Designated as Original
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October 6, 2011

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PWR Owners Group

Program Management Office
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June 20, 2011

OG-11-193

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 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.

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.

Enclosure 1 (Appendix A) to this letter provides the RAI responses to the questions received in Reference 3. Westinghouse is still working on providing some additional clarification to the response for DCI-RAI-7 and that will be provided in a subsequent letter. Enclosure 1 (Appendix B) also contains the proposed changes that are to be incorporated in the revised WCAP.

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NRR

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.

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

MLA:JPM:las

Enclosures: (1) – LTR-AMLRS-11-42

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cc:

Date: June 6, 2011

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Our ref: LTR-AMLRS-11-42

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. 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.

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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-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		
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.

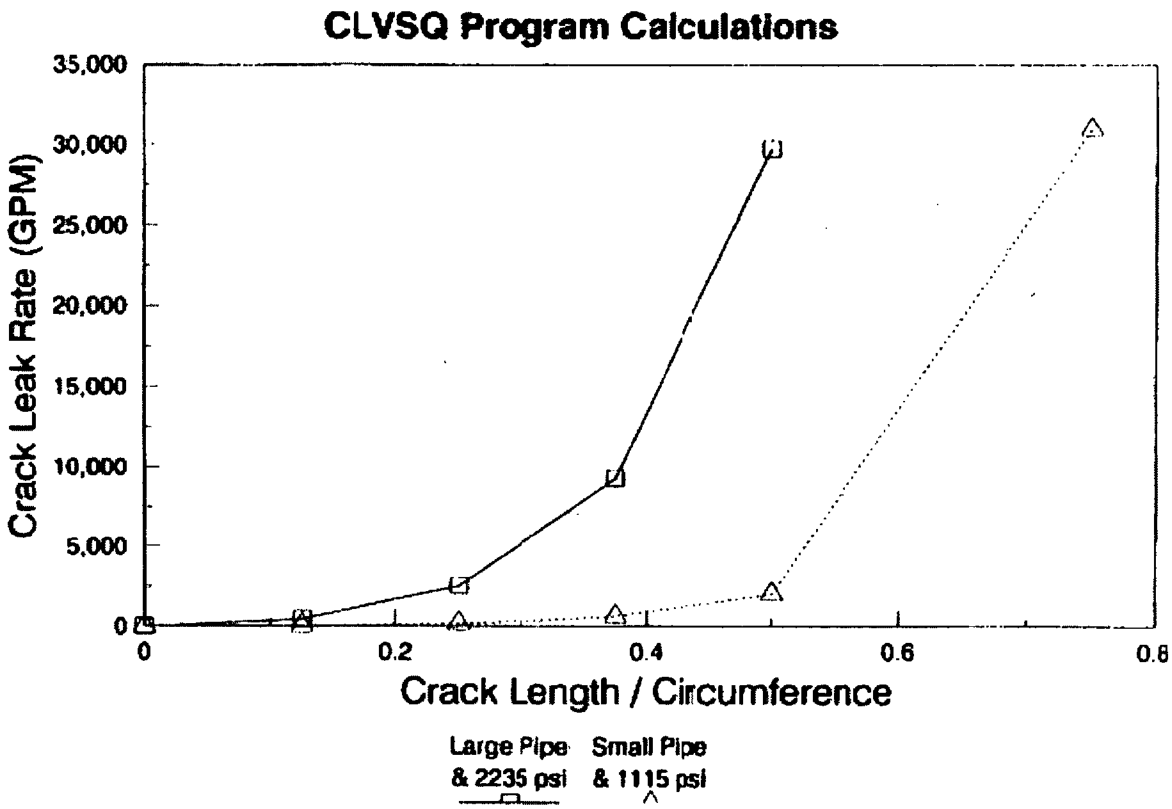


Figure DCI-RAI-2-1 – Leak Rate Program Calculations (from Reference 4)

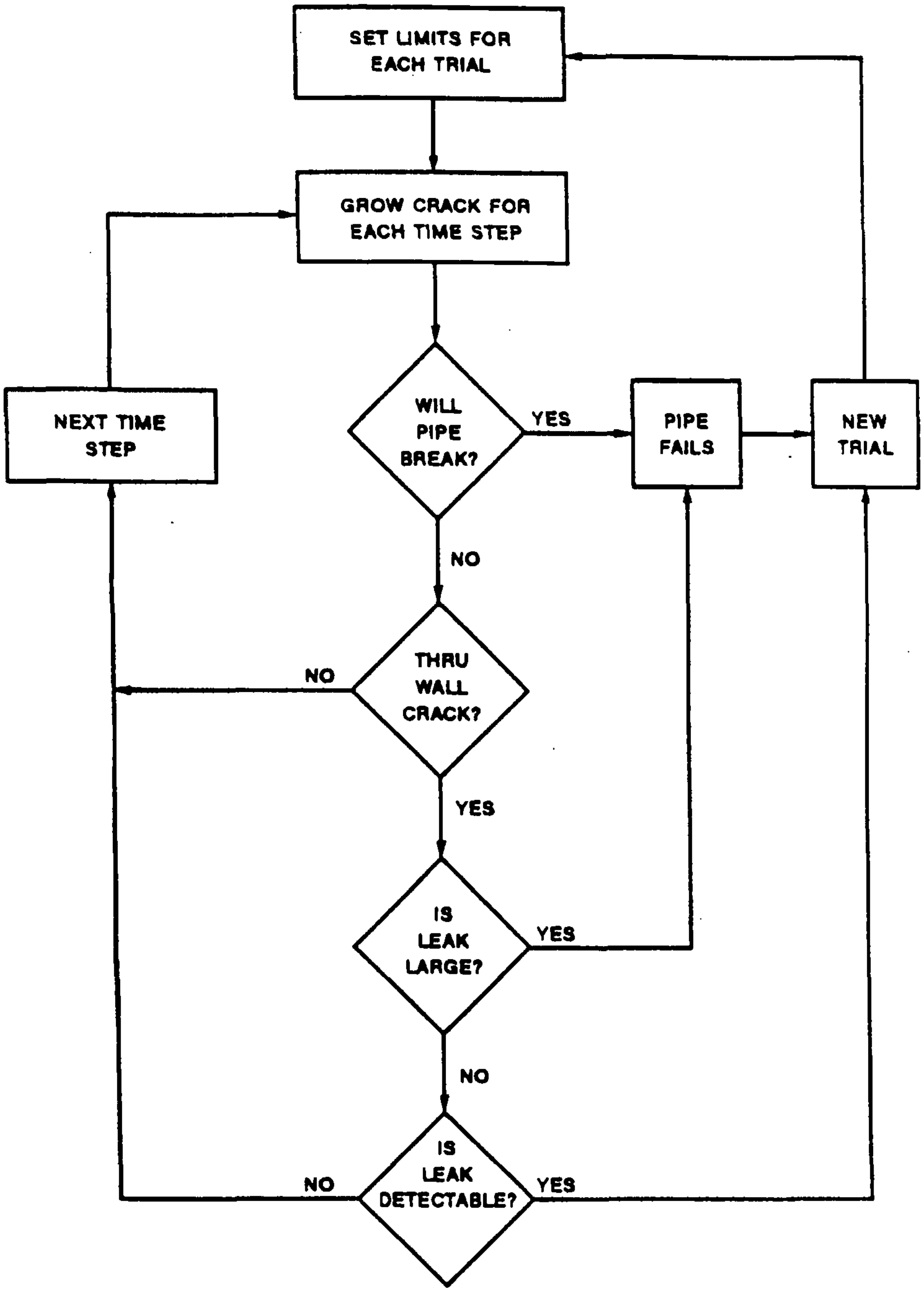


Figure DCI-RAI-2-2 – Flow Chart for Piping Weld Failure Modes (from Reference 4)

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.

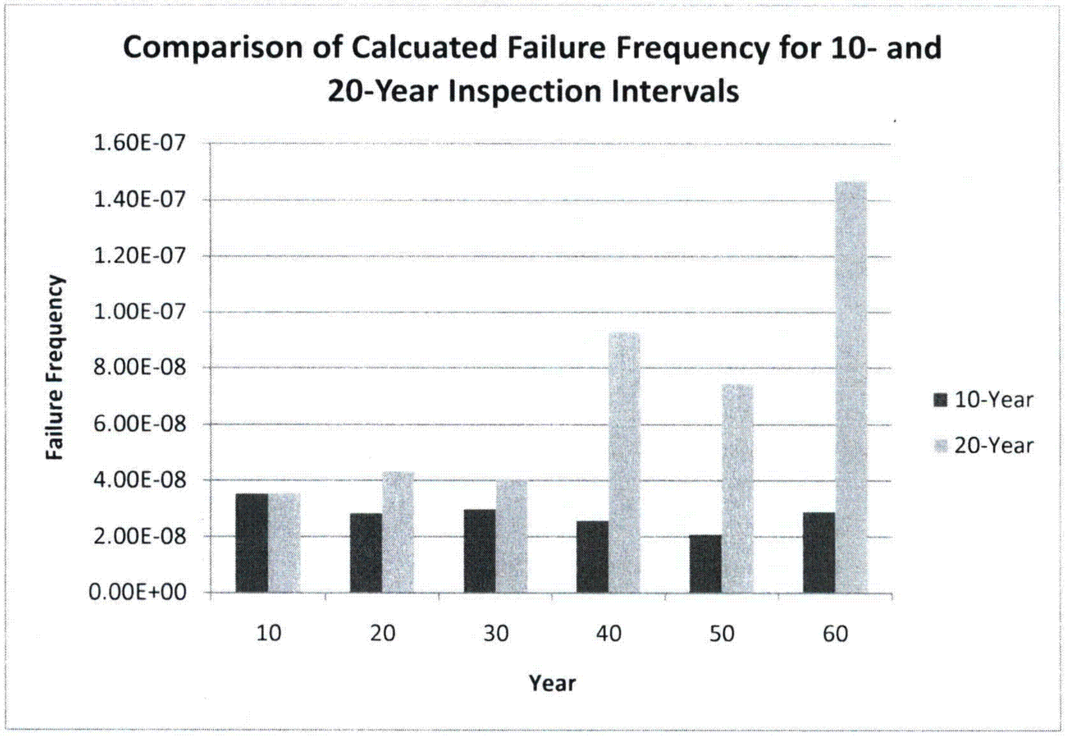
DCI-RAI-4 ***Section 3.2.3 of TR WCAP-17236 states under "Method," "[t]his difference in failure probability is then converted to a change in failure frequency by dividing the difference in failure probability by the respective number of years, 40 or 60." Using the first row of results in Table 3-7 as 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

Input Parameter	Value
Minimum Detectable Leak Rate (GPM)	0.0
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" option 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.

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.

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. 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.

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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:

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

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

Change-in-Risk Results and Conclusions

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

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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.

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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 RV Nozzle ISI Interval Extension

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

Alternative Change-in-Risk Evaluation Methods

If the 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

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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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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.

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

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

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:

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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:

   - $\Delta CDF_{SegMaxRCS}$ = 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.

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

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   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.

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

$$\Delta CDE_{All\text{-}RI\text{-}ISI} = \Delta CDE_{SegMax,All} * (SXI_{All} - RIISI_{All}) \qquad (3\text{-}4)$$

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.

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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 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. 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.

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

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:

   $\Delta CDF_{SegMaxj}$ = 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.

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

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

   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.

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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:

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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.

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

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.

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

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. 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.

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 HSS segment

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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. 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.

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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 ¶

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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.¶

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

$RISI_{i,j}$ = Number of RISI inspection elements for risk element i of system j,
$CCDP_{i,j}$ = Conditional core damage probability given a break in risk element i of system j.

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

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

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

This approach is consistent with the second approach discussed above but this approach allows for an increase in the probability of detection based on the use of an inspection strategy that is based on the postulated degradation mechanism. This is illustrated in equation 3-12, 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, 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:

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

and

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

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-ISI\} = hazard rate (time-dependent rupture frequency) for weld subjected to the RISI inspection strategy,

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

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

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

Method A

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