Westinghouse Non-Proprietary Class 3

WCAP-17236-NP-A
Revision 0

July 2012

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

Westinghouse

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Revision 0

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

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July 2012

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This work was performed for the PWR Owners Group under PWROG Project Authorization MSC-0440.

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

May 23, 2012

Mr. W. Anthony Nowinowski, Program Manager
PWR Owners Group, Program Management Office
Westinghouse Electric Company
1000 Westinghouse Drive, Suite 380
Cranberry Township, PA 16066

SUBJECT: FINAL SAFETY EVALUATION FOR PRESSURIZED WATER REACTOR OWNERS GROUP TOPICAL REPORT WCAP-17236-NP, REVISION 0, "RISK-INFORMED EXTENSION OF THE REACTOR VESSEL NOZZLE INSERVICE INSPECTION INTERVAL" (TAC NO. ME4878)

Dear Mr. Nowinowski:

By letter dated October 4, 2010 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML102790086), the Pressurized Water Reactor Owners Group (PWROG) submitted Topical Report (TR) WCAP-17236-NP, Revision 0, "Risk-Informed Extension of the Reactor Vessel Nozzle Inservice Inspection Interval," to the U.S. Nuclear Regulatory Commission (NRC) staff for review. By letter dated January 26, 2012, an NRC draft safety evaluation (SE) regarding our approval of TR WCAP-17236-NP, Revision 0, was provided for your review and comment (ADAMS Accession No. ML120240450). By letter dated March 22, 2012 (ADAMS Accession No. ML12083A195), the PWROG commented on the draft SE. The NRC staff's disposition of the PWROG's comments on the draft SE are discussed in the attachment to the final SE enclosed with this letter.

The NRC staff has found that TR WCAP-17236-NP, Revision 0, is acceptable for referencing in licensing applications for nuclear power plants to the extent specified and under the limitations delineated in the TR and in the enclosed final SE. The final SE defines the basis for our acceptance of the TR.

Our acceptance applies only to material provided in the subject TR. We do not intend to repeat our review of the acceptable material described in the TR. When the TR appears as a reference in license applications, our review will ensure that the material presented applies to the specific plant involved. License amendment requests that deviate from this TR will be subject to a plant-specific review in accordance with applicable review standards.

In accordance with the guidance provided on the NRC website, we request that the PWROG publish an approved version of this TR within three months of receipt of this letter. The approved version shall incorporate this letter and the enclosed final SE after the title page. Also, it must contain historical review information, including NRC requests for additional information and your responses. The approved version shall include an "-A" (designating approved) following the TR identification symbol.

A. Nowinowski - 2 -

If future changes to the NRC's regulatory requirements affect the acceptability of this TR, the PWROG and/or licensees referencing it will be expected to revise the TR appropriately, or justify its continued applicability for subsequent referencing.

Sincerely,

Sher Bahadur, Deputy Director
Division of Policy and Rulemaking
Office of Nuclear Reactor Regulation

Project No. 694

Enclosure:
Final Safety Evaluation

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

FINAL SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

TOPICAL REPORT WCAP-17236-NP, REVISION 0, "RISK-INFORMED EXTENSION OF THE REACTOR VESSEL NOZZLE INSERVICE INSPECTION INTERVAL"

PRESSURIZED WATER REACTOR OWNERS GROUP

PROJECT NO. 694

1.0 INTRODUCTION AND BACKGROUND

By letter dated October 4, 2010, the Pressurized Water Reactor Owners Group (PWROG) submitted Topical Report (TR) WCAP-17236-NP, Revision 0, "Risk-Informed Extension of the Reactor Vessel Nozzle Inservice Inspection Interval" (Reference 1), for U.S. Nuclear Regulatory Commission (NRC) staff review. By letter dated August 26, 2011 (Reference 2), the PWROG submitted responses to the NRC staff's request for additional information (RAI) questions on TR WCAP-17236-NP, Revision 0 (hereafter referred to as the TR), but did not expand its scope as originally submitted for NRC staff review. Attached to the August 26, 2011, letter is a revised TR WCAP-17236-NP, Revision 0, incorporating part of the PWROG's responses to the NRC's RAI questions.

In the TR, the PWROG provided the technical and regulatory basis for decreasing the frequency of inspections by extending the American Society of Mechanical Engineers (ASME) *Boiler and Pressure Vessel Code* (ASME Code) Section XI inservice inspection (ISI) interval from the current 10 years to 20 years for ASME Code Section XI, Category B-F and B-J reactor vessel (RV) nozzle welds that do not contain Alloy 82/182.

The TR described a risk-informed methodology that relies on the probabilistic fracture mechanics (PFM) methodology which is similar to that used in the approved PWROG risk-informed ISI (RI-ISI) methodology for piping welds (ASME Code, Section XI, Appendix R, Method A (Reference 3)). The extension of the ISI interval from 10 to 20 years is also consistent with the methodology used in the approved application for extension of the ISI interval for RV welds (Reference 4) from 10 to 20 years.

The proposed changes may affect the RI-ISI program for each licensee who has implemented a RI-ISI program. In addition to the PWROG RI-ISI methodology, the NRC has endorsed plant-specific RI-ISI methodology based on the Electric Power Research Institute (EPRI) methodology (ASME Code, Section XI, Appendix R, Method B (Reference 5)), and has accepted relief requests based, in part, on the methodology in ASME Code Case N-716, "Alternative Piping Classification and Examination Requirements, Section XI, Division 1" (Reference 6). The effect of extending the ISI interval for nozzle welds for all three RI-ISI methodologies is addressed in the TR and this safety evaluation (SE).

ENCLOSURE

2.0 REGULATORY EVALUATION

ISI of ASME Code Class 1, 2, and 3 components is performed in accordance with Section XI of the ASME Code and applicable Addenda as required by Title 10 of the *Code of Federal Regulations* (10 CFR) 50.55a(g), except where specific relief has been granted by the NRC pursuant to 10 CFR 50.55a(g)(6)(i). The regulation at 10 CFR 50.55a(a)(3) states that alternatives to the requirements of paragraph (g) may be used, when authorized by the NRC, if:
(i) the proposed alternatives would provide an acceptable level of quality and safety or
(ii) compliance with the specified requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

The regulations require that ISI of components and system pressure tests conducted during the first 10-year interval and subsequent intervals comply with the requirements in the latest edition and addenda of Section XI of the ASME Code incorporated by reference in 10 CFR 50.55a(b) 12 months prior to the start of the 120-month interval, subject to the limitations and modifications listed therein. The current requirements for the inspection of RV nozzle-to-pipe (RV nozzle) welds have been in effect since the 1989 Edition of ASME Code, Section XI. Article IWB-2000 of the ASME Code, Section XI establishes an ISI interval of 10 years. The TR proposed a methodology that can be used by individual licensees to demonstrate that extending the ISI interval on their Category B-F or B-J RV nozzle welds that do not contain Alloy 82/182 from 10 to 20 years would provide an acceptable level of quality and safety.

The NRC staff based its review of the risk information on NUREG-0800, "Standard Review Plan [(SRP)] for the Review of Safety Analysis Reports for Nuclear Power Plants," Chapter 19.2, "Review of Risk Information Used to Support Permanent Plant-Specific Changes to the Licensing Basis: General Guidance" (Reference 7). SRP Chapter 19.2 directs the NRC staff to review each of the four elements suggested in Section 2 of Regulatory Guide (RG) 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis" (Reference 8). These elements are: (1) define the proposed changes, (2) conduct engineering evaluations, (3) develop implementation and monitoring strategies, and (4) document the evaluations and submit the request. RG 1.174 also provides five key principles and numerical risk acceptance guidelines.

3.0 TECHNICAL EVALUATION

The objective of ISI is to identify conditions, such as flaw indications, that are precursors to leaks and ruptures which violate pressure boundary integrity principles for plant safety.

The TR contains a methodology based on the risk-informed approach to assess the change in core damage frequency (ΔCDF) and the change in large early release frequency (ΔLERF) due to extension of the ISI interval from 10 years to 20 years for RV nozzle welds of four configurations. This part of the methodology follows the basic steps of RG 1.174. Many plants have implemented RI-ISI programs for piping, which considered RV nozzle welds as piping welds. Consequently, extension of the ISI interval for RV nozzle welds may affect the current RI-ISI assessment. Evaluation of this effect is the second part of the proposed methodology. This TR provides calculations for Beaver Valley Power Station, Unit 1 (BV-1), and Three Mile Island Nuclear Station, Unit 1 (TMI-1); illustrating the application of the proposed methodology to these two pilot plants.

3.1 Define the Proposed Change

The TR proposed to extend the ISI interval for ASME Code, Section XI, Category B-F and B-J RV nozzle-to-safe-end and safe-end-to-pipe welds (excluding welds of Alloy 82/182 materials) from 10 years to a maximum of 20 years. The change will be accomplished through plant-specific requests for an alternative pursuant to 10 CFR 50.55a(a)(3)(i) on the basis that the alternative ISI interval provides an acceptable level of quality and safety.

The PWROG provided in the TR a proposed RV nozzle weld inspection schedule for participating PWROG plants, with the intent to achieve a somewhat uniform number of inspections per year from 2011 to 2050. The dates in this implementation plan are consistent with those in the plan for implementation of the RV ISI interval extension in TR WCAP-16168-NP-A, Revision 3 (Reference 4). Thus, the NRC staff determined that in its request for an alternative, each licensee shall identify the years in which future inspections will be performed. The dates provided must be within plus or minus one refueling cycle of the dates identified in the implementation plan referenced in the most recent Revision of TR WCAP-16168-NP-A. This will be listed as a condition in Section 4 of this SE.

3.2 Risk-Informed Evaluations

According to the guidelines in RG 1.174 and SRP Chapter 19.2, a RI application is an analysis of the proposed change using a combination of traditional engineering analysis with supporting insights from a risk assessment. The RI analysis in this TR proposes to verify that a reduction in the frequency of volumetric examination of the RV nozzle welds can be accomplished with an acceptably small change in risk.

The engineering evaluations include the PFM analysis to estimate the change in weld failure frequency caused by extending the ISI interval, and the change in risk caused by the change in failure frequency. The PFM engineering evaluations in the TR were based on results from applying the Westinghouse Structural Reliability and Risk Assessment (SRRA) Code (Reference 9), which is also the tool supporting the approved PWROG RI-ISI methodology for piping (Reference 3). These evaluations utilized the PFM methodology to model changes in failure frequency caused by change to the ISI interval. The change-in-risk evaluations are similar to the change-in-risk evaluations supporting the approved RI-ISI methodologies. The proposed methodology includes modifications to the RI-ISI change-in-risk evaluations to incorporate the increased failure frequency expected from the extended ISI interval.

3.2.1 PFM Methodology Evaluation

The ISI interval extension methodology is based, in part, on a PFM analysis of the effect of different ISI intervals on the frequency of postulated RV nozzle weld failure modes (i.e., Small, Medium, and Large Loss of Coolant Accident, or SLOCA, MLOCA, and LLOCA with leakage rates of 100, 1500, and 5000 gallons per minute (GPMs)). The likelihood of RV nozzle weld failure was postulated to increase with increasing time of operation due to the growth of pre-existing fabrication flaws by fatigue. The PFM methodology allowed for the consideration of distributions and uncertainties in flaw density and depth, material properties, crack growth resulting from fatigue, failure modes, stresses, and the effectiveness of inspections. For each of the four RV nozzle weld configuration types, the PFM approach was used to estimate a bounding change in failure frequency for each failure mode, considering the change of ISI interval from 10 years to 20 years. The change-in-risk calculation can then be performed for a

plant to determine the ΔCDF and ΔLERF associated with the increased ISI interval and changes to the RI-ISI program.

Validation of the Flaw Characteristics

The flaw characteristics used in the SRRA Code had already been accepted because this code was used in supporting the approved PWROG RI-ISI applications. The flaw characteristics were developed using the PRODIGAL Code, which relies on artificial intelligence rules that are based on experience to simulate each step in the weld fabrication, considering the various types of inspections used in the process. It is stated in Section 2.2 of the TR, "[t]he limiting flaw depth specified above [a through-wall depth of greater than six percent of the wall thickness and a length equal to six times the depth] 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." To validate this flaw depth distribution, DCI-RAI-1 requested the PWROG discuss the characteristics of the five recordable indications shown in Table 3-1 of the TR from the past RV nozzle ISI findings to justify the initial flaw depth distribution used in the PFM analyses in this application. The PWROG clarified in its August 26, 2011, response that all five indications identified in Table 3-1 of the TR are sub-surface flaws. Therefore, the NRC staff determined that using surface flaws with this initial through-wall depth distribution in the PFM analyses is conservative and bounds operating experience for RV nozzle welds.

Regarding flaw density, the PFM analyses supporting the TR were based on the assumption of one surface flaw per weld. The TR directs a licensee (Section 2.2) to validate that at most one surface breaking flaw is present based on past ISI results. If multiple surface breaking flaws have been detected in past inspections, the TR directs that the frequency be multiplied by the number of surface flaws. If the total flaw size from this method exceeds the dimension assumed above, a weld-specific PFM analysis should be performed to develop a weld-specific change-in-frequency value. Validation of this flaw assumption must also be performed in the future through continued monitoring every 20 years.

3.2.1.1 PFM and Leakage Analysis in the SRRA Code

Since the TR contains no details of the PFM methodology used in the application, DCI-RAI-2 requested the PWROG provide a summary of the PFM analysis methodology used in the TR, including the analysis methodology type (elastic plastic fracture mechanics or linear elastic fracture mechanics), failure criteria, and the growth law for a flaw with an initial flaw depth to a critical size or through-wall flaw, and eventually to a long flaw corresponding to SLOCA, MLOCA, or LLOCA. DCI-RAI-2 also requested information regarding the establishment of fracture toughness and other material properties critical to failure resistance for each of the two failure periods for the RV nozzle welds and the key parameters which affect through-wall flaw leakage, the leakage that is considered detectable, and how leak detection was credited.

The PWROG provided a summary in the August 26, 2011, response covering all aspects of the PFM analysis methodology that the NRC staff mentioned in DCI-RAI-2. This PFM analysis methodology was used in supporting the approved TR on PWROG RI-ISI for piping (Reference 3). The summary helped the NRC staff accept the inputs for the current application to the SRRA Code and identify additional conservatisms in the PFM analyses, such as the surface flaw assumption and the instant change from a semi-elliptic flaw to a circular through-wall flaw when leaking starts. Due to low neutron fluence and benign coolant condition, fatigue crack growth was identified as the only growth mechanism of concern in this application. The interface of leakage determination and PFM analysis is also consistent with the industry

approach that has been used in other areas such as leak-before-break applications. In addition, the PWROG's response to DCI-RAI-3 confirmed that "there were no parts of the SRRA Code used in generating PFM results for this application that were not needed in generating PFM results for the prior risk-ranking application [approved by the NRC]." This statement further supported the NRC staff's decision of not repeating a full, detailed, rigorous review of the PFM and leakage methodology documented in Reference 9.

To gain additional confidence in applying the SRRA Code in this application, the NRC staff requested additional information. DCI-RAI-4 inquired about the adequacy of obtaining an "average" change in failure frequency by dividing the difference in failure probability by 40 or 60 years. DCI-RAI-5 inquired about the use of engineering insights in certain places of the application. DCI-RAI-6 inquired about the RV nozzle diameter input. DCI-RAI-7 inquired about the difference between two flaw related inputs: "X-ray nondestructive examination (NDE)" and "One Flaw." DCI-RAI-8 inquired about the selection of the crack inspection accuracy parameter of 0.24 in adjusting the probability of detection (POD) curves used in the SRRA Code.

The response to DCI-RAI-4 included a histogram of the calculated failure frequencies corresponding to the first row of results in Table 3-7 of the TR. For the case of the 20-year ISI interval, the NRC staff estimated that the average failure frequency applicable between Year 50 and Year 60 would be 5.17E-7 based on the PWROG's failure frequency of 7.3E-8 at Year 50 and 1.47E-7 at Year 60. Similarly, for the case of the 10-year ISI interval, the NRC staff estimated that the average failure frequency applicable between Year 50 and Year 60 would be 7.52E-8 based on the PWROG's failure frequency of 2.0E-8 at Year 50 and 2.92E-8 at Year 60. Hence, the average change of failure frequency in the time between Year 50 and Year 60 due to the ISI interval change would be 4.42E-7/year, about three times the change in failure frequency based on averaging over 60 years as reported in the first row of Table 3-7. RG 1.174 directs that annual frequencies be estimated and used while the method of simulating lifetimes in PFM analysis results in failure probabilities which can vary over times that extend far beyond one year. Averaging the results over the full life of the facility is a reasonable approximation provided that the risk does not substantively increase toward the end of facility life. The factor of three differences in the annual frequency results is small compared to the generally large margin between the calculated changes in risk and the acceptable guideline values. Therefore, the NRC staff finds that the proposed conversion of the PFM results to annual frequency is acceptable because the evaluation in the TR indicates that other methods of conversion are not expected to substantively change the results.

The response to DCI-RAI-5 clarified that the fatigue stress range and design limiting stress, two of the SRRA Code inputs, were determined considering engineering (operating) experience. Also, when steam generator snubber lock-up is evaluated, the worst type of snubber was assumed in the analysis. The response stated that the heat-up and cool-down transients are the primary drivers for fatigue crack growth. This is appropriate because it is consistent with operating experience. Also, considering the current industry practice of having a refueling cycle of 1.5 years and the rare scenario of experiencing several heat-ups and cool-downs before a defective component is successfully repaired during a scheduled or forced outage, the NRC staff considers the assumed 5 cycles per year for heat-up and cool-down transients (specified in the accompanying table) conservative. Therefore, DCI-RAI-5 is resolved.

The response to DCI-RAI-6 clarified that the input of RV nozzle diameter may not reflect the real nozzle geometry. Instead, "all grouping of thickness and diameter inputs were evaluated....the grouping that provided the highest change in failure (MLOCA) frequency between 10-year and 20-year inspection intervals was selected as being limiting for that nozzle type." Therefore,

DCI-RAI-6 is resolved because the PWROG's approach of using the nozzle geometry that gave limiting results is conservative. Response to DCI-RAI-7 clarified that regardless what the SRRA input on flaw was called, "the SRRA Code simulate a maximum of one flaw at the worst stress location that could result in the first failure of the nozzle weld." Therefore, DCI-RAI-7 is resolved because the PWROG's approach of selecting the worst stress location for evaluation is conservative.

The response to DCI-RAI-8 provided PWROG's viewpoint regarding use of the crack inspection accuracy parameter of 0.24 versus 0.1. Since the NRC staff's conclusion does not depend on the results based on one particular performance factor, DCI-RAI-8 is resolved.

Based on the above evaluation and aided by the resolution of the eight DCI-RAIs, the NRC staff determined that the PWROG's use of SRRA Code in this application is appropriate and the PWROG's inputs for the SRRA Code are acceptable.

3.2.1.2 Change in Failure Frequencies Due to Extending the ISI Interval from 10 to 20 Years

The likelihood of RV nozzle weld failure was postulated to increase with increasing time of operation between inspections due to the growth of pre-existing fabrication flaws by fatigue. The likelihood of failure after an inspection decreased reflecting the possibility of identifying and repairing a flaw. The PFM approach in the TR simulated the growth of flaws over time between inspections and the repair of flaws that are detected during each ISI. The largest cracks were expected to exist at the end of the plant's operating life because, even with periodic inspection, flaws may be missed during an inspection. These flaws would remain in service and grow until eventually detected by ISI, failed in SLOCA, MLOCA, and LLOCA, or the end of plant life is reached. Therefore, the change in the likelihood of the event of concern is evaluated individually in the TR for SLOCA, MLOCA, and LLOCA.

Section 3.2.3 of the TR provides the bounding change-in-failure-frequency analysis results for all four types (Types A, B, C, and D) inlet and outlet nozzles for the failure modes of SLOCA, MLOCA, and LLOCA with 40 and 60 years' plant operation when the crack inspection accuracy parameter was assumed to be 0.24 (Tables 3-3 to 3-6). Detailed information supporting the MLOCA case in Tables 3-5 and 3-6 is provided in Tables 3-7 and 3-8, along with additional results for a crack inspection accuracy parameter of 0.1. The PWROG established the bounding nature of the results by first performing simulations at the highest and lowest weld temperatures and at different nozzle dimensions to determine the limiting case for the MLOCA. Subsequently, additional results using the identified limiting case were generated for the SLOCA and LLOCA for the normal and off-normal conditions.

During the implementation of a related TR, TR WCAP-16168-NP-A (Reference 4), which extended the ISI interval for RV welds, the NRC staff has concluded that relief from ASME Code 10 year inspection requirements should be requested every 20 years. Consistent with the requirement that relief be requested every 20 years, licensees need to determine whether the 40 or 60 year change in failure frequencies are most representative of the end of the requested 20 year extension.

In response to DRA-RAI-9, Westinghouse clarified that selecting whether the 40 or the 60 year failure frequencies should also include consideration of the plant life that has been used in the RI-ISI program. RI-ISI programs may have been based on the failure frequency after a 40 year plant life. If necessary, the plant life used in the RI-ISI program should be adjusted to match that required by the extension request. The examples in the TR sometimes use the 40 year

values and sometimes the 60 year values but the NRC staff does not endorse the examples - only the estimated change in failure frequencies and the general methodology. Each licensee should identify in its relief request which failure frequencies were selected and why.

Based on the NRC staff's evaluation of the PFM methodology in the SRRA Code, the associated key SRRA Code input parameters for this application, and the reasonable approach for determining the limiting case, as described above, the NRC staff accepts the PWROG's change-in-failure-frequency analysis results when used as described in the NRC staff endorsed version of this TR to evaluate the risk increase from extending the ISI interval for RV nozzle welds from 10 to 20 years.

3.2.2 Risk Assessment

In its response to DRA-RAI-1 and modifications to the TR, Westinghouse confirmed that at least one, and normally two, plant-specific changes in risk will be required to extend the RV nozzle welds ISI interval from 10 to 20 years: 1) the change in risk from the ASME Code, Section XI ISI program, and 2) the modified change in risk from the RI-ISI program if one is implemented.

The current ASME Code, Section XI requirements call for inspection of 100 percent of the RV nozzle welds every 10 years. The change in risk from the ASME Code, Section XI ISI program is required to identify the change in risk associated with relief from the 10 year inspection requirements in the ASME Code. Most licensees have, however, implemented a PWROG, EPRI, or ASME Code Case N-716 RI-ISI program to replace their ASME Code, Section XI ISI program. In this case, the change in risk from the RI-ISI program is required to be modified to include any additional change in risk associated with extending the interval.

The TR provides a methodology and part of the risk assessment inputs (the change in weld failure frequencies in Tables 3-3, 3-4, 3-5, and 3-6) for both risk assessments. The plant-specific risk assessment inputs to the change-in-risk calculations are the conditional core damage probabilities (CCDPs) and the conditional large early release probabilities (CLERPs) for SLOCA, MLOCA, and LLOCAs.

3.2.2.1 Change in Risk Associated with Relief from ASME Code, Section XI Inspection Interval Requirements

The change in risk is estimated by combining the appropriate change in weld failure frequencies from the TR with the plant-specific CCDPs and CLERPs. All change in failure frequency values are found in Tables 3-3 through 3-6. The TR proposes that failure frequency values without leak detection should be used for comparison to the ASME Section XI ISI interval. As discussed previously, the licensee will need to select, and justify, either the 40 or the 60 year life. The estimated change in risk for each LOCA size is estimated by multiplying the change in failure frequency, the number of welds in the nozzle, and the CCDP and CLERP for each size. The total change in risk from the increased interval is obtained by summing the risk from all LOCA sizes. The NRC staff concurs with the TR's direction that each licensee estimate the change in risk associated with extending the interval on the inspection of 100 percent of the welds from 10 to 20 years in each relief request that includes a request to extend the ISI intervals.

The NRC staff finds that the use of change in failure frequency without leak detection is conservative and therefore acceptable. The proposed calculations include the risk contribution for each possible weld failure and therefore yield estimates of the ΔCDF and ΔLERF that reflect

the change in risk from the increased intervals. The NRC staff concurs that an estimated change that is less than the guidelines from RG 1.174 indicates that any increase in risk caused by changing the ASME Code, Section XI ISI program to extend the ISI interval for nozzle welds from 10 to 20 years is small and satisfies Principle 4 in RG 1.174.

3.2.2.2 Change in Risk Associated with Relief from RI-ISI Inspection Interval Requirements

Most plants have implemented RI-ISI and no longer inspect 100 percent of the RV nozzle welds. The RI-ISI program development selects welds to inspect based on the risk significance of piping segments. One or more welds within high-safety-significant (HSS) piping segments are generally selected for inspection. Since failure in the primary reactor coolant loops can lead to un-isolable LLOCAs, these segments are often HSS. Some plants select welds other than the RV nozzle welds in the primary coolant loops to fulfill RI-ISI inspection requirements. Some plants select RV nozzle welds. If a plant has selected no RV nozzle welds for inspection, the risk of discontinuing inspections in those locations is already included in the RI-ISI change in risk estimates. Plants which have included inspection of one or more RV nozzle welds in their RI-ISI program should include the increased risk from extending the ISI interval in the RI-ISI program's change in risk estimate. The TR provides the change in failure frequencies and the methodology to include the increased risk from extending the ISI interval in the RI-ISI program change-in-risk estimate.

The TR proposes that the "with leak detection" failure frequencies be used in the RI-ISI change-in-risk calculations. Primary coolant leak detection capability in containment is mandated by regulation and the NRC staff finds that crediting this capability is acceptable and consistent with the RI-ISI methodologies.

The TR proposes a total of seven different methods to include the increased interval in the RI-ISI change-in-risk estimates; four of which could be used with the PWROG RI-ISI methodology, three of which could be used with the EPRI methodology.

PWROG RI-ISI

The PWROG RI-ISI methodology is based on weld failure frequencies developed using the same methods and computer programs used in this TR. The PWROG RI-ISI methodology uses a single, worst case, weld frequency to represent a segment failure frequency for each LOCA size regardless of the number of welds in the segment. A change in risk is only estimated when all inspections in a segment are discontinued, when one or more inspection is introduced in a previously uninspected segment, or when augmented inspections are improved. Changing the number of welds inspected within a segment does not result in an estimated change in risk. As described in the NRC SE on the PWROG RI-ISI methodology (Reference 3), the change-in-risk calculations were not intended to "precisely estimate the magnitude of the change, [but] the calculation can illustrate whether resulting change will be a risk increase or a risk decrease." The lack of precision in the risk increase estimate was found acceptable, in part, because the PWROG RI-ISI method included acceptance guidelines that called for a neutral change in risk or a risk decrease instead of the risk increases permitted according to the RG 1.174 guidelines.

The TR proposes one traditional method and three alternative methods to estimate the change in risk between the ASME program and a PWROG RI-ISI program that includes an extended ISI interval for selected RV nozzle welds. The traditional method is consistent with the PWROG RI-ISI change-in-risk methodology, but includes the addition of the increase in risk associated with the RV nozzle weld ISI interval extension. In response to DRA-RAI-7, Westinghouse

provided detailed equations describing the variables and the manipulations required to implement each of the three alternative methods. All three alternative methods modify the PWROG RI-ISI change-in-risk methodology by assigning the segment failure frequency to each weld in the segment, and accounting for changing the number of inspections within each segment. The three alternative methods differ by increasing the resolution of the CCDPs and CLERPs assigned to each segment from a worst case plant-wide estimate to a worst case system estimate and finally to a segment-specific estimate. Increasing the resolution will result in lower change in risk estimates.

The NRC staff finds that all four methods may be used. When using the traditional PWROG RI-ISI change-in-risk methodology each nozzle is considered a segment. Therefore the failure frequency of a nozzle segment with two welds can be represented by a single most limiting failure frequency. When using one of the alternative methods, the failure frequency of all welds must be included as described in the detailed equations. Therefore the failure frequency of a nozzle with two welds is twice the failure frequency of one weld as illustrated in the equations provided for each of the alternative methods. These differences are reflected in the different acceptance guidelines discussed below.

When the traditional PWROG RI-ISI change-in-risk methodology is used, the original change-in-risk methodology is used and therefore the original risk neutral acceptance guidelines are applied. However, when one of the three alternative methods is used, the acceptance guidelines are increased from risk neutral to reactor coolant system and total risk increases that would meet the very small risk increase guidelines in RG 1.174. This modification of acceptance guidelines is consistent with the alternative methods which now account for the changes in the number of welds inspected instead of the number of segments inspected. If the risk increase guidelines cannot be met with the current RI-ISI program, the TR directs the licensee to add inspections until the guidelines are met. The NRC staff finds the methodology and the associated acceptance guidelines acceptable because they incorporate any risk increase from extending the interval into the RI-ISI program. The resolution and thereby the precision of the change-in-risk estimates are increased in the alternative methods by accounting for the changes in the number of welds inspected and therefore changing the acceptance guidelines to the larger acceptable risk increases continue to provide confidence that the increase in risk is acceptable.

EPRI/N-716 RI-ISI

The EPRI/N-716 RI-ISI methodology is based on weld failure likelihood 'bins' determined only by the presence or absence of potential degradation mechanisms. Identification of segment safety significance and determination of the number of inspections is based on which degradation mechanism may be present and the CCDP and CLERP in each segment. The change-in-risk estimates in the EPRI/N-716 method use a single break size frequency and single values for CCDP and CLERP. The change-in-risk estimate is the product of the failure frequency of an uninspected weld associated with the potential degradation mechanism, the estimated CCDP and CLERP (EPRI/N-716 Method 1), and, optionally, an inspection effectiveness (IE) or probability of detection (POD) factor between 0 and 1 that characterizes the likelihood that inspections will identify flaws before weld failure (EPRI/N-716 Method 2). This IE factor is similar to the crack inspection accuracy parameter discussed in Section 3.2.1.1 of this SE and included in the frequency estimates in Tables 3-3 through 3-6 of the TR. Therefore, any calculation that combines frequencies from Tables 3-3 through 3-6 together with an IE factor would incorrectly account twice for inspections.

The TR repeats a variety of equations for calculating the change in risk between the ASME program and an EPRI/N-716 RI-ISI program (Equations 3-11, 3-12, 3-13, and 3-14). Inspection of the TR reveals three proposals for including the increased risk from extending the ISI interval into the EPRI/N-716 RI-ISI program.

The first method uses the EPRI/N-716 qualitative method augmented by the change-in-risk caused by extending the ISI interval for RV nozzle welds. When used to develop a RI-ISI program, the qualitative method concludes that there is no expected increase in risk from implementing the RI-ISI program. Therefore, the total change-in-risk from implementing the RI-ISI program and extending the ISI interval for RV nozzle weld inspection interval is not expected to be greater than the risk increase from extending the ISI interval. The risk increase from extending the ISI interval can be directly compared to the acceptance guidelines.

The second method estimates the increased risk from extending the ISI interval using the change in failure frequency developed in this TR and adds that increase in risk to the EPRI/N-716 RI-ISI change in risk. The increased risk is the product of the increased weld failure frequency (from Tables 3-3 through 3-6) and the plant specific CCDP and CLERP for reactor coolant loop LOCAs as described in the TR. Simply adding this risk increase to the increase in risk from implementing an EPRI/N-716 RI-ISI program is consistent with adding the increased risk from the extended interval with the increased risk from implementation of the RI-IS program and therefore acceptable.

The third method estimates the increased risk from extending the ISI interval by modifying the IE or POD factors that would be applied to the welds with the extended ISI interval instead of using the change in failure frequencies in Tables 3-3 through 3-6. Equations 3-12 and 3-13 provide the EPRI/N-716 formulas and parameters for the change in CDF calculations that use the IE or POD factor. Modifying the POD (currently done, for example, to reflect increased inspection volume) must be done by directly changing the value because there is no time component in POD. There is a time component in the Markov model which can be used to calculate the IE. The Markov model has been found acceptable for use in developing an EPRI/N-716 RI-ISI program. The NRC staff concurs that these equations and models can be used to reflect changing the inspection interval. However, modifying the POD or the IE require the use of parameters not discussed in the TR including, for example, estimates for the uninspected nozzle weld failure frequency.

The NRC staff finds that the first two methods (qualitative and using the change in failure frequencies developed in this TR) to incorporate the extension of the ISI interval into the EPRI/N-716 RI-ISI program change-in-risk estimates are consistent with the EPRI methodology and the development of failure frequencies in this TR and therefore acceptable. The failure frequencies in Tables 3-3 through 3-6 of the TR may be used in the first two methods and this TR referenced. Using the third method requires the use of parameters not discussed in the TR. Therefore, Section 4.0 of this SE, "Conditions and Limitations," states that licensees that use change to the POD or the IE to incorporate the extension of the ISI interval must identify and justify all input parameters.

Unlike the PWROG RI-ISI methodology, the change-in-risk acceptance guidelines are not changed. The NRC staff finds this is appropriate and acceptable because the EPRI/N-716 RI-ISI methodology uses changes in the number of welds inspected and these additional risk calculations also use changes in the number of welds inspected.

3.2.2.3 Evaluation of PRA Technical Adequacy

Technically adequate is defined, at the highest level, as an analysis that is performed correctly, in a manner consistent with accepted practices, commensurate with the scope and level of detail required to support the proposed change. The TR does not address the technical adequacy of the PRA.

The TR requires CCDPs and CLERPs for SLOCA, MLOCA, and LLOCA. The acceptance guidelines are comparable to the acceptance guidelines for a RI-ISI program. The NRC staff finds that a PRA that is adequate to support the development of a RI-ISI program is adequate to support the change-in-risk evaluations described in the TR because the PRA calculations required by the TR are fewer than, or equivalent to, those required to develop a RI-ISI program. However, the licensee will need to discuss any changes to the PRA or associated reviews and dispositions of findings since the RI-ISI application. Any licensee that has no RI-ISI program that requests relief to extend the ISI interval would need to justify that its PRA is technically adequate to support the request.

3.3 Submit Proposed Change

The fourth and final element in the RG 1.174 approach is the development and submittal of the proposed change to the NRC. Since the 10-year ISI interval is required by Section XI, IWB-2412, as codified in 10 CFR 50.55a, a relief for an alternative, in accordance with 10 CFR 50.55a(a)(3)(i), must be submitted and approved by the NRC to extend the ISI interval. Licensees that submit a request for an alternative based on the TR need to submit plant-specific information summarizing which methods from the TR were used and addressing each of the limitations and conditions in Section 4.0 of this SE.

3.4 Conformance to RG 1.174

In addition to the four element approach discussed above, RG 1.174 states that RI plant changes are expected to meet a set of five key principles. This section summarizes these principles and the NRC staff findings related to the conformance of the TR methodology with changes to ISI programs in general and with the extension of the ISI interval proposed in the TR.

Principle 1 states that the proposed change must meet the current regulations unless it is explicitly related to a requested exemption or rule change. ISI of ASME Code Class 1, 2, and 3 components is performed in accordance with Section XI of the ASME Code and applicable addenda as required by 10 CFR 50.55a(g), except where specific relief has been granted by the NRC pursuant to 10 CFR 50.55a(g)(6)(i). This RI application requires a request for an alternative under 10 CFR 50.55a(a)(3)(i) which meets the current regulations and, therefore, satisfies Principle 1.

Principle 2 states that the proposed change shall be consistent with the defense-in-depth philosophy[1]. The NRC staff believes that ISI is an integral part of defense-in-depth and extending the interval may change the robustness of the reactor coolant pressure boundary, albeit very slightly. However, the NRC staff concludes that increasing the failure frequency by

1 The NRC staff finds the defense-in-depth discussion in, and following, Table 3-12 of the TR, while supportive of defense-in-depth, is more descriptive of the strategies that will be used to monitor the impact of the proposed change and addresses the TR discussion under Principle 5.

extending the ISI interval is similar to increasing the failure frequency by discontinuing inspections in RI-ISI. Unlike RI-ISI, these increases are not offset by inspecting new locations but, also unlike RI-ISI, the scope of the change is limited to the small, well defined, population of nozzle welds. Therefore, consistent with the NRC staff conclusions endorsing RI-ISI, the NRC staff concludes that there is a reasonable assurance that the resulting ISI program will provide a substantive ongoing assessment of piping condition and therefore Principle 2 is met.

Principle 3 states that the proposed change shall maintain sufficient safety margins. The TR states that no safety analyses are changed. The NRC staff concurs that there are no changes to the evaluations of design-basis accidents in the Final Safety Analysis Report (FSAR). This proposal is only to extend the ISI interval and no other portions of the current inspection requirements are eliminated. The NRC staff finds that extending the ISI interval may permit some flaws to remain undetected and thereby reduce the margin to failure of these welds. However, the proposal does not, for example, change the acceptance criteria used to determine whether any identified flaws are acceptable and therefore the NRC staff finds that sufficient safety margins are maintained and Principle 3 is met.

Principle 4 states that when proposed changes result in an increase in CDF or risk, the increases should be small and consistent with the intent of the Commission's Safety Goals. The TR provides methods to estimate the change in risk associated with changing the ASME Code, Section XI inspection program for RV nozzle welds from 10 to 20 years, and from changing the ISI interval for RV nozzles in an existing RI-ISI program from 10 to 20 years. Provisions to increase the number of welds for inspection if the acceptance guidelines are not met are provided. Therefore, Principle 4 is met.

Principle 5 states that the impact of the proposed change should be monitored using performance measurement strategies. The TR states that nondestructive examinations will still be conducted, but on a less frequent basis not to exceed 20 years and that indications of potential generic degradation mechanisms of RV nozzle welds will still be available during this extended ISI interval (e.g., foreign experience, inspection of other similar locations, and periodic testing with visual examinations). To demonstrate that there will be a sampling of inspections performed over the 20-year interval that will provide an indication of emerging issues; a somewhat optimized implementation schedule was 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 shell weld inspections, the schedule is based on the schedule developed for the RV shell weld ISI interval extension as discussed in the most recent Revision of TR WCAP-16168-NP-A. 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. Any indications that are found during the inspections will be treated as flaw indications and evaluated under ASME Code, Section XI, and so there is no change to this monitoring aspect. Therefore, Principle 5 is met.

4.0 CONDITIONS AND LIMITATIONS

This section summarizes the conditions and limitations that should be addressed by all applicants in their relief requests to increase the ISI interval for RV nozzle welds from 10 years to 20 years:

- The PFM analyses supporting the TR were based on a key assumption - one surface flaw per weld. Therefore, consistent with the TR guidance in Section 2.2, the NRC staff

requires applicants to validate that at most one surface breaking flaw is present based on past ISI results. If multiple surface breaking flaws have been detected in past inspections, then the resulting change in failure frequency shall be multiplied by the number of surface flaws. If the total flaw size from this method exceeds the dimension assumed in the TR (i.e., a through-wall depth of greater than six percent of the wall thickness and a length equal to six times the depth), a weld-specific PFM analysis should be performed to develop a weld-specific change-in-frequency value. Validation of this flaw assumption must also be performed in the future through continued monitoring every 20 years.

- Licensees must identify the years in which future inspections will be performed. The dates provided must be within plus or minus one refueling cycle of the dates identified in the implementation plan referenced in the most recent Revision of TR WCAP-16168-NP-A.

- The NRC staff accepts the PWROG's change-in-failure-frequency analysis results in Tables 3-3 through 3-6 when used as described in the NRC staff endorsed version of this TR to evaluate the risk increase from extending the ISI interval for RV nozzle welds from 10 to 20 years. Licensees must select the 40 or 60 year change-in-failure-frequency results, clarify the relationship between the selected life time and the values used in the RI-ISI, and justify the selected life time values.

- Licensees must submit plant-specific change-in-risk results in the relief requests as described in the TR. A change in risk between the ASME requirements and the extended ISI interval must always be provided. If the licensee has a RI-ISI program, the change in RI-ISI risk results including the extended intervals should be provided. If any change in risk exceeds the applicable risk guidelines in the TR, the licensee should identify and justify the deviation.

- Licensees must identify specifically which of the change-in-risk equations and methods in the TR were used. Any deviations from the selected equations and\or methods must be identified and justified.

- The use of the changes to the IE or the POD to reflect changes in risk caused by extending the inspection interval may not use the change in failure frequencies in Tables 3-3 through 3-6. Each licensee that uses this method must identify and justify all parameter values used.

- Licensees should address PRA quality in their relief request. Licensees relying on a NRC staff approved RI-ISI program to demonstrate PRA quality should provide this statement in their submittal and provide any updated information appropriate for the application since the RI-ISI application. Licensees without a NRC staff approved RI-ISI program must describe the technical adequacy of their PRA in the relief request.

- The NRC staff does not endorse the BV-1 and TMI-1 examples or the use of any quantative results from any tables besides Tables 3-3 through 3-6 of the TR. Licensees (including BV-1 and TMI-1) may not reference the examples to justify any evaluation or calculation.

5.0 CONCLUSION

The NRC staff has reviewed TR WCAP-17236-NP and concludes that the TR, as modified by the conditions and limitations summarized in Section 4.0 of the SE, provides an acceptable methodology that can be used to support a request to extend the ISI interval for Category B-F or B-J RV nozzle welds that do not contain Alloy 82/182 from 10 to 20 years.

Section 3.2.1.1 of this SE mentioned that due to low neutron fluence and benign coolant condition, fatigue crack growth was identified as the only growth mechanism of concern in this application. Also discussed in this section are the postulated surface crack, the fatigue stress range, number of fatigue cycles, and design limiting stresses. Since extending the RV ISI interval could increase the risk of RV failure from such cracks, the SRRA Code was used to perform the fatigue crack growth analysis to produce PFM results for the subsequent risk-informed calculations. Based on the NRC staff evaluation of Section 3.2.1.1, the NRC staff has concluded that the TR has appropriately postulated and modeled the potential change in failure frequency risk that could be caused by fatigue crack growth over the life of operating facilities. Therefore the NRC staff accepts the PWROG's change-in-failure-frequency analysis results (in Tables 3-3 through 3-6) when used as described in the NRC staff endorsed version of this TR to evaluate the risk increase from extending the ISI interval for RV nozzle welds from 10 to 20 years.

The evaluation in the TR illustrates the variability in the estimated annual failure frequencies. This variability is incorporated into all the methodologies approved for the development of RI-ISI programs. The analysis that was performed to support this TR does not reduce this variability and therefore the NRC staff does not endorse any changes to PWROG or the EPRI/N-716 RI-ISI program methodology development.

Based on the above conclusions, the ASME Code Section XI ISI interval for examination categories B-F and B-J welds in PWR RVs can be extended from 10 years to a maximum of 20 years. Since the 10 year ISI interval is required by Section XI, IWB-2412, as codified in 10 CFR 50.55a, a request for an alternative, in accordance with 10 CFR 50.55a(g)(6)(i), must be submitted and approved by the NRC to extend any facility's ISI interval. During the implementation of a related TR WCAP-16168-NP-A (Reference 4) which extended the ISI interval for RV welds, the NRC staff has concluded that relief from ASME Code 10 year inspection requirements should be requested every 20 years. Similarly, relief from the ASME Code 10 year inspection requirement should be requested every 20 years when applying TR WCAP-17236-NP, Revision 0, in coordination with the TR WCAP-16168-NP-A application. Each licensee shall identify the years in which future inspections will be performed. The dates provided must be within plus or minus one refueling cycle of the dates identified in the implementation plan referenced in the most recent Revision of TR WCAP-16168-NP-A.

The NRC staff does not endorse the BV-1 and TMI-1 examples. Licensees (including BV-1 and TMI-1) may not refer to the examples to justify any evaluation or calculation. The NRC staff will not repeat its review of the matters described in the TR WCAP-17236-NP, as modified by the attachment to the supplement dated August 18, 2011, when the report appears as a reference in a request for an alternative, except to ensure that the material presented applies to the specific plant involved and the licensee has submitted all the information requested in Section 4.0 of this SE.

6.0 REFERENCES

1. WCAP-17236-NP, Revision 0, "Risk-Informed Extension of the Reactor Vessel Nozzle Inservice Inspection Interval," September 2010 (Agencywide Document Access and Management System (ADAMS) Accession No. ML102790088).

2. Letter from Melvin L. Arey Jr., PWR Owners Group, "Responses to the NRC Supplemental Request for Additional Information (RAI) on PWR Owners Group (PWROG) WCAP-17236-NP, Revision 0, 'Risk Informed Extension of the Reactor Vessel Nozzle Inservice Inspection Interval,' (TAC NO. ME4878) PA-MSC-0440," August 26, 2011 (ADAMS Accession No. ML11280A084).

3. WCAP-14572, Revision 1-NP-A, "Westinghouse Owners Group Application of Risk-Informed Methods to Piping Inservice Inspection Topical Report," February 1999 (ADAMS Accession Nos. ML042610469).

4. WCAP-16168-NP-A, Revision 3, "Risk-Informed Extension of the Reactor Vessel In-Service Inspection Interval," October, 2011 (ADAMS Accession No. ML11306A084).

5. EPRI Topical Report TR-112657, Revision B-A, "Revised Risk-Informed Inservice Inspection Evaluation Procedure," December 1999 (ADAMS Accession No. ML013470102).

6. ASME Code Case N-716, "Alternative Piping Classification and Examination Requirements, Section XI, Division 1," April 19, 2006.

7. U.S. NRC, NUREG-0800, Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," Section 19.2, "Review of Risk Information Used to Support Permanent Plant-Specific Changes to the Licensing Basis: General Guidance," June 2007 (ADAMS Accession No. ML071700658).

8. U.S. NRC, Regulatory Guide 1.174, Revision 1, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," November 2002 (ADAMS Accession No. ML023240437).

9. WCAP-14572, Revision 1-NP-A, Supplement 1, "Westinghouse Structural Reliability and Risk Assessment (SRRA) Model for Piping Risk-Informed Inservice Inspection," February 1999 (ADAMS Accession No. ML042610375).

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Date: May 23, 2012

RESOLUTION OF COMMENTS BY THE OFFICE OF NUCLEAR REACTOR REGULATION

REGARDING THE DRAFT SAFETY EVALUATION FOR

TOPICAL REPORT WCAP-17236-NP, REVISION 0, "RISK-INFORMED EXTENSION OF THE

REACTOR VESSEL NOZZLE INSERVICE INSPECTION INTERVAL"

PRESSURIZED WATER REACTOR OWNERS GROUP

PROJECT NO. 694

This Attachment provides the U.S. Nuclear Regulatory Commission (NRC) staff's review and disposition of the comments made by the Pressurized Water Reactor Owners Group (PWROG) on the draft safety evaluation for Topical Report WCAP-17236-NP, Revision 0, 'Risk-Informed Extension of the Reactor Vessel Nozzle Inservice Inspection Interval' (Agencywide Documents and Management System (ADAMS) Accession No. ML120240450). The PWROG provided its comments by a letter dated March 22, 2012 (ADAMS Accession No. ML12083A195).

ATTACHMENT

Suggested Changes on NRC Draft Safety Evaluation for WCAP-17236-NP, Revision 0

Comment #	Page	Section	Location*	Line(s)	Editorial (E) or Technical (T)	Description of Suggested Change	NRC Staff Comment Resolution
1	1	1.0	P3, S1	32	E	Before the text "Reference 3" add the text "(ASME-XI, Appendix R, Method A)"	Accepted. The SE now states, (ASME Code, Section XI, Appendix R, Method A (Reference 3))."
2	1	1.0	P4, S3	38-39	E	Before the text "Reference 5" add "(ASME-XI, Appendix R, Method B), "	Accepted. The SE now states, (ASME Code, Section XI, Appendix R, Method B (Reference 5))."
3	2	3.0	S1	33	E	Delete the word "and" before the word "which"	Accepted.
4	3	3.1	P2	4-9	T	The SER states that licensees must identify in their requests for relief the dates in which they plan to perform their inspections and they must be within plus or minus one outage of the dates provided in Table 3-13 of the TR. Table 3-13 of the TR is based on the PWROG plan for implementing the RV ISI interval extension as documented in PWROG letter OG-10-238. This plan is referenced in the recently revised SER for WCAP-16168-NP-A, Revision 3. Since these RV nozzle exams will be performed at the same time as the RV exams, it would be more efficient for industry and the NRC to manage implementation based on one schedule rather than two. It is suggested that the SER be revised to reference WCAP-16168-NP-A as the schedule for RV nozzle ISI interval extension implementation. The PWROG proposes to revise the sentence on Page 3-22 of the TR starting with "Since the RV nozzle weld inspections are..." to	Accepted. The SE now states that, "The dates provided must be within plus or minus one refueling cycle of the dates identified in the implementation plan referenced in the most recent Revision of TR WCAP-16168-NP-A."

Suggested Changes on NRC Draft Safety Evaluation for WCAP-17236-NP, Revision 0

Comment #	Page	Section	Location*	Line(s)	Editorial (E) or Technical (T)	Description of Suggested Change	NRC Staff Comment Resolution
						read "Since the RV nozzle weld inspections are performed at the same time as the RV inspections, the proposed inspection dates in the implementation plan are consistent with those in the plan for implementation of the RV ISI interval extension in the latest revision of WCAP-16168-NP-A, (Reference 6)." Furthermore, Reference 6 will be revised to reference WCAP-16168-NP-A, Revision 3, rather than WCAP-16168-NP-A, Revision 2.	
5	4	3.2.1	P1	4	E	Delete "fracture mechanics" and the parenthesis of "PFM" since they are not part of the TR sentence that the SE quoted from.	Accepted.
6	4	3.2.1	P1, last S	10	E	It is recommended that the text "of an aspect ratio of 6 to 1" be replaced with the text "with this initial through-wall depth distribution"	Accepted.
7	4	3.2.1	P2, last S	20	E	Add "every 20 years" after "continued monitoring"	Accepted.
8	7	3.2.2	P2, S5	16	T	It is believed that the intent of the text "ASME" in "EPRI/ASME" is to refer to ASME Section XI Code Case N-716. If so, it is suggested that the text be revised to "PWROG, EPRI, or ASME Code Case N-716". "EPRI/ASME" should also be changed throughout the SER to "EPRI/N-716". It is understood that Code Case N-716 is an ASME Code Case, but using only the word "ASME" leads the reader to believe that you are referring to the traditional ASME Section XI approach or	Accepted.

Suggested Changes on NRC Draft Safety Evaluation for WCAP-17236-NP, Revision 0

Comment #	Page	Section	Location*	Line(s)	Editorial (E) or Technical (T)	Description of Suggested Change	NRC Staff Comment Resolution
						one of the ASME Section XI Nonmandatory Appendix R methods.	
9	7	3.2.2.1	P1, S1	33	E	Change "discussed above" to "discussed previously"	Accepted.
10	8	3.2.2.2	P1, S1	3	E	Change "RV nozzle welds" to "RV nozzle-to-pipe (RV nozzle) welds" since it is repeated several times	Not accepted. "RV nozzle welds" was first used in Section 2.0 (Page 2) to represent "RV nozzle-to-pipe welds" and, since its first occurrence, appeared numerous times throughout the SE text.
11	8	3.2.2.2	Last P	22	T	It is stated that the TR proposes a total of seven different methods. Based on the comment # 13 (below), there should be a total of 8 different methods, 4 for PWROG and 4 for EPRI.	Accepted. Resolution of #12 and #13 added a PWROG method, but resolution of #16 indicated that two of the EPRI methods were, in fact, the same so there is still a total of 7 methods.
12	8	3.2.2.2	All	42-51	T	This paragraph says that "The TR proposes three alternative methods to estimate the change in risk between the ASME program and a PWROG RI-ISI program that includes an extended ISI interval for selected RV nozzle welds." It is stated later in the paragraph that "All three methods modify the PWROG RI-ISI change-in-risk methodology by assigning the segment failure frequency to each weld in the segment, and accounting for changing the number of inspections within each segment." However, the SER does not mention that the TR also proposes a methodology that is consistent with the	Accepted, the original PWROG methodology may be applied to the nozzles when the original acceptance guidelines can be satisfied.

Suggested Changes on NRC Draft Safety Evaluation for WCAP-17236-NP, Revision 0

Comment #	Page	Section	Location*	Line(s)	Editorial (E) or Technical (T)	Description of Suggested Change	NRC Staff Comment Resolution
						PWROG change-in-risk methodology in that the number of inspections within each segment is not considered. This original approach is discussed in Section 2.4.1 and in Section 3.2.5.1, Page 3-31, "Evaluation of Effect of RV Nozzle ISI Interval Extension." An example of this approach is shown in Table 3-15 for Beaver Valley Unit 1. The first sentence of Section 3.2.5.1, Page 3-31, "Alternative Change-in-Risk Evaluation Methods," states "If the PWROG original change-in-risk acceptance 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". The original PWROG change-in-risk method needs to be added as an acceptable method throughout the SER.	
13	9	3.2.2.2	P1	1-11	T	This paragraph states "...in response to DRA-RAI-2 and DRA-RAI-4, Westinghouse states that nozzles should be treated as segments and therefore nozzles with two welds should only use a single weld frequency (i.e., segment basis). This is inconsistent with the modified PWROG methodology..." As noted in Comment 13, the SER does not mention the original PWROG methodology in which the number	Accepted, the original PWROG methodology may be applied to the nozzles when the original acceptance guidelines can be satisfied.

Suggested Changes on NRC Draft Safety Evaluation for WCAP-17236-NP, Revision 0

Comment #	Page	Section	Location*	Line(s)	Editorial (E) or Technical (T)	Description of Suggested Change	NRC Staff Comment Resolution
						of welds is not considered. Further, the response to DRA-RAI-2 says "However, when evaluating the impact on the RI-ISI program for plants that have implemented the PWROG RI-ISI methodology and that are using the PWROG original change-in-risk evaluation, the evaluation is conducted on a per-segment basis. Thus, as discussed in the response to DRA-RAI-4, the change in risk added to the change in risk from the RI-ISI element selection should be calculated based on one weld per nozzle." It is suggested that the text in the SER be removed and to provide clarification, the PWROG proposes to add the text "...and the calculations are conducted on a per segment basis." to the end of the first sentence of the second paragraph of Section 2.4.1 of WCAP-17236-NP. It is agreed that if one of the 3 alternative methods are used, in which the number of welds is considered, the nozzles should be treated as two welds when two welds exist.	
14	9	3.2.2.2	P4	19	E	The word "associated" is missing the "d"	Accepted.
15	9 10	3.2.2.2	P3	47-50 31-33	T	It is stated that "The first method is a qualitative method. As stated in the TR, "[t]his method implicitly assumes that all inspections are performed on the same interval." The discussion in the TR does not provide any alternative to this assumption which is no longer valid if the ISI interval is extended and therefore the NRC staff does	Accepted, if the original RI-ISI program was determined to be risk neutral, the only increase would come from extending the nozzle weld inspection intervals.

Suggested Changes on NRC Draft Safety Evaluation for WCAP-17236-NP, Revision 0

Comment #	Page	Section	Location*	Line(s)	Editorial (E) or Technical (T)	Description of Suggested Change	NRC Staff Comment Resolution
						not approve the use of the qualitative method." However, the TR does state on Page 3-39 that "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." The PWROG proposes to replace "Regulatory Guide 1.174" with "EPRI RI-ISI". With this change, the PWROG believes that the qualitative method should be an acceptable method for evaluating the acceptability of the effect on the RI-ISI program. The SER should therefore be revised to allow the use of the qualitative method.	
16	10	3.2.2.2	P5	14-20	T	It is stated that "In the discussion following these equations (3-2 and 3-2), the TR states that changes in failure frequency from Tables 3-3 through 3-6 should somehow be used in the equations. This discussion is inconsistent with the definitions of the parameters in the equations and would yield incorrect results when combined with changes in the IE factors. Therefore, licensees that use the frequencies from Tables 3-3 through 3-6 cannot use these equations and parameter	Accepted. The general, mutually agreed observation that the change-in-frequencies from Table 3-3 through 3-6 should not be used with an IE or POD is accurate. The second and the third (of the original four) EPRI methods appeared to propose using an IE or a POD at all welds, including the nozzles. However, the PWROG clarified that the equations

Suggested Changes on NRC Draft Safety Evaluation for WCAP-17236-NP, Revision 0

Comment #	Page	Section	Location*	Line(s)	Editorial (E) or Technical (T)	Description of Suggested Change	NRC Staff Comment Resolution
						definitions and must report this deviation and identify and justify their proposed method and input values." It is assumed that the text that is being referred to is in the section "Method B" on page 3-43 of the TR. It was never the intention of the TR to propose that the change-in-failure frequencies be used to calculate inspection effectiveness factors and we do not believe that the text in the TR implies this. We agree that this would be incorrect. What is proposed is that even if the Markov Model had been used to originally calculate the change-in-risk for the RI-ISI program, the change-in-failure frequencies in Tables 3-3 through 3-6 could be used to calculate the incremental increase in risk from the RV nozzle ISI interval extension. This incremental increase in risk for the nozzles would be added to the total plant and RC system risk as determined for the RI-ISI program. This approach is similar to the approach defined for Method 2. The PWROG suggests that the quoted text from the SER be removed because we do not believe that it implies the use of the bounding change-in-failure frequencies in the determination of inspection effectiveness factors. However, it would be acceptable to the PWROG for the NRC wants to place a limitation in Section 4 stating that the bounding-change-in-failure frequencies may not be used to calculate	illustrated the EPRI method at all welds other than the nozzles - the nozzles would use the change in nozzle weld frequencies in the TR. Thus the previous second and third methods both treat the nozzles the same way and are one method with respect to the TR. The third method (previously the fourth method) does require the use of frequencies other than those in the TR and the requirement that each licensee fully report were it obtained all its parameters when using this last method is retained.

Suggested Changes on NRC Draft Safety Evaluation for WCAP-17236-NP, Revision 0

Comment #	Page	Section	Location*	Line(s)	Editorial (E) or Technical (T)	Description of Suggested Change	NRC Staff Comment Resolution
						inspection effectiveness factors, since we have no intention to do so.	
17	12	3.4	P3	26	T	As stated in Comment 4, the PWROG proposes to revise the TR to refer to WCAP-16168-NP-A as the basis for the implementation schedule.	Accepted. Consistent with NRC resolution of Comment 4.
18	12	4.0	B1	47	E	Because satisfaction of all Section 4.0 items is required for NRC acceptance in Section 5.0, please add "every 20 years" after "continued monitoring" to avoid any confusion in the future.	Accepted.
19	13	4.0	B1	2-5	T	The PWROG is of the opinion that the basis for the failure frequencies, whether 40 or 60 years, should be consistent with the piping RI-ISI program at all times. The suggestion to always be conservative is in contradiction with other TR requirements. It is recommended that the last sentence of this paragraph be removed.	Accepted.
20	13	4.0	B3	15-17	T	As noted in comment 15, the PWROG believes that this condition\limitation for the qualitative method should be removed.	Accepted.
21	13	4.0	B4	19-20	T/E	It is suggested that this condition \ limitation be revised to read as follows: "Licensees must identify specifically which of the change-in-risk equations and methods in the TR were used. Any deviations from the selected equations and\or methods must be identified and justified."	Accepted.
22	13	4.0	B6	36	E	It is requested that the text "...may not refer to the examples to justify any evaluation or calculation." be changed to	Accepted.

Suggested Changes on NRC Draft Safety Evaluation for WCAP-17236-NP, Revision 0

Comment #	Page	Section	Location*	Line(s)	Editorial (E) or Technical (T)	Description of Suggested Change	NRC Staff Comment Resolution
						"may not reference the examples as a basis for a plant specific request for alternative." The use of the word "refer" gives the impressions that the examples are not suitable for serving their intended purpose, which is to illustrate the method.	
23	14	5.0	P3	26-28	T	As stated in Comment 4, the PWROG proposes to revise the TR to refer to WCAP-16168-NP-A as the basis for the implementation schedule.	Accepted. Consistent with NRC resolution of Comment 4.
24	15	6.0	R4	1-2	E	WCAP-16168-NP-A, Revision 2 has been revised and is now Revision 3.	Accepted.
25	15	6.0	R6	8-9	E	No ASME approval date is specified for Code Case N-716.	Accepted.
26	15	6.0	R10	24-27	T	As stated in Comment 4, the PWROG proposes to revise the TR to refer to WCAP-16168-NP-A as the basis for the implementation schedule. Therefore, this reference is no longer needed and can be removed.	Accepted.

*Note: B is for bullet, P is for paragraph, R is for reference, and S is for sentence.

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PWR Owners Group
Member Participation[1] for PA-MSC-0440

Utility Member	Plant Site(s)	Participant	
		Yes	No
AmerenUE	Callaway (W)		X
American Electric Power	D.C. Cook 1&2 (W)	X	
Arizona Public Service	Palo Verde Unit 1, 2, & 3 (CE)		X
Constellation Energy Group	Calvert Cliffs 1 & 2 (CE)		X
Constellation Energy Group	Ginna (W)		X
Dominion Connecticut	Millstone 2 (CE)	X	
Dominion Connecticut	Millstone 3 (W)		X
Dominion Kewaunee	Kewaunee (W)	X	
Dominion VA	North Anna 1 & 2, Surry 1 & 2 (W)	X	
Duke Energy	Catawba 1 & 2, McGuire 1 & 2 (W)	X	
Duke Energy	Oconee 1, 2, 3 (B&W)	X	
Entergy	Palisades (CE)	X	
Entergy Nuclear Northeast	Indian Point 2 & 3 (W)		X
Entergy Operations South	Arkansas 2, Waterford 3 (CE), Arkansas 1 (B&W)	X	
Exelon Generation Co. LLC	Braidwood 1 & 2, Byron 1 & 2 (W), TMI 1 (B&W)	X	
FirstEnergy Nuclear Operating Co.	Beaver Valley 1 & 2 (W)	X	
	Davis-Besse (B&W)	X	
Florida Power & Light Group	St. Lucie 1 & 2 (CE)	X	
Florida Power & Light Group	Turkey Point 3 & 4, Seabrook (W)	X	
Florida Power & Light Group	Pt. Beach 1&2 (W)		X
Luminant Power	Comanche Peak 1 & 2 (W)		X
Xcel Energy	Prairie Island 1&2 (W)	X	
Omaha Public Power District	Fort Calhoun (CE)		X
Pacific Gas & Electric	Diablo Canyon 1 & 2 (W)		X
Progress Energy	Robinson 2, Shearon Harris (W),		X
	Crystal River 3 (B&W)		X
PSEG – Nuclear	Salem 1 & 2 (W)		X
Southern California Edison	SONGS 2 & 3 (CE)	X	
South Carolina Electric & Gas	V.C. Summer (W)		X
So. Texas Project Nuclear Operating Co.	South Texas Project 1 & 2 (W)		X
Southern Nuclear Operating Co.	Farley 1 & 2, Vogtle 1 & 2 (W)		X
Tennessee Valley Authority	Sequoyah 1 & 2, Watts Bar (W)	X	
Wolf Creek Nuclear Operating Co.	Wolf Creek (W)		X

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
International Member Participation[1] for PA-MSC-0440

Utility Member	Plant Site(s)	Participant	
		Yes	No
British Energy	Sizewell B		X
Electrabel (Belgian Utilities)	Doel 1, 2 & 4, Tihange 1 & 3		X
Hokkaido	Tomari 1 & 2 (MHI)		X
Japan Atomic Power Company	Tsuruga 2 (MHI)		X
Kansai Electric Co., LTD	Mihama 1, 2 & 3, Ohi 1, 2, 3 & 4, Takahama 1, 2, 3 &4 (W & MHI)		X
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Korea Hydro & Nuclear Power Corp.	Yonggwang 3, 4, 5 & 6 Ulchin 3, 4 , 5 & 6(CE)		X
Kyushu	Genkai 1, 2, 3 & 4, Sendai 1 & 2 (MHI)		X
Nuklearna Electrarna KRSKO	Krsko (W)		X
Nordostschweizerische Kraftwerke AG (NOK)	Beznau 1 & 2 (W)		X
Ringhals AB	Ringhals 2, 3 & 4 (W)		X
Shikoku	Ikata 1, 2 & 3 (MHI)		X
Spanish Utilities	Asco 1 & 2, Vandellos 2, Almaraz 1 & 2 (W)		X
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 provide examples which demonstrate that the effect of the extended inspection interval on the pilot plant's risk-informed inservice inspection program for piping is acceptable.

ACKNOWLEDGEMENTS

The authors acknowledge with appreciation those utility representatives and Westinghouse personnel who provided support in developing this risk-informed application and topical report. In particular, the authors would like to acknowledge David Grabski of FENOC and Gene Navratil of Exelon for providing information for the pilot plants.

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 Ed Fernandez, 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
HSS	High-Safety Significance
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	Nondestructive 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

ACRONYMS (cont.)

SER	NRC Safety Evaluation Report
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.

This "-A" version of WCAP-17236-NP, Revision 0, incorporates the NRC Safety Evaluation (SE). The PWROG responses to NRC requests for additional information (RAI) are also included in Appendix A. In addition, conditions and limitations, as specified in Section 4.0 of the SE, have been incorporated into the applicable sections of this "-A" version. Most instances of these conditions and limitations are in Section 2.0, Regulatory Evaluation, which provides the general implementation methodology. For clarity, all conditions and limitations are also listed in Section 4 of 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 (Category 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 a disconnect 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.

Note – The following condition and\or limitation is noted in Section 4.0 of the NRC Safety Evaluation and is applicable to this pilot plant example:

> *The NRC staff does not endorse the BV-1 and TMI-1 examples or the use of any quantitative results from any tables besides Tables 3-3 through 3-6 of the TR. Licensees (including BV-1 and TMI-1) may not reference the examples to justify any evaluation or calculation.*

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.

Note – The following condition and\or limitation is noted in Section 4.0 of the NRC Safety Evaluation and is applicable to Step 2:

> *The PFM analyses supporting the TR were based on a key assumption - one surface flaw per weld. Therefore, consistent with the TR guidance in Section 2.2, the NRC staff requires applicants to validate that at most one surface breaking flaw is present based on past ISI results. If multiple surface breaking flaws have been detected in past inspections, then the resulting change in failure frequency shall be multiplied by the number of surface flaws. If the total flaw size from this method exceeds the dimension assumed in the TR (i.e., a through-wall depth of greater than six percent of the wall thickness and a length equal to six times the depth), a weld-specific PFM analysis should be performed to develop a weld-specific change-in-frequency value. Validation of this flaw assumption must also be performed in the future through continued monitoring every 20 years.*

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 (L.D.) 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.

Note – The following conditions and\or limitations are noted in Section 4.0 of the NRC Safety Evaluation and are applicable to Step 3:

> *The NRC staff accepts the PWROG's change-in-failure-frequency analysis results in Tables 3-3 through 3-6 when used as described in the NRC staff endorsed version of this TR to evaluate the risk increase from extending the ISI interval for RV nozzle welds from 10 to 20 years. Licensees must select the 40 or 60 year change-in-failure frequency results, clarify the relationship between the selected life time and the values used in the RI-ISI, and justify the selected life time values.*
>
> *Licensees must submit plant-specific change-in-risk results in the relief requests as described in the TR. A change in risk between the ASME requirements and the extended ISI interval must always be provided. If the licensee has a RI-ISI program, the change in RI-ISI risk results including the extended intervals should be provided. If any change in risk exceeds the applicable risk guidelines in the TR, the licensee should identify and justify the deviation.*

Licensees should address PRA quality in their relief request. Licensees relying on a NRC staff approved RI-ISI program to demonstrate PRA quality should provide this statement in their submittal and provide any updated information appropriate for the application since the RI-ISI application. Licensees without a NRC staff approved RI-ISI program must describe the technical adequacy of their PRA in the relief request.

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).

Note – The following conditions and\or limitations are noted in Section 4.0 of the NRC Safety Evaluation and are applicable to Step 4:

> *The NRC staff accepts the PWROG's change-in-failure-frequency analysis results in Tables 3-3 through 3-6 when used as described in the NRC staff endorsed version of this TR to evaluate the risk increase from extending the ISI interval for RV nozzle welds from 10 to 20 years. Licensees must select the 40 or 60 year change-in-failure frequency results, clarify the relationship between the selected life time and the values used in the RI-ISI, and justify the selected life time values.*
>
> *Licensees must submit plant-specific change-in-risk results in the relief requests as described in the TR. A change in risk between the ASME requirements and the extended ISI interval must always be provided. If the licensee has a RI-ISI program, the change in RI-ISI risk results including the extended intervals should be provided. If any change in risk exceeds the applicable risk guidelines in the TR, the licensee should identify and justify the deviation.*
>
> *Licensees should address PRA quality in their relief request. Licensees relying on a NRC staff approved RI-ISI program to demonstrate PRA quality should provide this statement in their submittal and provide any updated information appropriate for the application since the RI-ISI application. Licensees without a NRC staff approved RI-ISI program must describe the technical adequacy of their PRA in the relief request.*
>
> *Licensees must identify specifically which of the change-in-risk equations and methods in the TR were used. Any deviations from the selected equations and\or methods must be identified and justified.*

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. Additionally, the basis of 40 or 60 years for the failure frequencies is consistent with the 40 or 60 year basis used in the current RI-ISI program. 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 (RCS) 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 PWROG original change-in-risk acceptance criteria discussed in the following section. If the PWROG original change-in-risk acceptance criteria cannot be met, additional inspections shall be added to the RI-ISI program until the criteria are met.

If the PWROG original change-in-risk acceptance criteria cannot be met by adding additional inspections, or it is impractical to do so, an alternative 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. If one of the alternative methods described in greater detail in Section 3.2.5.1 is used to perform the change-in-risk evaluation, the PWROG alternative change-in-risk acceptance criteria, which is the same as the criteria from the EPRI RI-ISI methodology, must be met.

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 change in risk added to the change in risk from the RI-ISI element selection should be calculated based on one weld per nozzle.

Acceptance Criteria

<u>PWROG Original Change-in-Risk Acceptance Criteria</u>

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. 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 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.

PWROG Alternative Change-in-Risk Acceptance Criteria

The PWROG alternative change-in-risk acceptance criteria can only be used for the alternative change-in-risk methods.

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.

Note – The following condition and\or limitation is noted in Section 4.0 of the NRC Safety Evaluation and is applicable to Methods 2 and 3 below:

> *The use of the changes to the IE or the POD to reflect changes in risk caused by extending the inspection interval may not use the change in failure frequencies in Tables 3-3 through 3-6. Each licensee that uses this method must identify and justify all parameter values used.*

1. Qualitative

   If the qualitative change-in-risk method from the EPRI topical report (Reference 8) were 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. The basis of 40 or 60 years for the change in failure frequencies is consistent with the 40 or 60 year basis used in the current RI-ISI program. 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 and 40 or 60 year basis to be consistent with the RI-ISI program. 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, determined on a 40- or 60-year basis consistent with the RI-ISI program, 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. The bounding change in failure frequencies are based on the 40- or 60-year basis consistent with the RI-ISI program. 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 (Category 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 four 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.

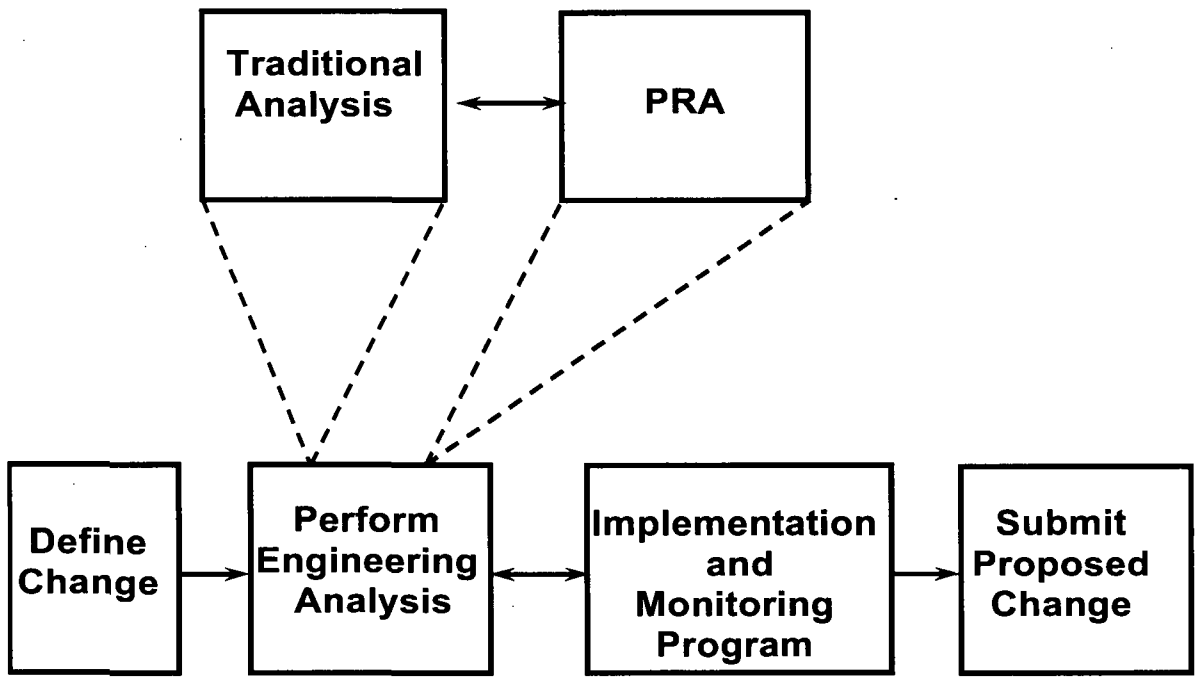

Figure 3-1 Basic Steps in (Principal Elements of) Risk-Informed, Plant-Specific Decision Making (from NRC RG 1.174)

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.

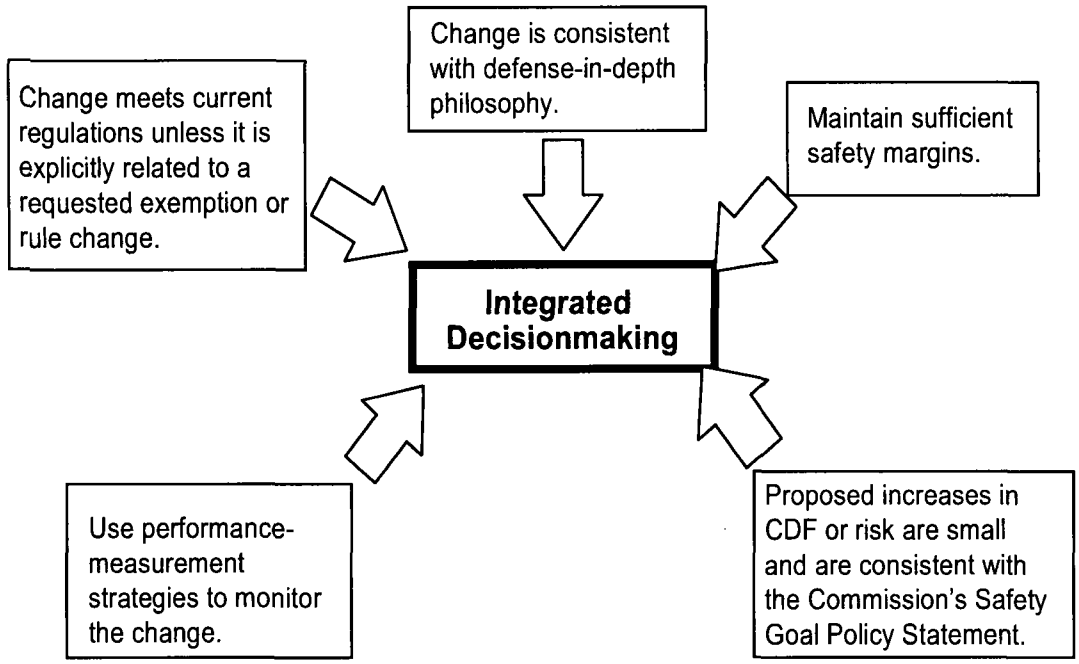

Figure 3-2 Principles of Risk-Informed Regulation (from NRC RG 1.174)

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 changes 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.

Principle 5: Use Performance-Measurement Strategies to Monitor the Change

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 < 1×10^{-6} per reactor year,
- Change in LERF < 1×10^{-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 two 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.

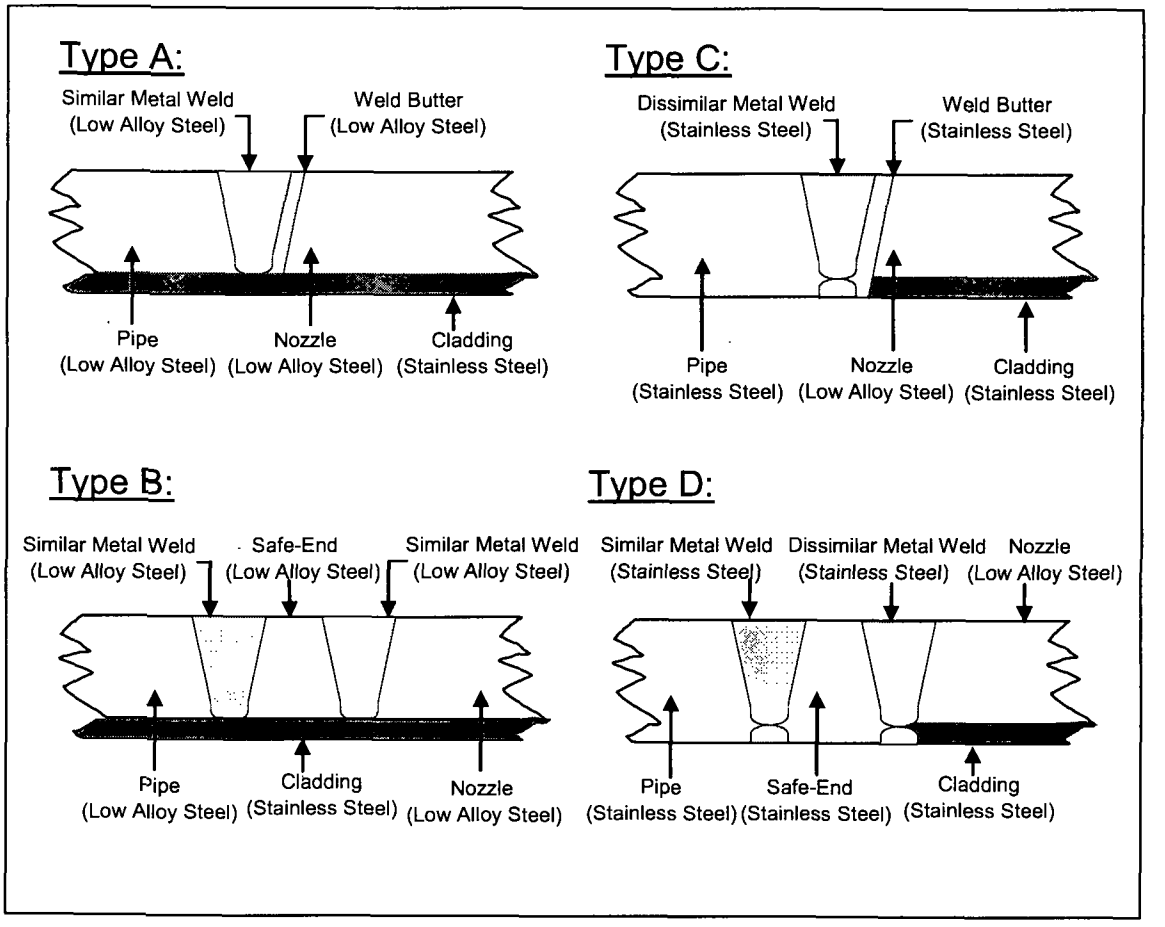

Figure 3-3 Nozzle Weld Configuration Types

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.

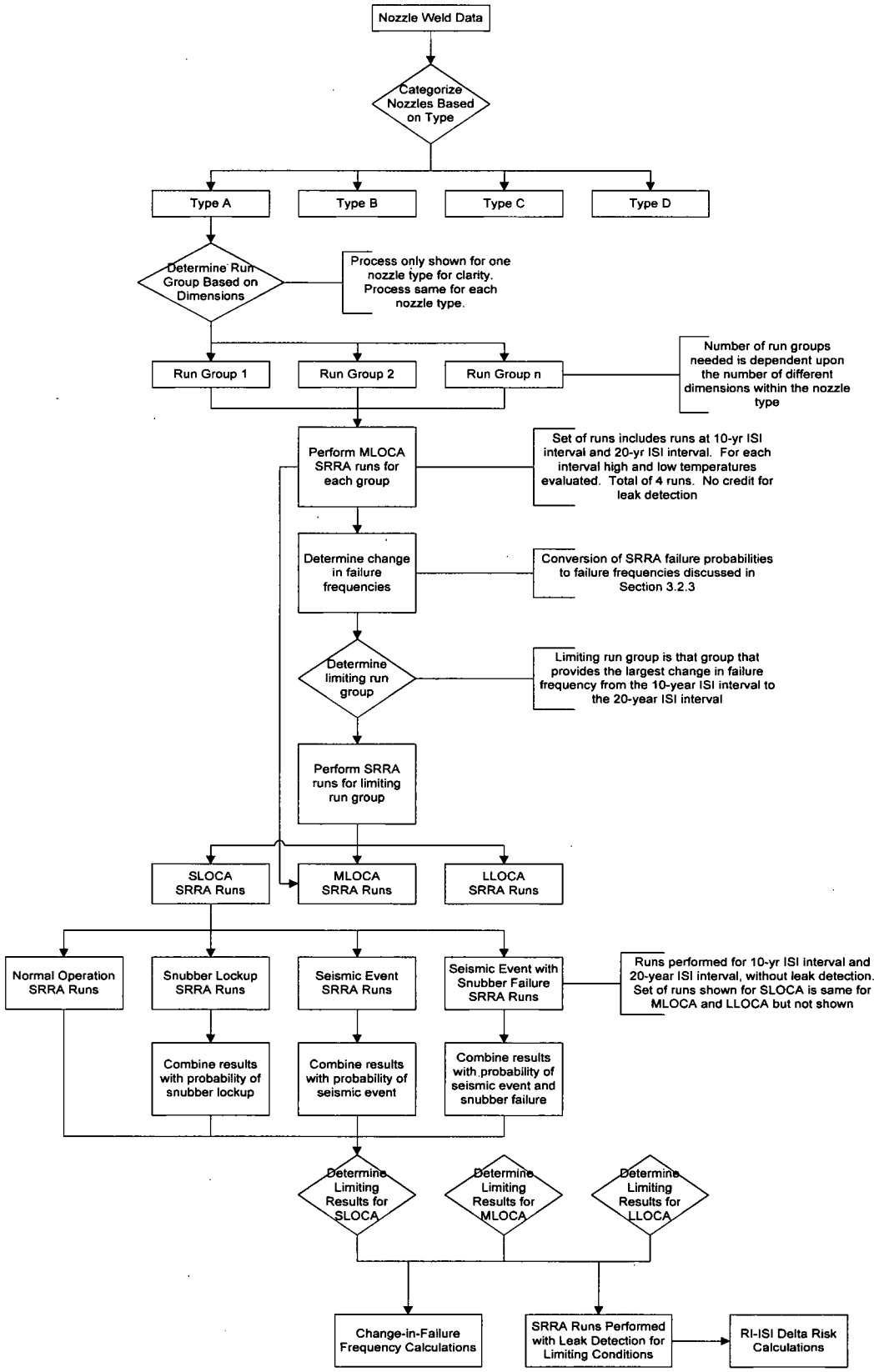

Figure 3-4 SRRA Run Process Flowchart

- 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 heatup and cooldown (Nozzle Types A and B),
    - A medium value of 0.50 for heatup and cooldown 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 for initial flaw

conditions was selected such that one flaw was simulated at the worst stress location in each 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 are 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 were not available to justify a specific crack depth corresponding to a 50% POD, so a depth of 1/10 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 3.2.2, 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 60-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.

Note – The following condition and\or limitation is noted in Section 4.0 of the NRC Safety Evaluation and is applicable to this pilot plant example:

> *The NRC staff does not endorse the BV-1 and TMI-1 examples or the use of any quantitative results from any tables besides Tables 3-3 through 3-6 of the TR. Licensees (including BV-1 and TMI-1) may not reference the examples to justify any evaluation or calculation.*

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 and 60-Year Basis)**	**CCDP**	**ΔCDF (/ year)**	**CLERP**	**ΔLERF (/ year)**
Outlet Nozzles					
SLOCA	1.18E-07	1.38E-05	1.63E-12	7.61E-12	8.97E-19
MLOCA	1.18E-07	1.68E-03	1.98E-10	4.70E-08	5.53E-15
LLOCA	1.18E-07	2.15E-03	2.53E-10	5.30E-08	6.23E-15
# of Welds Examined	3	Total ΔCDF	1.36E-09	Total ΔLERF	3.53E-14
Inlet Nozzles					
SLOCA	1.13E-07	1.93E-04	2.18E-11	2.90E-10	3.27E-17
MLOCA	1.23E-07	1.68E-03	2.07E-10	4.70E-08	5.78E-15
LLOCA	1.23E-07	2.15E-03	2.65E-10	5.30E-08	6.53E-15
# of Welds Examined	3	Total ΔCDF	1.48E-09	Total ΔLERF	3.70E-14
All Nozzles					
Total Change-in-Risk Results		Total ΔCDF	2.84E-09	Total ΔLERF	7.23E-14

Three Mile Island Unit 1

Three Mile Island Unit 1 is a B&W NSSS design and has Type A RV Nozzle welds. The 60-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.

Note – The following condition and\or limitation is noted in Section 4.0 of the NRC Safety Evaluation and is applicable to this pilot plant example:

> *The NRC staff does not endorse the BV-1 and TMI-1 examples or the use of any quantitative results from any tables besides Tables 3-3 through 3-6 of the TR. Licensees (including BV-1 and TMI-1) may not reference the examples to justify any evaluation or calculation.*

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 and 60-Year Basis)**	**CCDP**	**ΔCDF (/ year)**	**CLERP**	**ΔLERF (/ year)**
Outlet Nozzle					
SLOCA	3.93E-10	1.83E-03	7.20E-13	2.53E-04	9.95E-14
MLOCA	1.20E-11	2.23E-03	2.68E-14	2.55E-04	3.06E-15
LLOCA	5.42E-12	3.93E-02	2.13E-13	8.06E-04	4.37E-15
# of Welds Examined	2	Total ΔCDF	1.92E-12	Total ΔLERF	2.14E-13
Inlet Nozzle					
SLOCA	1.97E-10	1.83E-03	3.61E-13	2.53E-04	5.00E-14
MLOCA	6.32E-12	2.23E-03	1.41E-14	2.55E-04	1.61E-15
LLOCA	5.84E-12	3.93E-02	2.29E-13	8.06E-04	4.71E-15
# of Welds Examined	4	Total ΔCDF	2.42E-12	Total ΔLERF	2.25E-13
All Nozzles					
Total Change-in-Risk Results		Total ΔCDF	4.34E-12	Total ΔLERF	4.39E-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.

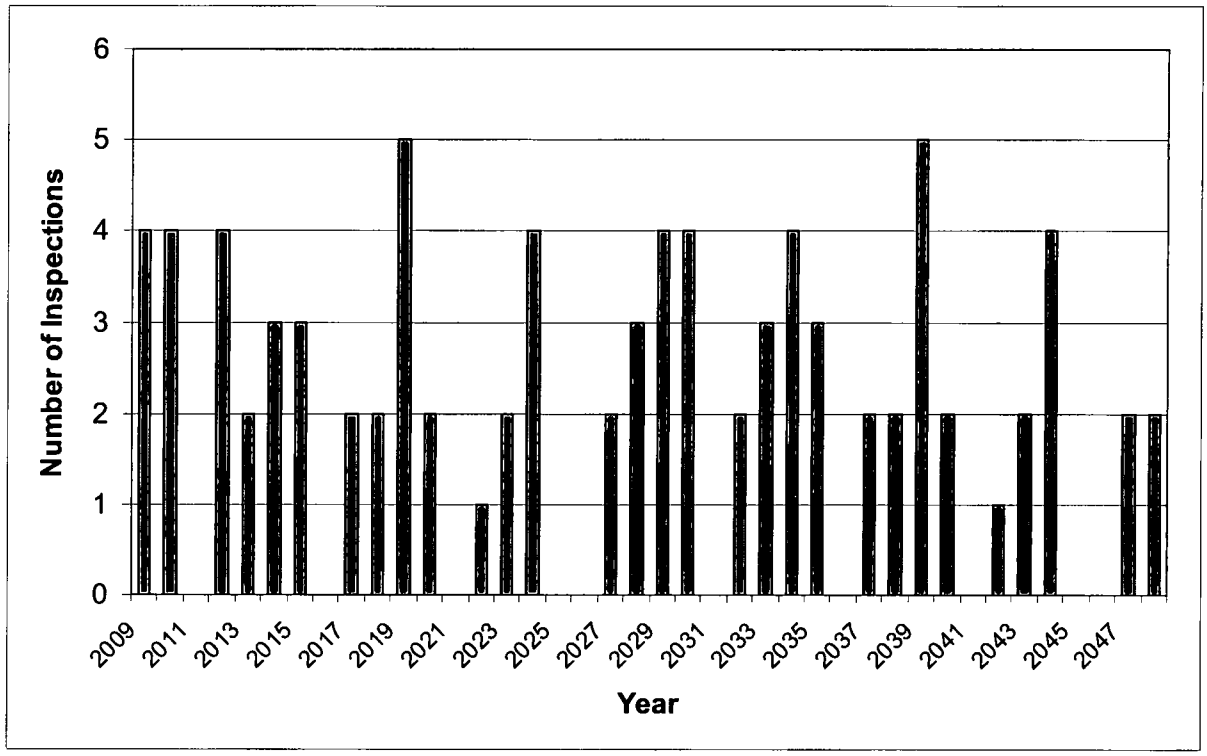

Figure 3-5 Number of Inspections per Year for Proposed Implementation Schedule

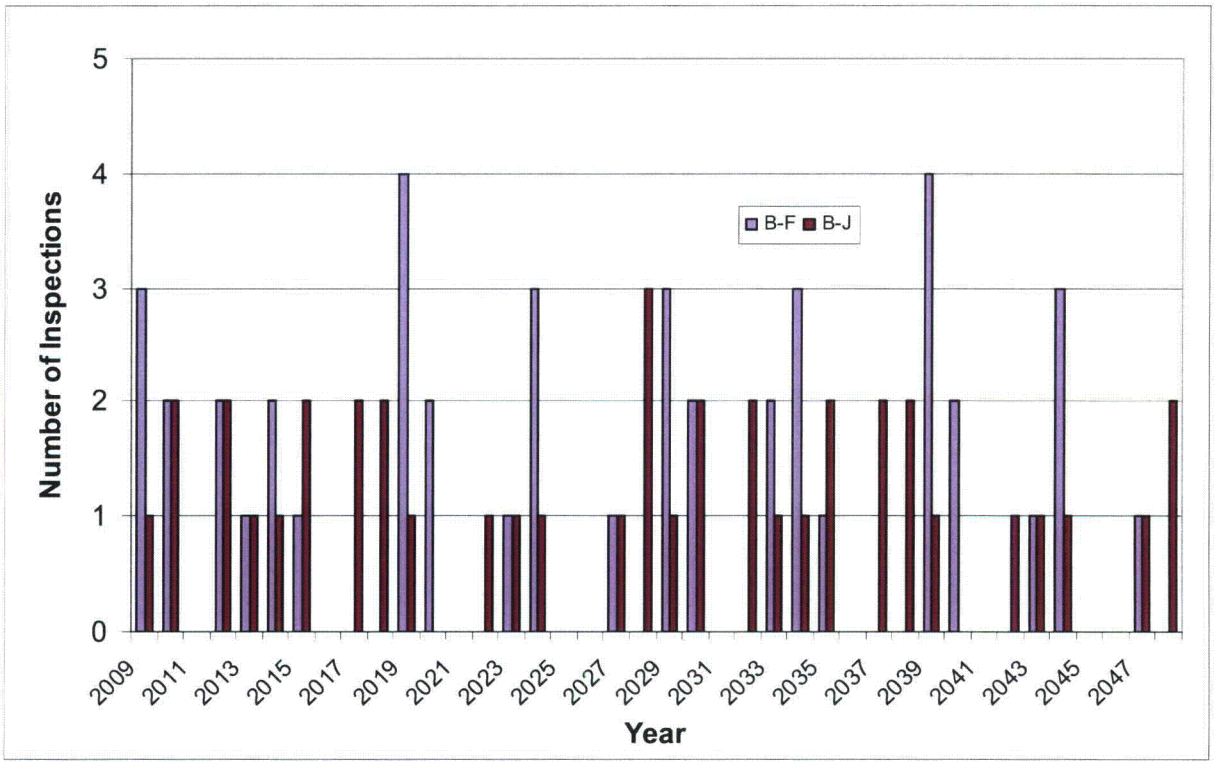

Figure 3-6 Number of Category B-F and B-J Inspections per Year for Proposed Implementation Schedule

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.

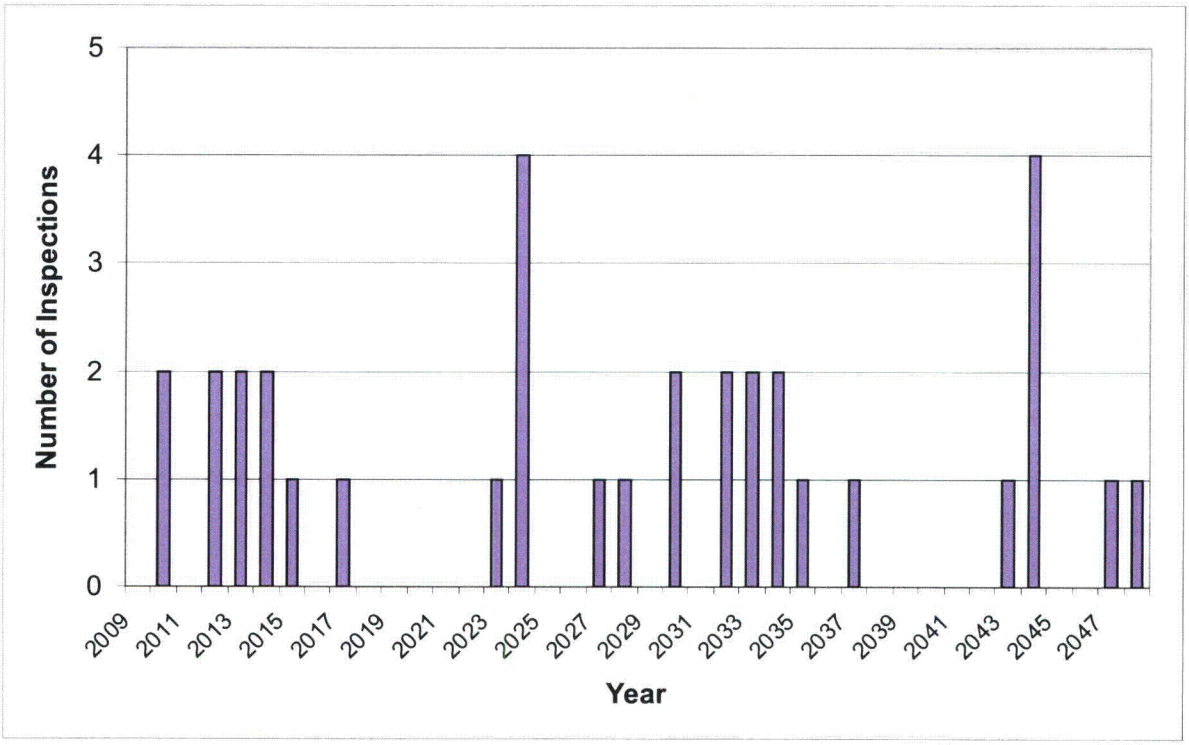

Figure 3-7 Number of Inspections per Year – Assuming Inspection of Alternative Locations

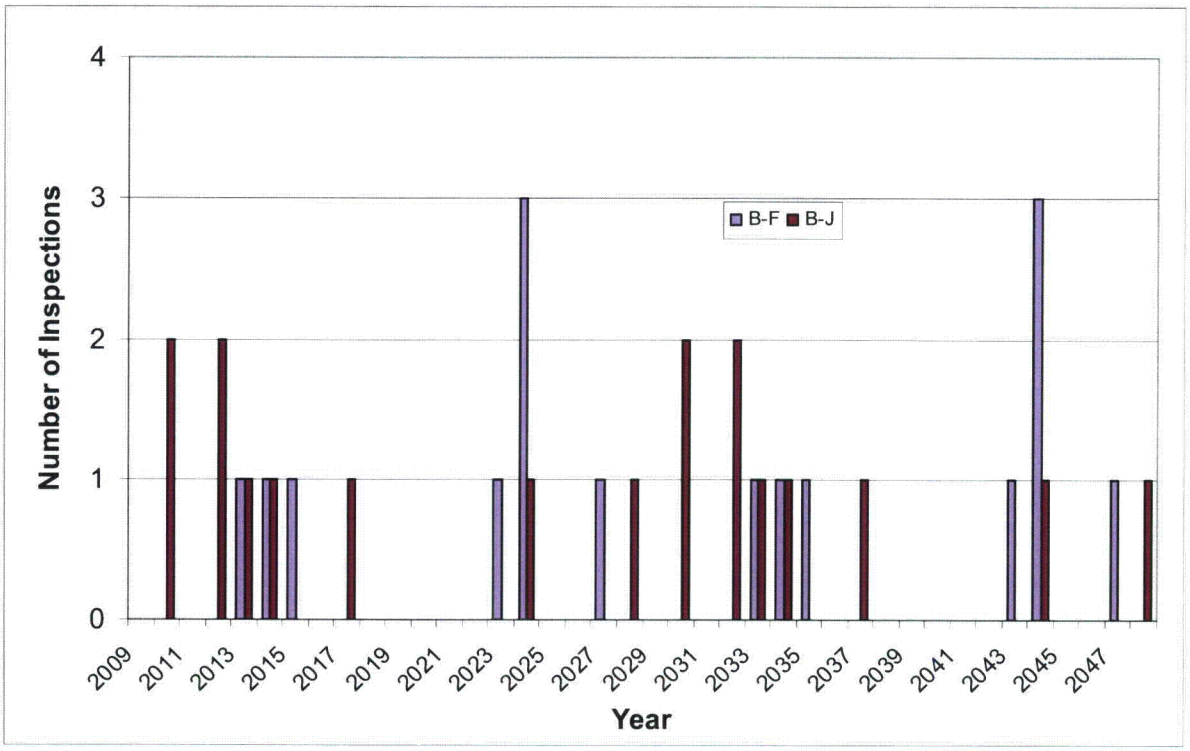

Figure 3-8 Number of Category B-F and B-J Inspections per Year – Assuming Inspection of Alternative Locations

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.

Note – The following condition and\or limitation is noted in Section 4.0 of the NRC Safety Evaluation and is applicable to the scheduling of inspections, as discussed above:

> *Licensees must identify the years in which future inspections will be performed. The dates provided must be within plus or minus one refueling cycle of the dates identified in the implementation plan referenced in the most recent Revision of TR WCAP-16168-NP-A.*

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 startup, 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 at 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 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 PWROG original change-in-risk acceptance criteria described in Section 2.4.1. 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 PWROG original change-in-risk acceptance criteria.

Evaluation of Effect of 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. The basis of 40 or 60 years for the failure frequencies is consistent with the 40-or 60-year basis used in the current RI-ISI program. 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 B and D, 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 PWROG original change-in-risk acceptance 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 when compared against the PWROG alternative change-in-risk acceptance criteria or additional exams are added to make the change in risk acceptable. These methods are discussed in more detail in the following sections.

First Alternative Evaluation Method – Examined Weld Counts Using Largest Change In Risk

In the first alternative evaluation method, the change in risk is based on the largest applicable segment change in risk. The reactor coolant system change in risk is evaluated using equation 3-1. The total plant change in risk is evaluated using equation 3-2.

$$\Delta CDF_{RCS} = \Delta CDF_{Nozzles} + \Delta CDF_{RCS\text{-}RI\text{-}ISI} \quad (3\text{-}1)$$

$$\Delta CDF_{All} = \Delta CDF_{Nozzles} + \Delta CDF_{All\text{-}RI\text{-}ISI} \quad (3\text{-}2)$$

Where:

ΔCDF_{RCS} = Change in CDF in the reactor coolant system between the ASME and RI-ISI programs including the effect of the reactor vessel nozzle ISI interval extension,

$\Delta CDF_{Nozzles}$ = Change in CDF from the reactor vessel nozzle ISI interval extension,

$\Delta CDF_{RCS\text{-}RI\text{-}ISI}$ = Change in CDF in the reactor coolant system between the ASME and RI-ISI programs excluding the effect of the reactor vessel nozzle ISI interval extension,

ΔCDF_{All} = Change in CDF in the total plant between the ASME and RI-ISI programs including the effect of the reactor vessel nozzle ISI interval extension,

$\Delta CDF_{All\text{-}RI\text{-}ISI}$ = Change in CDF in the total plant between the ASME and RI-ISI programs excluding the effect of the reactor vessel nozzle ISI interval extension,

Similar equations are conducted for the LERF. The equations are solved using the following steps:

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.

   Where:

   $\Delta CDF_{SegMaxRCS}$ = Maximum segment change in risk for the reactor coolant system segments that are in the scope of the RI-ISI program,

   $\Delta CDF_{SegMaxAll}$ = Maximum segment change in risk for all segments that are in the scope of the RI-ISI program including the reactor coolant system.

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, identify all welds examined excluding welds with visual only examinations.

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

   Where:

   SXI_{RCS} = Number of reactor coolant system welds within the scope of the RI-ISI program that are examined per the ASME Section XI program excluding visual-only exams,

   SXI_{All} = Number of all welds within the scope of the RI-ISI program that are examined per the ASME Section XI program, including the reactor coolant system, excluding visual-only exams,

   $RIISI_{RCS}$ = Number of reactor coolant system welds that are examined per the RI-ISI program, excluding visual only exams,

   $RIISI_{All}$ = Number of all welds that are examined per the RI-ISI program, including the reactor coolant system, excluding visual-only exams.

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. See equations 3-3 and 3-4.

$$\Delta CDF_{RCS\text{-}RI\text{-}ISI} = \Delta CDF_{SegMaxRCS} * (SXI_{RCS} - RIISI_{RCS}) \quad (3\text{-}3)$$

$$\Delta CDF_{All\text{-}RI\text{-}ISI} = \Delta CDF_{SegMaxAll} * (SXI_{All} - RIISI_{All}) \quad (3\text{-}4)$$

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. See equations 3-1 and 3-2.

5. Compare the results of step 4 against the PWROG alternative change-in-risk acceptance criteria,

   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.

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

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. Use equation 3-1 for estimating the reactor coolant system change in risk. The total plant change in risk is evaluated using equation 3-5.

$$\Delta CDF_{All} = \Delta \mathrm{CDF}_{\mathrm{Nozzles}} + \sum_j \Delta CDF_j \tag{3-5}$$

Where:

$\Delta \mathrm{CDF}_j$ = Change in CDF in the system j between the ASME and RI-ISI programs excluding the effect of the reactor vessel nozzle ISI interval extension.

A similar equation is used for LERF. The equation is solved using the following steps:

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.

   Where:

   $\Delta \mathrm{CDF}_{\mathrm{SegMax}j}$ = Maximum segment change in CDF for system j segments that are 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.

   Where:

   SXI_j = Number of system j welds within the scope of the RI-ISI program that are examined per the ASME Section XI program excluding visual only exams,

   RIISI_j = Number of system j welds that are examined per the RI-ISI program excluding visual only exams.

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. See equation 3-6.

$$\Delta \mathrm{CDF}_j = \Delta \mathrm{CDF}_{\mathrm{SegMax}j} * (\mathrm{SXI}_j - \mathrm{RIISI}_j) \tag{3-6}$$

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. See equation 3-1.

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. See equation 3-6.

6. Compare the results of step 5 against the PWROG alternative change-in-risk acceptance 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 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 reactor coolant system change in risk is evaluated using equation 3-7. The total plant change in risk is evaluated using equation 3-8.

$$\Delta CDF_{RCS} = \Delta CDF_{Nozzles} + \Delta CDF_{j\ for\ RCS} \tag{3-7}$$

$$\Delta CDF_{All} = \Delta CDF_{Nozzles} + \sum_{j} \Delta CDF_{j} \tag{3-8}$$

Where:

ΔCDF_j = Change in CDF for system j accounting for the number of welds examined in system j,

$\Delta CDF_{j\ for\ RCS}$ = Change in CDF for system j accounting for the number of welds examined in system j where system j is limited to the reactor coolant system.

A similar equation is used for LERF. The equations are solved using the following steps:

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.

   Where:

   ΔCDF_{jj} = Change in CDF for segment i (of system j) for the segments that are 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 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.

   Where:

   $SXI_{i,j}$ = Number of welds in segment i (in system j) within the scope of the RI-ISI program that are examined per the ASME Section XI program excluding visual-only exams,

   $RIISI_{i,j}$ = Number of welds in segment i (in system j) that are examined per the RI-ISI program excluding visual-only exams.

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. See equation 3-9.

$$\Delta CDF\#_{i,j} = \Delta CDF_{i,j} * (SXI_{i,j} - RIISI_{i,j}) \quad (3\text{-}9)$$

   Where:

   $\Delta CDF\#_{j,j}$ = Change in CDF for segment i (of system j) accounting for the number of welds examined in segment i.

4. Sum the individual segment change in risk for each segment in a system to obtain the system change in risk. See equation 3-10.

$$\Delta CDF_j = \sum_i \Delta CDF\#_{i,j} \quad (3\text{-}10)$$

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. See equation 3-7.

6. Add the reactor vessel nozzle ISI interval extension risk increase as calculated on a weld basis to the sum of the change in risk for each system in the scope of the RI-ISI program to obtain the total plant change in risk. See equation 3-8.

7. Compare the results of step 6 against the PWROG alternative change-in-risk acceptance 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 whereas RI-ISI exams are inspections for cause. In addition, per WCAP-14572 Supplement 2 (Reference 21), all postulated degradation mechanisms on a high-safety

significance (HSS) segment must be addressed in the RI-ISI program. 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 conservatism.

- 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.

PWROG Alternative Change-in-Risk Acceptance Criteria

The PWROG alternative change-in-risk acceptance criteria for the alternative change-in-risk evaluation methods are the same as the change-in-risk criteria used for the EPRI methodology. Refer to Section 2.4.1. 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 and the failure frequencies were based on 40 years to be consistent with the current RI-ISI program. 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.

Note – The following condition and\or limitation is noted in Section 4.0 of the NRC Safety Evaluation and is applicable to this pilot plant example:

> *The NRC staff does not endorse the BV-1 and TMI-1 examples or the use of any quantitative results from any tables besides Tables 3-3 through 3-6 of the TR. Licensees (including BV-1 and TMI-1) may not reference the examples to justify any evaluation or calculation.*

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 and 40-Year Basis)	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; 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 PWROG original change-in-risk acceptance criteria. 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 PWROG original change-in-risk acceptance criteria. 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 PWROG original change-in-risk acceptance criteria, 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 PWROG alternative 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 PWROG alternative 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 PWROG alternative 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 be 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.

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 of 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 then calculated for each system by summing the change in risk for all welds within that system. This calculation is shown in equation 3-11:

$$\Delta CDF_j = \sum_i [FR_{i,j} * (SXI_{i,j} - RISI_{i,j}) * CCDP_{i,j}]. \quad (3\text{-}11)$$

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 Tables 3-3, 3-4, 3-5, or 3-6. The basis of 40 or 60 years for the change in failure frequencies is consistent with the 40-or 60-year basis used in the current RI-ISI program. 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 it 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 3-12, 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:

$$\Delta CDF_j = (F_{rj} - F_{ej}) * CDF_j = (I_{rj} - I_{ej}) * F_{0j} * CCDP_j . \quad (3\text{-}12)$$

Where the subscript "rj" refers to the risk-informed inspection program at location j, and the subscript "ej" refers to the existing inspection program at location j. I is the inspection effectiveness factor. F_{0j} is the frequency of pipe rupture at location j, if no inspection is performed. $CCDP_j$ 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 and a basis of 40 or 60 years to be consistent with the RI-ISI program, 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 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 3-13 and 3-14:

$$\Delta CDF_j = \sum_{i=1}^{N} n_i \lambda_i P_i \langle R|F \rangle (I_{i,new} - I_{i,old}) CCDP_i \tag{3-13}$$

and

$$\Delta LERF_j = \sum_{i=1}^{N} n_i \lambda_i P_i \langle R|F \rangle (I_{i,new} - I_{i,old}) CLERP_i . \tag{3-14}$$

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 3-15 and 3-16:

$$I_{i,new} = \frac{h_{40}\{RI-ISI\}}{h_{40}\{noinsp\}} \text{ and} \tag{3-15}$$

$$I_{i,old} = \frac{h_{40}\{SecXI\}}{h_{40}\{noinsp\}}. \tag{3-16}$$

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, determined on a 40- or 60-year basis consistent with the RI-ISI program, 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 and a 40- or 60-year basis consistent with the RI-ISI program 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.

Note – The following condition and\or limitation is noted in Section 4.0 of the NRC Safety Evaluation and is applicable to this pilot plant example:

The NRC staff does not endorse the BV-1 and TMI-1 examples or the use of any quantitative results from any tables besides Tables 3-3 through 3-6 of the TR. Licensees (including BV-1 and TMI-1) may not reference the examples to justify any evaluation or calculation.

The results of the evaluations for the two methods are discussed below:

Method A

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 and a 40-year basis, consistent with the RI-ISI program. 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	
LLOCA CCDP	3.93E-02	
LLOCA CLERP	8.06E-04	
$\Delta CDF = \lambda_f P_i\langle R \mid F\rangle(\Delta I)CCDP$ (per nozzle)	3.01E-09	
$\Delta LERF = \lambda_f P_i\langle R \mid F\rangle(\Delta I)CLERP$ (per nozzle)	6.18E-11	
Number of RV Nozzle Welds Examined	6	
Total Nozzle ΔCDF (/year)	1.81E-08	
Total Nozzle ΔLERF (/year)	3.71E-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)	2.48E-08	
New RC System ΔLERF (/year)	1.49E-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 and a 40-year basis, 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 and 40-Year Basis)		CCDP	ΔCDF (/ year)	CLERP	ΔLERF (/ year)
Outlet Nozzle						
LLOCA	2.17E-12		3.93E-02	8.53E-14	8.06E-04	1.75E-15
# of Welds Examined		2	Total	1.71E-13	Total	3.50E-15
Inlet Nozzle						
LLOCA	1.39E-12		3.93E-02	5.48E-14	8.06E-04	1.12E-15
# of Welds Examined		4	Total	2.19E-13	Total	4.49E-15
All Nozzles						
Nozzle Change-in-Risk Results				3.90E-13		7.99E-15
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

Note – The following conditions and limitations are noted in Section 4.0 of the NRC Safety Evaluation for WCAP-17236-NP, Revision 0. These conditions and limitations should be addressed by all applicants in their relief requests to increase the ISI interval for RV nozzle welds from 10 years to 20 years:

- *The PFM analyses supporting the TR were based on a key assumption - one surface flaw per weld. Therefore, consistent with the TR guidance in Section 2.2, the NRC staff requires applicants to validate that at most one surface breaking flaw is present based on past ISI results. If multiple surface breaking flaws have been detected in past inspections, then the resulting change in failure frequency shall be multiplied by the number of surface flaws. If the total flaw size from this method exceeds the dimension assumed in the TR (i.e., a through-wall depth of greater than six percent of the wall thickness and a length equal to six times the depth), a weld-specific PFM analysis should be performed to develop a weld-specific change-in-frequency value. Validation of this flaw assumption must also be performed in the future through continued monitoring every 20 years.*

- *Licensees must identify the years in which future inspections will be performed. The dates provided must be within plus or minus one refueling cycle of the dates identified in the implementation plan referenced in the most recent Revision of TR WCAP-16168-NP-A.*

- *The NRC staff accepts the PWROG's change-in-failure-frequency analysis results in Tables 3-3 through 3-6 when used as described in the NRC staff endorsed version of this TR to evaluate the risk increase from extending the ISI interval for RV nozzle welds from 10 to 20 years. Licensees must select the 40 or 60 year change-in-failure frequency results, clarify the relationship between the selected life time and the values used in the RI-ISI, and justify the selected life time values.*

- *Licensees must submit plant-specific change-in-risk results in the relief requests as described in the TR. A change in risk between the ASME requirements and the extended ISI interval must always be provided. If the licensee has a RI-ISI program, the change in RI-ISI risk results*

including the extended intervals should be provided. If any change in risk exceeds the applicable risk guidelines in the TR, the licensee should identify and justify the deviation.

- *Licensees must identify specifically which of the change-in-risk equations and methods in the TR were used. Any deviations from the selected equations and\or methods must be identified and justified.*

- *The use of the changes to the IE or the POD to reflect changes in risk caused by extending the inspection interval may not use the change in failure frequencies in Tables 3-3 through 3-6. Each licensee that uses this method must identify and justify all parameter values used.*

- *Licensees should address PRA quality in their relief request. Licensees relying on a NRC staff approved RI-ISI program to demonstrate PRA quality should provide this statement in their submittal and provide any updated information appropriate for the application since the RI-ISI application. Licensees without a NRC staff approved RI-ISI program must describe the technical adequacy of their PRA in the relief request.*

- *The NRC staff does not endorse the BV-1 and TMI-1 examples or the use of any quantitative results from any tables besides Tables 3-3 through 3-6 of the TR. Licensees (including BV-1 and TMI-1) may not reference the examples to justify any evaluation or calculation.*

5 REFERENCES

1. ASME Boiler and Pressure Vessel Code, Section XI, 1989 Edition with the 1989 Addenda up to and including the 2010 Edition, American Society of Mechanical Engineers, New York.

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16. ASME Code Case N-648-1, "Alternative Requirements for Inner Radius Examinations of Class 1 Reactor Vessel Nozzles, Section XI, Division 1," September 7, 2001.

17. 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, Section XI, Division 1," January 10, 2007.

18. Draft NUREG-1661, "Technical Elements of Risk-Informed Inspection Programs for Piping – Draft Report," 1999.

19. NUREG/CR-6986, "Evaluations of Structural Failure Probabilities and Candidate Inservice Inspection Programs," March 2009.

20. OG-09-454, "Revised Plan for Plant Specific Implementation of Extended Inservice Inspection Interval per WCAP-16168-NP, Revision 1, 'Risk-Informed Extension of the Reactor Vessel In-Service Inspection Interval', PA-MSC-0120," December 1, 2009.

21. WCAP-14572, Revision 1-NP-A, Supplement 2, "Westinghouse Owners Group Application of Risk-Informed Methods to Piping Inservice Inspection Topical Report Clarifications," May 2004.

APPENDIX A
RESPONSES TO THE NRC REQUEST FOR ADDITIONAL INFORMATION (RAI) REGARDING THE REVIEW OF WCAP-17236-NP, REVISION 0

PWROG
PWR Owners Group

Program Management Office
102 Addison Road
Windsor, Connecticut 06095

August 26, 2011

OG-11-257

WCAP-17236-NP, Rev. 0
Project Number 694

U.S. Nuclear Regulatory Commission
Document Control Desk
Washington DC 20555-0001

Subject: Pressurized Water Reactor Owners Group
Responses to the NRC Supplemental Request for Additional Information (RAI) on PWR Owners Group (PWROG) WCAP-17236-NP, Revision 0 "Risk Informed Extension of the Reactor Vessel Nozzle Inservice Inspection Interval" (TAC NO. ME4878) PA-MSC-0440

References:

1. PWROG Letter from Melvin Arey to Document Control Desk, Request for Review and Approval of WCAP-17236-NP, Revision 0, entitled "Risk Informed Extension of the Reactor Vessel Nozzle Inservice Inspection Interval," dated September 2010, OG-10-342, October 4, 2010.
2. Acceptance for Review of PWR Owners Group (PWROG) Topical Report WCAP-17236-NP, Revision 0, entitled "Risk Informed Extension of the Reactor Vessel Nozzle Inservice Inspection Interval (TAC NO. ME4878) PA-MSC-0440, OG-11-75, March 1, 2011.
3. Request of Additional Information Pressurized Owners Group Topical Report WCAP-17236-NP, Revision 0 "Risk Informed Extension of the Reactor Vessel Nozzle Inservice Inspection Interval" (TAC NO. ME4878) PA-MSC-0440, OG-11-135, April 26, 2011.
4. Responses to the NRC Request for Additional Information (RAI) on PWR Owners Group (PWROG) WCAP-17236-NP, Revision 0 "Risk Informed Extension of the Reactor Vessel Nozzle Inservice Inspection Interval" (TAC NO. ME4878) PA-MSC-0440, OG-11-193, June 20, 2011.
5. Supplement Request for Additional Information RE: Topical Report WCAP-17236-NP, Revision 0 "Risk Informed Extension of the Reactor Vessel Nozzle Inservice Inspection Interval" (TAC NO. ME4878) PA-MSC-0440, OG-11-237, August 3, 2011.

In October 2010, the Pressurized Water Reactor Owners Group (PWROG), submitted WCAP-17236-NP, Revision 0, entitled "Risk Informed Extension of the Reactor Vessel Nozzle Inservice Inspection Interval," for review and approval (Reference 1). In March 2011, the NRC accepted the topical report (Reference 2) and provided a Request for Additional Information (RAI) (Reference 3) on April 11, 2011. The PWROG provided responses to the RAIs on June 20, 2011 (Reverence 4). A follow-up question was asked via email in regard to the response for DCI-RAI-7. In July 2011, the NRC provided a second set of RAIs (Reference 5).

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August 26, 2011
Page 2 of 2
OG-11-257

Enclosure 1 (Appendix A) to this letter provides the RAI responses to the questions received in Reference 5 and the follow-up question on DCI-RAI-7 in addition to those provided in Reference 4. Also, contained in Enclosure 1 (Appendix B) are the proposed changes that are to be incorporated in the revised WCAP. These changes include those originally provided in Reference 4 along with new changes made to address the questions in Reference 5.

If you have any questions, please do not hesitate to contact me at (704) 382-8619, or if you require further information, please contact Mr. Jim Molkenthin of the PWR Owners Group Project Management Office at (860) 731-6727.

Sincerely yours,

Melvin L. Arey, Jr., Chairman
PWR Owners Group

MLA:JPM:las

Enclosures: (1) -- LTR-AMLRS-11-42, Revision 1

cc: PWROG Steering Committee
PWROG Licensing Subcommittee
PWROG Program Management Office
J. Rowley, USNRC
M. Mitchell, USNRC
J. Andrachek, Westinghouse
A. Lloyd, Westinghouse
PWROG Management Committee
PWROG Materials Subcommittee
C. Brinkman, Westinghouse
N. Palm, Westinghouse
B. Bishop, Westinghouse
P. Stevenson, Westinghouse
S. Parker, Westinghouse

Westinghouse Non-Proprietary Class 3

Westinghouse

To: J. P. Molkenthin
cc:

Date: August 18, 2011

From: S. M. Parker
Ext: (412) 374-2652
Fax: (724) 940-8565

Our ref: LTR-AMLRS-11-42
Revision 1

Subject: Response to NRC Request for Additional Information and Mark-Up Pages for WCAP-17236-NP, Revision 0, "Risk-Informed Extension of the Reactor Vessel Nozzle Inservice Inspection Interval"

Attachment A to this letter contains the Westinghouse responses to requests for additional information (RAI) issued by the NRC Staff for WCAP-17236-NP, Revision 0. After delivering Revision 0 of these responses to the NRC Staff, a follow-up question was received regarding DCI-RAI-7. Additionally, two new requests, DRA-RAI-9 and DRA-RAI-10, were received. The Westinghouse responses to the DCI-RAI-7 follow-up and DRA-RAI-9 and DRA-RAI-10 are included in this revision. These responses have required additional changes to the WCAP. Attachment B contains the marked-up pages to this topical report that reflect changes proposed to address the NRC RAI.

Please contact the undersigned should you have any questions or concerns. Page 2 of this letter contains a matrix of the authors and reviewers for each RAI response.

Authors/Reviewers:

* S. M. Parker
  Aging Management and License
  Renewal Services

* N. A. Palm
  Aging Management and License
  Renewal Services

* B. A. Bishop
  Aging Management and License
  Renewal Services

* P. R. Stevenson
  Risk Applications and Methods I

Acting Manager:

* M. G. Semmler
  Aging Management and License
  Renewal Services

**Electronically approved records are authenticated in the electronic document management system.*

2

Westinghouse Non-Proprietary Class 3

RAI Response Author/Reviewer Matrix				
RAI Responses	B. A. Bishop	N. A. Palm	S. M. Parker	P. R. Stevenson
DCI-RAI-1	R	A		
DCI-RAI-2	A	R		
DCI-RAI-3	R		A	
DCI-RAI-4	R		A	
DCI-RAI-5.1	A	R		
DCI-RAI-5.2		R	A	
DCI-RAI-6	R	A		
DCI-RAI-7	R	A		
DCI-RAI-7 Follow-up	A	R		
DCI-RAI-8	A	R		
DCI-RAI-9		R	A	A
DCI-RAI-10		R		A
DCI-RAI-11		R		A
DCI-RAI-12	R	A		
DCI-RAI-13	R	R	A	
DCI-RAI-14	R	A		
DRA-RAI-1	R	R	A	R
DRA-RAI-2	R		A	
DRA-RAI-3		R		A
DRA-RAI-4	R	A		
DRA-RAI-5		R		A
DRA-RAI-6		R		A
DRA-RAI-7		R		A
DRA-RAI-8	R	A		
DRA-RAI-9		R		A
DRA-RAI-10		A	R	
A = Author, R = Reviewer				

Attachment A: RAI Responses

PWROG Responses to NRC RAIs

Related to

TOPICAL REPORT (TR) WCAP-17236-NP

"RISK-INFORMED EXTENSION OF THE REACTOR VESSEL NOZZLE INSERVICE INSPECTION INTERVAL"

DCI-RAI-1 ***It is stated in Section 2.2 of the Topical Report (TR) WCAP-17236, "[t]he limiting flaw depth specified above [a through-wall depth of greater than six percent of the wall thickness and a length equal to six times the depth] is based upon the upper 2-sigma bound on the log-normally distributed median value of the initial flaw depth used for the probabilistic fracture mechanics (PFM) analyses." Discuss the characteristics of the five recordable indications (Table 3-1) from the past reactor vessel (RV) nozzle inservice inspection (ISI) findings to justify the initial flaw depth distribution used in the PFM analyses in this application.***

Response

The intent of the data presented in Table 3-1 was to demonstrate that the number of flaws found in reactor vessel nozzle welds is small. This information was not explicitly used in the selection of the initial flaw depth. As stated on page 3-13 of topical report (TR) WCAP-17236-NP, the initial flaw depth and its uncertainty used in the SRRA Code are consistent with Figure 4.1 of Draft NUREG-1661. The values in Draft NUREG-1661 are also the same as those shown in Figure 2.13 of NUREG/CR-6986. It should be noted, however, that the initial flaw postulated by the SRRA Code is a surface flaw that is consistent with these NUREGs. All five indications identified in Table 3-1 of the TR were characterized as sub-surface flaws in accordance with Section XI of the ASME Boiler and Pressure Vessel Code. Therefore, all five of the recordable indications in Table 3-1 would satisfy the requirements on the initial flaw conditions for review of previous inservice inspection results that are specified in Section 2.2 of the TR.

DCI-RAI-2 ***Section 2.4.1 of TR WCAP-17236 mentions that "the calculations are performed using the change in failure frequencies with credit for leak detection." Provide a summary of the PFM analysis methodology used in TR WCAP-17236, including:***

1. ***The analysis methodology (elastic plastic fracture mechanics or linear elastic fracture mechanics), failure criteria, and the growth law for a flaw with an initial flaw depth that is randomly selected to grow from the initial size to a critical size or to a through-wall flaw, as applicable.***

2. ***The analysis methodology, failure criteria, and the growth law for a through-wall flaw to progress into a long flaw corresponding to small, medium, or large loss-of-coolant accident (LOCA).***

3. ***Establishment of fracture toughness and other material properties critical to failure resistance for each of the two failure periods for the RV nozzle-to-pipe welds.***

4. ***The key parameter or parameters which affect through-wall flaw leakage, the leakage that is considered detectable, and how leak detection was credited.***

The purpose of this summary is to minimize the staff's effort in looking for information in TR WCAP-14572, Revision 1-NP-A report (Reference 1) and TR WCAP-14572, Revision 1-NP-A, Supplement 1 (Reference 4), to support the current review.

Response

The PFM methodology used to calculate the change in failure frequencies with credit for leak detection is summarized as follows:

1. The PFM analysis methodology in the Westinghouse Structural Reliability and Risk Assessment (SRRA) Program LEAKPROF, per Reference 4, is linear-elastic fracture mechanics for circumferentially oriented initial fabrication flaws that grow to a through-wall flaw due to stress corrosion cracking (SCC) or fatigue crack growth (FCG). For the RV nozzle-to-pipe welds in PWR Plants that are not Alloy 82/182 material, the only growth mechanism of concern is FCG. The change in stress intensity factor (K) values is calculated separately for growth in the depth direction (ΔK_a) and length direction (ΔK_b), assuming that the circumferential fabrication flaw on the inside surface is located circumferentially where the membrane and bending stress is a maximum and that this maximum stress is uniform through the weld wall thickness. The values of ΔK, which are directly proportional to the change in uniform axial stress and the square root of the crack depth, are calculated in LEAKPROF as a function of a/w and a/b, where a is the crack depth, w is the weld wall thickness and b is half the length of the crack, L. The crack growth equations for FCG used by LEAKPROF are as follows:

$$da/dN = C\,(\Delta K_a)^n \text{ and } dL/dN = 2\,db/dN = 2\,C\,(\Delta K_b)^n$$

Where da is the change in crack depth, dN is the number of fatigue cycles in one year, C is the FCG coefficient, n is the FCG exponent, and dL is the change in crack length. For type 304 or 316 stainless steel welds, exponent n is a constant value of 4, while for carbon (ferritic) steel welds the exponent is 5.95 for ΔK values < 19 Ksi-(inch)$^{0.5}$ and 1.95 for ΔK values $\geq$ 19 Ksi-(inch)$^{0.5}$. The calculation of ΔK, the median values and uncertainty of coefficient C, and the constant values of exponent n in LEAKPROF are the same as those in pc-PRAISE. The PRAISE computer code was developed by Lawrence Livermore National Laboratory (LLNL) for NRC use in PFM analyses of piping welds and was used to benchmark the SRRA PFM models for FCG of fabrication flaws that are used in LEAKPROF. The development of the different versions of the PRAISE Code is summarized in the 1992 User's Manual for the pc-PRAISE Code by LLNL (NUREG/CR-5864). The FCG rate test data that was used to develop the median values of the coefficients and their log-normal uncertainties for both the PRAISE and LEAKPROF models is the same data that was used to develop the upper-bound growth rate equations in Section XI of the ASME Code. The equations for carbon-steel welds are in Appendix A, while the equations for stainless-steel welds are in Appendix C.

There are separate criteria for the two failure modes of concern for these linear elastic FCG calculations. For a small leak, which is used to compare the calculated probability with the industry piping failure experience, the criterion is that the crack depth is equal to the weld wall thickness. Although the only leaks from vessel nozzle-to-pipe welds were observed in Alloy 82/182 material, welds with this material were excluded from the TR. The other failure mode is a full break due to ductile rupture before a small leak is detected. Here the failure criterion is exceeding the weld material flow stress in the un-cracked portion of the piping weld cross section when the primary design limiting stress is applied. Note that the increase in primary stress due to cracking of the piping weld cross section is also estimated in the evaluation of this type of failure mode. The value of weld material flow stress that is used for this evaluation is discussed in more detail in the Response to Part 3 of this RAI.

2. The PFM analysis methodology in the Westinghouse SRRA Program LEAKPROF, per Reference 4, is linear-elastic fracture mechanics for the growth of through-wall flaws to a critical length due to FCG. The change in stress intensity factor (K) values are calculated for growth in the length direction (ΔK_b) only, with the same assumptions that the circumferential fabrication flaw on the inside surface is located circumferentially, where the membrane and bending stress is a maximum, and that this maximum stress is uniform through the weld wall thickness. It is also conservatively assumed that the initial through-wall crack length is the maximum length at the inside surface (weld ID) at the time the semi-elliptical crack just went through the wall at the outside surface (weld OD). The value of ΔK_b is directly proportional to the change in uniform axial stress and the square root of the crack length. The crack growth equations for FCG used by LEAKPROF are:

$$dL/dN = 2\ db/dN = 2\ C\ (\Delta K_b)^n$$

Again, the calculation of ΔK_b, the median values and uncertainty of coefficient C, and the constant values of exponent n in LEAKPROF are the same as those in pc-PRAISE (NUREG/CR-5864, 1992). The criterion for the failure mode of concern for these linear elastic FCG calculations is exceeding the critical through-wall flaw length for a small, medium or large LOCA before a small leak is detected. However, the calculation of the critical crack length for the leak rates corresponding to the different LOCA sizes and the crack length for the small detectable leak rate uses elastic-plastic fracture mechanics calculations involving the plastic stress intensity factor J. These elastic-plastic calculations are described in detail in the Response to Part 4 of this RAI. The Response to Part 4 also describes how leak detection is credited in the calculation of the probabilities of this failure mode.

3. The other material property critical to failure resistance for each of the two failure periods (40 years and 60 years) for the RV nozzle-to-pipe welds is the flow stress, which is a function of the weld material and its temperature, but does not change as a function of time. As discussed in Part 1 of this RAI Response, the failure criterion for full break due

to ductile rupture is exceeding the material flow stress in the uncracked portion of the pipe weld cross section. The values of material flow stress at various operating temperatures from 50°F to 650°F for stainless and carbon steel welds are provided in Table 3-3 of the SRRA Supplement (Reference 4). For the stainless steel welds, a statistical evaluation of measured flow stresses at room and operating (550°F) temperatures for various types of welds contained in a 1986 EPRI Report (NP-4768) was performed by Westinghouse to determine the mean values and their uncertainty. For carbon steel welds, utility responses to NRC Bulletin 87-01 which are contained in a 1988 EPRI Report (NP-6066) and a 1992 ASME Piping and Pressure Vessel Conference Paper by Phillips on the PRA risk significance of passive component failures indicated that the flow stress in a carbon steel weld should be approximately 6.6 Ksi higher than that in a stainless steel weld at the same temperature.

4. To calculate the failure probabilities for the failure modes of specified leak rates, such as a small-break loss-of-coolant accident (LOCA), a medium or large-break LOCA or the effects of leak detection on a full break due to ductile rupture, the pre-processor program CLVSQ is used. CLVSQ stands for crack length versus the leak flow Q, and typical calculated results are shown in Figure DCI-RAI-2-1 (Figure 2-4 in the SRRA Supplement, Reference 4). The crack flow rate, dQ/dt, in gallons per minute (GPM) is calculated in CLVSQ using the equation:

$$dQ/dt = 0.06\ P\ A\ /\ w^{0.5}$$

Where P is pressure in psi, A is the crack opening area in square inches, and w is the weld wall thickness in inches. This equation is based upon calculated results from the pc-PRAISE Code (NUREG/CR-5864, LLNL, 1992) with an improved leak leak-rate model similar to that in the PICEP Computer Code contained in the 1984 EPRI Report, NP-3596-SR. The constant value of 0.06 in the equation was derived for the improved pc-PRAISE results for five different sets of pipe sizes and operating conditions. However, the base case was for large-diameter piping welds at PWR reactor coolant system (RCS) operating conditions that would be directly applicable to the RV nozzle-to-pipe welds. The CLVSQ program also requires weld material properties for calculating the crack opening area as a function of crack length for the applied pressure, deadweight and thermal stresses per the elastic-plastic analysis methods contained in the 1984 EPRI Report NP-3607. Specifically, the crack opening displacement is calculated by numerically integrating the plastic stress intensity factor (J) along the length of the crack. The elastic-plastic weld material properties that are used in the CLVSQ program for type A-106B carbon and types 304 or 316 stainless steel, respectively, are the same as the default input values to the improved pc-PRAISE Code. Finally, the crack leak rates for small, medium, or large loss-of-coolant accidents (LOCA) are 100 GPM, 1,500 GPM and 5,000 GPM, respectively, per Sandia National Laboratory Report NUREG/CR-4550, Revision 1, Volume 1, 1990. A minimum detectable leak rate of 1 GPM is used based upon PWR plant Technical specifications for unidentified leakage.

How leak detection is credited is best shown in Figure DCI-RAI-2-2 (Figure 2-3 in the SRRA Supplement, Reference 4). For each random trial, the initial flaw size and crack size for a full break due to ductile rupture are set and the crack grows in depth and length due to the fatigue loading (stress range and number of cycles) for each time step, which is one year of operation. The growth in both directions to a through-wall flaw, and only in the length direction thereafter, was previously discussed in the Responses to Parts 1 and 2 of this RAI, respectively. Because a break due to ductile rupture from application of the design limiting stress could occur without a through-wall flaw, it is checked first to determine if failure occurs, and if so a new trial is initiated. If it does not fail, a check is made to determine if the flaw has grown through the wall. If not, the crack is grown for additional years of operation until 40 or 60 years is achieved and a new random trial is initiated. If it is through the wall, the length is checked to see if it will cause the specified large leak rate, such as that for a small, medium or large LOCA, and if so the pipe fails. If not, the crack length is checked to see if it exceeds that for a detectable leak of 1 GPM. If the leak is not detectable, the crack grows for another year due to fatigue loading and is then checked again for breaks and large leaks. If the leak is detectable, there is no failure and a new random trial is initiated.

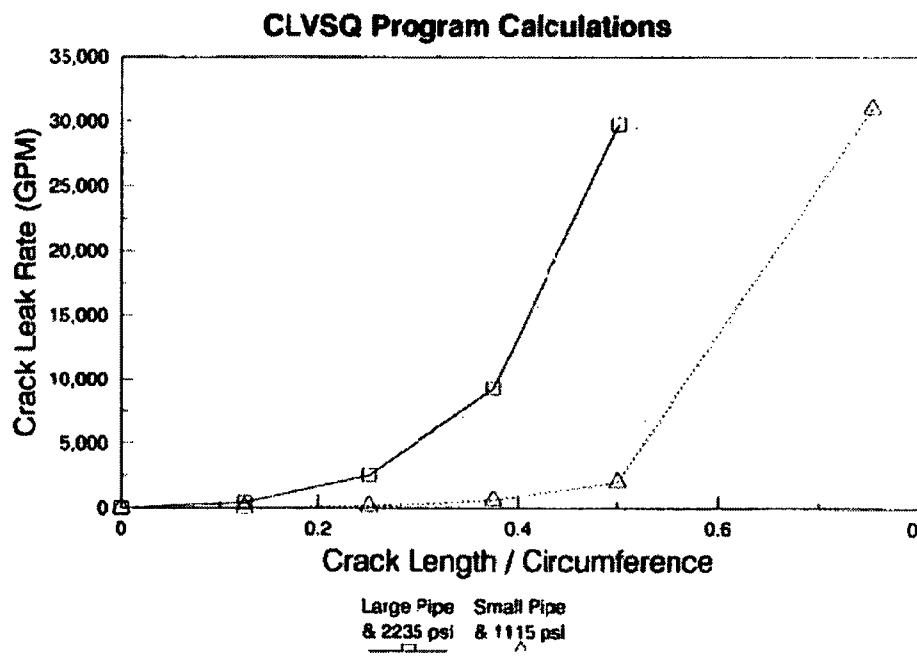


Figure DCI-RAI-2-1 – Leak Rate Program Calculations (from Reference 4)

- 6 -

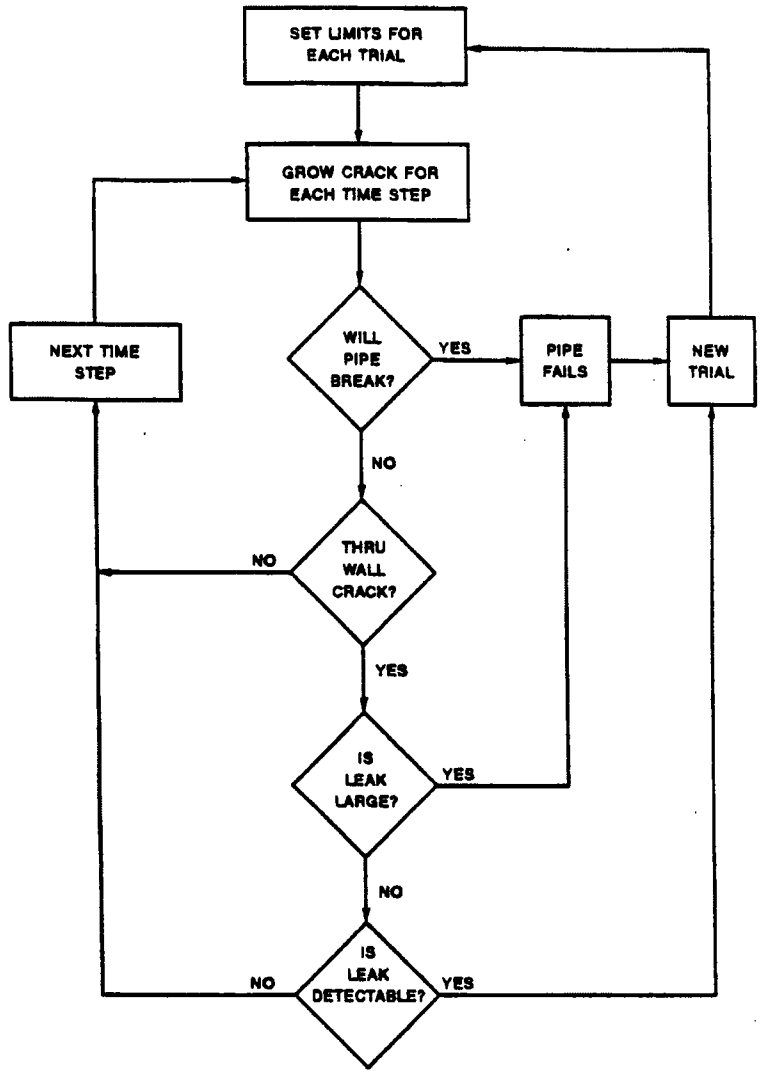


Figure DCI-RAI-2-2 – Flow Chart for Piping Weld Failure Modes (from Reference 4)

- 7 -

DCI-RAI-3 ***Section 3.2.3 of TR WCAP-17236 states, "[t]he SRRA [Structural Reliability and Risk Assessment] Code was developed for estimating piping failure probabilities to be used in relative risk-ranking of piping segments...." List and discuss any significant part of the SRRA Code which was not needed in generating results supporting prior applications, but are needed now to generate PFM results to support the current application for RV nozzle-to-pipe welds.***

Response

There were no parts of the SRRA Code used in generating PFM results for this application that were not needed in generating PFM results for the prior risk-ranking application. The same version of the SRRA Code that was used in previous applications for risk-informed ISI of piping welds was also used to generate the PFM results to support the current application for RV nozzle-to-pipe welds.

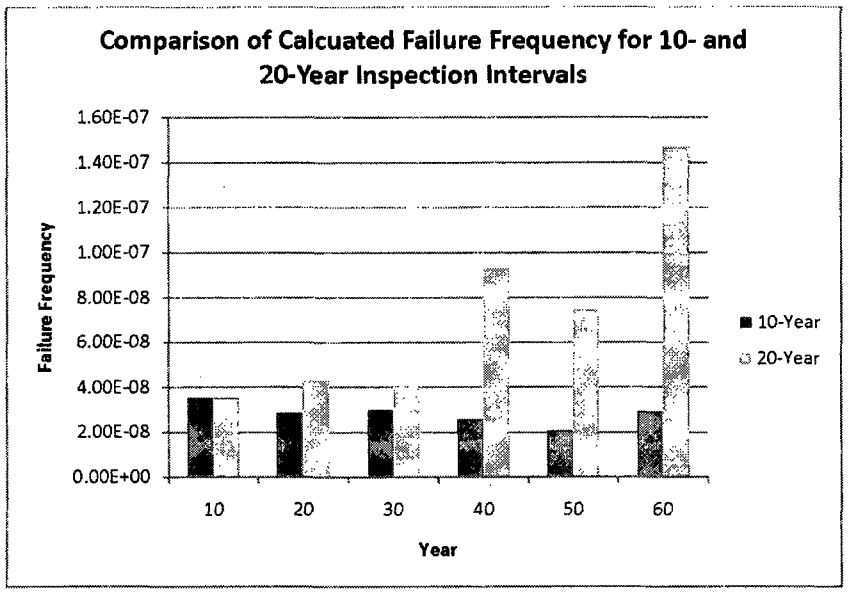
DCI-RAI-4 ***Section 3.2.3 of TR WCAP-17236 states under "Method," "[t]his difference in failure probability is then converted to a change in failure frequency by dividing the difference in failure probability by the respective number of years, 40 or 60." Using the first row of results in Table 3-7 as an example, provide the histogram of failures for the period of 60 years for the computer runs of the 10-year ISI interval and the 20-year ISI interval. The staff will use this information to determine whether the failure frequency obtained by averaging for 60 years is appropriate.***

Response

The SRRA Code output includes the probability of failure each year and the cumulative failure probability at the end of each year of operation. These calculated probabilities are typically much less than 10^{-4}, even after 40 or 60 years of operation. For calculating the change in risk, the cumulative failure probability needs to be converted to a failure frequency in events per year. This is consistent with Section 3.6.1 of WCAP-14572, Revision 1-NP-A, "Westinghouse Owners Group Application of Risk-Informed Methods to Piping Inservice Inspection Topical Report." The description of Equation 3-3 in this section states,

> *"Because the SRRA model generates a probability, the probability must be transformed into a failure rate. The cumulative failure probability at end of license is divided by the number of years at end of license."*

As an example, Table 3-7 in the TR contains the values used to calculate the change in failure frequency for the sensitivity studies. Below is a histogram of the calculated failure frequencies corresponding to the first row of results in Table 3-7. These failure frequencies were calculated at 10-year increments up to 60 years. The 10-year increment was selected because the frequency would change every 10 or 20 years due to the effects of inservice inspection.



In this example, the largest change in failure frequency occurs at the end of 40 or 60 years of operation due to extending the inspection interval from 10 years to 20 years. Therefore, use of the change in failure frequencies calculated at 40 and 60 years (the number of years at the end of license) for this topical report is both appropriate and consistent with WCAP-14572, Revision 1-NP-A.

DCI-RAI-5 ***Section 3.2.3 of TR WCAP-17236 states under "Inputs," "[o]perating 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 [Pressurized Water Reactor Owners Group] Method for piping risk-informed (RI) ISI. These inputs are based on a combination of design stress analysis results and engineering insights."***

1. ***Provide two examples for the critical input values where engineering insights were applied for their determination and demonstrate that these engineering insights have merit.***

2. ***Provide a complete set of SRRA Code inputs, using the first row of results in Table 3-7 as an example.***

Response

1. The 18 input parameters to the SRRA Computer Program and standard input values are listed in Table 3-1 in the Supplement 1 of the Piping RI-ISI WCAP-14572, Rev. 1-NP-A (SRRA Supplement, Reference 4) and example inputs are also included in the Response to

Part 2 of this RAI. Two critical parameters, where engineering insights were used to determine the input values for the RV nozzle welds, are the fatigue stress range and design limiting stress.

For the fatigue stress range, a low value of 0.3 of the flow stress (FS per Table 3-3 of the SRRA Supplement) was used as the input for plant heatup and cooldown of similar metal nozzle welds (Nozzle Types A and B). The value of 0.3 is based upon engineering insights and experience from ASME Code Stress Reports, where the fatigue stress range is twice the calculated value of the alternating stress amplitude, S_a. For normal and upset transients in similar material welds, the value of Sa is approximately 0.5 of the hot limit stress, which is typically 2/3 of the yield stress. If the yield stress is taken as 0.9 of the FS, then the median value of S_a is approximately 0.3 of the FS and the upper bound stress-report value on the fatigue stress range would be 0.6 of the FS. Engineering experience has also shown that the slow heatup and cooldown transients are the primary drivers for fatigue crack growth relative to the other design duty cycle transients. These other design duty cycle transients provide high skin stresses for fatigue crack initiation, which is the failure mode of concern in Section III of the ASME Code, but do not have sufficient energy to drive an initial inside-surface flaw through the thickness of the wall. For nozzles with dissimilar metal welds (Types C and D), the fatigue stress range is known from engineering experience to be from 50 to 75 percent higher due to the additional stresses due to the restraint of the differential thermal expansion of the two different materials. An above average increase of 67 percent was applied to the low value of 0.3 of the FS to give the medium input value of 0.5 of the flow stress for the welds in these types of nozzles. A high input value of 0.7 of the FS was used for the welds in all types of reactor vessel (RV) to pipe nozzles when the snubbers on the steam generators failed 10-percent of the time and locked up during normal heatup expansion of the reactor coolant loop piping. The 10-percent snubber failure rate is based upon the 90-percent snubber operability that is required with a high degree of confidence in the plant-specific snubber testing programs. Note that this type of snubber failure is typically associated with mechanical snubbers, which like ball-screws, lock up well dynamically. Moreover, hydraulic snubbers, like shock absorbers allow for slow movement, but also sometimes do not lock up for rapid movement due to dynamic events. To make the nozzle stresses used for the PFM analyses with the SRRA Code applicable to RV nozzle-to-pipe welds in a variety of PWR plant designs, the worst type of snubber was used in this evaluation. Finally, it should be noted that these input stress range (minimum to maximum) values are median values with a 2-sigma upper bound factor of 2 in a log-normal distribution. That is, if the snubber locks up when it is not supposed to, then significant plastic deformation would result in the upper bound case, which would also limit the actual stress range values to approximately 1.4 of the FS.

For the design-limiting stress, which includes only the primary stresses that could lead to a full break due to ductile rupture, a low value of 0.1 of the FS is used for normal operation in larger pipe welds, like those in the RPV nozzles, since the only loads of concern are deadweight and pressure. For the maximum seismic loading due to a design basis safe-

shutdown earthquake (SSE), a median stress value of 0.26 of the FS is used for all the nozzle welds. With an upper 2-sigma bound factor of 2 higher in a log-normal distribution, the upper bound value is 0.52, or approximately 50% of the ASME Code limits. This upper bound value, which corresponds to the SSE stress from an ASME Code stress report, is judged to be low based upon engineering insights and experience from typical ASME Code stress reports for RCS piping nozzle welds. These stress report values are low because snubbers are typically provided near the top of the massive and tall steam generators in PWR plants to limit the overturning bending moments during any seismic event. The frequency of the design-limiting SSE event that is used for the RPV nozzle welds is 0.001 (10^{-3}) events per year, a conservative high value per Section A.7 of the NRC Safety Evaluation Report (SER) for the SRRA Supplement to the Piping RI-ISI WCAP-14572, Rev. 1-NP-A. For a 10 percent chance that the steam generator snubbers fail to lock during the SSE event, the frequency of the combination of the two conditions would be 0.0001 (10^{-4}) events per year and the stress level would be much higher. The RV nozzle-to-pipe weld stress level for these conditions, which is not calculated in the ASME Code stress report, is taken to be a 2-sigma upper bound value between 0.8 of the FS (estimated proportional limit) and 0.9 of the FS (estimated yield strength), where the plastic flow would limit the stress values. The median value would be approximately 50% of the upper-bound value, or 0.42 of the FS. Note that this type of snubber failure to not lock up dynamically is typically associated with hydraulic snubbers. To make the nozzle weld stresses used for the PFM analyses with the SRRA Code applicable to RV nozzle-to-pipe welds in a variety of PWR plant designs, the type of snubber providing the most conservative input was used in this evaluation.

2. SRRA Input values for Type C Outlet Nozzle Sensitivity Run – MLOCA

Input Parameter	Value
Type of Piping Steel Material	316 SS
Pipe Weld Failure Mode	Large Leak
Years Between Inspections	10.0 and 20.0
Wall Fraction for 50% Detection[1]	0.24 and 0.10
Degrees (F) at Pipe Weld	620.1
Nominal Pipe Size (NPS, inch)	36.0
Thickness / Outside Diameter	0.0977
Operating Pressure (KSI)	2.250
Uniform Residual Stress (KSI)	10.0
Flaw Factor (<0 for 1 Flaw)	-10.80
DW & Thermal Stress / Flow Stress	0.17
SCC Rate / Rate for BWR Sens. SS	0.00
Factor on .0095 in/yr CS Wastage	0.00
P-P Vib. Stress (KSI for NPS of 1)	0.0
Cyclic Stress Range / Flow Stress	0.50
Fatigue Cycles per Year	5.0
Design-Limit Stress / Flow Stress	0.10
System Disabling Leak Rate (GPM)	1500
Minimum Detectable Leak Rate (GPM)	0.0

Input Parameter	Value
Value of Weld Metal Flow Stress in KSI	50.50

Note 1: Input used to adjust the probability of detection (POD) for sensitivity studies in Section 3.2.3 of the TR

DCI-RAI-6 ***Figure 3-3 of TR WCAP-17236 shows schematics of the four RV nozzle weld configurations. In reality, the diameter varies from the nozzle to the pipe. Identify the diameter that the PWROG used as input to the SRRA Code (nozzle diameter, pipe diameter, or something else) and justify the selection.***

Response

Nozzle and piping fabrication drawings were reviewed for all four nozzle types and it was determined that the diameter varies from the nozzle to the pipe only for Types C and D. For Types A and B, the diameter and thickness are constant from the end of the nozzle to the pipe. Therefore, for Types A and B this pipe diameter and thickness were used as input to the SRRA Code. For the type C and D nozzles, it was acknowledged, as stated in the last paragraph on page 3-9 of the TR, that "...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)..." It could be further stated that these different thicknesses may coincide with different diameters. As further stated in the WCAP, "...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." Therefore, run groupings were developed for each unique thickness and diameter combination (i.e. two groupings for Nozzle Type C and three groupings for Nozzle Type D). All groupings of thickness and diameter inputs were evaluated using the SRRA Code. As shown in Figure 3-4 in the TR, the grouping that provided the highest change in failure (MLOCA) frequency between 10-year and 20-year inspection intervals was selected as being limiting for that nozzle type.

DCI-RAI-7 ***Section 3.2.3 of TR WCAP-17236 states under "Inputs," "[t]he initial flaw conditions contained in the SRRA Code, including the median flaw depth and its uncertainty and the flaw density are ...the same as those shown in Figures 2.13 and 2.15 of NUREG/CR-6986 ["Evaluations of Structural Failure Probabilities and Candidate Inservice Inspection Programs"] (Reference 19). An input value of either X-ray NDE [nondestructive examination] or One Flaw was used." Please define the "X-ray NDE" and "One Flaw" and demonstrate that either input can be used in this application to produce similar results.***

Response

The SRRA Code simulates a maximum of one flaw at the worst stress location that could result in the first failure of the nozzle weld. This one flaw can result from either selecting the "One Flaw" option in the "Flaw Factor" SRRA input field or by selecting the "X-ray NDE" and entering a thickness and diameter such that the flaw density (flaws/inch) multiplied by the weld length

exceeds one. The flaw density used by the SRRA Code for welds that have been examined using X-Ray NDE, such as the reactor vessel nozzle welds, is shown in Figure 2-1 of WCAP-14572, Revision 1-NP-A, Supplement 1. The thinnest weld evaluated for WCAP-17236-NP is approximately 2.4". Based on Figure 2-1, a 2.4" thick weld would contain 2.206E-02 flaws per inch of weld length. Based on this density, 45.33" of weld length would be required for the SRRA Code to simulate one flaw. This 45.33" of weld length would be equivalent to a weld inside diameter of 14.44". The smallest inside diameter evaluated for WCAP-17236-NP was greater than 27". Given that the density and number of flaws increases with respect to increasing thickness and diameter, all welds evaluated in WCAP-17236-NP would result in one flaw. Therefore, the use of either "X-ray NDE" or "One Flaw" in the SRRA Code input will produce identical results. However, to avoid any confusion and to clarify the intended result, the bullet in Section 3.2.3 of WCAP-17236-NP dealing with initial flaw conditions will be revised to only state that the input was selected such that one flaw was simulated at the worst stress location in each weld.

Follow-up Information Requested by the NRC

> **The response stated that "The thinnest weld evaluated for WCAP-17236-NP is approximately 2.4"...Based on Figure 2-1, a 2.4" thick weld would contain 2.206E-02 flaws per inch of weld length." The response further stated that, "The smallest inside diameter [ID] evaluated for WCAP-17236-NP was greater than 27"." Based on the above stated information, the corresponding weld length for RV nozzle weld would be 84.82 inches (πx27), giving a flaw number of 1.87 flaws (2.206E-02x84.82). Since for most RV nozzle welds, the weld thickness is more than 2.4" and the ID is more than 27", the number of flaws per weld is most likely to be 2 or greater. This contradicts to your one flaw per weld assumption. Please clarify.**

Additional clarification was provided on June 28, 2011 as follows:

If the flaw density is 2 per weld (as described in DCI-RAI-7 Follow-up) and if there are multiple worst stress locations in the weld (for instance, a peak-valley-peak-valley distribution), then the resulting changes in failure frequencies may be much greater than what you calculated in the WCAP-17236 report. Please address this concern.

Response to Follow-up Request

In reality, there is not a singular worst stress location around the nozzle-to-pipe weld for the initial fabrication flaw. For the primary degradation mechanism of fatigue crack growth, the worst location of concern would be where the alternating stress range due to heat up or cool down is a maximum value. For calculating the through-wall flaw length to give a specified leak rate (LOCA size), the worst location would be where the combined pressure, deadweight and thermal stresses during normal operation are a maximum because this would give the largest crack opening displacement and corresponding leak rate. For a very large leak or full break due to ductile rupture, the worst location would be where the primary stresses due a safe-shutdown earthquake (SSE), including the vector combination of

stresses due to seismic loading in both the horizontal and vertical directions, are a maximum value.

The likelihood of any one of these three worst stress locations being the same as that of the randomly located initial flaw is very small. Since the stress conditions are all independent of one another, the likelihood of all three worst stress locations being the same as that of the initial flaw would be expected to be extremely small. However, that is what was conservatively assumed in the SRRA-PFM models and calculations for the nozzle-to-pipe welds evaluated in WCAP-17236-NP. It is highly unlikely that any one flaw would ever have a failure probability as high as that calculated with this overly conservative coincidence assumption, and even much less so for two flaws. That is why a maximum of only one flaw was assumed in the SRRA-PFM models and calculations.

To provide high confidence that only one flaw will control the weld failure probability, implementation of the ISI interval extension would require the following requirement from Section 2.2 of WCAP-17236-NP on review of previous inservice inspection results be satisfied:

> *"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."*

However, if the above requirement for only one weld flaw indication cannot be satisfied, then Section 2.2 also requires that the effects of multiple flaw indications be considered. This requirement provides high confidence that the effects of potential multiple weld flaw indications on failure probability would also be evaluated:

> *"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."*

More details on the requirements for evaluating indications for both one flaw and multiple flaws are also provided in Section 2.2 of WCAP-17236-NP.

DCI-RAI-8 ***Section 3.2.3 of TR WCAP-17236 states under "Inputs," "[t]he probability of detection [(POD)] 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 [inservice inspection] accuracy input." It was then stated that this value was 0.24. Although the results from a sensitivity study on the inspection accuracy are provided in Tables 3-7 and 3-8 of TR WCAP-17236 , there is no information about the actual***

performance level that the licensees can deliver. Please provide data, demonstrating that the selected performance parameter value of 0.24 is based on the licensees' actual performance, not just an arbitrarily assumed value.

Response

The selected performance parameter value of 0.24 for ISI accuracy was first used in the pilot application of the Westinghouse risk-informed program for piping at Surry Unit 1. The value was proposed for ultrasonic (UT) examinations by the NDE engineer at Surry in the late 1990s. This value was then reviewed and accepted by members of the ASME Research Task Force on Risk-Based Inspection Guidelines that authored NUREG/GR-0005, Volumes 1 and 2. It was also discussed at meetings of the Task Force that were held in conjunction with the public meetings of ASME Section XI, including the Working Group on Implementation of Risk Based Examination. However, this value of 0.24 was used prior to the implementation of qualified inspection per Appendix VIII of ASME Section XI, which is a requirement of the NRC and ASME for piping RV nozzle weld examinations. Since the implementation of the Appendix VIII requirements, a value of 0.10 is believed to be more realistic based upon actual piping weld inspection experience. That is why this new value of 0.10 was used in the sensitivity studies of Tables 3-7 and 3-8 in the WCAP TR.

DCI-RAI-9 ***Section 3.2.4 of TR WCAP-17236 states under "What are the Consequences," "[t]he likelihood of core damage and large early release, given a LOCA, can be quantified by the PRA [Probabilistic risk assessment] in terms of the conditional core damage probability (CCDP) and [conditional] large early release probability (CLERP), respectively." Sample analyses results are provided for Beaver Valley Power Station, Unit 1 and Three Mile Island, Unit 1 (Tables 3-10 and 3-11). The last sentence in the Executive Summary, "[f]urther, 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" implies a generic conclusion. What is the purpose of including these results in TR WCAP-17236? Would they be used generically to support a relief request or is each request to include plant-specific results? The results for Beaver Valley Power Station, Unit 1, indicate conditional containment failure probabilities as low as 1E-6 which seems unreasonable.***

Response

The pilot plant results calculated and reported in the TR are included to provide an example of how the methodology is applied and how it affects the respective risk-informed inservice inspection program for piping. These results are not meant to be bounding or applicable to the results of any plant applying this methodology other than the pilot plants.

The pilot-plant results would not be used to generically support a relief request. Each request would need to provide plant-specific risk results using the appropriate change in failure frequency from Tables 3-3 to 3-6. These results would then be compared to the guidelines for

an acceptably small change in risk as defined in R.G. 1.174 and any risk requirements for the applicable risk-informed inservice inspection program for piping.

The final sentence of the Executive Summary will be changed to read:

> *"Further, the pilot-plant results provide examples which demonstrate that the effect of the extended inspection interval on the pilot plant's risk-informed inservice inspection program for piping is acceptable."*

The Beaver Valley Power Station, Unit 1 results are consistent with the Beaver Valley plant-specific PRA results. Ninety-four percent (94%) of the Large Early Release Frequency (LERF) calculated in the Beaver Valley Unit 1 PRA results from containment bypass through a steam generator tube rupture (SGTR) or interfacing system LOCA. The remaining LERF contributors are high-pressure melt ejection, temperature-induced SGTR and rocket mode containment failures. None of these containment failures result from LOCA initiators. The Beaver Valley containment was originally designed as a sub-atmospheric containment and continues to be operated at a slightly negative pressure which prevents large pre-initiator containment bypass. The results of the Beaver Valley Unit 1 plant-specific PRA indicate conditional containment failure probabilities in the range of 1E-4 to 1E-6 for the loss-of-coolant initiators.

DCI-RAI-10 ***Section 3.2.5.1 of TR WCAP-17236 lists under "Alternative Change-in-Risk Evaluation Methods" five steps for executing the proposed Method 1 to assess the impact of RV nozzle ISI interval extension on the existing RI-ISI program.***

1. ***For Step 2a, it is stated, "[f]or the welds examined per the [American Society of Mechanical Engineers (ASME) Code, Section XI] program, conservatively identify all welds examined by a volumetric and surface exam and by a surface exam only." Why are the examination methods limited to a volumetric and surface exam and a surface exam only? The ASME Code, Section XI examination methods also include volumetric only and visual only.***

2. ***For Step 3, it is stated, "[m]ultiply the applicable largest segment change in risk times the difference in the number of welds examined per [ASME Code, Section XI] and the RI-ISI programs for the reactor coolant system and the total plant." What is the interval (e.g., every 10 years) for this weld counting?***

Response

The intent was not to exclude volumetric-only examination. The text in step 2a of Section 3.2.5.1 of TR WCAP-17236-NP will be revised from:

> *"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."*

To:

> *"For the welds examined per the ASME Section XI program, identify all welds examined excluding welds with visual only examinations."*

Visual-only exams are excluded from the change-in-risk evaluation since there is no change to the Section XI visual exams when switching to a RI-ISI program.

The interval for weld counting is every 10 years.

DCI-RAI-11 ***At the end of the subsection "Alternative Change-in-Risk Evaluation Methods" of Section 3.2.5.1, TR WCAP-17236 summarizes in four bullets the reasons for conservatism in the three alternative change-in-risk evaluation methods. The first bullet states, "[t]he 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." The fourth bullet states, "...the RI-ISI exams typically address more risk than the [ASME Code, Section XI] exams on a per weld basis, since the RI-ISI exams are inspections for cause." Please explain why these two reasons are different, considering both reasons emphasize that the RI-ISI program inspects locations of potential degradation mechanism.***

Response

The first and fourth bullets both use the reason that the RI-ISI program is an inspection for cause. The fourth bullet was meant to emphasize this reason when considering that not all RI-ISI programs will result in the largest ISI change in segment risk.

For clarification, the following changes will be made to WCAP-17236-NP.

- Information from the fourth bullet will be merged with the first bullet, and the fourth bullet will be removed. Specifically, the text, "whereas RI-ISI exams are inspections for cause" will be added to the end of the first sentence in the first bullet, and the following sentence will be inserted after the first sentence in the first bullet.

  > *"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."*

- In the sentence between the second and third bullet, "conservatisms" will be changed to "conservatism".

DCI-RAI-12 ***Section 3.2.5.2 of TR WCAP-17236 states under "Change-in-Risk Criteria" for the qualitative method, "[t]he RI-ISI program must provide for an increase 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 [Regulatory Guide] 1.174 are met, the effect of the extended [ISI] interval on the RI-ISI program is acceptable." The above change-in-risk criteria for the qualitative method are for the approved Electric Power Research Institute (EPRI) RI-ISI methodology when the inspections are converted from the ASME Code, Section XI to a RI-ISI program. Evaluation of the effect caused by RV nozzle ISI interval extension is discussed later in TR WCAP-17236 under "Evaluation of Effect RV Nozzle ISI Interval Extension." Therefore, please clarify that (1) the last paragraph in the quote regarding the effect of the extended ISI interval on the RI-ISI program is not part of the approved EPRI RI-ISI methodology, and (2) by including this paragraph, TR WCAP-17236 is proposing a modification to the qualitative method of the approved EPRI RI-ISI methodology.

Response

The last sentence of the cited paragraph, "Provided that the risk acceptance criteria of RG [Regulatory Guide] 1.174 are met, the effect of the extended [ISI] interval on the RI-ISI program is acceptable," is not part of the approved EPRI RI-ISI methodology. This TR is not proposing a modification to the qualitative method of the approved EPRI RI-ISI methodology. The cited sentence will be removed from Section 3.2.5.2 of the TR.

DCI-RAI-13 ***Section 3.2.5.2 of TR WCAP-17236 presents, under "Evaluation of Effect RV Nozzle ISI Interval Extension," Equation (3-2) to be used in the third EPRI RI-ISI method, "Bounding with Credit for Increase in POD." The definition for parameter "I_{rj}" in the equation is not given, even though the meaning of the subscript "rj" is given. Please make appropriate revision.***

Response

Parameter "I_{rj}" in Equation (3-2), now Equation 3-12), of the TR is the inspection effectiveness factor for the risk-informed inspection program at location j. The parameter "I_{ej}" is the inspection effectiveness factor for the existing inspection program at location j. The definition of "I" will be added to Section 3.2.5.2 of the WCAP TR.

DCI-RAI-14 ***Table 3-18 of TR WCAP-17236 presents input and calculated values for parameters used in estimation of the reactor coolant system change in core damage frequency (ΔCDF) and change in large early release frequency (ΔLERF) values for Three Mile Island Nuclear Station, Unit 1, when Method A of the Markov model (the proposed EPRI RI-ISI methodology) is applied. Please provide information on calculation of the hazard rates (i.e., the first two rows with input values), failure rate (the fifth row), and conditional probability of rupture (the sixth row) for this plant-specific example.***

Response

The hazard rates were calculated using the Markov model that was used in the development of the piping risk-informed inservice inspection program for Three Mile Island Unit 1. The hazard rate values corresponding to a 10-year inspection interval and no inspection are identical to those values used for the RV nozzle welds in the Three Mile Island Unit 1 RI-ISI program risk impact assessment. The hazard rate value for the 20-year interval was calculated using the same Markov model equations and input as those used to calculate the 10-year inspection interval and no inspection values with the exception that the inspection interval input was changed from 10 years to 20 years.

The failure rate and conditional probability of rupture are also the same values that were used for the RV nozzle welds in the development of the Three Mile Island risk-informed inservice inspection program. These values can be found in Table 5 of Enclosure 2 to the Three Mile Island relief request to implement the risk-informed inservice inspection program (ML022830211). The values used are those for construction defects in the reactor coolant system. Also indicated in Table 5 is the basis for these values, which is the following Reference:

T.J. Mikschl and K.N. Fleming, "Piping System Failure Rates and Rupture Frequencies for Use in Risk informed Inservice Inspection Applications," EPRI TR-111880, 1999, September 1999.

DRA-RAI-1 ***TR WCAP-17236 first develops the change in risk associated with extending the inspection interval from 10 to 20 years. This methodology seems to assume that all reactor nozzle welds will be inspected each 20 years instead of each 10 years. TR WCAP-17236 then evaluates the effect of the increased inspection interval on a RI-ISI program. The relationship between changing both the programs is unclear. Are all reactor nozzle welds inspected every 10 years under Section XI? Will they all be inspected after 20 years under a RI-ISI program? If all welds were inspected under Section XI, but not all welds will be inspected under a RI-ISI program, how does this affect the examples and the conclusions in TR WCAP-17236.***

Response

TR WCAP-17236 describes two different evaluations for change in risk. The first evaluation is applicable to all plants implementing TR WCAP-17236 and uses the bounding failure frequencies without leak detection in Tables 3-3 to 3-6 for the appropriate type of nozzle and compares the calculated change in risk to the guidelines for an acceptably small change in risk in Regulatory Guide 1.174. The second evaluation is only applicable to plants that have implemented a piping RI-ISI program and uses the bounding failure frequencies with leak detection in Tables 3-3 to 3-6 for the appropriate type of nozzle and compares the calculated change in risk to the requirements for the plant-specific piping RI-ISI Program or the alternative

criteria proposed in TR WCAP-17236. In both evaluations, the calculated change in risk is proportional to the number of welds examined.

According to ASME Boiler & Pressure Vessel Code Section XI inspection requirements in Table IWB-2500-1, 100% of all Examination Category B-F RV welds must be inspected. Examination Category B-J RV nozzle welds are terminal ends in piping connected to the RV and 100% of these welds must also be inspected. TR WCAP-17236 does not modify inspection locations, just the inspection interval. For plants following Section XI that apply this TR, all reactor nozzle welds will be inspected on a 20-year interval and evaluated in only the first change-in-risk evaluation.

For plants with a RI-ISI Program for piping, not all RV nozzle welds may be required to be inspected. If any of the reactor nozzle welds are included in the piping RI-ISI Program, then the number of welds examined would be the same for calculating the change in risk for both evaluations. If all the nozzle welds are included in the piping RI-ISI Program, then all the nozzle welds will be inspected after 20 years and included in both change-in-risk evaluations. If only 50% of the nozzle welds are included in the Piping RI-ISI Program, then only 50% the nozzle welds will be inspected after 20 years and included in both change-in-risk evaluations. If none of the nozzle welds are included in the piping RI-ISI Program, then TR WCAP-17236 would not be used.

The inclusion of only some of the reactor nozzle welds in a piping RI-ISI Program has no affect on the conclusions in TR WCAP-17236 because both examples used all the vessel nozzle welds, which gives the maximum values for the calculated change in risk. Also, the methods provided in TR WCAP-17236 would still be applicable, even if only some of the reactor vessel welds had been examined per the plant-specific piping RI-ISI Programs.

DRA-RAI-2 ***Tables 2-1 and 3-9 (and other tables) of TR WCAP-17236 direct that the change in risk includes a multiplier characterized as "(# of welds examined)." Figure 3-3 illustrates the different nozzle types where it appears that weld Types B and D have two welds per nozzle implying that the frequency results for these types of welds should be multiplied by 2. Other discussion and examples in TR WCAP-17236 imply that a frequency estimate is developed for each nozzle, not each weld in the nozzle. Is the "(# of welds examined)" more appropriately labeled "(# of nozzles examined)"? If not, please clarify what the relationship is between the (# of welds examined) and the number of welds in a nozzle.***

Response

In order to calculate the change in risk for a specific plant, the number of welds examined must be multiplied by the summed changes in CDF and LERF results for all three failure modes (LOCA leak rates) as described in Section 3.2.4 of the TR. This change in risk is determined on a per weld examined basis and not a per nozzle examined basis. If there are two welds within a single nozzle and both of these welds are examined, then they must be included in the change-in-risk calculation. For example, if a 3-loop plant with Type D nozzles, as illustrated in Figure 3-3

of the TR, inspects every RV inlet and outlet nozzle weld, there would be a total of 6 inlet nozzle welds and 6 outlet nozzle welds examined. These totals would be used to calculate the total change in risk, as represented by the total change in CDF and LERF.

However, when evaluating the impact on the RI-ISI program for plants that have implemented the PWROG RI-ISI methodology and that are using the PWROG original change-in-risk evaluation, the evaluation is conducted on a per-segment basis. Thus, as discussed in the response to DRA-RAI-4, the change in risk added to the change in risk from the RI-ISI element selection should be calculated based on one weld per nozzle.

DRA-RAI-3 ***Section 2.4 of TR WCAP-17236 begins by stating, "...the analysis described above is sufficient for showing that the extension in inspection interval is acceptable. However, if the plant has implemented a risk-informed inservice inspection (RI-ISI) program, which includes the reactor vessel nozzle welds, additional evaluation is required. The following sections detail the evaluations ..." Sections 2.4.1 and 2.4.2 proceeded to describe a proposal to incorporate the changes in nozzle failure frequency into TR WCAP-14572 (Reference 1) and EPRI/N-716 (References 2 and 3) RI-ISI programs. Later, in Section 3.2.5 of TR WCAP-17236, a proposal to incorporate the changes in nozzle failure frequency into TR WCAP-14572 and EPRI/N-716 RI-ISI programs is again described. Why is the impact on the existing RI-ISI program methods discussed in two different sections of TR WCAP-17236? If the two sections are not combined into one section, please confirm that there are no differences in the methodologies described in the two sections.***

Response

The layout of the TR is based on discussions between Westinghouse, the PWROG and the NRC. The intent of this layout was to assist the NRC in the preparation of the SER. Section 2 of the TR is intended to contain the methodology that the PWROG wants to be approved by the NRC in their SER. Section 3 contains the technical basis for the methodology proposed in Section 2. It is understood that this layout results in some redundancy. Therefore, the text has been modified to make redundant sections more consistent. In some cases, the redundancy has been eliminated by deleting a paragraph in Section 3 and referring to the applicable paragraph in Section 2.

The following changes will be made to WCAP-17236-NP.

- A sentence will be added to the end of the first paragraph in Section 2.4.1 under Implementation Method.
- A new paragraph will be inserted following the above sentence.

Refer to response to DRA-RAI-5 for additional changes that also provide clarification.

DRA-RAI-4 ***Section 2.4.1 of TR WCAP-17236, subsection "Implementation Method", states that "when 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 RI-ISI element selection." The "most limiting" frequency as defined in TR WCAP-14572 is developed by assuming that all degradation mechanisms in a segment are present at the weld and imposing the most severe operating conditions on that weld. It states in Section 3.2.3 of TR WCAP-17236 that the objective of the failure frequency evaluation "was to determine a single run group that could provide a bounding change in failure frequency for all welds for each nozzle type." Therefore it appears that TR WCAP-14572 and TR WCAP-17236 both develop a most limiting failure frequency. If this is correct, what adjustment is referred to in Section 2.4.1? If this is incorrect, please compare the most limiting estimates in TR WCAP-14572 with those in TR WCAP-17236 and clarify what "adjustment" is to be made.***

Response

The intent of the sentence was to identify that even though a nozzle-to-pipe connection may contain two welds, the WCAP-14572 method only considers one weld per RI-ISI piping segment. In the WCAP-14572 method, all degradation mechanisms in a particular segment are combined and placed on one weld at the worst stress location for the SRRA analysis. The same approach was used to determine the bounding change in failure frequencies for WCAP-17236. Therefore, only one of the two welds needs to be considered. Since the bounding change in failure frequency is the same for either of the two welds, there is no adjustment that needs to be made to the bounding change in failure frequencies. Therefore the referenced sentence from Section 2.4.1 of the TR will be revised to read:

> *"Therefore, for nozzle configurations (see Figure 3-3 in Section 3.2.3) where there are two welds per nozzle, the change in risk added to the change in risk from the RI-ISI element selection should be calculated based on one weld per nozzle."*

DRA-RAI-5 ***Section 2.4.1 of TR WCAP-17236, subsection Acceptance Criteria, appears to change the acceptance guidelines for a RI-ISI program developed according to the TR WCAP-14572 method. Specifically, the last two sentences (including the four bullets) (1) define insignificant as a factor of 10 higher than the definition of insignificant in TR WCAP-14572 and (2) state that the total change in risk from implementing a RI-ISI program can be an increase in CDF and LERF of up to 10^{-6} and 10^{-7}, respectively. These changes appear to be proposed even if the standard changes in risk calculations in TR WCAP-14572 are used. In contrast, Section 3.2.5 of TR WCAP-17236 first introduces three alternative risk calculations and then proposes changes to the acceptance guidelines in TR WCAP-14572, but only after using the alternative methods. Does TR WCAP-17236 intentionally propose changing the acceptance guidelines from TR WCAP-14572 for all TR WCAP-14572 RI-ISI programs in Section 2.4.1, or only propose new guidelines to be used after application of the alternative methods in Section 3.2.5? If changes to the acceptance guidelines are***

proposed without application of the alternative methods, please justify these new guidelines.

Response

The intent of the topical report is to propose new acceptance criteria if the alternative change-in-risk methods proposed in TR WCAP-17236-NP are also used, and not to propose new acceptance criteria for the change-in-risk in TR WCAP-14572.

For clarification, the following changes will be made to WCAP-17236-NP.

- The acceptance criteria of WCAP-14572 will be referred to as the "PWROG Original Change-in-Risk Acceptance Criteria."
- The alternative acceptance criteria will be referred to as the "PWROG Alternative Change-in-Risk Acceptance Criteria."
- The following sentence will be added prior to the PWROG alternative acceptance criteria description in Section 2.4.1:

  "The PWROG alternative change-in-risk acceptance criteria can only be used for the alternative change-in-risk methods."

- The paragraph beginning with "If the acceptance criteria cannot be met..." between the original and alternative acceptance criteria will be removed.
- The following sentence will be added to the end of the second to last paragraph of Section 2.4.1 in the Implementation Method subsection:

  "If the acceptance criteria cannot be met, additional inspections shall be added to the RI-ISI program until the criteria are met."

- The following paragraph will be added following the above sentence.

  "If the PWROG original acceptance criteria cannot be met by adding additional inspections, or it is impractical to do so, an alternative 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. If one of the alternative methods described in greater detail in Section 3.2.5.1 is used to perform the change-in-risk evaluation, the PWROG alternative change-in-risk acceptance criteria, which is the same as the criteria from the EPRI RI-ISI methodology, must be met."

- The PWROG original acceptance criteria in Section 2.4.1 will be revised to more closely match the criteria listed in WCAP-14572 Revision 1-NP-A.
- The PWROG original acceptance criteria will be removed from Section 3.2.5.1 and reference will be made to Section 2.4.1.

Refer to DRA-RAI-3 for changes to Section 3.2.5.1.

DRA-RAI-6 ***The first alternative method in Section 3.2.5.1 of TR WCAP-17236 seems to address two overlapping populations of welds, those in the RCS and those in the total plant. When referring to welds "in the total plant," does this include or exclude the RCS welds?***

Response

Welds in the total plant include the reactor coolant system welds since the total plant welds include all welds within the scope of the RI-ISI program for comparison with the total change-in-risk criterion. Welds counts are also conducted for just the reactor coolant system for comparison with the system change-in-risk criteria.

DRA-RAI-7 ***Please define variables and provide equations for the three alternative methods in Section 3.2.5.1 of TR WCAP-17236. This will reduce the possibility of misunderstanding.***

Response

Equations for the three alternative change-in-risk methods will be added to Section 3.2.5.1 of the TR along with definitions of the variables. Clarifying text associated with the equations will also be added. Equation numbers for Section 3 will be revised to reflect the additional equations.

DRA-RAI-8 ***In Section 3.2.5.2, TR WCAP-17236 summarizes the change in risk calculations in the EPRI/N-716 methodology. One EPRI/N-716 method estimates the change in risk using a probability of detection and the uninspected failure frequency of a weld and seems to have no provision to evaluate changing inspection intervals. An alternative EPRI/N-716 method (Markov) does provide the capability to evaluate a change in the inspection interval but changes to inspection intervals were not envisioned and therefore not used in the RI-ISI programs.***

a) What differences between the previously approved and the proposed Markov method exist?

b) Is the use of the Markov method required to use the EPRI/N-716 RI-ISI program together with TR WCAP-17236? Alternatively, how can TR WCAP-17236 be used together with the EPRI/N-716 change in risk method that uses a probability of detection and the uninspected failure frequency of a weld?

Response

a) There is no difference between the approved and proposed Markov methods. The only difference is that the inspection interval input is changed to 20 years when calculating the hazard rate corresponding to the 20-year inspection interval.

b) No, the use of the Markov method is not required to use the EPRI/N-716 RI-ISI program together with TR WCAP-17236-NP. For a plant that has not used the Markov model to evaluate the change in risk associated with implementing their RI-ISI program (i.e., they have used the qualitative method of one of the two bounding methods), the method for evaluating the effect of the nozzle ISI interval extension is discussed starting on page 3-39 and ending in the top paragraph on page 3-41 of the WCAP TR. These methods make use of the bounding change in failure frequencies from Tables 3-3 to 3-6 that were calculated using the SRRA Code. For a plant that has used the Markov model to evaluate the change in risk associated with implementing their RI-ISI program, as stated on page 3-42 of TR WCAP-17236-NP, there are two different methods that could be used; Method A or Method B. Method A makes use of the Markov model to develop new hazard rates and inspection effectiveness factors corresponding to a 20-year interval for the RV nozzle welds. Method B makes use of the bounding change in failure frequencies calculated using the SRRA Code. Method B is similar to the approach proposed on pages 3-39 through 3-41 for the plants that have used the qualitative or bounding methods.

For plants that have used the EPRI/N-716 change-in-risk method that uses a probability of detection and the uninspected failure frequency of a weld, the method for evaluating the effect of the extension in inspection interval for the RV nozzles is the same as that described on page 3-40 of the TR that is used for the "Bounding without any Credit for Increase in Probability of Detection (POD)" and "Bounding with Credit for Increase in Probability of Detection (POD)" methods. As stated on page 3-40 of the TR, 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. The resulting 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.

DRA-RAI-9 **In Section 3.2.4, "Change-in-Risk Calculation," subsection "Change-in-Risk Calculation Method," the Topical Report (TR) states "The change in failure frequency should be taken for the licensed life of the plant (40 or 60 years)." Both the examples, Beaver Valley Power Station, Unit 1, and Three Mile Island Nuclear Station, Unit 1 (TMI-1), have 60 year licenses yet use the change in failure frequencies at 40 years. Please clarify which failure frequency estimates should be used.**

Response

Whether the failure frequencies are based on 40 or 60 years is dependent upon how the failure frequencies are being used. In Step 3: Perform Change-in-Risk Evaluation (Section 2.3 and/or 3.2.4), the failure frequencies are based on the licensed life of the plant (i.e., 40 or 60 years, as applicable). The calculations for RG 1.174 are conducted once and therefore should be based on the licensed period of operation. In Step 4: Evaluated Effect on Risk-Informed Inservice inspection Program (Section 2.4 and/or 3.2.5), the failure frequencies use the same basis of 40 or 60 years as is used in the current RI-ISI program. The intent of this step is to evaluate the

impact of the reactor vessel nozzle ISI interval extension on the RI-ISI program and therefore should use the same basis of 40 or 60 years as is used in the RI-ISI program. The RI-ISI programs are living programs and are re-evaluated on a periodic basis. The impact of the reactor vessel nozzle ISI interval extension on the RI-ISI program is repeated every time that the change-in-risk calculation is repeated for the RI-ISI program. For some plants the failure frequencies used in Step 3 may be based on 60 years while in Step 4 they may be based on 40 years.

Based on the above the following changes will be made to WCAP-17236-NP:

- In Section 3.2.4, the pilot plant change-in-risk calculations will be revised to reflect a failure frequency basis of 60 years instead of 40 years.
- Various other changes will be made throughout WCAP-17236-NP to clarify the failure frequency basis of 40 or 60 years as indicated above. Refer to the highlighted changes in the attached marked-up copy of WCAP-17236-NP for the specific changes.

DRA-RAI-10 While describing Method A for the Electric Power Research Institute (EPRI) risk-informed inservice inspection (RI-ISI) methods on Page 3-40, the TR states "In some applications of the EPRI RI-ISI methodology, the change-in-risk calculation may use only one LOCA [loss-of-coolant accident]-initiating event (the one that is determined in the risk evaluation to be the most limiting in terms of CDF [core damage frequency] and LERF [large early release frequency])." A similar statement is included for Method B on Page 3-43. However, in the examples provided, only a small LOCA conditional core damage probability (CCDP) and conditional large early release probability (CLERP) is combined with the rupture failure frequency. On Page 8 of 18 in the TMI-1 submittal (Agencywide Documents Access and Management System accession number ML022830211) that is referenced as the example states "The failure rates and rupture frequencies that were used in this evaluation are from Reference 4." Reference 4 is "Piping System Failure Rates and Rupture Frequencies for Use in Risk Informed Inservice Inspection Applications," EPRI TR-111880, 1999, September 1999 EPRI Licensed Material. Reference 4 provides a single rupture frequency estimate for welds in each system and degradation mechanism. Given a single rupture frequency, the larger CCDP and CLERP for large LOCAs would appear to be most limiting in terms of CDF and LERF. Please explain this apparent discrepancy.

Response

It is agreed that the Large LOCA CCDP and CLERP values are the most limiting in terms of CDF and LERF. The TMI-1 examples in Tables 3-18 and 3-19 of WCAP-17236-NP will be revised to use the CCDP and CLERP values for Large LOCA. The TMI-1 example in Table 3-19 will also be revised to use the bounding change in failure frequencies for Large LOCA.

Attachment B: WCAP-17236-NP Mark-Up

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 provide examples which demonstrate that the effect of the extended inspection interval on the pilot plant's risk-informed inservice inspection program for piping is acceptable.

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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 a disconnect 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.

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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

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. Additionally, the basis of 40 or 60 years for the failure frequencies is consistent with the 40 or 60 year basis used in the current RI-ISI program. 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 PWROG original change-in-risk acceptance criteria discussed in the following section. If the PWROG original change-in-risk acceptance criteria cannot be met, additional inspections shall be added to the RI-ISI program until the criteria are met.

if the PWROG original change-in-risk acceptance criteria cannot be met by adding additional inspections, or it is impractical to do so, an alternative 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. If one of the alternative methods described in greater detail in Section 3.2.5.1 is used to perform the change-in-risk evaluation, the PWROG alternative change-in-risk acceptance criteria, which is the same as the criteria from the EPRI RI-ISI methodology, must be met.

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 change in risk added to the change in risk from the RI-ISI element selection should be calculated based on one weld per nozzle.

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Acceptance Criteria

PWROG Original Change-in-Risk 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. 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 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.

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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.

PWROG Alternative Change-in-Risk Acceptance Criteria

The PWROG alternative change-in-risk acceptance criteria can only be used for the alternative change-in-risk methods.

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

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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. The basis of 40 or 60 years for the change in failure frequencies is consistent with the 40 or 60 year basis used in the current RI-ISI program. 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 and 40 or 60 year basis to be consistent with the RI-ISI program. 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, determined on a 40 or 60 year basis consistent with the RI-ISI program, 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.

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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. The bounding change in failure frequencies are based on the 40 or 60 year basis consistent with the RI-ISI program. 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.

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.

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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 two 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.

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- 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 for initial flaw

conditions was selected such that one flaw was simulated at the worst stress location in each weld. All flaws are surface breaking and circumferentially oriented.

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- 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

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 3.2.2, 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

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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 60-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.

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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 and 60-Year Basis)	CCDP	ΔCDF (/year)	CLERP	ΔLERF (/year)
Outlet Nozzles					
SLOCA	1.18E-07	1.38E-05	1.63E-12	7.61E-12	8.97E-19
MLOCA	1.18E-07	1.68E-03	1.98E-10	4.70E-08	5.53E-15
LLOCA	1.18E-07	2.15E-03	2.53E-10	5.30E-08	6.23E-15
# of Welds Examined	3	Total ΔCDF	1.36E-09	Total ΔLERF	3.53E-14
Inlet Nozzles					
SLOCA	1.13E-07	1.93E-04	2.18E-11	2.90E-10	3.27E-17
MLOCA	1.23E-07	1.68E-03	2.07E-10	4.70E-08	5.78E-15
LLOCA	1.23E-07	2.15E-03	2.65E-10	5.30E-08	6.53E-15
# of Welds Examined	3	Total ΔCDF	1.48E-09	Total ΔLERF	3.70E-14
All Nozzles					
Total Change-in-Risk Results		Total ΔCDF	2.84E-09	Total ΔLERF	7.23E-14

Three Mile Island Unit 1

Three Mile Island Unit 1 is a B&W NSSS design and has Type A RV Nozzle welds. The 60-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.

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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 and 60 Year Basis)		CCDP	ΔCDF (/year)	CLERP	ΔLERF (/year)
Outlet Nozzle						
SLOCA	3.93E-10		1.83E-03	7.20E-13	2.53E-04	9.95E-14
MLOCA	1.20E-11		2.23E-03	2.68E-14	2.55E-04	3.06E-15
LLOCA	5.42E-12		3.93E-02	2.13E-13	8.06E-04	4.37E-15
# of Welds Examined		2	Total ΔCDF	1.92E-12	Total ΔLERF	2.14E-13
Inlet Nozzle						
SLOCA	1.97E-10		1.83E-03	3.61E-13	2.53E-04	5.00E-14
MLOCA	6.32E-12		2.23E-03	1.41E-14	2.55E-04	1.61E-15
LLOCA	5.84E-12		3.93E-02	2.29E-13	8.06E-04	4.71E-15
# of Welds Examined		4	Total ΔCDF	2.42E-12	Total ΔLERF	2.25E-13
All Nozzles						
Total Change-in-Risk Results			Total ΔCDF	4.34E-12	Total ΔLERF	4.39E-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.

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- 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 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

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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 PWROG original change-in-risk acceptance criteria described in Section 2.4.1. 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 PWROG original change-in-risk acceptance criteria.

Evaluation of Effect of 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. The basis of 40 or 60 years for the failure frequencies is consistent with the 40 or 60 year basis used in the current RI-ISI program. 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 B and D, 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 PWROG original change-in-risk acceptance 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

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The criteria for evaluating the change in risk for a PWROG-methodology-based RI-ISI program, as identified in Reference 4, is as follows:¶
<#>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.¶
<#>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.¶
<#>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:¶
<#>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.¶
<#>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.¶
<#> If any additional examinations are identified, the change-in-risk calculations should be revised to credit these additional examinations.¶

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with the first alternative method and move to the more complex methods until the results indicate an acceptable change in risk when compared against the PWROG alternative change-in-risk acceptance criteria or additional exams are added to make the change in risk acceptable. These methods are discussed in more detail in the following sections.

First Alternative Evaluation Method – Examined Weld Counts Using Largest Change In Risk

In the first alternative evaluation method, the change in risk is based on the largest applicable segment change in risk. The reactor coolant system change in risk is evaluated using equation 3-1. The total plant change in risk is evaluated using equation 3-2.

$$\Delta CDF_{RCS} = \Delta CDF_{Nozzles} + \Delta CDF_{RCS-RI-ISI} \quad (3\text{-}1)$$

$$\Delta CDF_{All} = \Delta CDF_{Nozzles} + \Delta CDF_{All-RI-ISI} \quad (3\text{-}2)$$

Where:

ΔCDF_{RCS} = Change in CDF in the reactor coolant system between the ASME and RI-ISI programs including the effect of the reactor vessel nozzle ISI interval extension.

$\Delta CDF_{Nozzles}$ = Change in CDF from the reactor vessel nozzle ISI interval extension.

$\Delta CDF_{RCS-RI-ISI}$ = Change in CDF in the reactor coolant system between the ASME and RI-ISI programs excluding the effect of the reactor vessel nozzle ISI interval extension.

ΔCDF_{All} = Change in CDF in the total plant between the ASME and RI-ISI programs including the effect of the reactor vessel nozzle ISI interval extension.

$\Delta CDF_{All-RI-ISI}$ = Change in CDF in the total plant between the ASME and RI-ISI programs excluding the effect of the reactor vessel nozzle ISI interval extension.

Similar equations are conducted for the LERF. The equations are solved using the following steps:

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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.

   Where:

   $\Delta CDF_{Segment-RCS}$ = Maximum segment change in risk for the reactor coolant system segments that are in the scope of the RI-ISI program.

   $\Delta CDF_{Segment-All}$ = Maximum segment change in risk for all segments that are in the scope of the RI-ISI program including the reactor coolant system.

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, identify all welds examined excluding welds with visual only examinations.

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

   Where:

   SXI_{RCS} = Number of reactor coolant system welds within the scope of the RI-ISI program that are examined per the ASME Section XI program excluding visual only exams.

   SXI_{All} = Number of all welds within the scope of the RI-ISI program that are examined per the ASME Section XI program, including the reactor coolant system, excluding visual only exams.

   $RIISI_{RCS}$ = Number of reactor coolant system welds that are examined per the RI-ISI program, excluding visual only exams.

   $RIISI_{All}$ = Number of all welds that are examined per the RI-ISI program, including the reactor coolant system, excluding visual only exams.

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. See equations 3-3 and 3-4.

$$\Delta CDF_{RCS-RI-ISI} = \Delta CDF_{Segment\ Max\ RCS} * (SXI_{RCS} - RIISI_{RCS}) \quad (3\text{-}3)$$

$$\Delta CDF_{All-RI-ISI} = \Delta CDF_{Segment\ Max\ All} * (SXI_{All} - RIISI_{All}) \quad (3\text{-}4)$$

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. Refer to equations 3-1 and 3-2.

5. Compare the results of step 4 against the PWROG alternative change-in-risk acceptance criteria.

   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.

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

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. Refer to equation 3-1 for estimating the reactor coolant system change in risk. The total plant change in risk is evaluated using equation 3-5.

$$\Delta CDF_{All} = \Delta CDF_{Nozzles} + \sum_{j} \Delta CDF_j \qquad (3\text{-}5)$$

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Where:

ΔCDF_j = Change in CDF in the system j between the ASME and RI-ISI programs excluding the effect of the reactor vessel nozzle ISI interval extension.

A similar equation is used for LERF. The equation is solved using the following steps:

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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.

   Where:

   $\Delta CDF_{System\ j}$ = Maximum segment change in CDF for system j segments that are 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.

   Where:

   SXI_j = Number of system j welds within the scope of the RI-ISI program that are examined per the ASME Section XI program excluding visual only exams.

   $RIISI_j$ = Number of system j welds that are examined per the RI-ISI program excluding visual only exams.

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. Refer to equation 3-6.

$$\Delta CDF_j = \Delta CDF_{System\ j} * (SXI_j - RIISI_j) \qquad (3\text{-}6)$$

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. Refer to equation 3-1.

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. Refer to equation 3-6.

6. Compare the results of step 5 against the PWROG alternative change-in-risk acceptance criteria

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   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 reactor coolant system change in risk is evaluated using equation 3-7. The total plant change in risk is evaluated using equation 3-8.

$$\Delta CDF_{RCS} = \Delta CDF_{Nozzles} + \Delta CDF_{Ter\,RCS} \quad (3\text{-}7)$$

$$\Delta CDF_{All} = \Delta CDF_{Nozzles} + \sum_{j} \Delta CDF_{j} \quad (3\text{-}8)$$

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Where:

ΔCDF_{j} = Change in CDF for system j accounting for the number of welds examined in system j.

$\Delta CDF_{j\,Ter\,RCS}$ = Change in CDF for system j accounting for the number of welds examined in system j where system j is limited to the reactor coolant system.

A similar equation is used for LERF. The equations are solved using the following steps:

Deleted: 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.

   Where:

   $\Delta CDF_{i,j}$ = Change in CDF for segment i (of system j) for the segments that are 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 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.

Where:

$SXI_{i,j}$ = Number of welds in segment i (in system j) within the scope of the RI-ISI program that are examined per the ASME Section XI program excluding visual only exams.

$RIISI_{i,j}$ = Number of welds in segment i (in system j) that are examined per the RI-ISI program excluding visual only exams.

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. Refer to equation 3-9.

$$\Delta CDF\#_{i,j} = \Delta CDF_{i,j} * (SXI_{i,j} - RIISI_{i,j}) \quad (3\text{-}9)$$

Where:

$\Delta CDF\#_{i,j}$ = Change in CDF for segment i (of system j) accounting for the number of welds examined in segment i.

4. Sum the individual segment change in risk for each segment in a system to obtain the system change in risk. Refer to equation 3-10.

$$\Delta CDF_j = \sum_i \Delta CDF\#_{i,j} \quad (3\text{-}10)$$

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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. Refer to equation 3-7.

6. Add the reactor vessel nozzle ISI interval extension risk increase as calculated on a weld basis to the sum of the change in risk for each system in the scope of the RI-ISI program to obtain the total plant change in risk. Refer to equation 3-8.

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Deleted: 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 PWROG alternative change-in-risk acceptance criteria:

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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 whereas 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. 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 conservatism:

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- 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.

PWROG Alternative Change-in-Risk Acceptance Criteria

Deleted: <#>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.¶

Deleted: Change-in-Risk Criteria for

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Deleted: 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.¶

The PWROG alternative change-in-risk acceptance criteria for the alternative change-in-risk evaluation methods are the same as the change-in-risk criteria used for the EPRI methodology. Refer to Section 2.4.1. 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 and the failure frequencies were based on 40 years to be consistent with the current RI-ISI program. 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 and 40-Year Basis)	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 PWROG original change-in-risk acceptance criteria. 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.

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

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Based on not being able to meet the PWROG original change-in-risk acceptance criteria, 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.

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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 PWROG alternative 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 PWROG alternative 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 PWROG alternative 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 be 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.

   Deleted: 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 of 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 3-11:

$$\Delta CDF_j = \sum_i [FR_{i,j} * (SXI_{i,j} - RISI_{i,j}) * CCDP_{i,j}]. \qquad (3\text{-}11)$$

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 Tables 3-3, 3-4, 3-5, or 3-6. The basis of 40 or 60 years for the change in failure frequencies is consistent with the 40 or 60 year basis used in the current RI-ISI program. 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 3-12, 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:

$$\Delta CDF_j = (F_{rj} - F_{ej}) * CDF_j = (I_{rj} - I_{ej}) * F_{0j} * CCDP_j \qquad (3\text{-}12)$$

Where the subscripts "rj" refer to the risk informed inspection program at location j, and the subscripts "ej" refer to the existing inspection program at location j. I is the inspection effectiveness factor. F_{0j} is the frequency of pipe rupture at location j, if no inspection is performed. $CCDP_j$ is the conditional core damage probability from a pipe rupture at location j, which is independent of the inspection strategy.

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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 and a basis of 40 or 60 years to be consistent with the RI-ISI program, 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 3-13 and 3-14:

$$\Delta CDF_j = \sum_{i=1}^{N} n_i \lambda_i P_i(R|F)(I_{i,new} - I_{i,old})CCDP_i \qquad (3\text{-}13)$$

and

$$\Delta LERF_j = \sum_{i=1}^{N} n_i \lambda_i P_i(R|F)(I_{i,new} - I_{i,old})CLERP_i \,. \qquad (3\text{-}14)$$

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,

$P_i(R|F)$ = 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 3-15 and 3-16:

$$I_{i,new} = \frac{h_{40}\{RI-ISI\}}{h_{40}\{noinsp\}} \text{ and} \tag{3-15}$$

$$I_{i,old} = \frac{h_{40}\{SecXI\}}{h_{40}\{noinsp\}}. \tag{3-16}$$

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, determined on a 40 or 60 year basis consistent with the RI-ISI program, 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 and a 40 or 60 year basis consistent with the RI-ISI program 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:

Method A

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 and a 40-year basis, consistent with the RI-ISI program. 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_d)	8.16E-06	
Cond. Prob. Rupture (P_r(R|F))	4.76E-02	
LLOCA CCDP	3.93E-02	
LLOCA CLERP	8.06E-04	

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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
$\Delta CDF = \lambda_f P_f(R\|F)(\Delta I)CCDP$ (per nozzle)	3.91E-08	
$\Delta LERF = \lambda_f P_f(R\|F)(\Delta I)CLERP$ (per nozzle)	6.18E-11	
Number of RV Nozzle Welds Examined	6	
Total Nozzle ΔCDF (/year)	1.81E-08	
Total Nozzle ΔLERF (/year)	3.71E-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)	2.48E-08	
New RC System ΔLERF (/year)	1.49E-09	

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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 and a 40-year basis, 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 and 40-Year Basis)		CCDP	ΔCDF (/year)	CLERP	ΔLERF (/year)
Outlet Nozzle						
LLOCA	2.17E-12		3.93E-02	8.53E-14	8.06E-04	1.75E-15
# of Welds Examined		2	Total	1.71E-13	Total	3.50E-15
Inlet Nozzle						
LLOCA	1.39E-12		3.93E-02	5.48E-14	8.06E-04	1.12E-15
# of Welds Examined		4	Total	2.19E-13	Total	4.49E-15
All Nozzles						
Nozzle Change-in-Risk Results				3.90E-13		7.99E-15
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

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