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

Dominion

November 10, 2003

RE: 10 CFR 50.55a(a)(3)(i)
10 CFR 50.55a(a)(3)(ii)

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

Serial No.:	03-462
	B18987
NL&OS/PRW	Rev 0
Docket No.:	50-336
License No.:	DPR-65

DOMINION NUCLEAR CONNECTICUT, INC. (DNC)
MILLSTONE POWER STATION UNIT NO. 2
REQUEST TO IMPLEMENT A RISK-INFORMED INSERVICE INSPECTION PROGRAM PLAN
AS AN ALTERNATIVE TO ASME CODE SECTION XI REQUIREMENTS

Dominion Nuclear Connecticut, Inc. (DNC) requests NRC approval to implement a Risk-Informed Inservice Inspection (RI-ISI) Program as an alternative to the American Society of Mechanical Engineers (ASME) Section XI inservice inspection requirements for Class 1 piping at Millstone Unit No. 2 (MPS 2). Additionally, DNC is requesting NRC approval to allow a pressure test and corresponding Visual, VT-2 examination in lieu of a volumetric examination for socket welds of any size and branch pipe connection welds Nominal Pipe Size (NPS) 2 inches and smaller that will be examined in accordance with the RI-ISI program. Pursuant to 10 CFR 50.55a(a)(3)(i), alternative request RR-89-40 for the proposed implementation of the RI-ISI program Plan is provided in Attachment 1. Pursuant to 10 CFR 50.55a(a)(3)(ii), alternative request RR-89-41 for the proposed use of a Visual, VT-2 examination of socket welds of any size and branch pipe connection welds NPS 2 and smaller is also provided as Attachment 2 for your review and approval. The information contained in this submittal has been reviewed and approved by the Site Operations Review Committee.

The RI-ISI program has been developed in accordance with the Westinghouse Owners Group Topical Report WCAP-14572, Revision 1-NP-A, "Westinghouse Owners Group Application of Risk-Informed Methods to Piping Inservice Inspection Topical Report." Attachment 1 is provided to document and to support the conclusion that the proposed alternative to implement a RI-ISI program for MPS 2 Class 1 piping provides an acceptable level of quality and safety as required by 10 CFR 50.55a(a)(3)(i).

Attachment 2 is provided to document and support the conclusion that a Visual, VT-2, examination is an acceptable alternative to performing a volumetric examination on socket welds of any size and branch pipe connection welds NPS 2 and smaller in accordance with the requirements of the approved Topical Report WCAP-14572, Revision 1-NP-A. Performing the required volumetric examination on these welds has been determined to be a hardship without a compensating increase in the level of quality and safety. Additional supporting documentation is available at the facility for the Nuclear Regulatory Commission (NRC) Staff's review.

A047

DNC requests NRC review and approval of the alternatives addressed in this letter supporting the implementation of the MPS 2 RI-ISI program be provided by September 30, 2004 to support the spring 2005 Refueling Outage (RFO16).

There are no regulatory commitments contained within this letter.

If you should have any questions regarding this submittal, please contact Mr. Paul R. Willoughby at (804) 273-3572.

Very truly yours,

Leslie N. Hartz
Vice President – Nuclear Engineering

Attachment (1)	Millstone Power Station, Unit No. 2 - Alternative Request RR-89-40 To Implement A Risk-Informed Inservice Inspection Program For Class1 Piping
Attachment (2)	Millstone Power Station, Unit No. 2 - Alternative Request RR-89-41 To Use A Visual, VT-2 Examination In Lieu Of A Volumetric Examination Under The Risk-Informed ISI Program For Socket Welds Of Any Size And Branch Pipe Connection Welds NPS 2 And Smaller

cc: U. S. Nuclear Regulatory Commission
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Washington, DC 20555-0001

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Attachment 1

Millstone Power Station, Unit No. 2

Alternative Request RR-89-40 To Implement A Risk-Informed Inservice Inspection Program For Class1 Piping

Millstone Power Station, Unit No. 2

Alternative Request RR-89-40
To Implement A Risk-Informed Inservice Inspection Program For Class1 Piping

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Alternative Request RR-89-40
To Implement a Risk-Informed Inservice Inspection Program for Class1 Piping

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

- Alternative Provides Acceptable Level of Quality and Safety -

1. INTRODUCTION/RELATION TO NRC REGULATORY GUIDE RG-1.174

1.1 Introduction

Dominion Nuclear Connecticut, Inc. (DNC) currently performs inservice inspections (ISI) at Millstone Power Station Unit No. 2 (MPS 2) on piping components using the requirements of the ASME Boiler and Pressure Vessel Code Section XI, 1989 Edition as required by 10CFR50.55a. The unit is currently in the third 10-year inspection interval as defined by the Code in accordance with IWA-2432 "Inspection Program B."

The objective of this submittal is to request NRC approval for a change to the ISI program plan for piping through the use of a risk-informed ISI program. The risk-informed process 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," and WCAP-14572, Revision 1-NP-A, Supplement 1, "Westinghouse Structural Reliability and Risk Assessment (SRRA) Model for Piping Risk-Informed Inservice Inspection," (referred to as "WCAP-14572, A-version" for the remainder of this document). "

As a risk-informed application, this submittal meets the intent and principles of Regulatory Guide 1.174. Further information is provided in Section 3.10 relative to defense-in-depth.

1.2 PRA Quality

The MPS 2 Level 1 and Level 2 probabilistic risk assessment (PRA) model, Version M2021114 dated October 2002 was used to evaluate the consequences of pipe ruptures during operation in Modes 1 and 2. The base core damage frequency (CDF) and base large early release frequency (LERF) from this version of the PRA model are 6.48E-05/yr and 7.26E-07/yr, respectively.

PRA model updates are scheduled at regular intervals not to exceed 36-months between major updates. Minor updates are performed as needed to support risk management activities. The guidance for model maintenance is contained in Dominion Resources Services, Inc. administrative procedures.

The RI-ISI evaluation includes a determination that the PRA model and supporting documentation accurately reflects the current plant configuration and operational practices

consistent with its intended application. The PRA model has been extensively assessed through internal reviews during the PRA model updates and a Combustion Engineering Owner's Group peer review in January 2000. An external consultant review was performed during the latest major update to the MPS 2 model in October 2002.

During the NRC's review of the individual plant examination (IPE), concerns were identified regarding plant specific data and the post-initiator human reliability analysis (HRA). Plant specific data was incorporated in the January 2000 PRA version. The HRA concern was that overly optimistic HRA probabilities and dependencies among multiple actions were not fully considered. The HRA analysis was revised to address these concerns in the June 2000 PRA version. The PRA was further updated for significant plant changes in October 2002 version used in the RI-ISI program.

2. PROPOSED ALTERNATIVE TO ISI PROGRAM

2.1 ASME Section XI

ASME Section XI Categories B-F and B-J currently contain the non-destructive examination (NDE) requirements for examining piping components. This current program is limited to ASME Class 1 piping. The alternative risk-informed inservice inspection (RI-ISI) program for piping is described in WCAP-14572, A-Version. The RI-ISI program will be substituted for the current examination program for piping 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. WCAP-14572, A-Version provides the requirements defining the relationship between the risk-informed examination program and the remaining unaffected portions of ASME Section XI.

2.2 Augmented Programs

There are no augmented inspection programs for MPS 2 Class 1 piping systems.

3. RISK-INFORMED ISI PROCESSES

The processes used to develop the RI-ISI program are consistent with the methodology described in WCAP-14572, A-Version and involve the following steps:

- Scope of Program
- Segment Definition
- Consequence Evaluation
- Failure Assessment
- Risk Evaluation
- Expert Panel Categorization
- Element/NDE Selection
- Implement Program
- Feedback Loop
- Deviations

There are no significant deviations to the process described in WCAP-14572, A-Version except as documented in Attachment (2) by the alternative request RR-89-41 to allow Visual, VT-2 examination in lieu of volumetric examination for socket welds of any size and branch pipe connection welds NPS 2 and smaller. As part of the risk evaluation described in Section 3.5, the uncertainty analysis as described on WCAP -14572, A-Version, page 125 was performed and is now included as part of the base process.

3.1 *Scope of Program*

The scope of this program is limited to Class 1 piping, and includes piping exempt from current NDE requirements. The Reactor Coolant System and portions of other systems which make-up the Class 1 piping included in the risk-informed ISI program are provided in Table 3.1-1.

3.2 Segment Definitions

Once the systems to be included in the program are determined, the piping for these systems is divided into segments.

The number of piping segments defined for the Class 1 piping systems is summarized in Table 3.1-1. The one-line drawings that were used to define the segments are included in the Millstone Calculation File PRA02NQA-03112S2, Rev. 0 including Calculation Change Notice 1. The one-line drawings are based on the as-operated piping and instrumentation diagrams (P&IDs) for the plant as referenced in the above calculation.

3.3 Consequence Evaluation

The consequences of pressure boundary failures are measured in terms of CDF and LERF. The impact on these measures, including direct and indirect effects, was considered and is summarized in Table 3.3-1.

3.4 Failure Assessment

Failure estimates were generated utilizing industry failure history, plant specific failure history and other relevant information. An engineering piping subpanel was established that had access to expertise from ISI, NDE, materials, stress analysis and system engineering. The team was trained in the failure probability assessment methodology and the Westinghouse Structural Reliability and Risk Assessment (SRRA) code, including identification of the capabilities and limitations as described in WCAP-14572, Revision 1-NP-A, Supplement 1. The SRRA code was used to calculate failure probabilities for the failure modes, materials, degradation mechanisms, input variables and uncertainties it was programmed to consider as discussed in the WCAP Supplement 1.

The engineering team assessed industry and plant experience, plant layout, materials, operating conditions and identified the potential failure mechanisms and causes. A

member of the engineering team gathered information from various sources to provide input for the SRRA model and results were discussed and reviewed by the team.

The SRRA code could not be used for all failure mechanisms or piping materials. In these instances, values were determined using alternative means. Generally, the SRRA code was used to determine where the possible ranges of failure probability would fall. For example thermal fatigue in socket welds is not modeled in the SRRA code. However, the code does model thermal fatigue in butt welds. This type of mechanism in butt welds bounds the probability for thermal fatigue in socket welds so that an upper and lower bound failure probability could be established. The team members understanding this bounding information and industry experience determined the appropriateness of the final failure probability results.

Sensitivity studies were performed to aid in determining representative input values when sufficient information was not available. Snubber failure history was also reviewed to identify any potential effects that could increase piping failure probability.

Table 3.4-1 summarizes the failure probability estimates for the dominant potential failure mechanism(s)/combination(s) by system. Table 3.4-1 also describes why the degradation mechanisms could occur at various locations within the system. Full break cases are shown only when pipe whip is of concern.

3.5 Risk Evaluation

Each piping segment within the scope of the program was evaluated to determine its CDF and LERF due to the postulated piping failure. Calculations were also performed with and without operator action.

Once this evaluation was completed, the total pressure boundary CDF and LERF were calculated by summing across the segments for each system.

The uncertainty analysis as described on WCAP -14572, A-Version, page 125 was performed and is now included as part of the base process. The results of these calculations are presented in Table 3.5-1. The CDF due to piping failure without operator action is 1.53E-05/year, and with operator action is 1.53E-05/year. The LERF due to piping failure without operator action is 3.53E-07/year, and with operator action is 3.52E-07/year. To assess safety significance, the risk reduction worth (RRW) and risk achievement worth (RAW) were calculated for each piping segment.

3.6 Expert Panel Categorization

The final safety determination (i.e., high and low safety significance) of each piping segment was made by the expert panel using both probabilistic and deterministic insights. The expert panel was comprised of personnel who have expertise in the following fields: probabilistic safety assessment, inservice examination, nondestructive

examination, stress and material considerations, plant operations, plant and industry maintenance, repair and failure history, system design and operation, and SRRA methods including uncertainty. Members associated with the Maintenance Rule were used to ensure consistency with the other PRA applications. Alternates were used if their expertise and training were sufficient.

The expert panel had the following positions represented by either the permanent or alternate member at all times during an expert panel meeting.

- Probabilistic Risk Assessment (PRA engineer)
- Operations (Senior Reactor Operator)
- Inservice Inspection (ISI)
- Plant & Industry Maintenance, Repair, and Failure History

A minimum of 5 members or alternates filling the above positions constituted a quorum. This core team of panel members was supplemented by other experts, including a metallurgist and piping stress engineer, as required for the piping system under evaluation.

The expert panel chairperson was appointed by the Station Management Team. The chairperson conducted and ruled on the proceedings of the meeting. The chairperson appointed an alternate chairperson from the panel if he was unable to attend a meeting.

Members and alternates received training and indoctrination in the risk-informed inservice inspection selection process. They were indoctrinated in the application of risk analysis techniques for ISI. These techniques included risk importance measures, threshold values, failure probability models, failure mode assessments, PRA modeling limitations and the use of expert judgment. Training documentation is maintained with the expert panel's records.

Worksheets were provided to the panel on each system for each piping segment, containing information pertinent to the panel's selection process. This information, in conjunction with each panel member's own expertise and other documents as appropriate, were used to determine the safety significance of each piping segment.

The expert panel used a consensus process. Consensus is defined as unanimous during first consideration and 2/3 (rounding conservatively) of members or alternates present in the second or subsequent considerations. The chairperson shall allow appropriate time duration between considerations for deliberation.

The chairperson appointed an individual to record the minutes of each meeting. The minutes included the names of members and alternates in attendance and whether a quorum was present. The minutes contained relevant discussion summaries and the results of membership voting. These minutes are available as program records.

3.7 Identification of High Safety Significant Segments

The number of high safety significant segments for each system, as determined by the expert panel, is shown in Table 3.7-1 along with a summary of the risk evaluation identification of high safety significant segments.

3.8 Structural Element and NDE Selection

The structural elements in the high safety significant piping segments were selected for inspection and appropriate non-destructive examination (NDE) methods were defined.

The initial program being submitted addresses the high safety significant (HSS) piping components placed in regions 1 and 2 of Figure 3.7-1 in WCAP-14572, A-Version. Segments considered as "high failure importance" (Region 1) were identified as all segments being affected by an active failure mechanism or analyzed to be highly susceptible to a failure mechanism (probability of large leak at 40 years greater than or equal to 1E-04). Region 3 piping components, which are low safety significant (LSS), are to be considered in an Owner Defined Program and are not considered part of the program requiring approval. Region 1, 2, 3 and 4 piping components will continue to receive Code required pressure testing, as part of the current ASME Section XI program. For the 129 piping segments that were evaluated in the RI-ISI program, Region 1 contains 17 segments, Region 2 contains 56 segments, Region 3 contains 0 segments, and Region 4 contains 56 segments.

The number of locations to be examined in a HSS segment was determined using a Westinghouse statistical (Perdue) model as described in section 3.7 of WCAP-14572, A-Version. None of the HSS piping segments in Region 1 and 31 of the HSS piping segments in Region 2 were evaluated using the Perdue model. The 42 segments that were not evaluated using the Perdue model included 8 segments that are subject to vibratory fatigue, 42 segments containing socket welds, and one segment subject to stress corrosion cracking, all of which are outside the applicability of the model. For these 42 segments, the guidance in Section 3.7.3 of WCAP-14572, A-Version was followed.

Table 4.1-1 in WCAP-14572; A-Version was used as guidance in determining the examination requirements for the HSS piping segments. Socket welds which were selected for examination for thermal fatigue and those that did not have any potential degradation mechanism identified will receive a Visual, VT-2 examination per the alternative requirements of RR-89-41 described in Attachment 2. Visual, VT-2 examinations are scheduled in accordance with the station's pressure test program for Class 1 piping and remain unaffected by the risk-informed inspection program.

Additional Examinations

Since the risk-informed inspection program will require examinations on a large number of elements constructed to lesser preservice inspection requirements, the program in all cases will determine through an engineering evaluation the root cause of any unacceptable flaw or relevant condition found during examination. The evaluation will include the applicable service conditions and degradation mechanisms to establish that the element(s) will still perform their intended safety function during subsequent operation. Elements not meeting this requirement will be repaired or replaced.

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

3.9 Program Relief Requests

Alternate methods are specified to ensure structural integrity in cases where examination methods cannot be applied due to limitations such as inaccessibility or radiation exposure hazard.

An attempt has been made to provide a minimum of >90% coverage (per Code Case N-460) when performing the risk-informed examinations. However, some limitations will not be known until the examination is performed, since some locations will be examined for the first time by the specified techniques.

At this time, all the risk-informed examination locations that have been selected provide >90% coverage. In instances where a location may be found at the time of the examination that it does not meet >90% coverage, the process outlined in Section 4.1 of WCAP-14572, A-Version will be followed.

3.10 Change in Risk

The risk-informed ISI program has been developed in accordance with Regulatory Guide 1.174 and 1.178, and the risk from implementation of this program is expected to slightly decrease when compared to that estimated from current requirements.

The change in risk calculations was performed according to the guidelines provided on page 213 of the WCAP-14572, A-Version. A comparison between the proposed RI-ISI program and the current ASME Section XI ISI program was made to evaluate the change in risk. The approach evaluated the change in risk with the inclusion of the probability of

detection as determined by the SRRA model. The four criteria for accepting the results discussed on page 214 and 215 in the WCAP were met and this evaluation resulted in no additional piping segments being added to the program.

The results from the risk comparison are shown in Table 3.10-1. As seen from the table, the RI-ISI program reduces the risk associated with piping CDF/LERF slightly more than the current Section XI program while reducing the number of examinations. Table 3.10-1 also includes the systems that are the main contributors to the risk reduction in moving from the current program to the RI-ISI program. The primary basis for this risk reduction is that examinations are now being placed on piping segments that are high safety significant and which are not inspected by NDE in the current ASME Section XI ISI program.

Defense-In-Depth

The reactor coolant piping will continue to receive a system pressure test and Visual, VT-2 examination as currently required by the Code. Volumetric examinations are proposed on smaller reactor coolant piping as part of the RI-ISI program. Larger reactor coolant loop piping segments were retained in the program for "defense-in-depth" considerations. The locations selected were associated with dissimilar metal welds at the reactor coolant pumps and at other connections to the main loop piping: the pressurizer surge line, and safety injection, charging, and shutdown cooling system nozzles.

4. IMPLEMENTATION AND MONITORING PROGRAM

Upon approval of the RI-ISI program, procedures that comply with the guidelines described in WCAP-14572, A-Version will be prepared to implement and monitor the program. The new program will be integrated into the existing ASME Section XI interval. No changes to the Final Safety Analysis Report are necessary for program implementation.

The applicable aspects of the 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 will be modified to address the RI-ISI process, as appropriate. Additionally, the procedures will be modified to include the high safety significant locations in the program requirements regardless of their current ASME class.

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

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

The RI-ISI program is a living program requiring feedback of new relevant information to ensure the appropriate identification of high safety significant piping locations. As a minimum, risk ranking of piping segments will be reviewed and adjusted on an ASME period basis. Significant changes may require more frequent adjustment as directed by NRC bulletin or generic letter requirements, or by plant specific feedback.

5. PROPOSED ISI PROGRAM PLAN CHANGE

A comparison between the RI-ISI program and the current ASME Section XI program requirements for piping is given in Table 5-1.

The plant will be performing examinations on elements not currently required to be examined by ASME Section XI. An example of these additional examinations are those similar and dissimilar butt welds located on piping less than NPS 4 where only a surface examination is currently required by Section XI, but have been determined under the RI-ISI process to be potentially subject to primary water stress corrosion cracking or thermal fatigue and will now receive a volumetric examination.

Upon approval of this request the initial RI-ISI program will be implemented in the inspection period current at the time of program approval and corresponding Class 1 Section XI examinations will cease. This is currently planned to take place in support of the second refueling outage of the second inspection period (RFO 16), during the current third 10-year inspection interval that will occur in the spring of 2005. (Note: The second inspection period of the third 10-year inspection interval for Unit 2 ends on November 30, 2005 and it is anticipated that the RI-ISI program will be in effect for the spring 2005 refueling outage RFO16). Some locations that have been selected for examination under the RI-ISI program will have already been examined for the existing ISI program in RFO15 scheduled to occur in the fall 2003. These previously examined locations will be credited under the new RI-ISI program provided the failure mechanisms(s) of concern were covered by the completed examinations. Regardless of the initial start date of the RI-ISI program all 100% of the examinations required by the RI-ISI program will be completed by the end of the current 10-year inspection interval based on the interval requirements of the current Section XI ISI program

6. SUMMARY OF RESULTS AND CONCLUSIONS

A partial scope Class 1 risk-informed ISI application has been completed for Millstone Unit 2. Upon review of the proposed risk-informed ISI examination program given in Table 5-1, an appropriate number of examinations are proposed for the high safety significant segments across the Class 1 portion of the plant piping systems. Resources to perform examinations currently required by ASME Section XI in the Class 1 portion of the plant piping systems, though reduced, are distributed to address the greatest amount of risk within the scope. Thus, the change in risk principle of Regulatory Guide 1.174 is maintained. Additionally, the examinations performed will address specific damage mechanisms postulated for the selected locations through appropriate examination selection and increased volume of examination.

MPS 2 is an older vintage 2 Loop Combustion Engineering (CE) PWR plant whose construction permit was issued on December 11, 1970 and that went into commercial operation December 26, 1975. The Class 1 piping is designed to various codes. Portions of the reactor coolant pressure boundary piping supplied by CE under the Nuclear Steam Supply System Vendor Technical Manual used ANSI B31.7 1968 Edition as the Construction Code. This design satisfies ASME Section III 1968 Edition, Summer 1969 Addenda. Other portions of the Class 1 piping were supplied by the Architect Engineer, Bechtel Corporation and its sub-vendors, and included shop fabrication under the requirements of USAS B31.7, 1969 Edition. Field fabrication and installation was completed under ASME Section III, 1971 Edition. The main loop piping is fabricated of low alloy steel and clad internally with austenitic stainless steel. The pressurizer surge line and reactor coolant pumps are fabricated from cast stainless steel. Because of the vintage of this plant an extensive amount of the small bore Class 1 piping is designed and fabricated using socket welds for piping NPS 2 and smaller. These socket welds have been accounted for in the development of the RI-ISI program and the alternative request RR-89-41 in Attachment 2 is provided to address these welds.

From a risk perspective, the PRA dominant accident sequences include small LOCAs and loss of a reactor coolant pump thermal barrier cooling train that leads to a reactor coolant pump seal LOCA.

For the RI-ISI program, appropriate sensitivity and uncertainty evaluations have been performed to address variations in piping failure probabilities and PRA consequence values along with consideration of deterministic insights to assure that all high safety significant piping segments have been identified.

As a risk-informed application, this submittal meets the intent and principles of Regulatory Guide 1.174.

7. REFERENCES/DOCUMENTATION

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

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

<u>Supporting Onsite Documentation</u>

Millstone Calculation File PRA02NQA-03112S2, Rev. 0 including CCN 1 "MP2 RI-ISI Piping Segment/Direct Consequence Determination".

Millstone Calculation File PRA02NQA-01326S2, Rev. 0 "MP2 RI-ISI Consequence and Risk Evaluation"

Millstone Calculation File 03-CP-01999M2, Rev. 0, "Failure Probabilities of Class 1 Piping for Risk-Informed Piping ISI Program for Unit 2".

Millstone Technical Evaluation, M2-EV-02-0028, Rev. 0, "Indirect Effects of Class 1 Piping for the Risk Informed ISI Project, Millstone Unit 2".

Millstone Calculation File PRA02NQA-03128S2, Rev. 0 "MP2 RI-ISI Change in Risk Calculation"

Millstone Calculation File PRA02NQA-03127S2, Rev. 0 "MP2 RI-ISI Expert Panel Worksheets"

Millstone Calculation File 03-CP-04038M2, Rev. 0, "Perdue Model Verification of Sample Size in Risk-Informed Piping ISI Program for Unit 2"

Table 3.1-1

SYSTEM SELECTION AND SEGMENT DEFINITION

System Description	PRA	Section XI	Number of Segments
1. RC - Reactor Coolant System	Yes	Yes	90
2. CH- Chemical & Volume Control System	Yes	Yes[1]	14
3. SI-High/Low Pressure Safety Injection System	Yes	Yes[1]	25
Total			129

Notes:

1. Portions of this system are not included in the Class 1 portion of the Section XI program.

Table 3.3-1

SUMMARY OF POSTULATED CONSEQUENCES BY SYSTEM

System	Summary of Consequences
RC - Reactor Coolant	The direct consequences associated with piping failures are large, medium and/or small loss of coolant accidents (LOCAs) and loss of ECCS flow to one loop.
SI - High/Low Pressure Safety Injection	The direct consequences associated with piping failures are the loss of accumulator injection, loss of one SI train for injection and recirculation from the safety injection system, loss of RWST inside containment and loss of containment sump recirculation inside containment.

Table 3.3-1
SUMMARY OF POSTULATED CONSEQUENCES BY SYSTEM

System	Summary of Consequences
CH - Chemical & Volume Control	The direct consequences associated with piping failures are isolable medium or small LOCA, loss of both charging trains for injection, recirculation, auxiliary spray and emergency boration, loss of the boron precipitation function via charging, loss of RWST inside containment.

Table 3.4-1
FAILURE PROBABILITY ESTIMATES (WITHOUT ISI)

System	Dominant Potential Degradation Mechanism(s)/ Combination(s)	Failure Probability range at 40 years with no ISI		Comments
		Small leak	Disabling leak (by disabling leak rate)*	
RC	• Thermal Fatigue • Vibratory Fatigue • Dissimilar Metal Weld • Stress Corrosion Cracking • No Significant mechanism	1.50E-05 – 6.64E-05 1.00E-04 7.00E-07 8.24E-03 2.71E-07 – 5.43E-04	• LLOCA 1.59E-05–3.36E-05 • MLOCA 1.70E-05 –3.65E-05 • SLOCA 1.97E-05 –3.80E-05 • SLOCA 1.53E-03 • LLOCA 3.54E-07 • MLOCA 3.56E-07 • SLOCA 3.80E-07 • SLOCA 3.62E-03 • LLOCA 3.10E-07 – 3.30E-06 • MLOCA 3.56E-07 –1.16E-05 • SLOCA 3.80E-07 –3.46E-04 • SYS 6.31E-06 – 8.38E-05	Surge line and safety injection subject to thermal stratification Socket welds near RCP Inconel buttering Socket welds. Inconel RPV vent line Butt welds and Socket welds
SI	• No Significant mechanism	4.54E-06 – 1.03E-03	4.65E-06 – 5.23E-04	Socket welds
CH	• No Significant mechanism	8.64E-06 –5.00E-03	3.33E-05 - 4.85E-03	Socket welds

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

Table 3.5-1
NUMBER OF SEGMENTS AND PIPING RISK CONTRIBUTION BY SYSTEM (WITHOUT ISI)

System	# of Segments	CDF without Operator Action (/yr)	CDF with Operator Action (/yr)	LERF without Operator Action (/yr)	LERF with Operator Action (/yr)
CH	14	5.78E-09	5.76E-09	2.08E-13	7.17E-15
RC	90	1.53E-05	1.53E-05	3.53E-07	3.52E-07
SI	25	4.44E-12	4.44E-12	2.04E-13	2.04E-13
TOTAL	129	1.53E-05	1.53E-05	3.53E-07	3.52E-07

Table 3.7-1
SUMMARY OF RISK EVALUATION AND EXPERT PANEL CATEGORIZATION RESULTS

System	Number of segments with any RRW >1.005	Number of segments with any RRW between 1.005 and 1.001	Number of segments with all RRW < 1.001	Number of segments with any RRW between 1.005 and 1.001 placed in HSS	Number of segments with all RRW < 1.001 selected for inspection	Total number of segments selected for inspection (High Safety Significant Segments)
RC	37	29	24	24	7	68
SI	0	0	25	0	4	4
CH	1	0	13	0	0	1
Total	38	29	62	24	11	73

Table 3.10-1
COMPARISON OF CDF/LERF FOR CURRENT SECTION XI
AND RISK-INFORMED ISI PROGRAMS
AND THE SYSTEMS WHICH CONTRIBUTED SIGNIFICANTLY TO THE CHANGE

Case (Systems Contributing to Change)	Current Section XI	Risk-Informed
CDF No Operator Action	5.01E-06	4.52E-06
• RC	5.01E-06	4.52E-06
• SI	1.18E-13	1.18E-13
• CH	1.72E-10	1.79E-10
CDF with Operator Action	4.99E-06	4.50E-06
• RC	4.99E-06	4.50E-06
• SI	1.18E-13	1.18E-13
• CH	1.70E-10	1.77E-10
LERF No Operator Action	1.17E-07	1.05E-07
• RC	1.17E-07	1.05E-07
• SI	5.45E-15	5.45E-15
• CH	2.28E-14	2.31E-14
LERF with Operator Action	1.16E-07	1.05E-07
• RC	1.05E-07	1.05E-07
• SI	5.45E-15	5.45E-15
• CH	4.36E-16	7.82E-16

Table 5-1

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

System	Number of High Safety Significant Segments (No. of HSS in Augmented Program / Total No. of Segments in Augmented Program)	Degradation Mechanism(s)	Class	ASME Code Category	Weld Count (Section XI / Exempt ≤ NPS1)		ASME XI Examination Methods (Volumetric (Vol) and Surface (Sur))		RI-ISI [a]	
					Butt	Socket	Vol & Sur	Sur Only	SES Matrix Region	Number of Exam Locations
CH	1 (0 / 0)	None [b]	Class 1	B-J	11 / 0	27 / 24	0	10	1A, 1B, 2	3 VIS [c]
SI	4 (0 / 0)	None [b]	Class 1	B-J	118 / 0	0 / 142	30	0	2	4
RC	68 (0 / 0)	None [b], VF, TF, Strip/Strat, PWSCC	Class 1	B-F	28	0	19	9	2	28
				B-J	303 / 22	41 / 329	76	11	1A, 1B, 2	12 + 1 VIS [c] / 10 + 23 VIS + 45 VIS [c]
			Class 1	B-F	28	0	19	9		28 NDE

Table 5-1

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

System	Number of High Safety Significant Segments (No. of HSS in Augmented Program / Total No. of Segments in Augmented Program)	Degradation Mechanism(s)	Class	ASME Code Category	Weld Count (Section XI / Exempt ≤ NPS1)		ASME XI Examination Methods (Volumetric (Vol) and Surface (Sur))		RI-ISI [a]	
					Butt	Socket	Vol & Sur	Sur Only	SES Matrix Region	Number of Exam Locations
TOTAL	73 (0 / 0)			B-J	432 / 22	68 / 495	106	21		16 NDE + 4 VIS [c] / 10 NDE + 23 VIS + 45 VIS [c]
			Total		482	563	125	30		54 NDE + 72 VIS

Summary: Current ASME Section XI selects a total of 155 non-destructive exams while the proposed RI-ISI program selects a total of 126 exams (126 - 72 visual exams), which results in a 65% reduction.

Degradation Mechanisms: VF – Vibratory Fatigue; TF – Thermal Fatigue, None [b]; PWSCC – Primary Water Stress Corrosion Cracking; Strip/Strat – Striping/Stratification

Notes for Table 5-1

a. *System pressure test requirements and VT-2 visual examinations shall continue to be performed in all ASME Code Class 1, 2, and 3 systems.*
b. *Where None is listed under the Degradation Mechanism(s) column elements will be treated as having TF.*
c. *VT-2 only socket weld (Refer To Attachment (2) Alternative Request RR-89-41).*

Attachment 2

Millstone Power Station, Unit No. 2

Alternative Request RR-89-41 To Use A Visual, VT-2 Examination In Lieu
Of A Volumetric Examination Under The Risk-Informed ISI Program
For Socket Welds Of Any Size And Branch Pipe Connection Welds NPS 2 And Smaller

Millstone Power Station, Unit No. 2

Alternative Request RR-89-41 To Use A Visual, VT-2 Examination In Lieu Of A Volumetric Examination Under The Risk-Informed ISI Program For Socket Welds Of Any Size And Branch Pipe Connection Welds NPS 2 And Smaller

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Millstone Power Station, Unit No. 2

Alternative Request RR-89-41 To Use A Visual, VT-2 Examination In Lieu Of A Volumetric Examination Under The Risk-Informed ISI Program For Socket Welds Of Any Size And Branch Pipe Connection Welds NPS 2 And Smaller

Proposed Alternative
In Accordance with 10 CFR 50.55a(a)(3)(ii)

- Hardship or Unusual Difficulty without Compensating Increase in Level of Quality or Safety

1. IDENTIFICATION OF COMPONENTS

ASME Class 1 socket welds of any size and branch pipe connection welds NPS 2 and smaller identified in High Safety Significant (HSS) segments.

2. RI-ISI PROGRAM PLAN EDITION AND ADDENDA

The proposed RI-ISI program plan described in Attachment 1 under alternative request RR-89-40 will comply with WCAP-14572, Rev. 1-NP-A and the ASME Code Section XI, 1989 Edition for the duration of the current third 10-year ISI interval for Millstone Unit No. 2 (MPS 2).

3. APPLICABLE REQUIREMENTS

It is recognized that ASME Code Case N-577 is listed in Regulatory Guide 1.193 as being not acceptable generically, however, as allowed by Regulatory Guide 1.193, DNC is requesting NRC approval to implement the provisions of Code Case N-577. DNC believes that this alternative results in an acceptable level of quality and safety. WCAP-14572, Rev. 1-NP-A, Table 4.1-1 "Examination Category R-A, Risk-Informed Piping Examinations" and requirements contained in Table 1 of ASME Code Case N-577 referenced by the WCAP require examination of selected HSS structural elements based upon the postulated failure mechanism. For selected HSS structural elements that are postulated to be subject to thermal fatigue a volumetric examination is required. If no mechanism is postulated then examinations generally default to thermal fatigue requirements based on normal and design basis loadings. In these cases socket welds and branch pipe connection welds NPS 2 and smaller require a volumetric examination. Additionally, Table 4.1-1 allows a Visual, VT-2 examination for socket welds with other postulated mechanisms such as vibratory fatigue or primary water stress corrosion cracking, but does not address socket weld examinations for other mechanisms or branch pipe connection welds NPS 2 and smaller.

4. BASIS FOR THE UNUSUAL DIFFICULTY WITHOUT A COMPENSATING INCREASE IN THE LEVEL OF QUALITY OR SAFETY

The requirements of WCAP-14572, Rev. 1-NP-A, Table 4.1-1 "Examination Category R-A, Risk-Informed Piping Examinations" and requirements contained in Table 1 of ASME Code Case N-577 referenced by the WCAP to perform volumetric examination when thermal fatigue or no mechanism is postulated does not account for the geometric limitations imposed by socket welds and branch pipe connection welds NPS 2 and smaller. Additionally, for other failure mechanisms addressed in the Tables, branch pipe connection welds NPS 2 and smaller that are subject to these same geometric limitations are not addressed.

Socket welds and branch pipe connection welds NPS 2 and smaller have been identified at MPS 2 to be included in HSS piping segments. Certain socket welds have been identified as HSS and require volumetric examination for their postulated failure mechanism. These instances are associated with a potential thermal fatigue failure mechanism either caused by local loading concerns or as a default mechanism for segments selected for their consequence of failure with no assumed or postulated active failure mechanism occurring. Additionally, there is no concern that the failure mechanism(s) evaluated would cause a flaw to be initiated from the outside diameter of these welds. Performing a volumetric examination on a socket weld or a branch pipe connection weld NPS 2 or smaller provides little or no benefit, because the flaws of concern would be initiated from the inside diameter, and the examination is limited by the joint configuration and the smaller pipe size.

The ASME Code Committee recognized this problem and revised Code Case N-577 to allow substitution of the VT-2 examination method for all damage mechanisms on socket welds selected as HSS structural elements. The revised version is noted as Code Case N-577-1 and provides for the substitution in Note 12 of Table 1. Incorporation of the branch pipe connection welds NPS 2 and smaller and a surface examination for these welds and socket welds that are subject to external chloride stress corrosion cracking is now under consideration by the committee. Millstone Unit No. 2 has no concerns related to having external stress corrosion cracking on any of its selected welds.

Performing a Visual, VT-2 type of examination on any socket weld or branch pipe connection weld NPS 2 and smaller, where volumetric examination is currently required, and where no concern exists that the failure mechanism would cause an outside diameter initiated flaw, is the most reasonable alternative to the volumetric examination requirement. Unless there is a concern due to an outside diameter flaw initiation, performing a volumetric examination on a socket weld of any size or a branch pipe connection weld NPS 2 and smaller, would result in unusual difficulty. Use of a volumetric examination would not provide any meaningful results, and there would be

no compensating increase in the level of quality and safety. Substituting a Visual, VT-2 examination as an alternative examination for these locations ensures reasonable assurance of continued component integrity.

5. PROPOSED ALTERNATIVE

All components of Class 1 HSS piping segments will receive a Visual, VT-2 type examination during each refueling outage. This examination will include HSS socket welds of any size and branch pipe connection welds NPS 2 and smaller. The VT-2 examination will be performed during the Class 1 system pressure test (leakage test) that will be conducted at nominal operating pressure (NOP) and nominal operating temperature (NOT) in accordance with the current ASME Code Section XI ISI program requirement. The examination will include looking for any boric acid residues. No hold time prior to the examination is required once at NOP and NOT. Identified through-wall or through-weld leakage will be corrected under the ASME Section XI Repair/Replacement program.

6. DURATION OF PROPOSED ALTERNATIVE

The alternative proposed in this request will be implemented upon approval of the alternative request R-89-40 to implement the proposed RI-ISI program at Millstone Unit No. 2 and this alternative request will be used for the duration of the approved RI-ISI program.

7. PRECEDENTS

The NRC approved a similar request for the examination of socket welds and branch connection welds for Watts Bar Nuclear Plant – Unit 1.