Northeast Nuclear Energy

Rope Ferry Rd. (Route 156), Waterford, CT 06385

Millstone Nuclear Power Station
Northeast Nuclear Energy Company
P.O. Box 128
Waterford, CT 06385-0128
(860) 447-1791
Fax (860) 444-4277

The Northeast Utilities System

JUL 25 2000

Docket No. 50-423
B18104

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

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

Millstone Nuclear Power Station, Unit No. 3
Risk-Informed Inservice Inspection Program Plan
Request For Relief From ASME Section XI

Northeast Nuclear Energy Company (NNECO) hereby requests to implement a Risk-Informed Inservice Inspection (RI-ISI) Program as an alternative to the current American Society of Mechanical Engineers (ASME) Section XI inservice inspection requirements for Class 1 piping at Millstone Unit No. 3. Therefore, pursuant to 10 CFR 50.55a(a)(3)(i), the proposed RI-ISI Program Plan described in Attachment 1 is provided for your review and approval.

The RI-ISI Program Plan 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." This program plan clearly supports the conclusion that the proposed alternative provides an acceptable level of quality and safety as required by 10 CFR 50.55a(a)(3)(i). Additional supporting documentation is available at the facility for the Nuclear Regulatory Commission (NRC) Staff's review.

It is requested that NRC approval be provided to support implementation of the Millstone Unit No. 3 RI-ISI Program prior to the Fall of 2002 to support the first refueling outage of the second period of the current 10 year inspection interval.

There are no regulatory commitments contained in this letter.

A047

Should you have any questions regarding this matter, please contact Mr. David W. Dodson at (860) 447-1791, extension 2346.

Very truly yours,

NORTHEAST NUCLEAR ENERGY COMPANY

Stephen E. Scace

Stephen E. Scace - Director
Nuclear Oversight and Regulatory Affairs

Attachment (1): Risk-Informed Inservice Inspection Program Plan Using The Westinghouse Owners Group Methodology (WCAP-14572, Revision 1-NP-A, February 1999)

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

Docket No. 50-423
B18104

Attachment 1

Millstone Nuclear Power Station, Unit No. 3

Risk-Informed Inservice Inspection Program Plan
Using The Westinghouse Owners Group Methodology
(WCAP-14572, Revision 1-NP-A, February 1999)

Risk-Informed Inservice Inspection Program Plan Using The Westinghouse Owners Group Methodology (WCAP-14572, Revision 1-NP-A, February 1999)

Table of Contents

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

1.1. Introduction

Inservice inspections (ISI) are currently performed on piping to the requirements of the ASME Boiler and Pressure Vessel Code Section XI, 1989 Edition which was the Code required by 10 CFR 50.55a at the beginning of the ISI interval. Millstone Unit No. 3 is currently in the first inspection period of its second 10-year interval as defined by the Code for Program B.

The objective of this submittal is to request a change to the ISI program plan for Class 1 piping only through the use of a risk-informed inservice inspection (RI-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 Guides 1.174 and 1.178. Further information is provided in Section 3.10 relative to defense-in-depth.

1.2. PRA Quality

The plant-specific Level 1 and Level 2 probabilistic risk assessment (PRA) model, Revision M3990927 dated October 1999 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 revision of the PRA model are 7.20E-05/yr and 8.98E-07/yr, respectively.

PRA model updates are scheduled either when a high priority change has been identified or within 90 days after startup from a refueling outage. The administrative guidance for this activity is contained in our administrative procedures.

The Millstone Unit No. 3 Class 1 RI-ISI evaluation included a determination that the PRA model and supporting documentation accurately reflects, to the extent possible, the current plant configuration and operational practices consistent with its intended application. The MP3 PRA model has had many reviews beginning with the MP3 PSS Study (1983) which used a three-tier review process with the third level being an industry peer review. Subsequently in 1990, the MP3 IPE model was reviewed and approved by the NRC.(1)(2) In 1996, an evaluation based on the Appendix B of the EPRI PSA Guide was performed to confirm that the PRA conforms to the industry standards. The results of this evaluation has been documented in the Westinghouse Topical Report, WCAP-14572 Revision 1-NP-A entitled "Westinghouse Owners Group Application of Risk-Informed Methods to Piping Inservice Inspection Topical Report." In 1999, an internal PRA self-assessment and a Westinghouse Owner's Group Peer Review Certification were conducted for the MP3 PRA model.

During the NRC's review of the IPE, concerns were identified regarding the smaller loss of offsite power contribution to core damage than that estimated in previous staff studies on station blackout. However, NNECO did commit to the installation of a third air-cooled diesel generator in accordance with the Station Blackout requirements. Because implementation of this third diesel would reduce the loss of offsite power/blackout contribution, the staff did not pursue this issue further during its review. The third SBO diesel generator has been installed and incorporated into the PRA model. Additional modifications to the PRA model following the NRC review included the addition of a total loss of service water as an initiator, the explicit modeling of HVAC and the explicit treatment of DC power.

2. PROPOSED ALTERNATIVE TO ISI PROGRAM

2.1. ASME Section XI

ASME Section XI, Class 1 Categories B-F and B-J, currently contain the requirements for examining via nondestructive examination (NDE), Class 1 piping components. This current program is limited to ASME Class 1 piping. The alternative RI-ISI program for piping is described in WCAP-14572, A-Version. The Class 1 RI-ISI program will be substituted for the current Class 1 examination program on 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.

---

(1) Letter from Mr. E. J. Mroczka to U.S. Nuclear Regulatory Commission dated August 31, 1990, entitled "Millstone Nuclear Power Station, Unit No. 3 Response to Generic Letter 88-20 Individual Plant Examination for Severe Accident Vulnerabilities Summary Report Submittal (B13596)."

(2) Letter from Mr. V. L. Rooney, Senior Project Manager, Project Directorate I-4, Division of Reactor Projects - I/II, Office of Nuclear Reactor Regulation to Mr. J. F. Opeka, dated May 5, 1992, entitled "Staff Evaluation Of Millstone 3 Individual Plant Examination (IPE) - Internal Events, GL 88-20 (TAC NO. M74434)."

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 the Millstone Unit No. 3 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.

The process that is being applied, involves the following steps:

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

There are no deviations to the process described in WCAP-14572, A-Version.

3.1. Scope of Program

The scope of this program is limited to Class 1 piping, and includes piping exempt from current 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 pipe segments defined for the Class 1 piping systems are summarized in Table 3.1-1. The as-operated piping and instrumentation diagrams (P&IDs) that were used to define the segments are included in the MP3 RI-ISI Calculation No. PRA99NQA-01741S3 Rev. 0.

3.3. Consequence Evaluation

The consequences of pressure boundary failures are measured in terms of core damage and large early release frequency. The impact on these measures due to both direct and indirect effects was considered.

A review of the license basis of Millstone Unit No. 3 (Millstone Unit No. 3 Final Safety Analysis Report and supporting documents) was performed to determine the potential impact of the indirect effects of pipe leak or rupture inside containment. As a result of the review, it was concluded that the containment structure and the safety related components inside containment are adequately protected from pipe failures such that the effects of a failure are limited to direct effects.

3.4. Failure Assessment

Failure estimates were generated utilizing industry failure history, plant specific failure history, and other industry relevant information.

The piping subpanel or engineering team that performed this evaluation used the Westinghouse structural reliability and risk assessment (SRRA) software program (described in WCAP-14572, Revision 1-NP-A, Supplement 1) to aid in the process. Generally, the SRRA code was used to estimate where the possible ranges of failure probability would fall. The final probability selected was determined by the team members using the relevant information and industry experience.

Table 3.4-1 summarizes the failure probability estimates by failure mechanism and also identifies the systems susceptible to these mechanisms.

No augmented inspections are performed for the Class 1 piping.

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 core damage frequency and large early release frequency were calculated by summing across the segments for each system. The results of these calculations are presented in Table 3.5-1. The expected value for core damage frequency due to piping failure without operator action is 7.16E-06/year, and with operator action is 7.10E-06/year. The expected value for large early release frequency due to piping failure without operator action is 2.33E-08/year, and with operator action is 2.31E-08/year. To assess safety significance, the risk reduction worth (RRW) and risk achievement worth (RAW) importance measures 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 risk assessment, inservice inspection, 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 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 members:

- Chairperson
- Probabilistic Risk Assessment (PRA engineer)
- Operations
- Inservice Inspection (ISI)
- Engineering/Component Performance

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

The expert panel chairperson was the chairperson for the Maintenance Rule expert panel. The chairperson conducted and ruled on the proceedings of the 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.

A consensus process was used by the expert panel. 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 allowed appropriate time between considerations for deliberation.

Meeting minute records were generated. The minutes included the names of members in attendance and whether a quorum was present. The minutes contained relevant discussion summaries and the results of membership voting.

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 5-1.

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 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 and described in Section 3.7.1 in WCAP-14572, A-Version. 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 NRC 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 113 piping segments that were evaluated in the RI-ISI program, Region 1 contains 4 segments, Region 2 contains 58 segments, no segments are contained in Region 3, and Region 4 contains 51 segments.

The number of locations to be inspected in applicable HSS segments was determined using a Westinghouse statistical (Perdue) model as described in section 3.7 of WCAP-14572, A-Version. The 58 piping segments in Region 2 and all the RCS primary loop piping were evaluated using this model. Only the welds of the 4 HSS CHS segments were found to be located in Region 1. Only 1 weld in each segment was located under Region 1A. These 4 welds are socket welds where only a VT-2 type of examination will be performed. The Perdue model was run for all HSS segments, including a pro-forma analysis of the reactor coolant pump seal injection lines (subject to vibration fatigue) for which the Perdue model is not directly applicable. The analysis used input from weld counts in each HSS segment, the SRRA reliability analysis, and other input. Based on this Perdue model analysis, all analyzed segments would be expected to meet the targeted mean leak rate with a confidence of nearly 100%. For those segments, where

the Perdue model was not applicable, the guidance in Section 3.7.3 of WCAP-14572, A-Version was followed to identify appropriate exam locations.

One segment RCS-65 had 2 elements selected based on 2 separate locations with potential thermal fatigue. This selection of 2 elements in this segment exceeded what would be required by the Perdue model.

Table 4.1-1 in WCAP-14752, A-Version, was used as guidance in determining the examination requirements for the HSS piping segments. VT-2 visual examinations are scheduled in accordance with the Unit's pressure test program which remains unaffected by the risk-informed inspection program.

3.9. <u>Additional Examinations</u>

Since the risk-informed inspection program will require examinations on a large number of elements constructed to lesser pre-service 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 on these elements up to a number equivalent to the number of elements required to be initially 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.

3.10. <u>Program Relief Requests</u>

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) for all of 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 should allow >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 4.0 (Inspection Program Requirements) of WCAP-14572, A-Version will be followed.

3.11. Change in Risk

The risk-informed ISI program was developed in accordance with Regulatory Guides 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.

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. This evaluation resulted in no additional piping segments being added for examination.

The results from the risk comparison are shown in Table 3.11-1. As seen from the table, the overall RI-ISI program slightly reduces the risk associated with piping CDF/LERF, with respect to the current Section XI program, while reducing the number of examinations. Table 3.11-1 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 exams will be required for highly safety significant piping segments which are not currently required to be inspected by NDE within the existing ASME Section XI ISI program.

3.12. Defense-In-Depth

The reactor coolant loop piping will continue to receive a system leakage test and visual VT-2 examination as currently required by the Code. Volumetric examinations and one surface examination will be performed on the smaller reactor coolant loop piping as part of the RI-ISI program. The larger reactor coolant loop piping will retain a requirement to perform a volumetric examination for "defense-in-depth" considerations to the extent that at least one structural element per HSS segment will be inspected each 10-year interval. Samples of dissimilar metal welds of the reactor vessel, steam generators, and pressurizer were all selected to continue to receive volumetric examinations under the RI-ISI program. Because of these structural element selections in the larger diameter RCS piping (at least one location per HSS segment), "defense-in-depth" has been maintained and no additional inspection locations are needed.

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 Technical Specifications or the Final Safety Analysis Report are necessary for program implementation.

The applicable aspects of the Code not affected by this change would 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 would be retained and would be modified to address the RI-ISI process, as appropriate. Additionally, the program will be modified to include the high safety significant locations.

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

A. Identify

B. Characterize

C. Evaluate

- Determine the cause and extent of the condition identified
- 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 expedited 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 volumetric examinations on elements not currently required to be volumetrically examined. An example of these additional examinations are those elements located on piping less than NPS 4 where only a surface exam is now required by Section XI, but have been determined under the RI-ISI process to be potentially subject to thermal fatigue.

Upon approval of this request, the initial Class 1 RI-ISI program will be started and all corresponding Class 1 Section XI examinations will cease. This is currently planned to take place in support of the first refueling outage of the second period during the current 10 year inspection interval which will occur in the Fall of 2002. Some locations that are selected for examination under the RI-ISI program will have already been

examined during RF06 and possibly RF07 for the existing ISI program. These previously examined locations will be credited under the new RI-ISI program provided the failure mechanism(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 inspection interval based on the interval requirements of the current Section XI ISI program.

6. REFERENCES/DOCUMENTATION

6.1 Primary

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

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

6.2. Supporting Onsite Documentation

6.2.1. PRA99NQA-01741S3, Revision 0, "MP3 RI-ISI Consequence and Risk Evaluation," May 2000.

6.2.2. PRA00NQA-01757S3, Revision 0, "MP3 RI-ISI: Indirect Effects," May 2000.

6.2.3. PRA00NQA-01783S3, Revision 0, "MP3 RI-ISI Change in Risk Calculation," May 2000.

6.2.4. PRA00NQA-01768S3, Revision 0, "MP3 RI-ISI Program Uncertainity," May 2000.

6.2.5. PRA00NQA-01767S3, Revision 0, "MP3 RI-ISI Program - Expert Panel Worksheets and Access Database," May 2000.

6.2.6. 00-CP-01755M-3, Revision 0, "Failure Probabilities of Class 1 Piping for Risk -Informed Piping ISI Program for Unit 3," March 2000.

6.2.7. 00-CP-01781M-3, Revision 0 CCN 1, "Perdue Model Verification of Sample Size in Risk-Informed Piping ISI Program for Unit 3," May 2000.

Table 3.1-1

System Selection and Segment Definition for Class 1 Piping

System Description	PRA	Section XI	Number of Segments
CHS - Chemical & Volume Control System	Yes	Yes	4
RCS - Reactor Coolant System	Yes	Yes	94
RHS - Residual Heat Removal System	Yes	Yes	2
SIH - High Pressure Safety Injection System	Yes	Yes	7
SIL - Low Pressure Safety Injection System	Yes	Yes	6
Total			113

Table 3.4-1

Failure Probability Estimates (without ISI) For HSS Segments

Failure Mechanism	Failure Probability Range (Leak Probability @ 40 years, no ISI)	Susceptible Systems
Thermal Fatigue(1) (TF)	5.8E-8 to 4.2E-5	RCS, RHS
Vibration(2) (V)	Up To 2.3E-4	CHS
None (Reverts To TF Exams)		CHS, RCS

NOTES:

(1) Including effects of dissimilar metal welds, thermal expansion, system operating transients, and thermal fatigue due to cyclic stratification and turbulence penetration.

(2) Vibration fatigue failure potential is highly dependent on sustained vibration levels. Only the upper value of the range was estimated.

Table 3.5-1

Number of Segments and Piping Risk Contribution by System (without ISI)
(values shown are expected values)

System	# of Segments	CDF without Operator Action (/yr)	CDF with Operator Action (/yr)	LERF without Operator Action (/yr)	LERF with Operator Action (/yr)
CHS	4	7.90E-07	7.90E-07	3.19E-10	3.19E-10
RCS	94	6.37E-06	6.31E-06	2.30E-08	2.28E-08
RHS	2	0.00	0.00	0.00	0.00
SIL	6	6.44E-11	6.25E-11	8.98E-13	8.98E-13
SIH	7	1.93E-10	1.85E-10	4.31E-12	4.31E-12
TOTAL	113	7.16E-06	7.10E-06	2.33E-08	2.31E-08

Table 3.11-1

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)	Piping CDF/LERF Current Section XI	Piping CDF/LERF Risk-Informed
CDF No Operator Action (RCS, CHS)	1.07E-06	1.05E-06
CDF with Operator Action (RCS, CHS)	1.05E-06	1.04E-06
LERF No Operator Action (RCS, CHS)	9.90E-10	9.77E-10
LERF with Operator Action (RCS, CHS)	9.43E-10	9.29E-10

Note: CDF/LERF values include credit for leak detection also.

Table 5-1

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

System	Number of High Safety-Significant Segments (No. in Augmented Program)	RI-ISI Program High Safety-Significant Structural Elements	ASME Section XI ISI Program 1989 Edition Examination Category Weld Selections		Total Number of Segments Credited in Augmented Programs
		CLASS 1	B-F	B-J	
CHS	4 (0)	4 + 4[b]	0	6	0
RCS	56 (0)	73	22	239	0
RHS	2 (0)	2	0	10	0
SIH	0 (0)	0	0	32	0
SIL	0 (0)	0	0	19	0
Total	62 (0)	79 + 4[b]	22	306	0

Summary: Total Class 1 welds equals 1218 which includes 22 B-F welds and 1196 B-J welds. Current ASME Section XI program selects a total of 328 Class 1 non-destructive exam locations while the proposed RI-ISI program selects a total of 79 exam locations (83 - 4 visual exam locations), which results in a 76% reduction. A total of 65 B-J welds or 5.4% of the B-J weld population is now scheduled for NDE in the Class 1 RI-ISI program. The remaining 14 welds receiving NDE are B-F welds. The selected 4 visual exam locations are B-J socket welds.

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. VT-2 area exam at specific location.