VF	Class 1	B-J	0	68	0	6	1, 2	4+4[b]	1, 2	4+4[b]
RCS	57 (0 / 0)	SCC,TF	Class 1	B-F	9	5	22	0	2	14	2	17+3[c]
		TF	Class 1	B-J	271	454	72	167	2	59	2	46+1[b]
RHS	2 (0 / 0)	None	Class 1	B-J	17	0	10	0	2	2	2	2
SIH	5 (0 / 0)	None	Class 1	B-J	12	227	0	32		0	2	5
SIL	0 (0 / 0)	None	Class 1	B-J	106	45	18	1		0		0
TOTAL	68 (0 / 0)	SCC	Class 1	B-F	9	5	22	0	14 NDE	14 NDE		20 NDE
		TF,VF		B-J	406	794	100	206		65 NDE + 4 VISUAL		57 NDE + 5 VISUAL
			Total		415	799	122	206		79 NDE + 4 VISUAL		77 NDE + 5 VISUAL

Table 1
MPS3 STRUCTURAL ELEMENT SELECTION
RESULTS AND COMPARISON TO ASME SECTION XI
1989 EDITION REQUIREMENTS

Summary: ASME Section XI selected a total of 328 welds while the proposed RI-ISI program selects a total of 77 welds (plus 5 visual exams), which results in a 76% reduction.

Degradation Mechanisms: SCC – Stress Corrosion Cracking; TF – Thermal Fatigue;
VF – Vibratory Fatigue. "X/X" indicates combination of mechanisms.

Systems:
CHS – Chemical and Volume Control System
RCS – Reactor Coolant System
RHS – Residual Heat Removal System
SIH –High Head Safety Injection System
SIL –Low Head Safety Injection System

Notes for Table 1

a. System pressure test requirements and VT-2 visual examinations shall continue in all ASME Code Class systems.
b. VT-2 or VE visual examinations as applicable at one location within segment.
c. Examinations on LSS segments having mitigated PWSCC locations.

ATTACHMENT 2

PRA TECHNICAL ADEQUACY FOR RISK-INFORMED INSERVICE INSPECTION PROGRAM

DOMINION NUCLEAR CONNECTICUT, INC.
MILLSTONE POWER STATION UNIT 3

PRA Technical Adequacy for RI-ISI Program

Introduction

Dominion employs a structured approach to establishing and maintaining the technical adequacy and plant fidelity of the probabilistic risk assessment (PRA) models for all Dominion nuclear generating sites. This approach includes both a proceduralized PRA maintenance and update process, and the use of self-assessments and independent reviews.

MAINTENANCE OF PRA

The Millstone Power Station (MPS3) PRA model and documentation have been maintained as a living program, and the PRA is routinely updated approximately every 3 years to reflect the current plant configuration and to reflect the accumulation of additional plant operating history and component failure data.

There are several procedures and GaRDs (Guidance and Reference Documentation) that govern Dominion's PRA program. Procedure NF-AA-PRA-101 controls the maintenance and use of the PRA documentation and the associated NF-AA-PRA Procedures and GaRDs. These documents define the process to delineate the types of calculations to be performed, the computer codes and models used, and the process (or technique) by which each calculation is performed.

The NF-AA-PRA series of GaRDs and Procedures provides a detailed description of the methodology necessary to

- Perform probabilistic risk assessments for the Dominion Nuclear Fleet, including Kewaunee, Millstone, North Anna and Surry Power Stations
- Create and maintain products to support licensing and plant operation concerns for the Dominion Nuclear Fleet
- Provide PRA model configuration control
- Create and maintain configuration risk evaluation tools for the Dominion Nuclear Fleet

The purpose of the NF-AA-PRA GaRDs and Procedures is to provide information and guidelines for performing probabilistic risk assessments. Nevertheless, non-routine risk assessments are often unique, requiring departure from these guidelines and information in order to correctly perform and meet the risk assessment objectives. Such departure must be evaluated and documented in accordance with applicable regulations and Dominion policies.

The previous MPS3 RI-ISI program was submitted in July 25, 2000 and obtained a Safety Evaluation Report (SER) from the NRC on March 12, 2002. The PRA model Revision M3999927, dated October 1999 was used to evaluate the consequences of pipe ruptures for the previous RI-ISI submittal. A summary of the MPS3 PRA history since October 1999 is as follows:

MPS3 Model Change History	
Date	**Model Change**
6/00	Incorporated loss of offsite power and offsite power restoration calculations
9/02	NUREG/CR-5750 used as source of general initiating event frequencies Incorporated some of the peer review level A and B Facts & Observations (F&Os).
2004	Added main feedwater and condensate systems to the secondary cooling function.
2005	Mitigating Systems Performance Index (MSPI) Model Update completed a) plant specific data b) reliability: 01/01/2000-12/31/2004 c) unavailability: January, 2002 to December, 2004 d) initiating events: 1990 to 12/31/2004 e) addressed remaining A and B level peer review F&Os
2006	2005 Mod A Model (M305 mod A) a) revised the cooling dependency for the Charging pump oil cooling system (CCE). Service Water (SW) is not required to cool Charging pumps if auxiliary building temperatures remain below 90F.
2006	2005 Mod B and C Model (M305 mod B & C) a) added internal flooding in mod B b) revised junction box flood damage logic in internal flooding model in mod C
2007	2005 Mod D Model (M305 mod D) in support of the Stretch Power Uprate a) added hot leg recirculation to large loss of coolant accident (LLOCA) b) added new pre-initiator Human Error Probabilities (HEPs) c) updated Human Reliability Analysis (HRA) using latest methodology [Cause Based Decision Tree (CBDT), Human Cognitive Reliability Correlation (HCR), Technique for Human Error Rate Prediction (THERP)] d) updated interfacing system Loss of Coolant Accident (LOCA) e) updated level 2 f) various other changes (e.g. replaced logic that assumed LOCA, Steam Generator Tube Rupture (SGTR) or Steamline Break (SLB) occurs in one Reactor Coolant System (RCS) loop or steam generator).

MPS3 Model Change History	
Date	**Model Change**
2008	M308A Model a) Data Update, including Generic and Plant-Specific Data for Failure Rates, Unavailability and Initiating Events b) Included Additional Loss of Single AC and DC buses to Support Regulatory Guide 1.200, rev. 1 c) Included Modeling of the Operator Action to Swap from Demineralized Water Storage Tank (DWST) to Condensate Storage Tank (CST), including Equipment Failures and DWST Refill via Fire Protection d) Upgraded Modeling of Safety Injection (SI)/Containment Depressurization Actuation (CDA)/Main Steam Isolation (MSI) Actuation Signals e) Crediting Room Heating, Ventilation and Air Conditioning (HVAC) as an additional CCE Heat Sink f) Upgraded Modeling of Any of the Four Steam Generators as the Faulted Steam Generator g) Subsumed Instrument Tube LOCA (ITLOCA) into Small-Small LOCA (SSLOCA) h) Included Modeling of the Spurious Safety Injection Signal Initiating Event i) Upgraded Component Boundaries to be Consistent with Generic Data as required by RG 1.200, rev. 1 j) Included Additional Flooding Initiating Events associated with test and maintenance activities k) Added RWST Low-Low Level Signal to Start the RSS Pumps Coincident with CDA signal

An administratively controlled process is used to maintain configuration control of the MPS3 PRA models, data, and software. In addition to model control, administrative mechanisms are in place to assure that plant modifications, procedure changes, system operation changes, and industry operating experiences (OEs) are appropriately screened, dispositioned, and scheduled for incorporation into the model in a timely manner. These processes help assure that the MPS3 PRA reflects the as-built, as-operated plant within the limitations of the PRA methodology.

This process involves a periodic review and update cycle to model any changes in the plant design or operation. Plant hardware and procedure changes are reviewed on an approximate semi-annually or more frequent basis to determine if they impact the PRA and if a PRA modeling and/or documentation change is warranted. These reviews are documented, and if any PRA changes are warranted, they are added to the PRA Configuration Control (PRACC) database for PRA implementation tracking.

Control of the PRA software is maintained in accordance with the Dominion software control guidelines. New versions of the PRA codes are released for use by the code manager after the version has been verified and processed in accordance with the configuration control program. The PRA models of each of the Dominion sites are also maintained by model managers who are responsible for control of the models used for the various PRA applications.

COMPREHENSIVE CRITICAL REVIEWS

The MPS3 PRA model has benefited from the following comprehensive technical reviews:

- MPS3 PRA Self-Assessment
- NEI PRA Peer Review

MPS3 PRA Self-Assessment

A self-assessment or independent review of the MPS3 PRA against the ASME PRA Standard was performed by Dominion with the support of a contracting company, MARACOR, in late 2007 using guidance provided in NRC Regulatory Guide RG 1.200, Revision 1, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results from Risk-Informed Activities". This self-assessment was documented and used as a planning guide for the M308A model updates.

Many of the Supporting Requirements (SRs) identified in the self-assessment as not meeting capability category II have been incorporated into the MPS3 M308A model of record. The improvements made to the model involved documenting sources of uncertainty/assumptions, systematic process for establishing common cause failure (CCF) groups, updating Modular Accident Analysis Program (MAAP) runs and improving success criteria documentation. In the M308A model update, nearly all of the remaining SRs were addressed by further upgrades to the model documentation as well as improvements to the model. Of the 321 SRs, the MPS3 PRA currently does not meet 47 SRs. Thirty-nine of the forty-seven "not met" requirements pertain to various documentation issues. Nineteen of the documentation issues are associated with completion of QU.4, Assumptions and Limitations. There are eight modeling issues associated with room cooling, incorporation of Severe Accident Management Guidance (SAMGs), plant-specific alignments, review of Type A inspection procedures and flooding from inadvertent fire protection actuation. The attached Table 1 provides the status of open gap items not meeting Capability Category II of the ASME PRA Standards.

NEI PRA Peer Review

In 1999, the MPS3 internal events PRA received a formal industry PRA peer review. The purpose of the PRA peer review process was to provide a method for establishing the technical quality of a PRA for the spectrum of potential risk informed plant licensing applications for which the PRA may be used. The PRA peer review process used a team composed of industry PRA consultants and utility peers, each with significant expertise in both PRA development and PRA applications. This review team provided both an objective review of the PRA technical elements and a subjective assessment, based on their PRA experience, regarding the acceptability of the PRA elements.

This review was performed using the Westinghouse Owners Group (WOG) implementation of the industry PRA peer review methodology as defined in NEI 00-02, "PRA Peer Review Process Guidance." The review team reviewed over 200 attributes of 11 different elements of the PRA. Reviewer questions or comments that could not be answered during the review were documented in Facts & Observations (F&O) forms and were categorized by level of significance as follows:

A – Extremely important, technical adequacy may be impacted
B – Important, but may be deferred to next model update
C – Less important, desirable to maintain model flexibility and consistency with the industry
D – Editorial, minor technical item
S – Strength / Superior Treatment (no follow-up required)

The peer review is documented in the Westinghouse PRA peer review report (Reference 6.15). Subsequent to the peer review, the model has been updated several times and F&Os were addressed during each model update. Currently, there are 6 Category C F&Os still open that do not impact the quality or results of the MPS3 PRA model.

<u>General Conclusion Regarding PRA Capability</u>

The quality of modeling and documentation of the MPS3 PRA model has been demonstrated by the foregoing discussions on the following aspects:

- Maintenance of the PRA
- Comprehensive critical reviews

The MPS3 Level 1 and Level 2 PRAs provide the necessary and sufficient scope and level of detail to allow the calculation of core damage frequency (CDF) and large early release frequency (LERF) changes due to the proposed configuration. As specific risk-informed PRA applications are performed, remaining "not-met" supporting requirements will be reviewed to determine which, if any, would merit application-specific sensitivity studies in the presentation of the application results.

In addition, the MPS3 internal events PRA has been used in support of various regulatory programs and relief requests that have received NRC Safety Evaluation Reports (SERs), further indication of the quality of the MPS3 internal events PRA and suitability for regulatory applications. This list includes:

- MPS3 Individual Plant Examination (IPE) Staff Evaluation Report (SER)
- Risk-Informed EDG 14 day License Amendment addressing AOT technical specification change
- Risk-Informed Inservice Inspection (RI-ISI) SER
- Integrated Leak Rate Test (ILRT) frequency extension SER
- Life Extension (Severe Accident Mitigation Alternative (SAMA)) License Amendment
- Cable Spreading Room Manual Fire Suppression
- Maintenance Rule
- Mitigating Systems Performance Index (MSPI)
- Significance Determination Process evaluations

<u>Assessment of PRA Capability Needed for Risk-Informed Inservice Inspection</u>

In the RI-ISI program at MPS3, the Pressurized Water Reactor Owners Group (PWROG), formerly Westinghouse Owners Group, RI-ISI methodology (Reference 1) is used to define alternative Inservice Inspection requirements. Plant-specific PRA-derived risk significance information is used during the RI-ISI plan development to support the consequence assessment, risk ranking and delta risk evaluation steps.

The importance of PRA consequence results, and therefore the scope of PRA technical capability, is tempered by three processes in the PWROG methodology.

- In the PWROG methodology, two sets of consequences are developed: One based on the operators taking no action to isolate or mitigate the piping failure and the other based on the operators being perfect in taking any credible operator action to isolate or mitigate the piping failure. Based on this, four risk evaluation workbooks are created for CDF and LERF. If the risk metrics from any of these four risk evaluation workbooks are quantitatively high safety significant (HSS), the segment is identified as quantitatively HSS.

- A simplified uncertainty analysis is performed to ensure that no low safety significant segments could move into high safety significance when reasonable variations in the pipe failure and conditional CDF/LERF probabilities are considered.

- The PWROG RI-ISI methodology is a risk-informed process and not a risk-based process. The quantitative results from the risk evaluation along with deterministic insights and other input data are presented to an expert panel in an integrated decision making process. The primary focus of the expert panel is to review all pertinent information and determine the final safety-significance category for each of the piping segments. The expert panel is comprised of plant personnel with a wide breadth and depth of experience as specified in WCAP-14572 (Reference 1). Segments that have been determined to be quantitatively HSS are typically categorized as HSS by the expert panel. The focus of the expert panel is to add segments to the higher classification. The MPS3 expert panel categorized 20 MPS3 segments, which were not quantitatively HSS, as HSS based on deterministic insights, high failure potential and/or high consequences. Additionally, as part of the integrated decision making process, the expert panel considers limitations in the process when categorizing segments as HSS or LSS. This may include PRA model limitations and limitations in modeling the consequences using the PRA model.

The limited manner of PRA involvement in the RI-ISI process is also reflected in the risk-informed license application guidance provided in Regulatory Guide 1.174 (Reference 2).

Section 2.2.6 of Regulatory Guide 1.174 provides the following insight into PRA capability requirements for this type of application:

There are, however, some applications that, because of the nature of the proposed change, have a limited impact on risk, and this is reflected in the impact on the elements of the risk model.

An example is Risk-Informed Inservice Inspection (RI-ISI). In this application, risk significance was used as one criterion for selecting pipe segments to be periodically examined for cracking. During the staff review it became clear that a high level of emphasis on PRA technical acceptability was not necessary. Therefore, the staff review of plant-specific RI-ISI typically will include only a limited scope review of PRA technical acceptability.

In the PWROG RI-ISI process, the PRA model is not used as the basis for the risk evaluation, but instead is used as an input to the risk evaluation process. The vast majority of the piping failure consequences are identified as loss of a system or train of a system. The PRA results are then used as an input to the risk evaluation for the relative ranking of the segments. Table 1.3-1 of the ASME PRA Standard (Reference 3) identifies the bases for PRA capability categories. The bases for Capability Category I for scope and level of detail attributes of the PRA states:

Resolution and specificity sufficient to identify the relative importance of the contributors at the system or train level including associated human actions.

Based on the above, in general, Capability Category I is suitable for PRA quality for a RI-ISI application.

In addition to the above, it is noted that segments and their associated welds determined to be low risk significant are not eliminated from the ISI program on the basis of risk information. For example, the risk significance of a segment may be determined by the expert panel to be low safety significant, resulting in it not being a candidate for inspection. However, it remains in the program and, if in the future the assessment of its ranking changes (either by damage mechanism, PRA risk, or deterministic insight), then it can again become a candidate for inspection.

Conclusion Regarding PRA Capability for Risk-Informed ISI

The MPS3 PRA models continue to be suitable for use in the RI-ISI application. This conclusion is based on:

- the PRA maintenance and update processes in place,
- the PRA technical capability evaluations that have been performed and are being planned, and
- the RI-ISI process considerations, as noted above, that demonstrate the relatively limited reliance of the process on PRA capability.

References

1. WCAP-14572, "Westinghouse Owners Group Application of Risk-Informed Methods to Piping Inservice Inspection Topical Report," Revision 1-NP-A, including: Supplement 1, "Westinghouse Structural Reliability and Risk Assessment (SRRA) Model for Piping Risk-Informed Inservice Inspection," Revision 1-NP-A.
2. U.S. Nuclear Regulatory Commission, *An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis*, Regulatory Guide 1.174, Revision 1, November 2002.
3. American Society of Mechanical Engineers, *Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications*, ASME RA-Sb-2005, New York, New York, December 2005.

Table 1 - Status of Open Gaps to Capability Category II of the ASME PRA Standards

SR	Description of Gap	Current Status/Comment	Importance to RI-ISI
SY-A4	The previous system notebooks describe the systems analysis but do not include any discussion of plant walkdowns or interviews with system engineers or operators. The current notebooks also refer to a previous version of MPS3 PRA documentation for historical purposes. However, this historical version of the PRA also does not include discussion of walkdowns performed and interviews conducted.	**Documentation enhancement:** Dominion PRA staff works closely with Millstone 3 system engineers and operators on nearly a daily basis while supporting the various risk informed programs such as Maintenance Rule, MSPI, RI-ISI, risk informed TS submittals and SDP resolution. The plant scheduling staff along with the Shift Technical Advisors use the PRA for assessing plant risk on a daily basis as required by the Maintenance Rule a(4) program. This kind of interaction with the plant staff provides valuable feedback on the PRA model. Also, the PRA staff has extensive plant knowledge with staff members who were previously system engineers, shift technical advisors, and senior reactor operators. Although formal interviews with the plant staff are not documented to allow closure of this ASME PRA Standard supporting requirement, it is not anticipated that not meeting this requirement will have an impact on the model. Formal interview checklists will be developed and completed in the future. (Ref. PRACC 8321).	Note 1
SY-A19	Several different conditions that can result in system failure are included in the system models. Examples include ventilation for rooms containing electrical equipment, EDGs, and certain pumps; and shedding of major loads to prevent potential EDG failure. Assumptions regarding whether a particular condition will result in system failure are documented in Table 1 of Volume SY.2. These assumptions however do not reference any basis for success given a certain system condition (e.g., the temperature in the RPCCW pump area is assumed to remain below the allowable 90-second limit with 2 charging pumps and RPCCW operating). Other assumptions note that ventilation is currently not modeled for turbine building or service building electrical equipment (including all the non-vital load centers, MCCs, and DC switchboards), nor for TPCCW, instrument air, or service air. The acceptability of these assumptions were to be determined by future analysis.	**Model enhancement:** Ventilation requirements have been documented in SY.3.HV *MP3 room heatup.xlsx*. There are two additional compartments that will require ventilation modeling. These are the normal switchgear and RPCCCW compartments. These HVAC will be included in the next model update (Ref. PRACC 8317).	Note 2
SY-B6	In a few instances support systems are assumed not to be required without an engineering analysis referenced to determine the systems are not needed, for example (from the assumptions in Table 1 of Volume SY.2: -- The temperature in the RPCCW pump area is assumed to remain below the allowable 90-second limit with 2 charging pumps and RPCCW operating -- It is assumed RHR seal cooling is not needed during large LOCA due to the short mission time -- It is assumed that ventilation is not needed for turbine building or service building electrical equipment (including all the non-vital load centers, MCCs, and DC switchboards), or for TPCCW, instrument air or service air, the acceptability of these assumptions was to be determined by future analysis.	**Model enhancement:** Room cooling dependencies have been updated and are documented in SY.3.HV HV *MP3 room heatup.xlsx*. Further information on room cooling dependencies: 1) HVAC dependency for RPCCW/CH pumps will be included in the next model update (See SY-A19). 2) RHR is required for LLOCA injection, then the pumps shut off. Recirculation is performed by the RSS pumps. SC.1 states "Sump recirculation must be established, in the worst case, within 33 minutes of SI initiation." Therefore, the pumps seals should not fail after 33 minutes of injecting cold RWST water. 3) Normal switchgear ventilation will be included in the next model update. (Ref. PRACC 8326).	Note 2

SR	Description of Gap	Current Status/Comment	Importance to RI-ISI
SY-B8	The current system notebooks reference separate fire, internal flood, and seismic analysis notebooks for discussion of spatial and environmental hazards. The current system notebooks do not include any discussion of plant walkdowns. Only the internal flooding notebooks are available for review (which is discussed with those SRs for the IF element), however the IF notebooks and model have not yet been approved and incorporated in the model of record. The current notebooks also refer to a previous version of MPS3 PRA documentation for historical purposes. However, this historical version of the PRA also does not include discussion of spatial and environmental hazards nor walkdowns performed.	**Documentation enhancement:** The internal flooding model has been updated and incorporated into the model as part of the mod C 2005 model. This SR remains open until the documentation issues relating to the spatial dependency are completely documented. The internal flooding results are not expected to be impacted by these open items. (Ref PRACC 8328)	Note 1
SY-C2	The MPS3 systems analysis is documented in an updated series of system notebooks that have a common content and format, and are structured to correspond to the requirements of this standard. In addition, the system dependencies, assumptions and success criteria are documented in a set of tables common to all systems. These notebooks include much of the information listed in this SR. However, in some areas the content of the system notebooks could be enhanced to better document compliance with particular SRs. In addition not all system notebooks have yet been approved.	**Documentation enhancement:** System notebooks have gone through a major enhancement process to meet the documentation items in the supporting requirement. Though there are place holders for all the above information, the notebooks may not contain documentation of all SR elements. For example, completed check lists for walkdowns. (Ref. PRACC 8329).	Note 1

SR	Description of Gap	Current Status/Comment	Importance to RI-ISI
HR-A1	The Type A HRE identification employed an initial review of PRA system P&IDs to identify components potentially susceptible to Type A realignment errors, followed by a review of surveillance procedures to identify those that require realignment. The documented methodology, as summarized below, doesn't appear to discuss realignment of equipment outside its normal operational or standby status for activities other than surveillance. Also, the system notebooks summarize the Type A HRE identification findings, but the number of components listed in the notebooks appears to be too few to represent a complete inventory of manual valves for PRA systems. The following steps were performed: 1) Systematic reviews of the plant systems were performed to identify misalignment HFEs. These reviews are summarized in each of the system model notebooks, although the supporting documentation is not referenced. 2) For mis-alignment of components, motor-operated valves and air-operated valves were screened out since there is indication in the control room of their position. They also generally receive an actuation signal during an accident and they can be realigned from the control room. Pumps, fans, compressors and other components that have controls and indication in the control room were also screened out since the misposition would be identified. The scope of mis-alignments was therefore limited to manual valves that are realigned during surveillance tests. 3) The P&ID drawings were reviewed to identify manual valves that could be realigned to disable a flow path in the model. 4) Surveillance procedures were searched to identify any that change the position of the manual valves from their normal position. If the position of the valve is not changed in any surveillance procedure, then a pre-initiator HFE is not required.	**Model enhancement:** As part of the 2007 revision of the model (2005 mod D model), numerous pre-initiator HEPs were added to the model after a systematic review of each system was performed. Each system notebook contains the reviews and the resulting pre-initiator HEPs that were determined to be required. The HR.1 notebook documents the quantification of the HEPs. No screening values were used. A review of inspection procedures is required to meet the additional RG 1.200 requirements. Ensure Type A assessment realignment of equipment outside its normal operational or standby status for activities other than surveillance, i.e., maintenance and inspection activities, or document why these activities do not pose credible Type A failures This will be addressed in the next model update (Ref. PRACC 8238)	Note 2
HR-B1	The basis for screening individual activities are summarized in the system notebooks. These rules for screening activities were generally found to be appropriate, however no consolidated list of rules is provided in the HR.1 notebook. Also, activities were screened in some instances on the basis that they have no or insignificant impact on PRA results. However, the quantitative basis for screening are not discussed in terms of CDF/LERF impact. For example, the potential to misalign multiple trains SG feed lines was evaluated in the MPS3 PRA Model Notebook, Feedwater System Analysis Model, Volume SY.3.FW. Since the success criteria for steam generator cooling is 2 of 4 SGs, a single misalignment HEP that causes a loss of SGC would require for failing align 3 or more valves, which is considered negligible (less than 4E-6). It is preferable to express quantitative arguments in terms of impact to CDF/LERF.	**Documentation enhancement:** GARDs 2051 and 2052 provide guidance on identifying and quantifying Type A HEPs. MPS3 Model notebook HR.1 needs to be updated in accordance with the aforementioned GARDs. (Ref. PRACC 8241).	Note 1

SR	Description of Gap	Current Status/Comment	Importance to RI-ISI
HR-B2	The documentation of the Type A HRE identification process does not indicate whether any activities that could simultaneously have an impact on multiple trains of a redundant system or diverse systems were identified and screened.	**Documentation enhancement:** The MPS3 model notebooks are developed in accordance with GARD 2051, which states, "No activities that could simultaneously have an impact on multiple trains of a redundant PRA system or diverse PRA systems shall be screened." HR.1 needs to be updated in accordance with the aforementioned GARD. (Ref. PRACC 8242).	Note 1
HR-C2	Potential failure modes considered in the analysis include failure to restore: (a) equipment to the desired standby or operational status, (b) initiation signal or set point for equipment start-up or realignment. However, no documentation was found of considerations of the failure to restore automatic realignment or power. Also, no discussion is provided in MPS3 PRA Model Notebook, Pre-initiator Human Failure Event Analysis, Volume HR.1.of a review for such failure modes as part of the collection of plant-specific or applicable generic operating experience.	**Model enhancement:** Examine and document plant-specific or applicable generic operating experience that leave equipment unavailable for response in accident sequences Include consideration of modes of unavailability resulting from failure restore: 1) automatic realignment or 2) electrical power. The above will be completed in the next model update. (Ref. PRACC 8243)	Note 2
HR-D3	No documentation was found that discusses the quality of written procedures, administrative controls or the quality of the human-machine interface.	**Documentation enhancement:** The MPS3 model notebooks are developed in accordance with GARD 2051, which provides guidance that addresses this SR. Improve the documentation of the quality of written procedures, administrative controls or the quality of the human-machine interface. (Ref. PRACC 8245).	Note 1
HR-E1	The methodology for identifying key human response actions is documented in HRA PRA Manual, PRAM-2E (Rev. 0, February 2005) and complies with this SR. However discussion of the implementation of this methodology for MPS3 is not provided in the HR.1 PRA notebook.	**Documentation enhancement:** The methodology for identifying human response actions is documented in the updated HR.1 notebook, which is developed in accordance with GARD 2051. HR.1 needs to be updated in accordance with the aforementioned GARD. (Ref. PRACC 8246).	Note 1
HR-G6	The PRA notebooks do not document a review of the HFEs and their final HEPs relative to each other to check reasonableness given the scenario context, plant history, procedures, operational practices, and experience.	**Documentation enhancement:** Document a review of the HFEs and their final HEPs relative to each other to confirm their reasonableness given the scenario context, plant history, procedures, operational practices, and experience. (Ref. PRACC 8248)	Note 1
DA-C8	Notebook DA.4 documents the development of the data for the PRAs alignment-specific events. The current approach used for these events meets Capability Category 1, in that the PRA assumes an overall average distribution of system alignments. The estimates used are reasonable. However, this approach does not meet Category 2 requirements.	**Model enhancement:** M308A currently does not meet this SR due to the following assumption: "Similar components should have the same unavailability. Therefore, similar components are grouped together to evaluate their average unavailability." (Ref. PRACC 8234).	Note 2
DA-D4	Notebook DA.2 includes an assessment of the difference between the updated mean values to the original generic (prior) data. An analysis is provided of the possible reasons for the larger variances that were observed is provided. Also, the notebook includes plots of the prior and updated distributions for each event, which would indicate a "single bin histogram" or multimodal condition and other anomalies. However, the documentation does not discuss whether or not a specific review was performed on the data for each of these various tests that are recommended in this SR.	**Documentation enhancement:** GARD 2061 provides guidance to address each of the various tests that are recommended in the SR. Need to review DA.2 to ensure GARD 2061 is met. (Ref. PRACC 8235).	Note 1

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DA-D6	While the alpha factors used for MP3 are based on recent generic estimates (and appear to be appropriate), there is no discussion in the DA.3 notebook to indicate that the alpha factors were checked for consistency with plant operating experience.	**Documentation enhancement:** This SR will be addressed in the next model update. (Ref. PRACC 8236)	Note 1
IF-B2	Three categories of flooding initiating events were evaluated for the potential flood sources identified: tank rupture, system pipe rupture, and maintenance-related events. However, it does not appear that failure of gaskets, expansion joints, fittings, seals or other components were considered. In addition, although a method for maintenance-induced flood events is described in the Flooding Initiating Events Notebook, no such scenarios were explicitly evaluated in the documentation. Inadvertent fire sprinkler actuation also does not appear to be considered.	**Model enhancement:** IF.3, rev.2 has been updated to include maintenance induced flooding events and expansion joints ruptures. However, the piping failure rates are based on EPRI TR-102266, which has been updated to EPRI TR- 1013141, which states, "All piping system pressure boundary failures have been included including failures in pipe base metal, welds, and other metallic pressure boundary components such as valve bodies, heat exchangers, and fittings. However, inadvertent sprinkler actuation is currently (M308A) not addressed or modeled." This will be completed in the next model update (PRACC 8263).	Note 2
IF-C3b	Based on a review of the draft IF PRA notebooks, inter-area propagation flow paths are appropriately identified and account for: 1) potential structural failure due to flooding loads, 2) propagation via penetrations, doors, stairwells, hatchways and HVAC ducts. No summary discussion of the locations of floor drain check valves and considerations for their potential failure was found. The potential for barrier unavailability was also not addressed.	**Documentation enhancement:** Update the propagation analysis and flood models to consider the potential for barrier unavailability. Provide a summary discussion of the locations of floor drain check valves and the considerations for their potential failure. IF.2, rev. 3, does not consider the potential for barrier unavailability. E.g., "...Flood Compartment CSW-3. The room is equipped with a water-tight door that is assumed to remain intact, and thus propagation to other compartments is not postulated." In addition, no mention of floor drain check valves are included in the IF.x notebooks. (PRACC 8265)	Note 1
IF-C4a	No credible multi-unit scenarios exist. 1) MPS2 and MPS3 share in common an electrical switchyard and a station blackout diesel. Internal flooding scenarios are not applicable to the switchyard. The SBO diesel is located in a stand-alone building in the yard that communicates with no other parts of the plant. No credible internal flooding initiating events were identified for the building that houses the SBO diesel. 2) Also, communication exists between the plants via the water treatment facility. More discussion is suggested to describe that no credible flood propagation paths exist via this pathway.	**Documentation enhancement:** Propagation paths Chemical Polishing Facility and MPS3 TB have been addressed. Path between MPS2 and MPS3 has been addressed. However, propagation from the water treatment facility and MPS3 has not been addressed in IF.3, rev. 2. (PRACC 8266)	Note 1
IF-D5	Based on a review of Draft B of PRA Notebook IE.1, "Initiating Events Analysis," the flood-initiating event frequency for each flood scenario group appears to be calculated using the applicable IE supporting requirements (Table 4.5.1-2c of Addendum B). However, the documentation does not indicate that the initiating event frequencies are computed in terms of reactor years.	**Documentation enhancement:** IF.2, rev. 3, states, "Rupture Frequency (per Reactor Critical Year) {Adjustments for Capacity Factor (if applicable) are handled in Notebook IE.2}". However, IE.2, rev 4, does not contain the information necessary to convert from "per reactor critical year" to "reactor year". In fact in M308A, the IF IEs are in "per reactor critical year". This is conservative since the IF IEs have not been multiplied by capacity factor. (PRACC 8269)	Note 1

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QU-D3	The current version (R2) of the QU.2 Model Quantification Results Notebook provides a list of plant features that influence risk, but no comparison of results with similar plants. The previous version of the notebook included a comparison of the CDF contributors with two other Dominion Westinghouse PWRs, some possible reasons for the differences, and the list of plant features that influence risk. In some cases only the difference is noted, and not the cause for the difference.	**Documentation enhancement:** Comparison of plants outside Dominion has not been completed yet. (Ref. PRACC 8293).	Note 1
LE-C2a	The SAMGs have not been discussed in the Level 2 analysis. Generally, this is conservative in not crediting actions, but realistic evaluation may decrease the releases from some sequences. No discussion of Level 2 operator actions could be found.	**Model enhancement:** As part of the 2007 model revision (2005 mod D model), the level II analysis was updated. Revision 0 of the LE.2 model notebook contains the details of the updated level II model. The SAMGs have not been completely incorporated into the MP3 Level 2 analysis, except for credit for initiating LPI after induced hot leg failure. Rev. 3: SAMGs were reviewed to determine which actions can be credited in the MPS3 PRA Level-2 model to reduce Large Early release Frequency (LERF). (Ref. LE.2, Rev 1) Open Issue: Still need to address PRACC 9863, depressurization for SGTR. **Documentation enhancement:** Level 2 operator actions are documented in LE.2, rev. 1, section 2.2.	Note 2
LE-C9b	No review of the dominant LERF sequences for such credit was discussed in the MP3 Level 2 documentation.	**Documentation enhancement:** LE.1, rev. 1, does contain a review of top 100 cutsets, however, no review was performed to credit additional equipment or operator mitigative features. In addition, a review of the non-significant sequences was not documented. (PRACC 8279)	Note 1

SR	Description of Gap	Current Status/Comment	Importance to RI-ISI
IE-D3 AS-C3 SC-C3 SY-C3 HR-I3 DA-E3 IF-E6 IF-F3 QU-B1 QU-E1 QU-E2 QU-E4 QU-F2 QU-F4 QU-F5 LE-F2 LE-F3 LE-G2 LE-G4	These Supporting Requirements are not met and awaiting a revision to QU.4, Rev. 0, *Model Assumptions And Uncertainties*. The purpose of QU.4 is to document modeling assumptions and uncertainties in the MPS3 PRA Model, M308A. The ASME PRA Standard, as modified by Regulatory Guide 1.200 requires the identification and characterization of sources of uncertainty with significant potential to influence PRA results and applications. NUREG-1855 notes that non-parametric sources of uncertainty include "modeling" uncertainties and "scope and level of detail" uncertainties. Modeling uncertainties must be considered in both the base PRA and in specific risk-informed applications.	Sources of uncertainties and assumptions have been identified and documented in the notebook(s). Assessment. Need to revise QU.4 based on the current model, M308A. QU.4 is scheduled to be revised in the second quarter of 2010. (PRACC 8296).	Note 3

Notes

1. This Supporting Requirement is a documentation issue only. The documentation is not expected to affect the RI-ISI program for Millstone Unit 3.
2. Updating the PRA model to meet this Supporting requirement could potentially impact the base PRA model results. However, since the PWROG RI-ISI methodology uses a relative ranking process to assess risk significance, the changes are not expected to affect the RI-ISI program for Millstone Unit 3.
3. This supporting requirement addresses uncertainty in the PRA model. The PWROG RI-ISI methodology includes an application specific uncertainty analysis, which should be sufficient for this application. Additional uncertainties are not expected to affect the RI-ISI program for Millstone Unit 3.