Westinghouse Non-Proprietary Class 3

WCAP-17236-NP
Revision 0

September 2010

Risk-Informed Extension of the Reactor Vessel Nozzle Inservice Inspection Interval

Westinghouse

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Risk-Informed Extension of the Reactor Vessel Nozzle Inservice Inspection Interval

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September 2010

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Note:
1. Project participants as of the date the final deliverable was completed. On occasion, additional members will join a project. Please contact the PWR Owners Group Program Management Office to verify participation before sending this document to participants not listed above.

PWR Owners Group
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Taiwan Power Co.	Maanshan 1 & 2 (W)		X
Electricite de France	54 Units		X

Note:
1. This is a list of participants in this project as of the date the final deliverable was completed. On occasion, additional members will join a project. Please contact the PWR Owners Group Program Management Office to verify participation before sending documents to participants not listed above.

EXECUTIVE SUMMARY

Section XI of the *American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code* specifies a 10-year interval between reactor vessel (RV) nozzle weld inspections. The industry has expended significant cost and man-rem exposure performing inspections that have found no service-induced flaws in ASME Section XI Category B-F or B-J RV nozzle welds that do not contain Alloy 82/182. Furthermore, many plants have implemented a 20-year inspection interval for the RV shell-to-shell and shell-to-nozzle welds in accordance with WCAP-16168-NP-A, Revision 2. For many of these plants, continuing to inspect the RV nozzle welds on a 10-year interval presents a significant hardship without a corresponding increase in safety from performing the inspections.

The objective of this report is to provide the technical basis and methodology for extending the Section XI inspection interval from the current 10 years to 20 years for Category B-F and B-J RV nozzle-to-safe-end and safe-end-to-pipe welds that are not fabricated with Alloy 82/182 materials. Bounding change-in-failure-frequency values have been calculated for use in plant-specific implementation of the extended inspection interval. Plant-specific pilot studies have been performed and the results show that the change in risk associated with extending the interval from 10 to 20 years after the initial 10-year inservice inspection satisfies the guidelines specified in Regulatory Guide 1.174 for an acceptably low change in risk for core damage frequency (CDF) and large early release frequency (LERF). Further, the pilot-plant results show that the effect of the extended inspection interval on the plant's risk-informed inservice inspection program for piping is acceptable.

ACKNOWLEDGEMENTS

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Finally, the authors would like to thank members of the PWR Owners Group (PWROG) Materials Subcommittee (MSC), Jim Molkenthin of the PWROG Project Office, MSC Chairman Jim Cirilli, and Chairmen of the PWROG Dennis Buschbaum and Mel Arey, for their continued support in the development of this risk-informed application and Topical Report.

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ACRONYMS

ASME	American Society of Mechanical Engineers
B&PV	Boiler and Pressure Vessel (Code)
B&W	Babcock & Wilcox
BV1	Beaver Valley Unit 1
CCDP	Conditional core damage probability
CDF	Core damage frequency
CE	Combustion Engineering
CFR	Code of Federal Regulations
CLERP	Conditional large early release probability
CS	Carbon steel
EPRI	Electric Power Research Institute
FENOC	First Energy Nuclear Operating Company
FF	Failure frequency
FSAR	Final Safety Analysis Report
GPM	Gallons per minute
GQA	Graded quality assurance
ID	Inner diameter
ISI	Inservice inspection
IST	Inservice testing
LD	Leak detection
LERF	Large early release frequency
LLOCA	Large loss-of-coolant accident
LOCA	Loss-of-coolant accident
MLOCA	Medium loss-of-coolant accident
NDE	Non-destructive examination
NRC	U.S. Nuclear Regulatory Commission
NSSS	Nuclear Steam Supply System
PA	PWR Owners Group Project Authorization
PDI	Performance Demonstration Initiative
PFM	Probabilistic fracture mechanics
PNNL	Pacific Northwest National Laboratories
POD	Probability of detection
PRA	Probabilistic risk assessment
PWR	Pressurized water reactor
PWROG	PWR Owners Group
QA	Quality assurance
RCS	Reactor coolant system
RG	NRC Regulatory Guide
RI-ISI	Risk-informed ISI
RPV	Reactor pressure vessel
RT_{NDT}	Reference nil-ductility transition temperature
RV	Reactor vessel
RV ISI	Reactor vessel inservice inspection
SER	NRC Safety Evaluation Report

ACRONYMS (cont.)

SLOCA	Small loss-of-coolant accident
SRP	Standard Review Plan
SRRA	Structural reliability and risk assessment
SS	Stainless steel
SSC	Systems, structures, and components
SSE	Safe shutdown earthquake
TMI	Three Mile Island
UT	Ultrasonic examination

1 INTRODUCTION AND BACKGROUND

1.1 INTRODUCTION

Section XI of the *American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code*, (Reference 1) specifies that reactor vessel (RV) nozzle welds are to be inspected on a 10-year interval. The manner in which these examinations are conducted has been augmented by Appendix VIII of Section XI, 1996 Addenda, as implemented by the NRC in an amendment to 10 CFR 50.55a effective November 22, 1999 (Reference 2). The industry has expended significant cost and man-rem exposure performing the required examinations for ASME Section XI, Table IWB-2500-1, Category B-F or B-J RV nozzle welds that do not contain Alloy 82/182 with no service-induced flaws being detected. These results indicate that the current ASME Code criteria for the selection of examination areas and the frequency of examinations are not an effective way to expend inspection resources.

The objective of the study described in this report was to:

1. Verify that the interval between volumetric examinations of non-Alloy 82/182 RV nozzle full-penetration welds can be extended from the current ASME Code requirement of 10 years to 20 years with an acceptably small change in risk and an acceptable effect on a plant's risk-informed inservice inspection (RI-ISI) program (if applicable).

2. Provide a methodology that can be used by licensees to justify implementation of the extended ISI interval on a plant-specific basis.

   Note: The terms "Inspection," "Examination," and "Exam" are used interchangeably within this report.

1.2 BACKGROUND

The original objective of the ASME B&PV Code, Section XI (Reference 1), ISI program was to assess the condition of pressure-retaining components in nuclear power plants to ensure continued safe operation. If non-destructive examination (NDE) found indications that exceeded the allowable standards, examinations were extended to additional welds in components in the same examination category. If the NDE found indications that exceeded the acceptance standards in those welds, then the examinations were extended even further to similar welds in similar components.

The original examination interval of 10 years was based on "wear-out" rate experience in the pre-nuclear utility and petrochemical process industries. As with some other Section XI ISI requirements, with no indications being found in the vessel welds under evaluation in this report, these inspections are decreasing in value with increasing industry experience to rely upon. The NRC has granted a number of exemptions to inspections for other areas and components, such as piping (Reference 4) and reactor coolant pump motor flywheels (Reference 5), based on inspection experience and man-rem reductions. This has been attributed to the combined design, fabrication, examination, and Quality Assurance (QA) rigor of the nuclear codes, and more careful control of plant operating parameters by the utilities.

WCAP-16168-NP-A, Revision 2, "Risk-Informed Extension of the Reactor Vessel In-Service Inspection Interval," (Reference 6) was approved by the NRC in May 2008 and provides a basis for the extension of the ASME Section XI (Reference 1) inspection interval from 10 years to 20 years. This interval extension applies to the reactor vessel (RV) shell-to-shell (ASME Section XI, Table IWB-2500-1 Category B-A) and shell-to-nozzle (ASME Section XI, Table IWB-2500-1, Category B-D) welds.

Typically, the reactor vessel nozzle welds are inspected using the same tooling as the shell-to-shell and shell-to-nozzle welds. Depending on the manufacturer of the reactor vessel and designer of the plant, the configurations of welds joining the reactor vessel nozzles to the piping vary. Some reactor vessels were fabricated with a safe-end welded to the nozzle. Depending on whether the reactor coolant main loop piping is stainless steel or low-alloy steel, a dissimilar metal weld (Catgeory B-F) or a similar metal weld (Category B-J), respectively, joins the safe-end to the nozzle. A similar metal weld (Category B-J) then joins the safe-end to the piping. For plants that do not have a safe-end, a single weld joins the nozzle to the piping. For plants with stainless steel reactor coolant main loop piping, this is a dissimilar metal weld (Category B-F) whereas it is a similar metal weld (Category B-J) for plants with low-alloy steel piping. These configurations are shown in detail in Section 3.2.3.

The effort to develop WCAP-16168-NP-A, Revision 2, originally included the ASME Category B-F and B-J welds discussed above. The Category B-F welds were removed from the scope of the effort during the development of the supporting ASME Code Case (Reference 7) because of concerns that Alloy 82/182 welds would be included. Therefore, as a resolution to a request for additional information from the NRC, the Category B-J welds were removed. This has created disconnectedness in that plants that have implemented the 20-year interval for the shell-to-shell and shell-to-nozzle welds may still be required to inspect the nozzle-to-pipe welds on a 10-year interval. This is a significant issue because the reactor core barrel will need to be removed from the vessel in order to gain access to inspect these welds.

For a number of reasons, removal of the core barrel is an activity that should be minimized to the extent practical. As with any heavy-lifting activity, there are significant safety risks. For the core barrel, this lift typically results in a high man-rem dose. Furthermore, the removal of the core barrel requires a full core offload, which typically consumes critical path outage time and always has the potential for fuel handling errors. For several plants, their refueling cavity is not deep enough to accommodate the core barrel and shielding must be erected around the core barrel after it has been removed because the upper portion is not submerged.

To develop a quantitative estimate of the cost of core barrel removal and RV nozzle inspection, a survey was performed by the PWROG. The results of this survey indicated an average cost of $515K per plant for inspecting the reactor vessel nozzles and an average dose of 1.65 man-rems of exposure.

While some plants with risk-informed programs for piping weld inservice inspection may be able to select welds other than the RV nozzle welds for inspection, which would eliminate the need to remove the core barrel, this is not an option for a significant number of plants. There are still several PWRs that do not have RI-ISI programs and must select locations for inservice inspection in accordance with Section XI, which includes the RV nozzle weld locations. Also, many plants that do have RI-ISI programs are limited in the availability of other locations for an alternate inspection. These other locations may only be inspected with limited coverage or may require the installation of scaffolding and shielding and the removal of insulation, and result in higher dose than inspecting the nozzle locations. These factors are likely the reasons why the RV nozzle locations were selected for inspection when the RI-ISI program was originally developed. For these plants, the best, and sometimes the only, solution is to inspect the RV nozzle welds on a 20-year interval.

2 REGULATORY EVALUATION

ASME Section XI currently requires that reactor vessel nozzle welds, including nozzle-to-pipe welds, nozzle-to-safe-end welds, and safe-end-to-pipe welds, be inspected on a 10-year interval. This interval may be extended for a particular plant, provided that the criteria of Regulatory Guide 1.174 (Reference 3) can be met and the effect on the plant's risk-informed inservice inspection program (if applicable) for piping can be shown to be acceptable. Approval of the process used to make this determination, described below, is requested. This process and two pilot-plant examples are described in detail in Section 3.

2.1 STEP 1: DETERMINE NOZZLE WELD CONFIGURATION TYPE

For the plants analyzed as part of this effort, nozzle weld geometries, dimensions, and operating conditions were reviewed to determine four different configurations; Types A, B, C, and D. For each configuration type, a set of bounding change in failure frequencies was determined. Figure 3-3 in Section 3.2.3 shows these weld types and Table 4-1 in Section 4 identifies the weld type for each plant analyzed. The first step in implementing the RV nozzle ISI interval extension is to determine which configuration type is applicable for a given plant.

2.2 STEP 2: REVIEW PREVIOUS INSERVICE INSPECTION RESULTS

The results from previous inservice inspections should be reviewed to confirm that there is no more than one ID surface flaw in each of the welds for which the ISI interval extension will be implemented. Furthermore, the surface flaw may not have a through-wall depth of greater than six percent of the wall thickness and a length equal to six times the depth.

If multiple surface-breaking flaws are present in a given weld, are in close proximity to one another (as defined by ASME Section XI proximity requirements), and can be bounded by the aforementioned flaw size, they may be treated as one flaw. If there are multiple flaws present in a given weld, and they are not bounded by the aforementioned flaw size, the bounding change in failure frequencies may need to be adjusted to account for the presence of multiple flaws. One way of making this adjustment would be to multiply the change in failure frequencies of Tables 3-3 through 3-6 by the number of surface flaws present in the weld. If the flaw size exceeds the dimensions specified above, a weld-specific probabilistic fracture mechanics (PFM) analysis, such as that described in Section 3.2.3, should be performed to develop a weld-specific change-in-failure-frequency value.

The limiting flaw depth specified above is based upon the upper 2-sigma bound on the log-normally distributed median value of the initial flaw depth used for the PFM analyses. Only about 2.5 percent of the flaws simulated in the PFM analyses would be expected to have a depth greater than the limiting value. The effects of flaw growth during operation are not included because the probability of the initial flaw growing through the wall and allowing a large leak is very small, typically less than 10^{-5} even after 40 years of operation.

2.3 STEP 3: PERFORM CHANGE-IN-RISK EVALUATION

The change in risk associated with extending the ISI interval for the RV nozzles can be calculated using the template shown in Table 2-1. The bounding change in failure frequencies for use in these calculations can be obtained from the appropriate table in Section 3.2 (Tables 3-3 through 3-6, depending on configuration type). The values should be those without credit for leak detection and either the 40-year or 60-year values may be used, depending on the licensed period of operation for a particular plant. Plant-specific conditional core damage probability (CCDP) and conditional large early release probability (CLERP) values, determined from the plant model for probabilistic risk assessment (PRA), should be used for the three failure modes for loss-of-coolant accident (LOCA) shown in Table 2-2. These LOCA failure modes are defined in Table 2-2. If additional LOCA sizes are modeled in the plant PRA, such as small-small LOCA, it is acceptable to use the small LOCA (SLOCA) change-in-failure-frequency values as an approximation for that failure mode. Examples of the change-in-risk calculations can be found in Section 3.2.4.

Table 2-1 Change-in-Risk Calculations for RG 1.174

Failure Mode	Bounding Change in Failure Frequency (w/o L.D.)	CCDP	ΔCDF (/year)	CLERP	ΔLERF (/year)
Outlet Nozzles					
SLOCA	ΔFF_{SLOCA}	$CCDP_{SLOCA}$	$= (\Delta FF_{SLOCA})(CCDP_{SLOCA})$	$CLERP_{SLOCA}$	$= (\Delta FF_{SLOCA})(CLERP_{SLOCA})$
MLOCA	ΔFF_{MLOCA}	$CCDP_{MLOCA}$	$= (\Delta FF_{MLOCA})(CCDP_{MLOCA})$	$CLERP_{MLOCA}$	$= (\Delta FF_{MLOCA})(CLERP_{MLOCA})$
LLOCA	ΔFF_{LLOCA}	$CCDP_{LLOCA}$	$= (\Delta FF_{LLOCA})(CCDP_{LLOCA})$	$CLERP_{LLOCA}$	$= (\Delta FF_{LLOCA})(CLERP_{LLOCA})$
# of Welds Examined	#	Total ΔCDF	= (sum of above)*(# of welds examined)	Total ΔLERF	= (sum of above)*(# of welds examined)
Inlet Nozzles					
SLOCA	ΔFF_{SLOCA}	$CCDP_{SLOCA}$	$= (\Delta FF_{SLOCA})(CCDP_{SLOCA})$	$CLERP_{SLOCA}$	$= (\Delta FF_{SLOCA})(CLERP_{SLOCA})$
MLOCA	ΔFF_{MLOCA}	$CCDP_{MLOCA}$	$= (\Delta FF_{MLOCA})(CCDP_{MLOCA})$	$CLERP_{MLOCA}$	$= (\Delta FF_{MLOCA})(CLERP_{MLOCA})$
LLOCA	ΔFF_{LLOCA}	$CCDP_{LLOCA}$	$= (\Delta FF_{LLOCA})(CCDP_{LLOCA})$	$CLERP_{LLOCA}$	$= (\Delta FF_{LLOCA})(CLERP_{LLOCA})$
# of Welds Examined	#	Total ΔCDF	= (sum of above)*(# of welds examined)	Total ΔLERF	= (sum of above)*(# of welds examined)
All Nozzles					
Total Change-in-Risk Results		Total ΔCDF	Sum of ΔCDF for inlet and outlet nozzles	Total ΔLERF	Sum of ΔLERF for inlet and outlet nozzles

Table 2-2 Failure Modes		
Failure Modes	**Acronym**	**Leak Rate (GPM)**
Small Loss-of-Coolant Accident	SLOCA	100
Medium Loss-of-Coolant Accident	MLOCA	1500
Large Loss-of-Coolant Accident	LLOCA	5000

The total change in risk associated with the extension in inservice inspection interval for the reactor vessel nozzles of the plant must satisfy the regulatory guidelines in RG 1.174 (Reference 3) for an acceptably small change in risk. These guidelines can be summarized as follows:

- Change in Core Damage Frequency (ΔCDF) < 1E-06/year, and
- Change in Large Early Release Frequency (ΔLERF) < 1E-07/year.

2.4 STEP 4: EVALUATE EFFECT ON RISK-INFORMED INSERVICE INSPECTION PROGRAM

If the plant has a traditional Section XI inservice inspection program for piping, rather than a risk-informed inservice inspection (RI-ISI) program, the analysis described above is sufficient for showing that the extension in inspection interval is acceptable. However, if the plant has implemented a RI-ISI program, which includes the RV nozzle welds, additional evaluation is required. The following sections detail the evaluations required for plants with PWROG (Reference 4) and EPRI (Reference 8) RI-ISI programs for piping. The evaluations for the EPRI RI-ISI programs are also applicable for plants with inspection programs based on ASME Section XI Code Case N-716 (Reference 9).

2.4.1 Effect on RI-ISI Program – PWROG Methodology

For plants that have applied the PWROG program for risk-informed inservice inspection (RI-ISI) of piping, the following steps and calculations are required for implementing the ISI interval extension for RV Category B-F and B-J nozzle welds that do not contain Alloy 82/182.

Implementation Method

To determine the effect on the piping risk-informed inservice inspection program of the plant, the change-in-risk calculations for the template in Table 2-1 are duplicated with the exception that the calculations are performed using the change in failure frequencies with credit for leak detection. These change-in-risk values, which represent the increase in risk associated with the extension of the ISI interval for the RV nozzles, are then added to the change-in-risk results of the RI-ISI program (Reference 4). These values are added to both the reactor coolant system change-in-risk values and also the total plant scope values for the ΔCDF, with and without operator action, and ΔLERF, with and without operator action cases. For each of these four cases, the system level and total change-in-risk values must be assessed against the change-in-risk acceptance criteria discussed in the following section.

It should be noted that the PWROG methodology as approved in WCAP-14572, Revision 1-NP-A (Reference 4) considers risk on a segment basis and the that risk is not dependent on the number of welds within a given piping segment. This is because the highest risk at the limiting location is controlling for that piping segment. Therefore, for nozzle configurations (see Figure 3-3 in Section 3.2.3) where there are two welds per nozzle, the risk should be adjusted to reflect only the most limiting weld prior to being added to the change in risk from the RI-ISI element selection.

Acceptance Criteria

The acceptance criteria of WCAP-14572, Revision 1-NP-A (Reference 4), which shall be used to determine the acceptability of the effect of the ISI interval extension on the RI-ISI program, can be summarized as follows:

1. The total change in piping risk should be risk neutral or a risk reduction in moving from Section XI to RI-ISI.

2. For dominant systems (e.g., system contribution to the total is greater than approximately 10%) the change in piping risk should be risk neutral or a risk reduction in moving from Section XI to RI-ISI.

3. For non-dominant systems:

   a. The CDF increase for the system should be less than a) than two orders of magnitude below the risk-informed ISI CDF for that system or b) 1E-08 (whichever is less).

   b. The LERF increase for the system should be less than a) two orders of magnitude below the risk-informed ISI LERF for that system or b) 1E-09 (whichever is higher).

If the acceptance criteria cannot be met, additional inspections shall be added to the RI-ISI program until the criteria are met. If the acceptance criteria cannot be met by adding additional inspections, or it is impractical to do so, the RI-ISI change-in-risk evaluation may be performed, consistent with the method used for the EPRI RI-ISI methodology, taking into account the number of welds per segment (see Equation 3-1 in Section 3.2.5.2). If this method is used, the following criteria from the EPRI RI-ISI methodology must be met.

The implementation of the RI-ISI program should be risk neutral, a decrease in risk, or, at most, an insignificant increase in risk. The increase in risk for each system shall meet the following criteria in order for it to be considered insignificant:

- Change in Core Damage Frequency (ΔCDF) < 1E-07/year, and
- Change in Large Early Release Frequency (ΔLERF) < 1E-08/year.

The total change for all systems must meet the criteria of RG 1.174 for an acceptably small change in risk which are as follows:

- Change in Core Damage Frequency (ΔCDF) < 1E-06/year, and
- Change in Large Early Release Frequency (ΔLERF) < 1E-07/year.

2.4.2 Effect on RI-ISI Program – EPRI or Code Case N-716 Methodology

For plants that have applied the EPRI program for risk-informed inservice inspection (RI-ISI) of piping or ASME Code Case N-716, the following steps and calculations are required for implementing the ISI interval extension for RV Category B-F and B-J nozzle welds that do not contain Alloy 82/182.

Implementation Method

To account for the extension in the inservice inspection interval for the reactor vessel nozzles, there are several methods that can be used depending on the method that was used to perform the change-in-risk evaluation for the original RI-ISI program development. These methods are discussed below based on the change-in-risk method.

1. Qualitative

   If the qualitative change-in-risk method from the EPRI topical report (Reference 8) was to show that there is no reduction in the number of inspections when moving from a Section XI inservice inspection program to a RI-ISI program, or if there is an increase in the number of inspections, the only increase in risk would be the result of the extension in inspection interval for the reactor vessel nozzle welds. Therefore, as long as the change in risk, as calculated per Section 2.3, meets the criteria of Regulatory Guide 1.174 (Reference 3) for an acceptably small change in risk, the extension in inspection interval would be acceptable.

2. Bounding with or without any Credit for Increase in Probability of Detection (POD)

   The effect of the ISI interval extension on the RI-ISI program may be evaluated by adding the bounding change in failure frequencies for the appropriate weld type (see Tables 3-3 through 3-6 in Section 3.2.3) to the bounding rupture frequencies from the EPRI topical report. These values would be added for each of the welds for which the ISI interval will be extended. For these calculations the bounding change in failure frequencies with credit for leak detection may be used. Using these revised bounding rupture frequencies, the system and total plant change-in-risk values would be calculated per the requirements of the EPRI topical report or Code Case N-716. The change-in-risk values for each system and for the total plant must be assessed against the change-in-risk acceptance criteria discussed in the following section.

   Alternatively, the CCDP and CLERP values for each of the welds, for which the ISI interval will be extended, can be multiplied by the bounding change in failure frequencies for the appropriate weld type. These change-in-risk values for each weld can then be summed to determine the total change in risk for the RV nozzle weld ISI interval extension. This total risk for the RV nozzle weld ISI interval extension can then be added to the system and total plant change-in-risk results of the RI-ISI program. The change in risk for each system and for the total plant must be assessed against the change-in-risk acceptance criteria discussed in the following section.

3. Markov Method

For plants that used the Markov method for evaluating the change in risk when moving from a Section XI inservice inspection program to a RI-ISI program, two methods are acceptable for evaluating the effect of the extension in inservice inspection interval for the RV nozzles.

Method A – Use Markov Model

For the reactor vessel nozzle welds for which the ISI interval is to be extended to 20 years, the hazard rate for the RI-ISI program would be calculated based on a 20-year interval. This hazard rate would then be used to calculate the inspection effectiveness factor for these particular welds. This inspection effectiveness factor would be used for the RV nozzle welds in the change-in-risk calculations, and the change in risk would be a result of the difference in inspection effectiveness between the Section XI exams performed on a 10-year interval and the RI-ISI exams performed on a 20-year interval. Therefore, the change in risk for the system and total plant would account for the increase in risk associated with the extension in inspection interval. The change in risk for each system and for the total plant must be assessed against the change-in-risk acceptance criteria discussed in the following section.

Method B – Blended Approach

The bounding change in failure frequencies in Tables 3-3 through 3-6 in Section 3.2.3 would be used to calculate the increase in risk from the RV nozzle ISI interval extension in lieu of the Markov model. Consistent with the discussion for the "Bounding" approach, CCDP and CLERP values for each of the welds, for which the ISI interval will be extended, can be multiplied by the bounding change in failure frequencies for the appropriate weld type. These change-in-risk values for each weld can then be summed to determine the total change in risk for the RV nozzle weld ISI interval extension. This total risk for the RV nozzle weld ISI interval extension can then be added to the system and total plant change-in-risk results of the RI-ISI program that have been calculated using the Markov method. The change in risk for each system and for the total plant must be assessed against the change-in-risk acceptance criteria discussed in the following section.

Acceptance Criteria

For the three methods discussed above, the acceptance criteria for change in risk from the EPRI RI-ISI topical report (Reference 8) or Code Case N-716 (Reference 9) can be stated as the implementation of the RI-ISI program should be risk neutral, a decrease in risk, or, at most, an insignificant increase in risk. The increase in risk for each system shall meet the following criteria in order for it to be considered insignificant:

- Change in Core Damage Frequency (ΔCDF) < 1E-07/year, and
- Change in Large Early Release Frequency (ΔLERF) < 1E-08/year.

The total change for all systems must meet the criteria of RG 1.174 for an acceptably small change in risk which are as follows:

- Change in Core Damage Frequency (ΔCDF) < 1E-06/year, and
- Change in Large Early Release Frequency (ΔLERF) < 1E-07/year.

If the scope of the RI-ISI program encompasses all Class 1 welds, the system level criteria shall be met. If the acceptance criteria cannot be met, additional inspections shall be added to the RI-ISI program until an acceptable change in risk is achieved.

3 TECHNICAL EVALUATION

3.1 BACKGROUND

Since its beginning, ASME B&PV Code, Section XI (Reference 1) has required inspections of weld areas of reactor vessels and other pressure-retaining nuclear system components. The selection of inspection locations was based on areas known to have high-service factors and additional areas to provide a representative sampling for the condition of pressure-retaining nuclear system components.

Applicable Weld Configurations

Depending on the manufacturer of the reactor vessel and designer of the plant, the configurations of welds joining the reactor vessel nozzles to the piping vary. Some reactor vessels were fabricated with a safe-end welded to the nozzle. Depending on whether the reactor coolant main loop piping is stainless steel or low-alloy steel, a dissimilar metal weld (Catgeory B-F) or a similar metal weld (Category B-J) joins the safe-end to the nozzle. A similar metal weld (Category B-J) then joins the safe-end to the piping. For plants that do not have a safe-end, a single weld joins the nozzle to the piping. For plants with stainless steel reactor coolant main loop piping, this is a dissimilar metal weld (Category B-F) whereas it is a similar metal weld (Category B-J) for plants with low-alloy steel piping. These configurations are shown in detail in Figure 3-3 in Section 3.2.3. For plants with no safe-end, this evaluation was limited to the single nozzle-to-pipe weld. For plants with safe-ends, this evaluation addresses both the nozzle-to-safe-end weld and the safe-end-to-pipe weld.

Examination Approaches

The preceding discussion of RV nozzle welds addresses the Category B-F and B-J welds of Table IWB-2500-1 of Section XI. The ultrasonic examination (UT) of these RV nozzle and piping welds, prior to the 1996 Addenda of Section XI, was conducted in accordance with Appendix I, I-2220. The 1996 Addenda and later editions/addenda require Appendix VIII inspections for welds in piping. The inspection volume for these welds is shown in Section XI, Figure IWB-2500-8 and requires inspection of the inner 1/3 of the weld thickness.

Service Experience

There has been no report of structural failure or leakage from any full-penetration weld being addressed in this report in a PWR RV nozzle, globally. In volumetric examinations of these welds via ISI performed in accordance with the requirements of Section XI, flaws identified in the original construction have been detected and were acceptable under Section XI requirements. These flaws have been monitored and to date, no growth has been identified. There also has been no evidence to date of inservice flaw initiation in these welds.

Location-Specific ISI Data from Participating Plants

While it is known that the number of flaws found in RV nozzle welds is very small, it is important to relate their number to the number of welds that have been examined over the past 30+ years with no evidence of the development of service-induced flaws.

To develop location-specific ISI data from nuclear plants, ISI data on the RV nozzle and piping weld categories noted above were gathered in a survey. The response to this survey is summarized in Table 3-1.

Table 3-1 Summary of Survey Results on RV Nozzle ISI Findings

# of Plant Inspections	# of RV Nozzles	# of Recordable Indications	# of Reportable Indications[1]
19	94	5	0

Notes:
1. Defined as an indication that does not meet the ASME Section XI acceptance standards of IWB-3514

3.2 ISI INTERVAL EXTENSION METHODOLOGY

The ISI interval extension methodology is primarily based on a risk analysis, including a PFM analysis of the effect of different inspection intervals on the frequency of failure. The quantitative change-in-risk assessment discussed below shows that extending the inspection interval from 10 to a maximum of 20 years has an acceptably small effect on the change in core damage frequency (ΔCDF) and large early release frequency (ΔLERF) per the guidelines of RG 1.174 (Reference 3). A summary of the RG 1.174 methodology and requirements is provided for information in Section 3.2.1.The ISI interval extension methodology that was developed was then applied to two pilot plants. The pilot plants utilized for the risk evaluations summarized in this report were FirstEnergy Nuclear Operating Company's (FENOC's) Beaver Valley Unit 1 (BV1) and Exelon Corporation's Three Mile Island (TMI) Unit 1.

3.2.1 Risk-Informed Regulatory Guide 1.174 Methodology

The NRC has developed a risk-informed regulatory framework. The NRC definition of risk-informed regulation is: "insights derived from probabilistic risk assessments are used in combination with deterministic system and engineering analysis to focus licensee and regulatory attention on issues commensurate with their importance to safety."

The NRC issued RG 1.174, *An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Current Licensing Basis* (Reference 3), to allow licensees to take advantage of this regulatory framework. In addition, the NRC issued application-specific RGs and Standard Review Plans (SRPs):

- RG-1.175 (Reference 10) and SRP Chapter 3.9.7, related to inservice testing (IST) programs,
- RG-1.176 (Reference 11) related to Graded Quality Assurance (GQA) programs,
- RG-1.177 (Reference 12) and SRP Chapter 16.1, related to Technical Specifications,
- RG-1.178 (Reference 13) and SRP-3.9.8, related to piping ISI programs.

These RGs and SRP chapters provide guidance in their respective application-specific subject areas to reactor licensees and the NRC staff regarding the submittal and review of risk-informed proposals that would change the licensing basis for a power reactor facility.

Regulatory Guide 1.174 Basic Steps

The approach described in RG 1.174 was used in each of the application-specific RGs/SRPs, and has 4 basic steps as shown in Figure 3-1. The four basic steps are discussed below.

Step 1: Define the Proposed Change

This element includes identifying:

1. Those aspects of the plant's licensing bases that may be affected by the change.

2. All systems, structures, and components (SSCs), procedures, and activities that are covered by the change and consider the original reasons for inclusion of each program requirement.

3. Any engineering studies, methods, codes, applicable plant-specific and industry data and operational experience, PRA findings, and research and analysis results relevant to the proposed change.

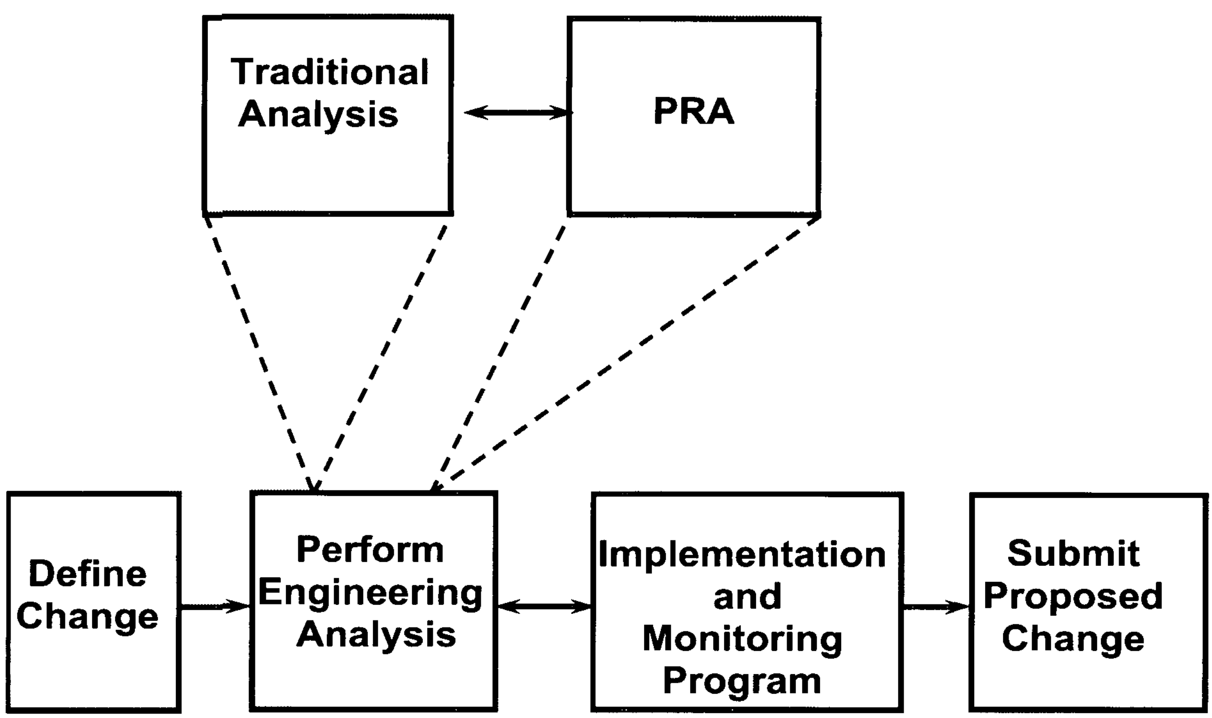

Figure 3-1 Basic Steps in (Principle Elements of) Risk-Informed, Plant-Specific Decision Making (from NRC RG 1.174)

Step 2: Perform Engineering Analysis

This element includes performing the evaluation to show that the fundamental safety principles on which the plant design was based are not compromised (defense-in-depth attributes are maintained) and that sufficient safety margins are maintained. The engineering analysis includes both traditional deterministic analysis and probabilistic risk assessment (PRA). The evaluation of risk effect should also assess the expected change in CDF and LERF, including a treatment of uncertainties. The results from the traditional analysis and the PRA must be considered in an integrated manner when making a decision.

Step 3: Define Implementation and Monitoring Program

This element's goal is to assess SSC performance under the proposed change by establishing performance monitoring strategies to confirm assumptions and analyses that were conducted to justify the change. This is to ensure that no unexpected adverse safety degradation occurs because of the changes. Decisions concerning implementation of changes should be made in light of the uncertainty associated with the results of the evaluation. A monitoring program should have measurable parameters, objective criteria, and parameters that provide an early indication of problems before becoming a safety concern. In addition, the monitoring program should include a cause determination and corrective action plan.

Step 4: Submit Proposed Change

This element includes:

1. Carefully reviewing the proposed change in order to determine the appropriate form of the change request.

2. Assuring that information required by the relevant regulation(s) in support of the request is developed.

3. Preparing and submitting the request in accordance with relevant procedural requirements.

Regulatory Guide 1.174 Fundamental Safety Principles

Five fundamental safety principles that each application for a change must meet are described. These are shown in Figure 3-2, and are discussed below.

Principle 1: Change Meets Current Regulations Unless it is Explicitly Related to a Requested Exemption or Rule Change

The proposed change is evaluated against the current regulations (including the general design criteria) to identify either where changes are proposed to the current regulations (e.g., Technical Specification, license conditions, and FSAR), or where additional information may be required to meet the current regulations.

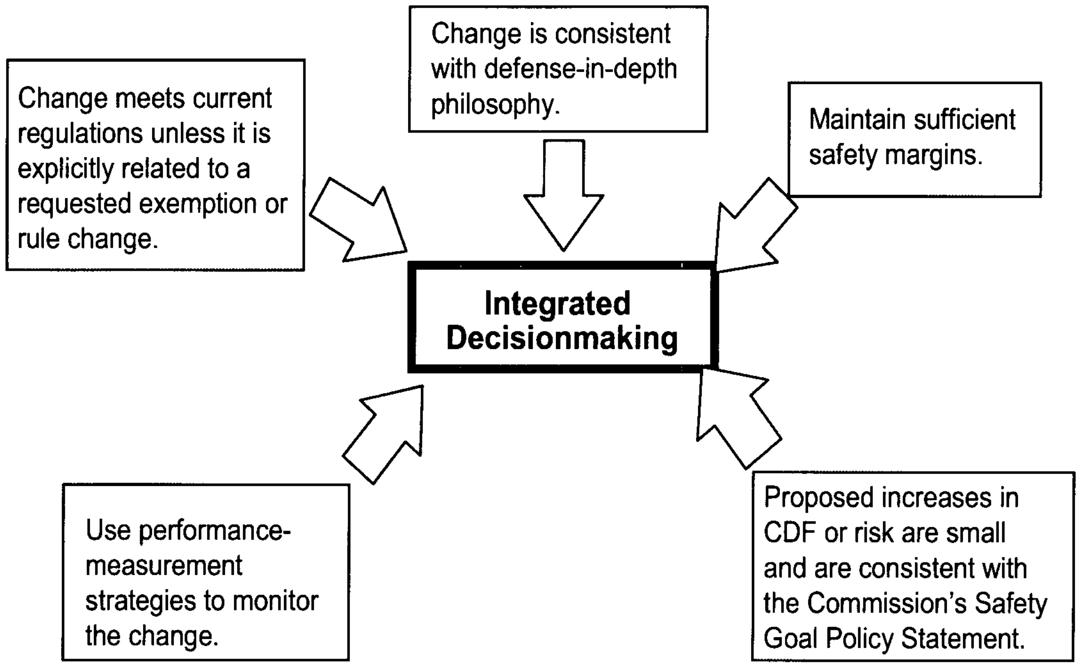

Figure 3-2 Principles of Risk-Informed Regulation (from NRC RG 1.174)

Principle 2: Change is Consistent with Defense-in-Depth Philosophy

Defense-in-depth has traditionally been applied in reactor design and operation to provide multiple means to accomplish safety functions and prevent the release of radioactive material. As defined in RG 1.174 (Reference 3), defense-in-depth is maintained by assuring that:

- A reasonable balance among prevention of core damage, prevention of containment failure, and consequence mitigation is preserved.
- Over-reliance on programmatic activities to compensate for weaknesses in plant design is avoided.
- System redundancy, independence, and diversity are preserved commensurate with the expected frequency and consequences to the system (e.g., no risk outliers).
- Defenses against potential common cause failures are preserved and the potential for introduction of new common cause failure mechanisms is assessed.
- Independence of barriers is not degraded (the barriers are identified as the fuel cladding, reactor coolant pressure boundary, and containment structure).
- Defenses against human errors are preserved.

Defense-in-depth philosophy is not expected to change unless:

- A significant increase in the existing challenges to the integrity of the barriers occurs.
- The probability of failure of each barrier changes significantly.
- New or additional failure dependencies are introduced that increase the likelihood of failure compared to the existing conditions.
- The overall redundancy and diversity in the barriers changes.

Principle 3: Maintain Sufficient Safety Margins

Safety margins must also be maintained. As described in RG 1.174, sufficient safety margins are maintained by assuring that:

- Codes and standards, or alternatives proposed for use by the NRC, are met.
- Safety analysis acceptance criteria in the licensing basis (e.g., FSARs, supporting analyses) are met, or proposed revisions provide sufficient margin to account for analysis and data uncertainty.

Principle 4: Proposed Increases in CDF or Risk are Small and are Consistent with the Commission's Safety Goal Policy Statement

To evaluate the proposed change with regard to a possible increase in risk, the risk assessment should be of sufficient quality to evaluate the change. The expected change in CDF and LERF are evaluated to address this principle. An assessment of the uncertainties associated with the evaluation is conducted. Additional qualitative assessments are also performed.

There are two acceptance guidelines, one for CDF and one for LERF, both of which should be used.

The guidelines for CDF are:

- If the application can be clearly shown to result in a decrease in CDF, the change will be considered to have satisfied the relevant principle of risk-informed regulation with respect to CDF.
- When the calculated increase in CDF is very small, which is taken as being less than 10^{-6} per reactor year, the change will be considered regardless of whether there is a calculation of the total CDF.
- When the calculated increase in CDF is in the range of 10^{-6} per reactor year to 10^{-5} per reactor year, applications will be considered only if it can be reasonably shown that the total CDF is less than 10^{-4} per reactor year.

- Applications that result in increases to CDF above 10^{-5} per reactor year would not normally be considered.

The guidelines for LERF are:

- If the application can be clearly shown to result in a decrease in LERF, the change will be considered to have satisfied the relevant principle of risk-informed regulation with respect to LERF.

- When the calculated increase in LERF is very small, which is taken as being less than 10^{-7} per reactor year, the change will be considered regardless of whether there is a calculation of the total LERF.

- When the calculated increase in LERF is in the range of 10^{-7} per reactor year to 10^{-6} per reactor year, applications will be considered only if it can be reasonably shown that the total LERF is less than 10^{-5} per reactor year.

- Applications that result in increases to LERF above 10^{-6} per reactor year would not normally be considered.

These guidelines are intended to provide assurance that proposed increases in CDF and LERF are small and are consistent with the intent of the Commission's Safety Goal Policy Statement.

<u>Principle 5: Use Performance-Measurement Strategies to Monitor the Change</u>

Performance-based implementation and monitoring strategies are also addressed as part of the key elements of the evaluation as described previously.

Risk-Acceptance Criteria for Analysis

For the purposes of this bounding analysis of the risk effect of the proposed extension in the RV nozzle weld inspection interval, the following criteria are applied with respect to Principle 4 (small change in risk):

- Change in CDF $< 1 \times 10^{-6}$ per reactor year,
- Change in LERF $< 1 \times 10^{-7}$ per reactor year.

These values are selected so that the proposed change may be later considered on a plant-specific basis regardless of the plant's baseline CDF and LERF.

3.2.2 Failure Modes and Effects

Failure Modes

Failure is defined for the purposes of this study as a leak rate large enough to result in a loss-of-coolant accident (LOCA) within the RV nozzle-to-safe-end and safe-end-to-pipe welds. There are three different failure modes, or leak rates, defined for this study and they are a small, medium, and large LOCA. These failure modes are defined in Table 3-2. The degradation mechanism of concern was thermal fatigue crack growth due to typical plant operation. The mechanism for failure is growth of an existing undetected fabrication flaw in the RV nozzle weld until it results in one of the LOCA leak rates identified in Table 3-2 or growth to the critical size that would lead to ductile rupture if a design limiting event, such as a seismic event, were to occur.

Table 3-2 Failure Modes

Failure Mode	Acronym	Leak Rate (GPM)
Small Loss-of-Coolant Accident	SLOCA	100
Medium Loss-of-Coolant Accident	MLOCA	1500
Large Loss-of-Coolant Accident	LLOCA	5000

Failure Effects

A LOCA due to piping failure was considered to result in core damage and a large early release. The failure modes specified in Table 3-2 correspond to leak rates for initiating events that are typically evaluated in the plant probabilistic risk assessment (PRA) model per NUREG/CR-4550 (Reference 15) and are considered to represent the spectrum of risk from failure (leakage) of the weld locations evaluated in this report.

3.2.3 Change-in-Failure-Frequency Calculations

A probabilistic fracture mechanics (PFM) methodology was used because it allows the consideration of distributions representing the uncertainties in key parameters, such as flaw size, material strength, crack growth rate, applied stresses, and the effectiveness of inspections. The PFM methodology also provides the failure frequency (probability per year) due to a given loading condition and a prescribed inspection interval.

The change-in-failure-frequency calculations for this study were performed using the Westinghouse Structural Reliability and Risk Assessment (SRRA) Code. The SRRA Code was developed for estimating piping failure probabilities to be used in relative risk-ranking of piping segments and for calculating the change in risk due to the different inspection schedules for the PWROG methodology for risk-informed inservice inspection (RI-ISI) of piping (Reference 4). Furthermore, as stated in the NRC's Safety Evaluation Report (SER) for the SRRA methodology (Reference 14), the program is consistent with the "state of the art" for calculating piping failure probabilities.

The SRRA Code has been used for estimation of failure probabilities in other ASME Code Cases and NRC-approved applications that have involved the reduction or relaxation of inservice inspection requirements. These ASME Code Cases and NRC approved applications include:

- WCAP-15666-A Revision 1, "Extension of Reactor Coolant Pump Motor Flywheel Examination" (Reference 5)

- ASME Code Case N-648-1, Alternative Requirements for Inner Radius Examinations of Class 1 Reactor Vessel Nozzles (Reference 16)

- ASME Code Case N-706-1, Alternative Examination Requirements of Table IWB-2500-1 and Table IWC-2500-1 for PWR Stainless Steel Residual and Regenerative Heat Exchangers (Reference 17)

The SRRA code for piping RI-ISI was developed in response to the NRC requirements for PFM calculations in a 1999 Draft Report, NUREG-1661 (Reference 18). These requirements included those for the initial flaw depth and its uncertainty, flaw density, and the effects of ISI. These same types of requirements for evaluating structural failure probabilities and candidate inspection programs were reevaluated in a recent study by NRC contractors at Pacific Northwest National Laboratories (PNNL). The results of this study, which are documented in a 2009 report, NUREG/CR-6986 (Reference 19), did not change any of the 1999 requirements used in developing the SRRA code for piping RI-ISI. This SRRA Code version has already been used in the past to calculate the failure probabilities of the piping-to-component dissimilar metal welds (Types A and C in Figure 3-3) and piping-to-safe-end welds (Types B and D in Figure 3-3) in a number of RI-ISI Programs. With the exclusion of the Alloy 82/182 welds that are susceptible to primary water stress corrosion cracking, there is no technical reason to preclude the application of the piping SRRA Code to the similar component-to-safe-end welds in the Type B and D configurations that are also evaluated in this RI-ISI Program.

In the previous piping RI-ISI Programs, the SRRA Code was used to calculate the change in failure probabilities and the associated change in risk for locations selected for an ASME Section XI ISI every 10 years relative to those with no ISI or a 10-year ISI at other locations. The same approach is also used in this particular risk-informed application of the SRRA Code, where it is used to calculate the change in failure probabilities and the associated change in risk for the locations shown in Figure 3-3 for an ASME Section XI ISI every 10 years relative to the same locations with ISI every 20 years.

Method

The first step was to review the nozzle and weld geometries and determine similarities between the nozzles of different plants. Based on these similarities, nozzles could be grouped and one set of runs could be performed for each grouping, rather than each plant individually. After reviewing fabrication drawings, the RV nozzles of the participating plants (as identified in Table 4-1 in Section 4) were categorized into four different types based on their weld configuration. These configurations can be seen below in Figure 3-3. Type A is typical for RV nozzles in Babcock and Wilcox Nuclear Steam Supply System (NSSS) designs. Type B is typical for RV nozzles in Combustion Engineering NSSS designs. Type C and D are applicable for RV nozzles in Westinghouse NSSS designs.

Based on the nozzle types identified in Figure 3-3, geometric data, and operating conditions of the participating plants, run groups were determined where each group could be evaluated by a single set of SRRA runs. Since each weld may join two different thicknesses (nozzle and pipe), or the nozzle type may contain 2 welds and three different thicknesses (nozzle, safe-end, and pipe), the objective was to determine a single run group that could provide a bounding change in failure frequency for all of the welds for each nozzle type.

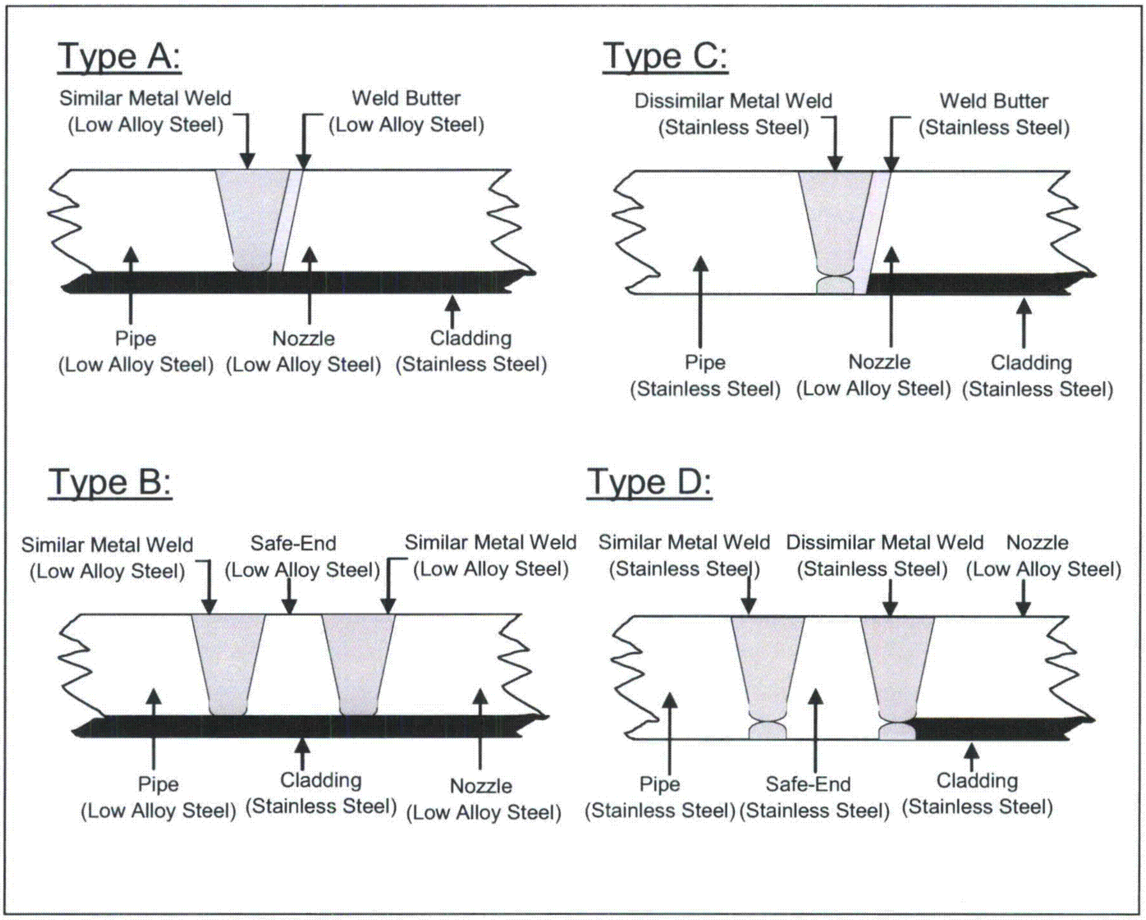

Figure 3-3 Nozzle Weld Configuration Types

The SRRA Code was used to calculate piping failure probabilities for a 60-year lifetime to correspond to a period of extended operation. Probabilities were calculated for the three different failure modes, or leak rates, shown in Table 3-2. The SRRA Code calculates and reports the cumulative failure probability for each year up to the input 60 years. For each combination of inputs, two cases are evaluated. One case considers inservice inspection performed on a 10-year interval while the other considers inspection performed on a 20-year interval. The difference in failure probabilities, output by the SRRA Code, is calculated by taking the difference between the 20-year-interval case and the 10-year-interval case at both 40 years and 60 years. This difference in failure probability is then converted to a change in failure frequency by dividing the difference in failure probability by the respective number of years, 40 or 60.

Initially, two sets of two runs (four runs) were made for each run group using the MLOCA failure mode during normal operation without credit for leak detection capability. Each set of runs consisted of a run using a 10-year ISI interval and a run using a 20-year ISI interval. For each set, one run was made at the highest temperature for the run group and one run was made for the lowest temperature of the run group.

It was expected that variation in failure frequencies between the SLOCA, MLOCA, and LLOCA failure modes would be small and that the MLOCA failure mode could be used to determine the relative importance of the different run groupings. This expectation was confirmed in subsequent evaluations.

Based on the results of the MLOCA runs, the limiting run group was determined for each nozzle type. For each of the limiting run groups, further SRRA runs were performed to determine the failure frequencies for the SLOCA and LLOCA failure modes during normal operation. These runs confirmed the expectation that there would be very little difference between the failure frequencies for the three different LOCA failure modes. Additional runs were then performed for the following off-normal conditions, with the calculated frequencies adjusted by the probability of the condition occurring in any one year of operation:

- Seismic Event (safe shutdown earthquake or SSE),
- Snubber Locking (during heat up or cool down),
- Seismic Event with Snubber Failure (not locking).

These off-normal runs, along with the normal operation runs, were all performed without leak detection. The decision to not perform the runs with leak detection was made so that the change in failure frequency between the 10- and 20-year intervals could be maximized. If leak detection were credited, the change in risk would have been minimized by the effects of leak detection.

The most limiting results for each of the LOCA failure modes and operating conditions were then determined by comparing the change in failure frequency for each run condition.

Since the risk-informed inservice inspection programs for piping use failure frequencies that are calculated with consideration of leak detection for the change-in-risk evaluations, additional runs were performed for the limiting conditions that considered leak detection.

The process described above is shown graphically in Figure 3-4, which also shows the intended uses for the bounding change in failure frequencies in following steps.

Inputs

The inputs to the SRRA Code are identified and discussed in detail in Reference 14. The input median values that were used in the calculation of the RV nozzle weld failure frequencies are discussed below.

- The geometry and temperature inputs to the SRRA Code were selected based on a review of the plant-specific records. These inputs included the inside diameter, outside diameter, and thickness. Likewise, plant-specific records were reviewed to determine the normal RV inlet and outlet operating temperatures.

- An operating pressure of 2.25 ksi was used for all cases.

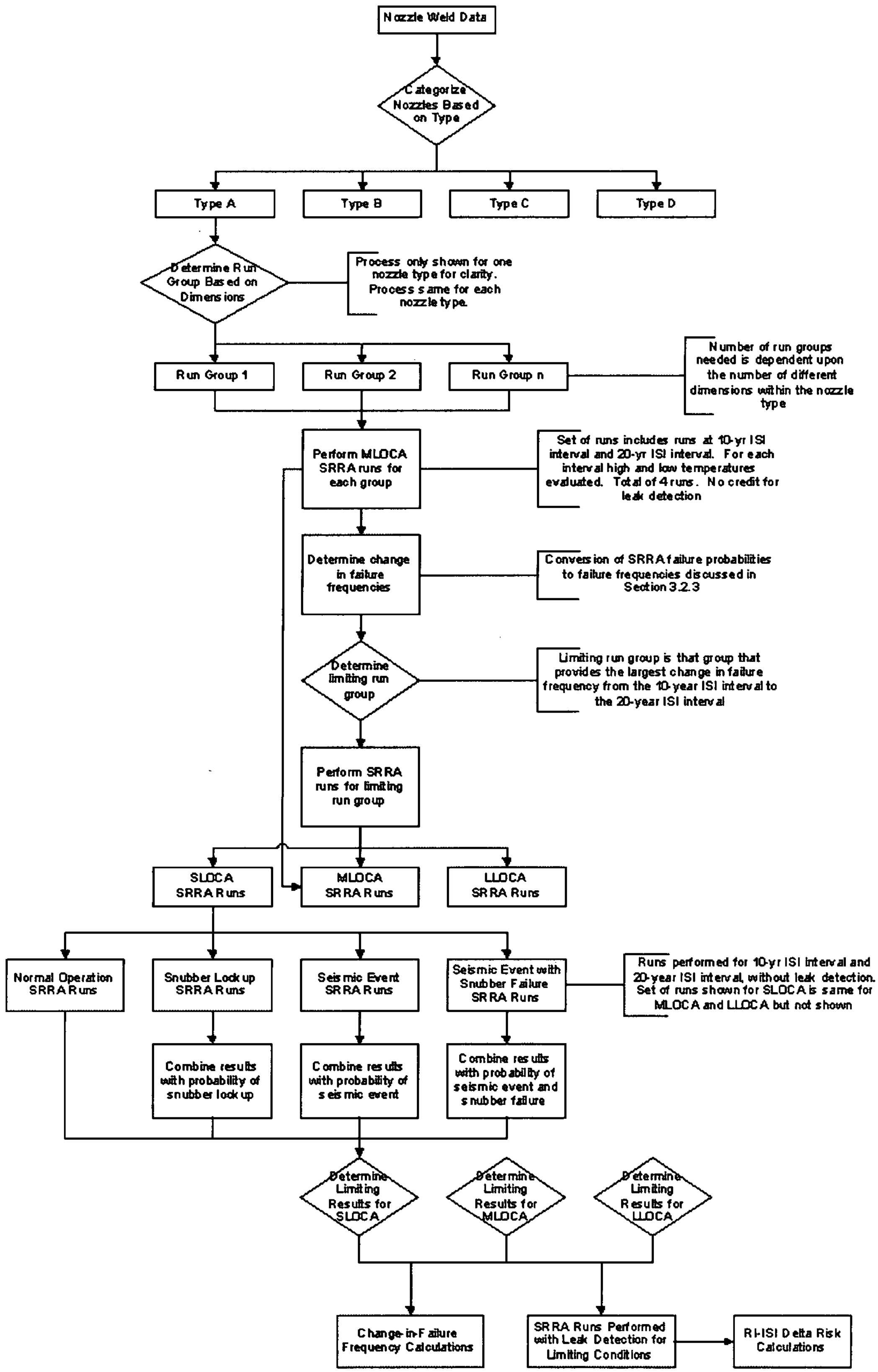

Figure 3-4 SRRA Run Process Flowchart

- Operating stress and other SRRA input values are consistent with those developed by the engineering teams for 19 U.S. plants and 10 other plants that used the PWROG Method for piping RI-ISI. These inputs are based on a combination of design stress analysis results and engineering insights. The stress input values are in terms of a fraction of the material flow stress. The material flow stress is dependent on temperature and the values used in the SRRA Code are included in Table 3-3 of Supplement 1 of the RI-ISI WCAP Report (Reference 14).
  - A high value of 0.17 was used for the deadweight and thermal stress level based on the high normal operating temperatures of these welds.
  - The following input values were used for the fatigue stress range:
    - A low value of 0.30 for heat up and cool down (Nozzle Types A and B),
    - A medium value of 0.50 for heat up and cool down of dissimilar metal welds (Nozzle Types C and D),
    - A high value of 0.70 for snubber locking (All Nozzle Types).
  - The following input values were used for the design limiting stress (primary stresses only):
    - A low value of 0.10 for normal operation,
    - A medium value of 0.26 for SSE,
    - A high value of 0.42 for SSE with failure of snubbers to lock.
- The low cycle fatigue frequency was set to 5 cycles per year. This is conservative based on the fatigue cycle count information that has been compiled on a plant-specific basis as part of the license renewal application process.
- Material Wastage Potential, Stress Corrosion Potential, and Vibratory Stress Range inputs were all set to zero since there is no service experience to indicate that these are degradation mechanisms that should be considered for these nozzle weld types.
- The snubber failure probability used in the evaluation was 0.1 and the seismic event (SSE) probability used was 0.001. As stated in the safety evaluation report for the SRRA Code (Reference 14), these values are conservative.
- The minimum leak detection rate was 1 gallon per minute per typical plant technical specifications.
- The initial flaw conditions contained in the SRRA Code, including the median flaw depth and its uncertainty and the flaw density are consistent with Figure 4.1 and Table 4.1 of Draft NUREG-1661 (Reference 18). Furthermore, these values are the same as those shown in Figures 2.13 and 2.15 of NUREG/CR-6986 (Reference 19). An input value of either X-ray NDE

or One Flaw was used. Either input value results in one flaw simulated per weld. All flaws are surface breaking and circumferentially oriented.

- The probability of detection curves used in the SRRA Code, for carbon and stainless steel, are consistent with those in NUREG/CR-6986 (Reference 19) but are adjusted based on the SRRA ISI accuracy input. This input corresponds to the ratio of crack depth to wall thickness that provides a 50% probability of detecting or not detecting the flaw. The input value, which was used for ultrasonic examination (UT) in the PWROG RI-ISI pilot plant application, and has been used in subsequent PWROG RI-ISI applications, was 0.24.

Results

The resulting bounding change in failure frequencies for each weld type are shown in Tables 3-3, 3-4, 3-5, and 3-6 for weld types A, B, C, and D, respectively. This information can be used to perform plant-specific change-in-risk calculations for extending the RV nozzle weld inspection interval from 10 to 20 years.

Table 3-3 Type A Bounding Change in Failure Frequencies (/year)

Results for	Failure Mode	Without Leak Detection	With Leak Detection
Outlet Nozzle – 40 Year	SLOCA	5.90E-10	2.84E-11
	MLOCA	1.80E-11	6.90E-12
	LLOCA	8.13E-12	2.17E-12
Outlet Nozzle – 60 Year	SLOCA	3.93E-10	1.89E-11
	MLOCA	1.20E-11	4.60E-12
	LLOCA	5.42E-12	1.45E-12
Inlet Nozzle – 40 Year	SLOCA	2.96E-10	1.34E-11
	MLOCA	7.87E-12	1.50E-12
	LLOCA	7.77E-12	1.39E-12
Inlet Nozzle – 60 Year	SLOCA	1.97E-10	8.93E-12
	MLOCA	6.32E-12	1.00E-12
	LLOCA	5.84E-12	9.29E-13

Table 3-4 Type B Bounding Change in Failure Frequencies (/year)			
Results for	Failure Mode	Without Leak Detection	With Leak Detection
Outlet Nozzle – 40 Year	SLOCA	5.36E-10	2.85E-11
	MLOCA	1.97E-11	8.09E-12
	LLOCA	1.97E-11	7.45E-12
Outlet Nozzle – 60 Year	SLOCA	3.57E-10	1.90E-11
	MLOCA	1.31E-11	5.39E-12
	LLOCA	1.31E-11	4.97E-12
Inlet Nozzle – 40 Year	SLOCA	5.28E-10	4.77E-11
	MLOCA	2.07E-11	6.10E-12
	LLOCA	1.99E-11	1.47E-12
Inlet Nozzle – 60 Year	SLOCA	3.52E-10	3.18E-11
	MLOCA	1.45E-11	4.07E-12
	LLOCA	1.40E-11	9.78E-13

Table 3-5 Type C Bounding Change in Failure Frequencies (/year)			
Results for	Failure Mode	Without Leak Detection	With Leak Detection
Outlet Nozzle – 40 Year	SLOCA	6.71E-08	4.49E-09
	MLOCA	6.68E-08	3.16E-09
	LLOCA	6.68E-08	3.04E-09
Outlet Nozzle – 60 Year	SLOCA	1.18E-07	3.96E-09
	MLOCA	1.18E-07	2.66E-09
	LLOCA	1.18E-07	2.58E-09
Inlet Nozzle – 40 Year	SLOCA	7.88E-08	3.52E-09
	MLOCA	7.54E-08	1.52E-09
	LLOCA	7.45E-08	1.40E-09
Inlet Nozzle – 60 Year	SLOCA	1.13E-07	3.61E-09
	MLOCA	1.23E-07	1.75E-09
	LLOCA	1.23E-07	1.67E-09

Table 3-6 Type D Bounding Change in Failure Frequencies (/year)			
Results for	**Failure Mode**	**Without Leak Detection**	**With Leak Detection**
Outlet Nozzle – 40 Year	SLOCA	7.09E-08	7.02E-09
	MLOCA	7.09E-08	7.08E-09
	LLOCA	7.03E-08	7.02E-09
Outlet Nozzle – 60 Year	SLOCA	1.55E-07	7.77E-09
	MLOCA	1.53E-07	1.07E-08
	LLOCA	1.52E-07	1.06E-08
Inlet Nozzle – 40 Year	SLOCA	8.74E-08	3.27E-08
	MLOCA	7.89E-08	2.60E-08
	LLOCA	7.83E-08	2.60E-08
Inlet Nozzle – 60 Year	SLOCA	2.02E-07	2.37E-08
	MLOCA	1.92E-07	1.89E-08
	LLOCA	1.91E-07	1.89E-08

One of the parameters in probabilistic fracture mechanics applications identified as a concern to the NRC was the probability of detection for ISI. The probability of detection (POD) is dependent on the size of flaw that is being investigated and therefore, POD data is typically expressed as a function of flaw size. In the case of the SRRA Code, the shape of the POD curve is defined in the detailed input, but its location is indexed by the "Crack Inspection Accuracy" parameter. This parameter establishes the point on the POD curve at which the crack depth, as a fraction of the wall thickness, equates to a 50% POD.

While an input value of 0.24 has been consistently used for ISI accuracy in application of the SRRA Code for RI-ISI piping inspections using ultrasonic testing, it was requested that a sensitivity study be performed for an increased accuracy of inspection. This was motivated by the industry implementation of ASME Section XI, Appendix VIII, inspection techniques that had been qualified via the Performance Demonstration Initiative (PDI). These techniques have been widely credited by industry and the NRC for providing better POD than previously used techniques. The purpose of the sensitivity studies was to determine the extent to which a change in inspection accuracy would affect the bounding change in nozzle weld failure frequency results provided in Tables 3-3 to 3-6.

Data was not available to justify a specific crack depth corresponding to a 50% POD, so a depth of 1/10th of the wall thickness was used. The limiting run groups for the Type C and D nozzles were reevaluated for the normal operation and snubber locking cases at the MLOCA failure mode and 60 years of operation. A comparison of the results for the 0.10 inspection accuracy cases and the 0.24 inspection accuracy cases is shown in Tables 3-7 and 3-8. This comparison shows that the bounding failure frequencies in the Tables 3-3 through 3-6 are conservative.

Table 3-7 Sensitivity Run Comparison – Type C – MLOCA Failure Mode							
Results For	Failure Conditions	Wall Fraction	10 Years		20 Years		Δ (FF_{20} - FF_{10})
			Cumulative Probability	Failure Frequency	Cumulative Probability	Failure Frequency	
Outlet Nozzle – 60 Year	Normal Operation	0.24	1.75E-06	2.92E-08	8.81E-06	1.47E-07	1.18E-07
		0.1	3.95E-07	6.58E-09	1.63E-06	2.72E-08	2.06E-08
	Snubber Locking	0.24	3.34E-05	5.57E-07	6.70E-05	1.12E-06	5.60E-08
		0.1	6.02E-06	1.00E-07	3.18E-05	5.31E-07	4.30E-08
Inlet Nozzle – 60 Year	Normal Operation	0.24	9.17E-07	1.53E-08	8.30E-06	1.38E-07	1.23E-07
		0.1	3.79E-07	6.31E-09	1.32E-06	2.20E-08	1.56E-08
	Snubber Locking	0.24	3.59E-05	5.99E-07	9.42E-05	1.57E-06	9.71E-08
		0.1	9.94E-06	1.66E-07	4.89E-05	8.16E-07	6.50E-08

Table 3-8 Sensitivity Run Comparison – Type D – MLOCA Failure Mode							
Results For	Failure Conditions	Wall Fraction	10 Years		20 Years		Δ (FF_{20} - FF_{10})
			Cumulative Probability	Failure Frequency	Cumulative Probability	Failure Frequency	
Outlet Nozzle – 60 Year	Normal Operation	0.24	1.44E-06	2.40E-08	7.96E-06	1.33E-07	1.09E-07
		0.1	4.36E-07	7.26E-09	1.27E-06	2.12E-08	1.39E-08
	Snubber Locking	0.24	6.11E-05	1.02E-06	1.53E-04	2.55E-06	1.53E-07
		0.1	1.20E-05	2.00E-07	6.47E-05	1.08E-06	8.78E-08
Inlet Nozzle – 60 Year	Normal Operation	0.24	3.96E-06	6.60E-08	1.17E-05	1.95E-07	1.29E-07
		0.1	4.26E-07	7.10E-09	1.72E-06	2.86E-08	2.15E-08
	Snubber Locking	0.24	4.99E-05	8.32E-07	1.65E-04	2.75E-06	1.92E-07
		0.1	1.21E-05	2.01E-07	6.61E-05	1.10E-06	9.00E-08

3.2.4 Change-in-Risk Calculations

The objective of the change-in-risk assessment was to evaluate the change in core damage and large early release risk from the extension of the inservice inspection interval of the RV nozzle welds relative to other plant risk contributors through a qualitative and quantitative evaluation.

NRC RG 1.174 (Reference 3) provided the basis for this evaluation as well as the acceptance guidelines to make a change to the current licensing basis.

Risk was defined as the combination of likelihood of an event and severity of consequences of an event. Therefore, the following two questions were addressed.

- What was the likelihood of the event?
- What would the consequences be?

For the purposes of extending the ISI interval for the RV nozzle welds, the change in likelihood as a result of the ISI interval extension needs to be evaluated rather than the absolute values. The following sections describe the likelihood and postulated consequences and the changes as a result of the extension in ISI interval. The change in likelihood and the consequences were then combined in the change-in-risk calculation and the results are presented in this report.

What is the Likelihood of the Event?

As identified in Section 3.2.2, the event of concern is a loss-of-coolant accident (LOCA). The likelihood of this event, and the change in the likelihood of this event, was addressed by the calculations in Section 3.2.3. These calculations are summarized in the change-in-failure-frequency results in Tables 3-3, 3-4, 3-5, and 3-6.

What are the Consequences?

As discussed in Section 3.2.2, a LOCA was considered to result in core damage and a large early release. The failure modes specified in Table 3-2 correspond to leak rates for initiating events that are typically evaluated in the plant probabilistic risk assessment (PRA) model per NUREG/CR-4550 (Reference 15) and are considered to represent the spectrum of risk from failure (leakage) of the weld locations evaluated in this report. The likelihood of core damage and large early release, given a LOCA, can be quantified by the PRA in terms of the conditional core damage probability (CCDP) and large early release probability (CLERP), respectively.

Change-in-Risk Calculation Method

As discussed in Section 3.2.1, the change in failure frequency associated with the extension of the inservice inspection interval was calculated for three failure modes (leak rates): SLOCA, MLOCA, and LLOCA. The change in failure frequency is the difference in failure frequencies for the licensed life of the plant (40 or 60 years). This change in failure frequency for each of these failure modes was multiplied by the conditional core damage probability (CCDP) and conditional large early release probability (CLERP) for that particular failure mode to determine the change in core damage frequency (ΔCDF) and the change in large early release frequency (ΔLERF), respectively. The total change in CDF and change in LERF for the reactor vessel nozzles were determined by adding the results from all three failure modes and then multiplying by the number of RV nozzle welds examined. This calculation is shown graphically in Table 3-9.

Table 3-9 Change-in-Risk Calculations

Failure Mode	Bounding Change in Failure Frequency	CCDP	ΔCDF (/ year)	CLERP	ΔLERF (/ year)
SLOCA	ΔFF_{SLOCA}	$CCDP_{SLOCA}$	$= (\Delta FF_{SLOCA})(CCDP_{SLOCA})$	$CLERP_{SLOCA}$	$= (\Delta FF_{SLOCA})(CLERP_{SLOCA})$
MLOCA	ΔFF_{MLOCA}	$CCDP_{MLOCA}$	$= (\Delta FF_{MLOCA})(CCDP_{MLOCA})$	$CLERP_{MLOCA}$	$= (\Delta FF_{MLOCA})(CLERP_{MLOCA})$
LLOCA	ΔFF_{LLOCA}	$CCDP_{LLOCA}$	$= (\Delta FF_{LLOCA})(CCDP_{LLOCA})$	$CLERP_{LLOCA}$	$= (\Delta FF_{LLOCA})(CLERP_{LLOCA})$
	# (No.) of Welds Examined	Total ΔCDF	= (sum of above)(# of welds examined)	Total ΔLERF	= (sum of above)(# of welds examined)

The calculations in Table 3-9 would need to be performed for both the RV inlet and outlet nozzles. For the change-in-risk calculation, the bounding change in failure frequencies with or without credit for leak detection from Tables 3-3, 3-4, 3-5, or 3-6 shall be used. To determine the total change in risk, the totals determined in Table 3-9 would need to be summed together for both the RV inlet and outlet nozzles.

To determine the acceptability of the change in risk associated with the extension in the inservice inspection interval, the total ΔCDF and total ΔLERF without credit for leak detection are compared to the criteria in Regulatory Guide 1.174 for an acceptably small change in risk. These criteria were discussed previously in Section 3.2.1.

Pilot Plant Change-in-Risk Calculations

Beaver Valley Unit 1

Beaver Valley Unit 1 is a Westinghouse NSSS design and has Type C RV Nozzle welds. The 40-year bounding change in failure frequencies from Table 3-5, without credit for leak detection, were used along with plant-specific CCDP and CLERP values to determine the change in risk associated with the extension in inspection interval for Beaver Valley Unit 1. The results of the change-in-risk calculations are shown in Table 3-10.

Table 3-10 Change-in-Risk Calculations – Beaver Valley Unit 1						
Failure Mode	**Bounding Change in Failure Frequency (From Table 3-5, No Leak Detection)**		**CCDP**	**ΔCDF (/ year)**	**CLERP**	**ΔLERF (/ year)**
Outlet Nozzles						
SLOCA	6.71E-08		1.38E-05	9.26E-13	7.61E-12	5.11E-19
MLOCA	6.68E-08		1.68E-03	1.12E-10	4.70E-08	3.14E-15
LLOCA	6.68E-08		2.15E-03	1.44E-10	5.30E-08	3.54E-15
# of Welds Examined		3	Total ΔCDF	7.71E-10	Total ΔLERF	2.01E-14
Inlet Nozzles						
SLOCA	7.88E-08		1.93E-04	1.52E-11	2.90E-10	2.28E-17
MLOCA	7.53E-08		1.68E-03	1.26E-10	4.70E-08	3.54E-15
LLOCA	7.46E-08		2.15E-03	1.60E-10	5.30E-08	3.95E-15
# of Welds Examined		3	Total ΔCDF	9.05E-10	Total ΔLERF	2.25E-14
All Nozzles						
Total Change-in-Risk Results			Total ΔCDF	1.68E-09	Total ΔLERF	4.26E-14

<u>Three Mile Island Unit 1</u>

Three Mile Island Unit 1 is a B&W NSSS design and has Type A RV Nozzle welds. The 40-year bounding change in failure frequencies from Table 3-3, without credit for leak detection, were used along with plant-specific CCDP and CLERP values to determine the change in risk associated with the extension in inspection interval for Three Mile Island Unit 1. The results of the change-in-risk calculations are shown in Table 3-11.

Table 3-11 Change-in-Risk Calculations – Three Mile Island Unit 1					
Failure Mode	Bounding Change in Failure Frequency (From Table 3-3, No Leak Detection)	CCDP	ΔCDF (/ year)	CLERP	ΔLERF (/ year)
Outlet Nozzle					
SLOCA	5.90E-10	1.83E-03	1.08E-12	2.53E-04	1.49E-13
MLOCA	1.80E-11	2.23E-03	4.01E-14	2.55E-04	4.59E-15
LLOCA	8.13E-12	3.93E-02	3.20E-13	8.06E-04	6.55E-15
# of Welds Examined	2	Total ΔCDF	2.88E-12	Total ΔLERF	3.21E-13
Inlet Nozzle					
SLOCA	2.96E-10	1.83E-03	5.42E-13	2.53E-04	7.49E-14
MLOCA	7.87E-12	2.23E-03	1.75E-14	2.55E-04	2.01E-15
LLOCA	7.77E-12	3.93E-02	3.05E-13	8.06E-04	6.26E-15
# of Welds Examined	4	Total ΔCDF	3.46E-12	Total ΔLERF	3.33E-13
All Nozzles					
Total Change-in-Risk Results		Total ΔCDF	6.34E-12	Total ΔLERF	6.54E-13

Change-in-Risk Results and Conclusions

The analysis shown above demonstrates that changes in CDF and LERF as a result of the extension in ISI interval for the RV nozzle welds for Beaver Valley Unit 1 and Three Mile Island Unit 1 do not exceed the NRC's RG-1.174 (Reference 3) acceptance guidelines for a small change in CDF and LERF ($<10^{-6}$ per year for Total ΔCDF, $<10^{-7}$ per year for Total ΔLERF).

As part of this evaluation, the key principles identified in RG-1.174 and summarized in Section 3.2.1 were reviewed and the responses based on the evaluation are provided in Table 3-12.

This evaluation concluded that extension of the RV nozzle weld inservice inspection interval from 10 to 20 years would not be expected to result in an unacceptable increase in risk. Given this outcome, and the fact that other key principles listed in RG-1.174 continue to be met, the proposed change in inspection interval from 10 to 20 years is acceptable.

Table 3-12 Evaluation with Respect to Regulatory Guide 1.174 (Reference 3) Key Principles	
Key Principles	**Evaluation Response**
Change meets current regulations unless it is explicitly related to a requested exemption or rule change.	Change to current ASME Section XI requirements, endorsed in 10 CFR 50.55a is proposed.
Change is consistent with defense-in-depth philosophy.	NDE examinations still conducted, but on less frequent basis not to exceed 20 years. Other indications of potential degradation of RV nozzle welds are available (e.g., foreign experience, inspection of other similar locations, and periodic testing with visual examinations). See the discussion below for additional information on defense in depth.
Maintain sufficient safety margins.	No safety analysis margins are changed.
Proposed increases in CDF or risk are small and are consistent with the Commission's Safety Goal Policy Statement.	Proposed increase in risk is estimated to be acceptably small.
Use performance-measurement strategies to monitor the change.	NDE examinations still conducted, but on less frequent basis not to exceed 20 years. Other indications of potential degradation of RV nozzle welds are available (e.g., foreign experience, inspection of other similar locations, and periodic testing with visual examinations).

Defense-in-Depth

Extending the RV nozzle weld ISI interval does not imply that generic degradation mechanisms will be ignored for 20 years. (With the number of PWR nuclear power plants in operation in the U.S. and globally, a sampling of plants inevitably undergo examinations in a given year.) This provides for early detection of any potential emerging generic degradation mechanisms, and would permit the industry to react with more frequent examinations if needed. Furthermore, similar welds in other locations, such as steam generator or pump nozzles, operating at similar service conditions will continue to be inspected on a 10-year interval and will provide an indication of any emerging issues that could also affect the RV nozzle welds.

To demonstrate that there will be a sampling of inspections performed over the 20-year interval that will provide an indication of emerging issues, example implementation schedules were developed. This schedule is for the period from 2009 to 2048 and applies to plants with non-alloy 82/182 Category B-F and B-J welds. Since the RV nozzle weld inspections are performed at the same time as the RV inspections, the schedule is based on the schedule developed for the RV Weld ISI interval extension provided in PWR Owners Group Letter OG-09-454 (Reference 20). The schedule is based upon every plant identified in Table 4-1 implementing the 10-to-20-year interval extension for the inspection of RV nozzle welds. The schedule also includes plants with non-Alloy 82-182 welds that were not evaluated as part of this project who will continue to use a 10-year inspection schedule. This inspection schedule can be seen below in Table 3-13.

Table 3-13 Proposed Reactor Vessel Nozzle Weld Inspection Schedule							
Utility	**Plant Name**	**Weld Type**	**Current ISI Date**	**Proposed ISI Dates**			
AEP	D. C. Cook Unit 2	D	2009	2019	2039		
Constellation	Calvert Cliffs Unit 1(1)	B	2008	2008	2018	2028	2038
	Calvert Cliffs Unit 2(1)	B	2009	2009	2019	2029	2039
	R. E. Ginna(1)	C or D	2009	2009	2019	2029	2039
Dominion	Kewaunee	C	2014	2014	2034		
	Millstone Unit 2(2)	B	2008	2028	2048		
	North Anna Unit 1	C	2009	2009	2019	2029	2039
	North Anna Unit 2	C	2010	2010	2020	2030	2040
	Surry Unit 1(2)	C	2013	2023	2043		
	Surry Unit 2(2)	C	2014	2024	2044		
Duke	Catawba Unit 1(2)	D	2014	2024	2044		
	McGuire Unit 2(2)	B	2014	2024	2044		
	Oconee Unit 1(2)	A	2012	2012	2032		
	Oconee Unit 2(2)	A	2013	2013	2033		
	Oconee Unit 3(2)	A	2014	2014	2034		
Entergy	Palisades(2)	B	2006	2010	2030		
	ANO Unit 1	A	2018	2028	2048		
	ANO Unit 2	B	2009	2018	2038		
	Waterford Unit 3	B	2008	2015	2035		
Exelon	Three Mile Island Unit 1	A	2011	2015	2035		
FENOC	Beaver Valley Unit 1(2)	C	2017	2027	2047		
	Davis-Besse(2)	A	N/A	2012	2032		
FPL	Point Beach Unit 1(1)	C or D	2018	2010	2020	2030	2040
	Point Beach Unit 2(1)	C or D	2018	2009	2019	2029	2039
	St. Lucie Unit 1(2)	B	2018	2017	2037		
	St. Lucie Unit 2(2)	B	2010	2010	2030		
	Turkey Point Unit 3(2)	C	2014	2013	2033		
	Turkey Point Unit 4(2)	C	2015	2014	2034		
Progress	Crystal River Unit 3(1)	A	2017	2017	2027	2037	2047

Table 3-13 (cont.)	Proposed Reactor Vessel Nozzle Weld Inspection Schedule						
Utility	**Plant Name**	**Weld Type**	**Current ISI Date**	**Proposed ISI Dates**			
SCE	San Onofre 2	B	2012	2022	2042		
	San Onofre 3	B	2013	2023	2043		
TVA	Sequoyah Unit 1(2)	C	2006	2015	2035		
	Sequoyah Unit 2(2)	C	2015	2024	2044		
Xcel	Prairie Island Unit 1	D	2014	2012	2033		
	Prairie Island Unit 2	C	2013	2012	2034		

Notes:
1. Plant does not have Alloy 82/182 RV nozzle welds but is not participating in this project (See Table 4-1).
2. Based on available data, this plant must inspect the RV nozzle welds to meet the requirements of their RI-ISI or Section XI ISI program (i.e. another weld location may not be selected for inspection).

The distribution of the inspections by dates specified in Table 3-13 can be seen in Figure 3-5. Figure 3-6 displays the nozzle weld inspection distribution by weld category (B-F or B-J) from 2009 to 2048. These figures display the number of inspections possible following this schedule.

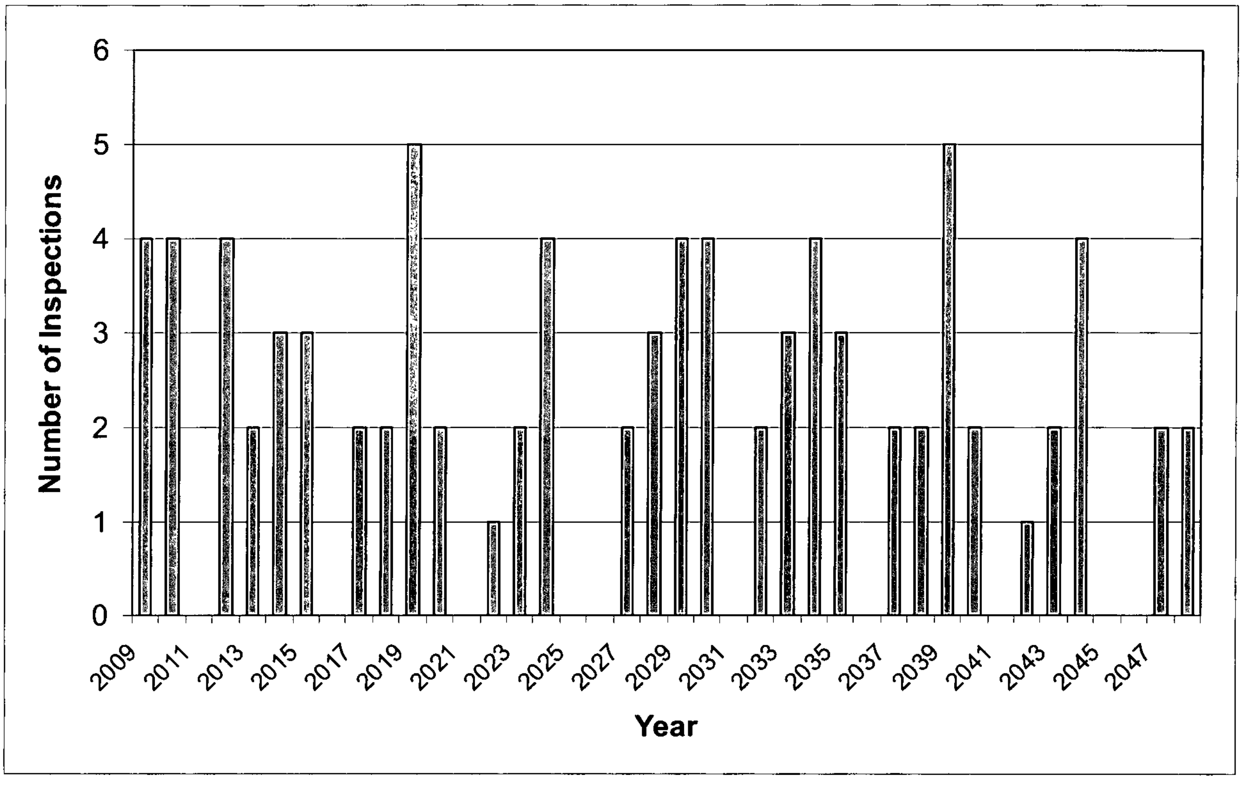

Figure 3-5 Number of Inspections per Year for Proposed Implementation Schedule

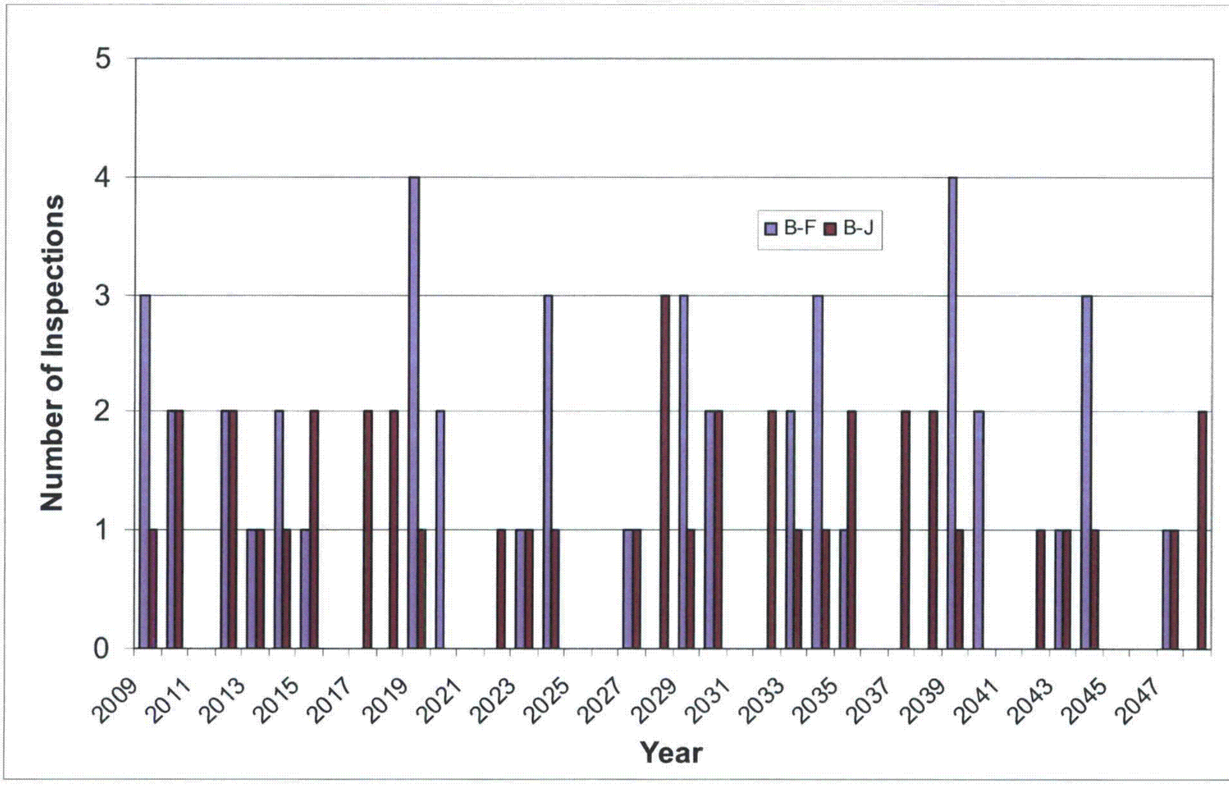

Figure 3-6 Number of Category B-F and B-J Inspections per Year for Proposed Implementation Schedule

Figures 3-7 and 3-8 display the number of inspections that may be performed for the non-Alloy 82/182 nozzle welds if it is assumed that all plants identified in Table 3-13 that have the ability to inspect another location for inspection within their RI-ISI program will do so. Plants that do not have this ability are assumed to implement the 10-to-20-year interval extension.

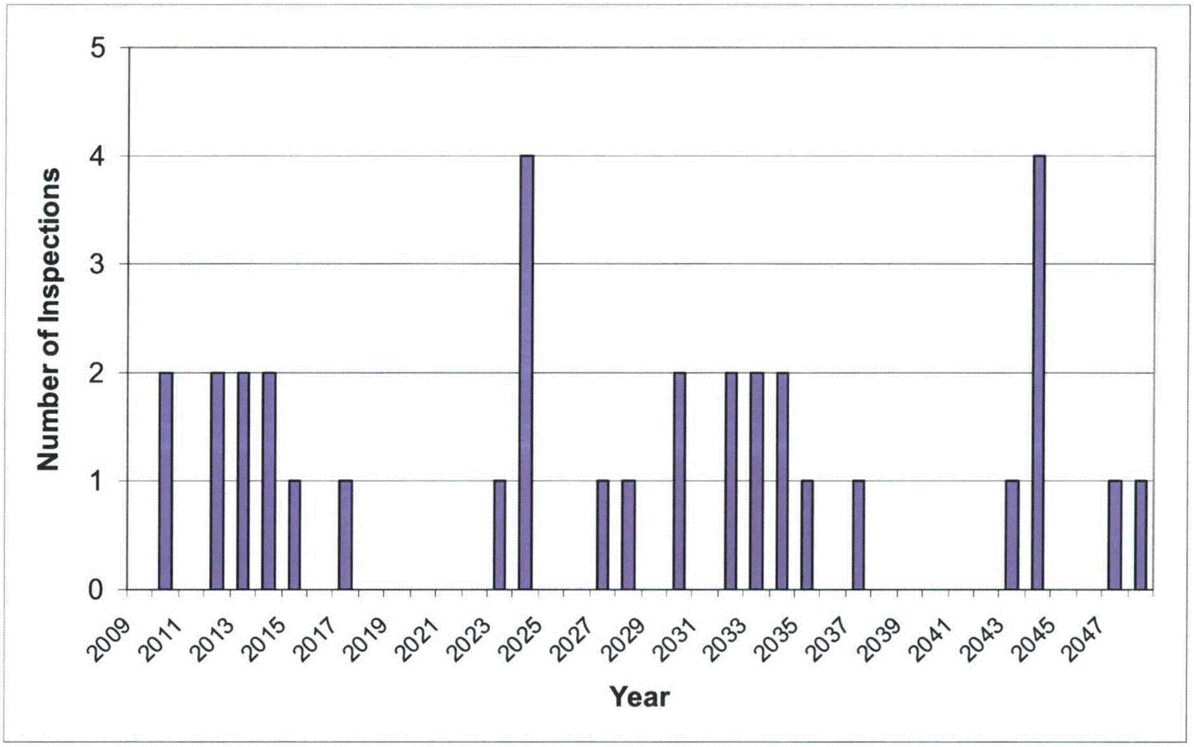

Figure 3-7 Number of Inspections per Year – Assuming Inspection of Alternative Locations

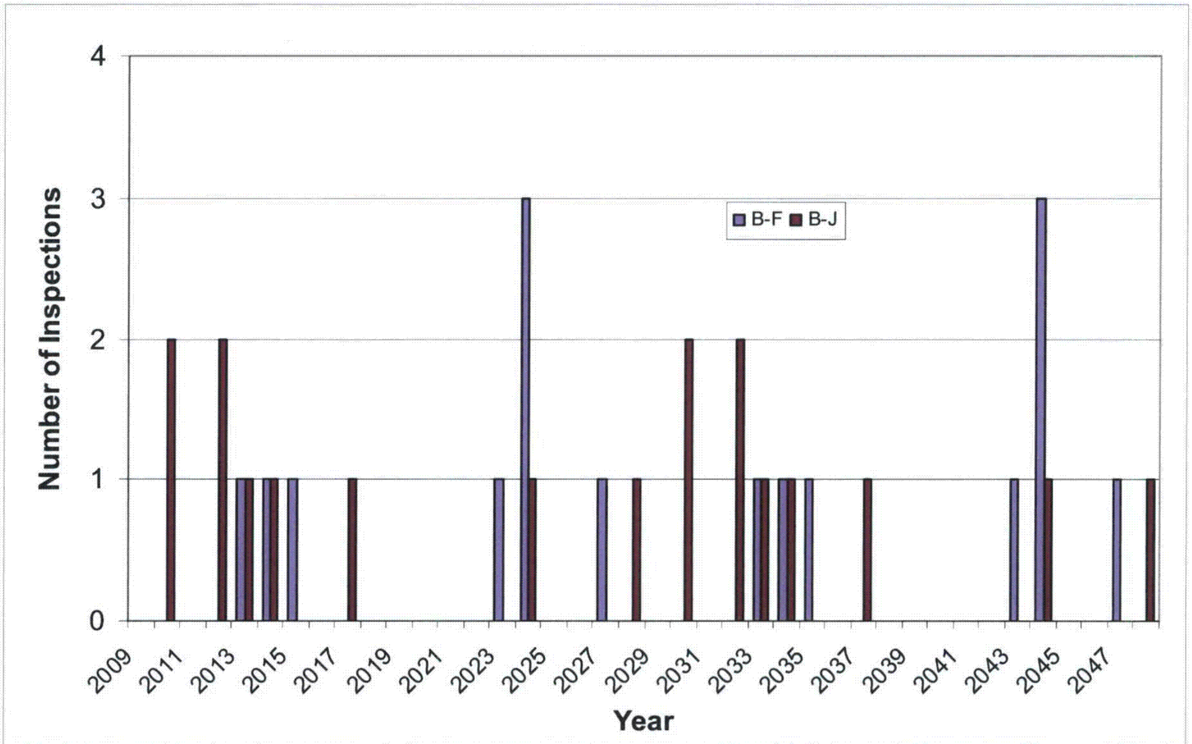

Figure 3-8 Number of Category B-F and B-J Inspections per Year – Assuming Inspection of Alternative Locations

Based on the data presented in Table 3-13 and Figures 3-5 and 3-6, an acceptable number of inspections for the RV nozzle welds can still be achieved by implementing the 20-year inspection schedule. Specifically, in the 40-year period evaluated, there are only five inspection gaps of one year and only two gaps of two years. As shown in Figures 3-7 and 3-8, if it is assumed that plants will select another location for inspection as an alternative to the RV nozzle weld, the total amount of RV nozzle weld inspections will be greatly reduced. Extension of the inspection interval from 10 to 20 years for RV nozzle welds would provide an effective means to maintain defense-in-depth in the form of a consistent and evenly distributed inspection schedule that provides for early detection of emerging degradation mechanisms.

It should also be recognized that all reactor coolant pressure boundary failures occurring to date have been identified as a result of leakage, and were discovered by visual examination. The proposed RV ISI interval extension does not alter the visual examination interval. The reactor vessel would undergo, as a minimum, the Section XI Examination Category B-P pressure tests and visual examinations conducted at the end of each refueling before plant start-up, as well as leak tests with visual examinations that precede each start-up following maintenance or repair activities.

Relative to Defense in Depth, Regulatory Guide 1.174 states that:

- "Defense-in-depth philosophy is not expected to change unless:
  - A significant increase in the existing challenges to the integrity of the barriers occurs.
  - The probability of failure of each barrier changes significantly.
  - New or additional failure dependencies are introduced that increase the likelihood of failure compared to the existing conditions.
  - The overall redundancy and diversity in the barriers changes."

The extension in inspection interval will not result in any of the changes identified above. Also identified in RG 1.174 and Section 3.2.1 are six elements for maintaining defense-in-depth. Due to the fact that the interval extension will not result in any of the changes identified above, it is expected that the defense in depth elements will not be affected. Additional assessment of the effect on each of the six defense-in-depth elements is provided below:

- A reasonable balance among prevention of core damage, prevention of containment failure, and consequence mitigation is preserved:

  The proposed increase in inspection would not cause an increased reliance on any of the identified elements. Therefore, the interval increase would not change the existing balance among prevention of core damage, prevention of containment failure, and consequence mitigation.

- Over-reliance on programmatic activities to compensate for weaknesses in plant design is avoided:

  The change in inspection interval does not change the robustness of the RV nozzle welds in any way. It is because of this robustness that the inspection interval can be doubled with no significant change in failure frequency.

- System redundancy, independence, and diversity are preserved commensurate with the expected frequency and consequences to the system (e.g., no risk outliers):

  The proposed inspection interval extension does not affect system redundancy, independence, or diversity in any way since it is not changing the plant design or how it is operated.

- Defenses against potential common cause failures are preserved and the potential for introduction of new common cause failure mechanisms is assessed:

  The proposed inspection interval extension does not affect any defenses against any common cause failures and there is no reason to expect the introduction of any new common cause failure mechanisms. This requirement applies to multiple active components, not to vessel nozzle welds that are passive components.

- Independence of barriers is not degraded (the barriers are identified as the fuel cladding, reactor coolant pressure boundary, and containment structure):

  The inspection interval extension does not change the relationship between the barriers in anyway and therefore does not degrade the independence of the barriers. The change in inspection interval does not change the robustness of the vessel nozzle design in any way. It is because of this robustness that the inspection interval can be doubled with no significant change in failure frequency.

- Defenses against human errors are preserved:

  The RV nozzle weld inspection interval extension does not affect any defenses against human errors in any way. The inspection interval extension reduces the frequency for which the lower internals need to be removed. Reducing this frequency reduces the possibility for human error and damaging the core but still provides for detection of emerging degradation mechanisms.

3.2.5 RI-ISI Program Effects

For plants that have a Risk-Informed Inservice Inspection (RI-ISI) program for piping, it is necessary to determine the effect of the ISI interval extension and ensure that the program still meets appropriate metrics for risk. The two most commonly applied methodologies in the U.S. for risk-informed inservice inspection (RI-ISI) of piping are the PWROG methodology and the Electric Power Research Institute (EPRI) methodology. These methodologies are included as Methods A and B in Nonmandatory Appendix R of Section XI of the ASME Code and are documented in more detail in References 4 and 8, respectively. One other methodology that has been applied is ASME Section XI Code Case N-716

(Reference 9). Relative to the extension of the inservice inspection interval, Code Case N-716 is very similar to the EPRI methodology. The fundamental steps involved with developing a RI-ISI program are identified below and, for the steps that are affected by the inservice inspection interval, a discussion of how inservice inspection of the RV nozzle welds is credited or used is provided.

1. Scope Determination

2. Segment Definition

3. Consequence Assessment

4. Failure Potential Assessment – For each piping segment, the likelihood for failure is determined.

   a. For the PWROG methodology, the failure potential is calculated in the form of a cumulative failure probability using probabilistic fracture mechanics. Separate failure probabilities are calculated with and without credit for inservice inspection. The probabilities without credit for inservice inspection are to be used in the risk evaluation (next step) while the failure probabilities with credit for inservice inspection are to be used in the change-in-risk evaluation. Therefore, for the PWROG methodology, the inservice inspection interval is considered in this step.

   b. For the EPRI methodology, failure potential is based solely on the postulated degradation mechanism. Therefore, for the EPRI methodology, the inservice inspection interval is not considered in this step.

5. Risk Evaluation

6. Element Selection

7. Change-in-Risk Evaluation – A comparison is made between the risk associated with the welds selected for inspection in the non-risk-informed, ASME Section XI program, and the risk associated with the welds to be inspected for the proposed risk-informed program. This comparison is performed by taking credit for inservice inspection for the welds that are selected for each program and by not taking credit for the benefits of inservice inspection for those that are not selected. Since this is the only step that takes credit for inservice inspection, this is the key step for determining whether the extended inservice inspection interval has an effect on the RI-ISI program. Each RI-ISI methodology has different criteria for an acceptable change in risk. These criteria are discussed in the following sections for each methodology. If these criteria cannot be satisfied, additional examinations are required until they are satisfied. It should be noted that the RI-ISI methodologies take credit for leak detection.

Based on the discussion above, the effect on the RI-ISI program can be determined during the change-in-risk evaluation. The method for doing so is discussed below for the different RI-ISI methodologies.

3.2.5.1 PWROG RI-ISI Methodology

Change-in-Risk Evaluation Method

In the PWROG RI-ISI methodology, the change in risk associated with the change in number of piping segments selected for inspection is calculated. The change in risk is calculated for each system by summing the change in risk for all segments within that system. The total change in risk is then calculated by adding the change in risk for all systems. The method for performing this change-in-risk assessment is discussed in detail in WCAP-14572, Revision 1-NP-A (Reference 4). The total change in risk and system level change in risk must then be compared to the criteria below. The PWROG methodology requires that the change-in-risk evaluation be performed for CDF and LERF with and without the effects of operator actions and all four delta risk cases are compared against the change-in-risk criteria.

Change-in-Risk Criteria

The criteria for evaluating the change in risk for a PWROG-methodology-based RI-ISI program, as identified in Reference 4, is as follows:

1. The total change in piping risk should be risk neutral or a risk reduction in moving from Section XI to RI-ISI. If not, the dominant system and piping segment contributors to the RI-ISI risk should be re-examined in an attempt to identify additional examinations which would make the application at least risk neutral. If additional examinations can be proposed, then the change-in-risk calculations should be revised to credit these additional examinations until at least a risk neutral position is achieved.

2. Once this is achieved, an evaluation of the dominant system contributors to the total risk for the RI-ISI (e.g., system contribution to the total is greater than approximately 10%) should be examined to identify where no improvement has been proposed (i.e., where moving from no ISI or Section XI ISI to RI-ISI, the risk has not changed and it is still a dominant contributor to the total CDF/LERF). If any systems are identified where this is the case, the dominant piping segments in that system should be reevaluated in an attempt to identify additional examinations which would reduce the overall risk for these systems and thus possibly the overall risk.

3. The results should be reviewed to identify any system in which there is a risk increase in moving from the current Section XI program to the RI-ISI program. The following guidelines are suggested to identify whether additional examinations are necessary:

    a. If the CDF increase for the system is approximately a) greater than two orders of magnitude below the risk-informed ISI CDF for that system, or b) greater than 1E-08 (whichever is higher), then at least one dominant segment in that system should be reevaluated to identify additional examinations.

    b. If the LERF increase for the system is a) greater than two orders of magnitude below the risk-informed ISI LERF for that system, or b) greater than 1E-09 (whichever is higher), then at least one dominant segment in that system should be reevaluated to identify additional examinations.

4. If any additional examinations are identified, the change-in-risk calculations should be revised to credit these additional examinations.

Evaluation of Effect RV Nozzle ISI Interval Extension

To determine the effect on the piping risk-informed inservice inspection program of the plant, the change-in-risk-calculations in Table 3-9 are duplicated with the exception that the calculations are performed using the change in failure frequencies with credit for leak detection from Table 3-3, 3-4, 3-5, or 3-6. These change-in-risk values, which represent the increase in risk associated with the extension of the ISI interval for the RV nozzles, are then added to the change-in-risk results of the RI-ISI program (Reference 4). These values are added to both the reactor coolant system change-in-risk values and also the total plant scope values for the CDF, with and without operator action, and LERF, with and without operator action cases. It should be noted that the PWROG methodology considers risk on a segment basis and that the risk is not dependent on the number of welds within a given piping segment. This is because the highest risk at the limiting location is controlling for that piping segment. Therefore, for Nozzle Types A and C, where there are two welds per nozzle, the risk should be adjusted to reflect only the most limiting weld prior to being added to the change in risk from the RI-ISI element selection.

Alternative Change-in-Risk Evaluation Methods

If the change-in-risk criteria cannot be met using the PWROG change-in-risk evaluation method in WCAP-14572 or an excessive number of exams would have to be added to meet the criteria, the following three alternative change-in-risk evaluation methods can be utilized to evaluate the effect on the RI-ISI program. In all three alternative evaluations methods, the change-in-risk evaluation is conducted on a weld-examined basis to address the underestimation of risk increases arising from the reduction in the number of inspections within each segment when the change-in-risk evaluation is conducted on a segment basis. The three alternative methods, in order of increasing complexity, are:

1. Examined Weld Counts Using Largest Change in Risk,
2. Examined Weld Counts Using Sum of System Change in Risk for Total Plant,
3. Examined Weld Counts Using Applicable Segment Change in Risk.

Licensees may select any of the three alternative methods, but it is expected that the licensee will start with the first alternative method and move to the more complex methods until the results indicate an acceptable change in risk or additional exams are added to make the change in risk acceptable. These methods are discussed in more detail in the following sections.

<u>First Alternative Evaluation Method – Examined Weld Counts Using Largest Change In Risk</u>

In the first alternative evaluation method, the change in risk is based on the largest applicable segment change in risk. The following steps are conducted:

1. Identify the applicable largest (i.e., most conservative) segment change in risk for the reactor coolant system and the total plant. The segment change in risk is based on the change between the segment being examined per the ASME Section XI or RI-ISI and no examination using the

guidelines in WCAP-14572 with consideration for leak detection, augmented ISI programs, and the factor of three.

2. Identify the number of welds examined per the ASME Section XI program and the RI-ISI program for the reactor coolant system and the total plant.

   a. For the welds examined per the ASME Section XI program, conservatively identify all welds examined by a volumetric and surface exam and by a surface exam only.

   b. For the welds examined per the RI-ISI program conservatively do not count the welds examined as part of a VT-2 visual examination.

3. Multiply the applicable largest segment change in risk times the difference in the number of welds examined per ASME Section XI and the RI-ISI programs for the reactor coolant system and the total plant.

4. Add the reactor vessel nozzle ISI interval extension risk increase as calculated on a weld basis to the current change in risk for the reactor coolant system and the total scope of the RI-ISI program.

5. Compare the results of step 4 against the criteria for the alternative change-in-risk methods

   a. If the change-in-risk criteria are met, no further analysis is required.

   b. If the change-in-risk criteria are met for the reactor coolant system but not the total plant, add exams or proceed to the second alternative evaluation.

   c. If the change-in-risk criteria are not met for the reactor coolant system, add exams or proceed to the third alternative evaluation method.

<u>Second Alternative Evaluation Method – Examined Weld Counts Using Sum of System Change in Risk for Total Plant</u>

The second alternative evaluation method is very similar to the first alternative evaluation method except that instead of using the largest overall change in risk to calculate the total plant change in risk, the change in risk from all the systems is summed. The following steps are conducted:

1. Identify the applicable largest (i.e., most conservative) segment change in risk for each system in the scope of the RI-ISI program. This is conducted in the same manner as the first alternative change-in-risk evaluation method with the exception that it is conducted only on a system basis for all systems in the scope of the RI-ISI program.

2. Identify the number of welds examined per the ASME Section XI program and the RI-ISI program for each system in the scope of the RI-ISI program. This is conducted in the same manner as the first alternative change-in-risk evaluation method.

3. Multiply the largest segment change in risk for each system times the difference in the number of welds examined per ASME Section XI and the RI-ISI programs for the respective system.

4. Add the reactor vessel nozzle ISI interval extension risk increase as calculated on a weld basis to the current change in risk for the reactor coolant system.

5. Sum the change in risk for each system, to obtain the total plant change in risk. Note that the reactor coolant system change in risk calculated in step 4 is used in this step.

6. Compare the results of step 5 against the criteria for the alternative change-in-risk methods

   a. If the change-in-risk criteria are met, no further analysis is required.

   b. If the change-in-risk criteria are not met, add exams or proceed to the third alternative evaluation method.

Third Alternative Evaluation Method – Examined Weld Counts Using Applicable Segment Change in Risk

In the third alternative evaluation method, the change in risk is based on the applicable segment change in risk instead of the largest segment change in risk for the system or plant. The following steps are conducted:

1. Identify the individual segment change in risk. This is conducted in the same manner as the first alternative change-in-risk evaluation method with the exception that it is conducted on a segment basis and is only required where there is a difference in the number of welds examined between the ASME Section XI program and the RI-ISI program.

2. Identify the number of welds examined per the ASME Section XI program and the RI-ISI program for each segment in the scope of the RI-ISI program. This is conducted in the same manner as the first alternative change-in-risk evaluation method with the exception that it is conducted on a segment basis.

3. Multiply the segment change in risk times the difference in the number of welds examined per ASME Section XI and the RI-ISI programs for that segment.

4. Sum the individual segment change in risk for each segment in a system to obtain the system change in risk.

5. Add the reactor vessel nozzle ISI interval extension risk increase as calculated on a weld basis to the change in risk for the reactor coolant system.

6. Sum the change in risk for each system in the scope of the RI-ISI program to obtain the total plant change in risk. Note that the reactor coolant system change in risk calculated in step 5 is used in this step.

7. Compare the results of step 6 against the following criteria:

   a. If the change-in-risk criteria are met, no further analysis is required.
   b. If the change-in-risk criteria are not met, add exams until the criteria are met.

All three alternative change-in-risk evaluation methods are conservative for the following reasons.

- All ASME Section XI exams are conservatively assumed to address the potential degradation mechanism of concern. The underestimation in risk reductions arising from changing inspection locations from a weld subject to no potential degradation mechanism to another with an identified potential degradation mechanism still applies.

- No credit is taken for visual (VT-2) examinations performed per the RI-ISI program.

In addition, the first and second alternative change-in-risk evaluation methods have the following conservatisms.

- The largest ISI change in segment risk is assumed to represent each weld examined in a system. The vast majority of welds that are examined per ASME Section XI will not result in the largest ISI change in segment risk. While it is also true that the vast majority of welds examined per the RI-ISI program will not result in the largest ISI change in risk, there are fewer welds examined per the RI-ISI program. Thus the overall effect is conservative.

- Although not all RI-ISI exams will result in the largest ISI change in segment risk, the RI-ISI exams typically address more risk than the ASME Section XI exams on a per weld basis, since the RI-ISI exams are inspections for cause. In addition per WCAP-14572 Supplement 2 (Reference 21), all postulated degradation mechanisms on a HSS segment must be addressed in the RI-ISI program.

Change-in-Risk Criteria for Alternative Change-in-Risk Evaluation Methods

The change-in-risk criteria for the alternative change-in-risk evaluation methods are the same as the change-in-risk criteria used for the EPRI methodology. The implementation of the RI-ISI program should be risk neutral, a decrease in risk, or, at most, an insignificant increase in risk. The increase in risk for each system shall meet the following criteria in order for it to be considered insignificant:

- Change in Core Damage Frequency (ΔCDF) < 1E-07/year, and
- Change in Large Early Release Frequency (ΔLERF) < 1E-08/year.

The total change for all systems must meet the criteria of RG 1.174 as stated in Section 3.2.1. If the scope of the RI-ISI program encompasses all Class 1 welds, the system level criteria shall be met. If the acceptance criteria cannot be met, additional inspections shall be added to the RI-ISI program until an acceptable change in risk is achieved.

Use of the alternative evaluations and criteria are acceptable since conducting the change-in-risk evaluation on a weld examined basis is consistent with how the change-in-risk evaluation is conducted for

EPRI and Code Case N-716 methodologies. The underestimation of risk increases arising from the reduction in the number of inspections within each segment is addressed. In addition, the three alternative change-in-risk evaluation methods are conservative since the underestimation of risk reductions arising from changing inspection locations from a weld subject to no degradation mechanism to another with an identified degradation mechanism is not addressed.

Pilot Plant Example

Beaver Valley Unit 1 has a RI-ISI program for piping that is based on the PWROG methodology. To determine the effect on the Beaver Valley Unit 1 piping risk-informed inservice inspection program, the change-in-risk calculations in Table 3-10 were duplicated with the exception that the calculations were performed using the change in failure frequencies from Table 3-5 (Type C), with credit for leak detection. These calculations are shown in Table 3-14. The change in risk calculated in Table 3-14 was then added to the change-in-risk results from the development of the RI-ISI program. The results of this evaluation are shown in Table 3-15.

Table 3-14 Change-in-Risk Calculations for RI-ISI Program Effects – Beaver Valley Unit 1

Failure Mode	Bounding Change in Failure Frequency (From Table 3-5, with Leak Detection)	CCDP	ΔCDF (/ year)	CLERP	ΔLERF (/year)
Outlet Nozzles					
SLOCA	4.49E-09	1.38E-05	6.19E-14	7.61E-12	3.42E-20
MLOCA	3.16E-09	1.68E-03	5.31E-12	4.70E-08	1.49E-16
LLOCA	3.04E-09	2.15E-03	6.54E-12	5.30E-08	1.61E-16
# of Welds Examined	3	Total ΔCDF	3.57E-11	Total ΔLERF	9.29E-16
Inlet Nozzles					
SLOCA	3.52E-09	1.93E-04	6.79E-13	2.90E-10	1.02E-18
MLOCA	1.52E-09	1.68E-03	2.55E-12	4.70E-08	7.15E-17
LLOCA	1.40E-09	2.15E-03	3.02E-12	5.30E-08	7.43E-17
# of Welds Examined	3	Total ΔCDF	1.87E-11	Total ΔLERF	4.40E-16
All Nozzles					
Total Change-in-Risk Results		Total ΔCDF	5.45E-11	Total ΔLERF	1.37E-15

Table 3-15 Effects of RV Nozzle ISI Interval Extension on the Beaver Valley RI-ISI Program Utilizing Change-In-Risk Criteria from WCAP-14572				
	Beaver Valley Unit 1 with Operator Action		**Beaver Valley Unit 1 without Operator Action**	
	ΔCDF (/year)	**ΔLERF (/year)**	**ΔCDF (/year)**	**ΔLERF (/year)**
RC System (Existing RI-ISI Program)	-2.58E-13	4.52E-19	-2.58E-13	4.52E-19
Additional Risk from ISI Int. Extension (From Table 3-14)	5.45E-11	1.37E-15	5.45E-11	1.37E-15
Total RC System Change in Risk	5.42E-11	1.37E-15	5.42E-11	1.37E-15
Acceptable System Change in Risk	0.0E+00	1.0E-09(1)	0.0E+00	1.0E-09(1)
Total Plant (Existing RI-ISI Program)	-3.94E-11	-7.88E-13	-2.02E-10	-9.36E-13
Additional Risk from ISI Int. Extension (From Table 3-14)	5.45E-11	1.37E-15	5.45E-11	1.37E-15
Total Plant Change in Risk	1.51E-11	-7.87E-13	-1.48E-10	-9.35E-13
Acceptable Total Change in Risk	0.0E+00	0.0E+00	0.0E+00	0.0E+00

Note:
1. The RC system is not a dominant system for LERF and therefore a small increase in risk is acceptable.

As can be seen in Table 3-15, when the increase in risk associated with extension of the ISI interval is added to the risk as a result of the risk-informed inservice inspection program element selection, the total change in risk does not meet the change-in-risk acceptance criteria for the PWROG RI-ISI methodology. Therefore, in order to implement the ISI interval extension for the RV nozzles, additional piping segments would need to be selected for inspection in the reactor coolant system until the total plant change in risk is either risk neutral or a risk reduction.

A review was conducted to see how many segments would have to be added for Beaver Valley Unit 1 to meet the change-in-risk criteria for the PWROG RI-ISI methodology. It was identified that even if all RCS segments were selected for examination, the criteria (absolute neutrality) could not be met.

Based on not being able to meet the change-in-risk criteria utilizing the change in risk from WCAP-14572, the first alternative evaluation, Examined Weld Counts Using Largest Change In Risk, was applied to Beaver Valley Unit 1. The results of this evaluation are presented in Table 3-16.

Table 3-16 Effects of RV Nozzle ISI Interval Extension on the Beaver Valley RI-ISI Program Utilizing First Alternative Change-in-Risk Evaluation				
	Beaver Valley Unit 1 with Operator Action		**Beaver Valley Unit 1 without Operator Action**	
	ΔCDF (/year)	**ΔLERF (/year)**	**ΔCDF (/year)**	**ΔLERF (/year)**
Total RC System Change in Risk	1.97E-09	5.03E-14	1.97E-09	5.03E-14
Acceptable System Change in Risk	1.0E-07	1.0E-08	1.0E-07	1.0E-08
Total Plant Change in Risk	7.75E-08	1.16E-08	7.86E-07	1.84E-08
Acceptable Total Change in Risk	1.0E-06	1.0E-07	1.0E-06	1.0E-07

As can be seen in Table 3-16, the change in risk for the Beaver Valley RI-ISI program, including the additional risk associated with the extension in inspection interval meets the system and total plant change-in-risk acceptance criteria. Therefore, using the first alternative evaluation for the change in risk, the effect of the extension in inspection interval for the RV nozzles on the Beaver Valley Unit 1 RI-ISI program is acceptable.

Although the change-in-risk criteria were met utilizing the first alternative evaluation, for additional information, the second alternative evaluation was applied to Beaver Valley Unit 1. The results of this evaluation are presented in Table 3-17.

Table 3-17 Effects of RV Nozzle ISI Interval Extension on the Beaver Valley RI-ISI Program Utilizing Second Alternative Change-in-Risk Evaluation				
	Beaver Valley Unit 1 with Operator Action		**Beaver Valley Unit 1 without Operator Action**	
	ΔCDF (/ year)	**ΔLERF (/ year)**	**ΔCDF (/ year)**	**ΔLERF (/ year)**
Total RC System Change in Risk	1.97E-09	5.03E-14	1.97E-09	5.03E-14
Acceptable System Change in Risk	1.0E-07	1.0E-08	1.0E-07	1.0E-08
Total Plant Change in Risk	2.99E-08	3.98E-09	2.85E-07	6.31E-09
Acceptable Total Change in Risk	1.0E-06	1.0E-07	1.0E-06	1.0E-07

The change in risk for the Beaver Valley RI-ISI program, including the additional risk associated with the extension in inspection interval meets the system and total plant change-in-risk acceptance criteria. Therefore, using the second alternative evaluation for the change in risk, the effect of the extension in inspection interval for the RV nozzles on the Beaver Valley Unit 1 RI-ISI program is acceptable. As expected, there was no change in the change in risk for the reactor coolant system between the first and second alternative evaluation. As anticipated, there was a reduction in the change in risk in the total plant when going from the first alternative evaluation to the second alternative evaluation.

3.2.5.2 EPRI RI-ISI Methodology

Change-in-Risk Evaluation Methods

The EPRI RI-ISI Methodology in Reference 8 provides four methods for evaluating the change in risk associated with implementing the RI-ISI program. These four methods in order of increasing complexity are:

1. Qualitative,
2. Bounding without any credit for increase in Probability of Detection (POD),
3. Bounding with credit for increase in Probability of Detection (POD),
4. Markov Model.

Licensees may select any of the four methods but it is expected that the licensee will start with the qualitative methodology and move to the more complex methods until the results indicate an acceptable change in risk or additional inspections are added to make the change in risk acceptable. These methods are discussed in more detail in the following sections.

It should be noted that the change-in-risk analysis methods for the EPRI RI-ISI methodology can also used with Code Case N-716 (Reference 9). Therefore, the discussion below would also be applicable for a plant that has implemented a Code Case N-716 based RI-ISI program.

Change-in-Risk Criteria

1. Qualitative Method (1)

   The RI-ISI program must provide for an increased number of inspections in each High- or Medium-risk category (Categories 1-3 and 4-5, respectively), or a comparable number of inspections are redirected to locations that are more likely to identify failure precursors on the basis of characteristics of the potential damage mechanisms. Provided that the risk acceptance criteria of RG 1.174 are met, the effect of the extended inservice inspection interval on the RI-ISI program is acceptable.

2. Quantitative Methods (2, 3, & 4) – Bounding, with and without Credit for POD, and Markov Method

   The implementation of the RI-ISI program should be risk neutral, a decrease in risk, or, at most, an insignificant increase in risk. The increase in risk for each system shall meet the following criteria in order for it to be considered insignificant:

   - Change in Core Damage Frequency (ΔCDF) < 1E-07/year, and
   - Change in Large Early Release Frequency (ΔLERF) < 1E-08/year.

The total change for all systems must meet the criteria of RG 1.174 as stated in Section 3.2.1. If the scope of the RI-ISI program encompasses all Class 1 welds, the system level criteria shall be met. If the acceptance criteria cannot be met, additional inspections shall be added to the RI-ISI program until an acceptable change in risk is achieved.

Evaluation of Effect RV Nozzle ISI Interval Extension

A discussion of the methods and how they would be affected by the change-in-inspection interval is provided below. It should be noted that all four methods include the assumption that there is a negligible increase in risk associated with the elimination of inspections of welds in piping segments in the lowest risk categories, 6 and 7.

1. Qualitative Method

   In some cases, the RI-ISI process can be shown to provide an increased number of inspections in each High- or Medium-risk category (Categories 1-3 and 4-5, respectively), or a comparable number of inspections are redirected to locations that are more likely to result in failure precursors on the basis of characteristics of the potential damage mechanisms. In these cases, the change in risk can qualitatively be shown to be a decrease in risk.

   This method implicitly assumes that all inspections are performed on the same interval. If this method were to show that there is no reduction, or there is an increase in the number of inspections, the only increase in risk would be as a result of the extension in inspection interval for the reactor vessel nozzle welds. Therefore, as long as the change in risk as calculated per Section 3.2.4 meets the Regulatory Guide 1.174 acceptance criteria, the extension in inspection interval would be acceptable.

2. Bounding without any Credit for Increase in Probability of Detection (POD)

   A quantitative estimate of the change in risk can be performed for all system locations in the high- and medium-risk categories. This evaluation is performed using bounding values for CCDPs and rupture frequencies as specified in the EPRI topical report (Reference 8). The bounding values for high, medium, and low failure potentials correspond to rupture frequencies of 1E-4, 1E-5, and 1E-6 per weld year, respectively. High-, medium-, and low-consequence categories correspond to CCDPs of 1, 1E-4, and 1E-6 per reactor year, respectively. The CCDP for the high consequence category can also be calculated from the plant-specific, as the highest value of CCDP. The change in risk is calculated for each weld and the change in risk is then calculated for each system by summing the change in risk for all welds within that system. This calculation is shown in equation 3-1:

$$\Delta CDF_j = \sum_i [FR_{i,j} * (SXI_{i,j} - RISI_{i,j}) * CCDP_{i,j}]. \quad (3\text{-}1)$$

Where:

ΔCDF_j	=	Change in CDF for system j,
$FR_{i,j}$	=	Rupture frequency per element for risk element i of system j,
$SXI_{i,j}$	=	Number of ASME Section XI inspection elements for risk element i of system j,
$RISI_{i,j}$	=	Number of RISI inspection elements for risk element i of system j,
$CCDP_{i,j}$	=	Conditional core damage probability given a break in risk element i of system j.

Similar calculations can be performed using the CLERP (conditional large early release probability) to determine the change in LERF for each system. The change in risk for each system and the total plant is compared to the EPRI acceptance criteria described above to determine the acceptability of the RI-ISI program.

To account for the extension in the inservice inspection interval for the reactor vessel nozzles, the change-in-risk calculations in Table 3-9 are duplicated with the exception that the calculations are performed using the change in failure frequencies with credit for leak detection from Table 3-3, 3-4, 3-5, or 3-6. These change-in-risk values, which represent the increase in risk associated with the extension of the ISI interval for the RV nozzles, are then added to the system and total plant change-in-risk results of the RI-ISI program. In some applications of the EPRI RI-ISI methodology, the change-in-risk calculation may use only one LOCA-initiating event (the one that is determined in the risk evaluation to be the most limiting in terms of CDF and LERF) to model the range of LOCA sizes. In these instances, the change in risk associated with the extension in interval for the limiting LOCA size shall be added to the system level change in risk.

3. Bounding with Credit for Increase in Probability of Detection (POD)

This approach is consistent with the second approach discussed above but this approach allows for an increase in the probability of detection based on the use of an inspection strategy that is based on the postulated degradation mechanism. This is illustrated in equation 3-2, which can be used to estimate the change in risk of core damage at location j that is affected by the changes in the RI-ISI program:

$$\Delta CDFj=(F_{rj}-F_{ej})*CDF_j=(I_{rj}-I_{ej})*F_{0j}*CCDP_j\,. \tag{3-2}$$

Where the subscripts "rj" refer to the risk informed inspection at location j, and the subscripts "ej" refer to the existing inspection program at location j. F0j is the frequency of pipe rupture at location j, if no inspection is performed. CCDPj is the conditional core damage probability from a pipe rupture at location j, which is independent of the inspection strategy.

For the reactor vessel nozzle welds addressed in this calculation, there is no expected increase in probability of detection associated with the implementation of the RI-ISI program because there is no change in the inspection strategy. Therefore, the method for determining the effect of the extended inservice inspection interval is consistent with the approach above in that the change in risk as calculated per Table 3-9, using change in failure frequencies with credit for leak detection, would be added to the system and total plant change-in-risk results of the RI-ISI program.

4. Markov Model

The Markov model attempts to make a more realistic model of the interactions between potential degradation mechanisms that cause pipe cracks and pipe inspections, and leak detection processes that mitigate pipe cracks, leaks, and ruptures. For the purposes of the change-in-risk evaluation, the Markov model is used to develop hazard rates that are in turn used to determine inspection effectiveness factors. The change in risk for each system j is calculated using equations 3-3 and 3-4:

$$\Delta CDF_j = \sum_{i=1}^{N} n_i \lambda_i P_i \langle R|F \rangle (I_{i,new} - I_{i,old}) CCDP_i \quad (3\text{-}3)$$

and

$$\Delta LERF_j = \sum_{i=1}^{N} n_i \lambda_i P_i \langle R|F \rangle (I_{i,new} - I_{i,old}) CLERP_i \, . \quad (3\text{-}4)$$

Where:

ΔCDF_j = change in core damage frequency due to changes in inspection strategy for the system j,

$\Delta LERF_j$ = change in large early release frequency due to changes in the inspection strategy for the system j,

i = index for risk element having the same potential degradation mechanisms and consequence of pipe ruptures,

N = number of risk elements in the system,

n_i = number of elements (welds) in risk element i,

λ_i = failure rate for welds in risk element i (including leak and rupture failure modes) assuming no inspections, estimated from service data,

$P_i \langle R | F \rangle$ = conditional probability of rupture given failure of welds in risk element i assuming no inspections, estimated from service data,

$I_{i,new}$ = inspection effectiveness factor for proposed risk informed inspection strategy for risk element i, calculated from Markov model,

$I_{i,old}$ = inspection factor for current ASME Section XI based inspection strategy for element i, calculated from Markov model,

$CCDP_i$ = conditional core damage probability due to pipe ruptures in risk element i, obtained from Consequence Evaluation,

$CLERP_i$ = conditional large early release probability due to pipe ruptures in risk element i, obtained from Consequence Evaluation.

As mentioned above, the Markov model is used to determine the inspection effectiveness factors, $I_{i,new}$ and $I_{i,old}$, associated with the new (RI-ISI) and old (ASME Section XI) inspection programs. Each factor represents the ratio of the rupture frequency with credit for inspections to that given no credit for inspections. Noting the solution of the Markov model is a set of time-dependent state probabilities and rupture frequencies; the hazard rate of the Markov model at the end of the 40-year design life is used to determine these factors. More specifically, the inspection factors are defined using equations 3-5 and 3-6:

$$I_{i,new} = \frac{h_{40}\{RI-ISI\}}{h_{40}\{noinsp\}} \text{ and} \tag{3-5}$$

$$I_{i,old} = \frac{h_{40}\{SecXI\}}{h_{40}\{noinsp\}}. \tag{3-6}$$

Where:

$h_{40}\{RI\text{-}ISI\}$ = hazard rate (time-dependent rupture frequency) for weld subjected to the RISI inspection strategy,

$h_{40}\{SecXI\}$ = hazard rate (time-dependent rupture frequency) for weld subjected to the ASME Section XI inspection strategy,

$h_{40}\{noinsp\}$ = hazard rate (time-dependent rupture frequency) for weld subjected to no inservice inspection.

To account for the extension in the inservice inspection interval for the reactor vessel nozzles, there are two different methods that could be used.

Method A

For the reactor vessel nozzle welds for which the ISI interval is to be extended to 20 years, the hazard rate for the RI-ISI program would be calculated based on a 20-year interval. This hazard rate would then be used to calculate the inspection effectiveness factor for these particular welds. In the change-in-risk calculations, the change in risk would be a result of the difference in inspection effectiveness between the Section XI exams performed on a 10-year interval and the RI-ISI exams performed on a 20-year interval. Therefore, the change in risk for the system would account for the increase in risk associated with the extension in inspection interval.

Method B

The bounding change in failure frequency calculated using the SRRA code would be used in lieu of the Markov model. The change-in-risk-calculations in Table 3-9 are duplicated with the exception that the calculations are performed using the change in failure frequencies with credit for leak detection from Table 3-3, 3-4, 3-5, or 3-6. This calculated change in risk would then be added to the change in risk for the system containing the reactor vessel nozzle welds. In instances where the change-in-risk calculation uses one LOCA initiating event (the one that is most limiting in terms of CDF and LERF) to model the range of LOCA sizes, the change in risk associated with the extension in interval for the limiting LOCA size shall be added to the system level change in risk.

Pilot Plant Example

Three Mile Island Unit 1 has a RI-ISI program that is based on the EPRI methodology. The Markov method was used for performing the TMI-1 RI-ISI change-in-risk evaluation. Therefore, the effect on the RI-ISI program was evaluated using the two methods described in the preceding sections. The results of the evaluations for the two methods are discussed below:

<u>Method A</u>

The TMI-1 Markov model ISI frequency input was changed to 20 years. New hazard rates for the RV nozzle welds were calculated by the Markov model based on this inspection interval. This hazard rate was used to calculate inspection effectiveness factors and determine the change in risk associated with extending the ISI interval for the RV nozzles from 10 to 20 years. The results of this evaluation are shown in Table 3-18

Table 3-18 Effects of RV Nozzle ISI Interval Extension on the TMI-1 RI-ISI Program – Method A		
ISI Interval	**10 Years**	**20 Years**
Hazard Rate with ISI (h_{40}\{xxyr\})	4.0238E-10	5.8499E-10
Hazard Rate without ISI (h_{40}\{noinsp\})	9.1872E-10	
Inspection Effectiveness Factor	0.438	0.637
Change in Inspection Effectiveness (ΔI)	0.199	
Failure Rate (λ_f)	8.16E-06	
Cond. Prob. Rupture ($P_i\langle R\mid F\rangle$)	4.76E-02	
SLOCA CCDP	1.83E-03	
SLOCA CLERP	2.53E-04	
$\Delta CDF = \lambda_f P_i\langle R\vert F\rangle(\Delta I)CCDP$ (per nozzle)	1.40E-10	
$\Delta LERF = \lambda_f P_i\langle R\vert F\rangle(\Delta I)CLERP$ (per nozzle)	1.94E-11	
Number of RV Nozzle Welds Examined	6	
Total Nozzle ΔCDF (/year)	8.41E-10	
Total Nozzle ΔLERF (/year)	1.16E-10	
RC System ΔCDF (/year) from RI-ISI	6.74E-09	
RC System ΔLERF (/year) from RI-ISI	1.12E-09	
New RC System ΔCDF (/year)	7.58E-09	
New RC System ΔLERF (/year)	1.24E-09	

As can be seen in Table 3-18, the change in risk for the RI-ISI program, including the additional risk associated with the extension in inspection interval still meets the system and total plant change-in-risk acceptance criteria for the EPRI RI-ISI methodology. Therefore, using Method A, the effect of the extension in inspection interval for the RV nozzles on the TMI-1 RI-ISI program is acceptable.

Method B

Method B uses the bounding failure frequencies from Table 3-3 (Type A), with credit for leak detection, in lieu of the Markov model. The calculations and results of this method are shown in Table 3-19.

As can be seen in Table 3-19, the change in risk for the RI-ISI program, including the additional risk associated with the extension in inspection interval still meets the system and total plant change-in-risk acceptance criteria for the EPRI RI-ISI methodology. Therefore, using Method B, the effect of the extension in inspection interval for the RV nozzles on the TMI-1 RI-ISI program is acceptable.

Table 3-19 Effects of RV Nozzle ISI Interval Extension on the TMI-1 RI-ISI Program – Method B

Failure Mode	Maximum Change in Failure Frequency (From Table 3-3, with Leak Detection)		CCDP	ΔCDF (/ year)	CLERP	ΔLERF (/ year)
Outlet Nozzle						
SLOCA	2.84E-11		1.83E-03	5.19E-14	2.53E-04	7.18E-15
# of Welds Examined		2	Total	1.04E-13	Total	1.44E-14
Inlet Nozzle						
SLOCA	1.34E-11		1.83E-03	2.45E-14	2.53E-04	3.39E-15
# of Welds Examined		4	Total	9.80E-14	Total	1.36E-14
All Nozzles						
Nozzle Change-in-Risk Results				2.02E-13		2.79E-14
RC System Change in Risk				6.74E-09		1.12E-09
New RC System Change in Risk				6.74E-09		1.12E-09
Plant Change in Risk				4.08E-08		5.36E-09
New RC Plant Change in Risk				4.08E-08		5.36E-09

4 LIMITATIONS AND CONDITIONS FOR ACCEPTANCE

The limitations for the acceptance and application of the RV nozzle weld ISI interval extension methodology described in this report are as follows:

- The ISI interval extension cannot be applied to plants where the full-penetration weld has been fabricated using Alloy 82 or 182 weld materials.

- The bounding change in failure frequencies identified in Tables 3-3 through 3-6 in Section 3.2.3 are applicable for RV nozzle welds of the plants identified in Table 4-1. The operating conditions and geometries for other plants were not reviewed as part of the development of the bounding change in failure frequencies and therefore, it has not been confirmed that the values in Tables 3-3 through 3-6 would be applicable to other plants.

Table 4-1 Plants Evaluated

Plant	Nozzle Weld Configuration Type	Weld Material
ANO Unit 1	A	CS
ANO Unit 2	B	CS
Beaver Valley Unit 1	C	SS
Catawba Unit 1	D	SS
Davis-Besse	A	CS
D.C. Cook Unit 2	D	SS
Kewaunee	C	SS
McGuire Unit 2	D	SS
Millstone Unit 2	B	CS
North Anna Unit 1	C	SS
North Anna Unit 2	C	SS
Oconee Unit 1	A	CS
Oconee Unit 2	A	CS
Oconee Unit 3	A	CS
Palisades	B	CS
Prairie Island Unit 1	D	SS
Prairie Island Unit 2	C	SS
San Onofre Unit 2	B	CS
San Onofre Unit 3	B	CS
Sequoyah Unit 1	C	SS
Sequoyah Unit 2	C	SS

Table 4-1 Plants Evaluated (cont.)

Plant	Nozzle Weld Configuration Type	Weld Material
St. Lucie Unit 1	B	CS
St. Lucie Unit 2	B	CS
Surry Unit 1	C	SS
Surry Unit 2	C	SS
Three Mile Island Unit 1	A	CS
Turkey Point Unit 3	C	SS
Turkey Point Unit 4	C	SS
Waterford Unit 3	B	CS

5 REFERENCES

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