ENERGY NORTHWEST

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GO2-00-199

Docket No. 50-397

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

Gentlemen:

Subject: **WNP-2, OPERATING LICENSE NPF-21**
REQUEST FOR APPROVAL OF ALTERNATE RISK-INFORMED INSERVICE INSPECTION (RI-ISI) REQUIREMENTS
(ADDITIONAL INFORMATION)

Reference: Letter, dated November 13, 2000, Jack Cushing (NRC) to JV Parrish (Energy Northwest), "Request for Additional Information (RAI) for WNP-2, (TAC NO. MA9827)"

In the reference, the staff requested additional information to support review of our pending request for approval of alternate risk-informed inservice inspection requirements.

The requested information is included as an attachment. Should you have any questions or desire additional information regarding the matter, please call me or PJ Inserra at (509) 377-4147.

Respectfully,

D. W. Coleman
for

RL Webring
Vice President, Operations Support/PIO
Mail Drop PE08

Attachment

cc: EW Merschoff - NRC RIV
JS Cushing - NRC NRR
NRC Resident Inspector - 988C
TC Poindexter - Winston & Strawn
DL Williams - BPA/1399

A047

Request for Additional Information Response
Energy Northwest
Docket No. 50-397

NRC Question 1

Please provide the following information

A. When does the current 10-year inspection interval start and end?
B. When does the current inspection period start and end?
C. What cumulative percentage of inspections have been completed for the current interval?

Energy Northwest Response to Question 1

A. The current ISI inspection interval began February 10, 1995 and ends December 12, 2005. This interval was extended per ASME Section XI 2430(d) to accommodate our transition to a 24-month fuel cycle.
B. We are in our second inspection period of the second inspection interval. The current period started December 13, 1998 and ends December 12, 2001.
C. We have completed 38% of the Code Category B-J weld examinations scheduled for the second inspection interval. These examinations were completed during the first and part of the second inspection periods.

NRC Question 2

The implementation of a RI-ISI program for piping should be initiated at the start of a plant's 10-year inservice inspection interval consistent with the requirements of the ASME Code Section XI, Edition and Addenda committed to by the Owner in accordance with 10 CFR 50.55a. However, the implementation may begin at any point in an existing interval as long as the examinations are scheduled and distributed to be consistent with ASME XI requirements, e.g., the minimum examinations completed at the end of the three inspections intervals under Program B should be 16%, 50%, and 100%, respectively, and the maximum examinations credited at the end of the respective periods should be 34%, 67%, and 100%.

It is our view that it is a virtual necessity that the programs for the RI-ISI inspections (RI-ISIs) and for the balance of the inspections be on the same interval start and end dates. This can be accomplished by either implementing the RI-ISIs at the beginning of the interval or merging RI-ISIs into the program for the balance of the inspections if the RI-ISIs are to begin during an existing ISI interval. One reason for this view is that it eliminates the problem of having different Codes of record for the RI-ISIs and for the balance of the inspections. A potential problem with using two different interval start dates and hence two different Codes of record would be having two sets of repair/replacement rules depending upon which program identified the need for repair (e.g., a weld inspection versus a pressure test).

NRC Question 2 (Continued)

In addition, with the change to a RI-ISI program the Code minimum and maximum percentages of examination per period still apply to the RI-ISIs. For example, if a licensee is interested in starting the RI-ISIs during the second period, either the RI-ISIs or the Code required inspections should satisfy the second period minimum/maximum percentages. The code required percentages would have already been satisfied for the first period.

Please describe your implementation plan with respect to the above discussion.

Energy Northwest Response to Question 2

We will merge the RI-ISI program into the existing ISI program. Our plan as stated in our submittal would have all RI-ISI welds completed by the end of the current inspection interval. Of the 84 welds selected for RI-ISI examination, 17 welds (20%) have already been examined in the first period under the existing Section XI program. We will complete between 50% and 67% of the 84 welds by the end of the second inspection period. The remaining RI-ISI welds will be completed during the third inspection period. This will result in examination of all the RI-ISI welds during the existing Section XI program or under the RI-ISI program by the end of the second inspection interval.

NRC Question 3

Will the RI-ISI program be updated every 10 years and submitted to the NRC consistent with the current ASME XI requirements?

Energy Northwest Response to Question 3

Yes, the RI-ISI program will be updated every 10 years and submitted to the NRC consistent with the current ASME XI requirements.

NRC Question 4

Under what conditions will the RI-ISI program be resubmitted to the NRC before the end of any 10-year interval?

Energy Northwest Response to Question 4

A revised RI-ISI program would be submitted to the NRC if significant impact on the RI-ISI program occurred due to any new risk insights, plant changes, industry information, or if the basis for NRC's approval of the program were discovered. Changes to the RI-ISI program would not take place without prior NRC approval.

NRC Question 5

Section 3.5 states that since there are no high-risk segments identified in the Main Steam (MS), Low Pressure Core Spray (LPCS) and the Standby Liquid Control (SLC) systems, no elements in these systems were included in the RI-ISI program volumetric inspection plan. However, it would be prudent to select some elements for inspection from these systems for defense-in-depth purposes and to comply with EPRI TR-112657, Rev. B-A, Section 3.6.5, which states that considerations that go into element selection include distribution of inspections among systems. Please explain the rationale for not including any elements for these systems.

Energy Northwest Response to Question 5

Risk categories for Main Steam (MS) piping segments are 4, 5, 6, and 7. Sixteen (16) MS welds are examined under the existing high-energy line break (HELB) program that remains unchanged by this RI-ISI submittal. All of these MS welds will be examined prior to the end of the current ISI inspection interval.

LPCS welds are risk category 5 and 6. In order to provide defense in depth for the system, one additional weld inspection is added from the system and will be reflected in our revised submittal.

SLC has only two 1-1/2 inch butt welds that have no degradation mechanism identified and these welds are classified as risk category 4. The rest of the SLC welds in the segment are socket welds and as such, are outside the scope of Code Case N-560 RI-ISI. The SLC system connects to the HPCS system that is selected for RI-ISI inspection. Please see Table 3.8-1.

NRC Question 6

Section 1.2 states that the NRC staff noted a weakness in the licensee's evaluation of human errors as part of their IPE and cautioned the use of the IPE for applications that are sensitive to human error analysis. How have you addressed this weakness as it applies to this submittal?

Energy Northwest Response to Question 6

The statement in Section 1.2 was made based on NRC review of the information provided in the "WNP-2 Individual Plant Examination" Revision 1, July 1994. Periodic updates of the Probablilistic Safety Assessment (PSA) model and documents have been performed to ensure that the PSA results and applications of the results are based on current plant configuration, operating and maintenance practices with associated human error modeling, and plant specific systems and component failure rate data. The RI-ISI consequence analysis was performed using the PSA, Revision 4, March 2000. This latest version included the following:

- An examination of plant procedures and practices of each pre-initiator human action was performed to confirm that the recovery factors were applicable to each human error probability (HEP). The value of the recovery factor was based on the assumption that each human action would be independently checked by an operator and/or subjected to a functional test. In addition, a data review was performed to ensure uniform application of the modeling of pre-initiator human actions to systems.

- Additional HEP was provided to account for RHR/LPCS common cause human error due to miscalibration of sensors.

- The Modular Accident Analysis Program (MAAP) results were used to quantify the time available for the operator to perform specific post-initiator actions. The expected implementation times were estimated by a licensed senior reactor operator and verified by the emergency operations program coordinator.

- A comparison of post-initiator HEPs and typical ranges of the risk significant human errors given in EPRI's "Final PSA Applications Guide" was performed to verify their applicability. A sensitivity study was performed for HEPs that did not fall into the recommended range by setting the failure probability to a value in the recommended range and confirming that the impact to the core damage frequency was small.

- The important pre-initiator and post-initiator human actions identified by the human reliability analysis have been communicated to the licensed operators and engineering support personnel via respective training programs.

Based on the above, our submittal will be revised to remove the statement regarding PSA weakness.

NRC Question 7

The first paragraph of Section 3.8 states that "... the risk from implementation of this program is expected to remain neutral or decrease when compared to that estimated from current requirements." A similar statement is made in the last sentence of the introductory portion of Section 3. However, from Table 3.8-1 the estimated CDF and LERF increased by 9.6E-8/year and 1.8E-9/year, respectively. Please explain this apparent discrepancy and/or correct the statements.

Energy Northwest Response to Question 7

The first paragraph will be changed to read " ... the risk increase from implementation of this program is expected to be negligible."

NRC Question 8

Do the RI-ISI Exams values presented in Table 3.8-1 include credit for some of the augmented inspection programs (e.g., HELB, IGSCC, and FAC) or are these inspections in addition to the augmented inspections? In addition, do the ΔCDF and ΔLERF values presented in Table 3.8-1 include the contribution from the augmented inspection programs?

Energy Northwest Response to Question 8

The Energy Northwest RI-ISI program credits 36 elements (welds) from the intergranular stress corrosion cracking (IGSCC) augmented inspection program (see response to question 8c). No credit was taken for any ongoing flow accelerated corrosion (FAC) program exams. All RI-ISI inspections are in addition to FAC program exams.

We have reassessed our handling of IGSCC in our original delta risk evaluation. All the current Section XI inspection locations in Class 1 piping susceptible to IGSCC will continue to be inspected as part of the plant's GL 88-01 program. We decided to not take credit for these continuing inspections when assessing the risk impact associated with the proposed RI-ISI program. Therefore, where present, the contribution of IGSCC will be included in the delta risk assessment. This will result in a conservative delta risk assessment that is independent of the IGSCC program inspection scope. This is described in more detail in our response to question 11a below.

Table 3.8-1 will be modified to follow the format of the EPRI template as previously approved for the James A. Fitzpatrick Nuclear Power Plant RI-ISI submittal to describe the augmented inspection programs considered in the RI-ISI program. For our discussion on HELB, please refer to our response to question 8b.

NRC Question 8a

Please explain how the IGSCC B-G and the FAC degradation mechanisms are accounted for during the ranking process. Also explain how they are accounted for in the change in risk calculation. For example, in Table 3.8-1, in the first entry for the RRC system it appears that the FAC degradation mechanism is included in the ranking to place the segment in Risk Category 1. What is the failure frequency used to estimate the change in risk due to the decrease of 8 inspections and from what Table and line is it taken from in EPRI TR-111880?

Energy Northwest Response to Question 8a

In accordance with 3.4.2.5 of EPRI TR-112657 Revision B-A, "EPRI System for Evaluation of Pipe Rupture Potential," the IGSCC degradation mechanism is assigned a "Medium" rupture potential and the FAC degradation mechanism is assigned a "High" pipe rupture potential.

The delta risk evaluation was performed in accordance with the quantitative evaluation guidelines provided in Section 3.7.2 of EPRI TR-112657,"Quantitative Evaluation of Risk Impact." The rupture frequencies used in this evaluation are taken from EPRI TR-111880, Table A-11, "Conditional Failure Rates and Rupture Frequencies for General Electric Plants." These rupture frequencies were applied as follows:

- The estimates assume that the ASME Section XI inspections are 100% effective in eliminating the potential for pipe rupture and no credit is taken for any added inspection volumes or increased probability of detection (POD) that may be associated with inspections performed as part of RI-ISI program. Hence the change in core damage frequency (CDF) is determined only by the increase in CDF for locations removed from the inspection program and not by any enhancements to the inspection program for any of the pipe segments.

- For segments with no assessed damage mechanism, the rupture frequencies per weld associated with design and construction (D&C) errors are used.

- For segments with one degradation mechanism, the rupture frequencies for that mechanism (with the exception of FAC) are summed with the rupture frequencies for design and construction defects that are assumed to occur at any location.

- For segments with two or more degradation mechanisms, the associated rupture frequencies for these mechanisms (with the exception of FAC) are summed with the rupture frequency for design and construction defects. However, when there are two or more damage mechanisms, including IGSCC or FAC, the failure rates and rupture frequencies are increased by a factor of 3 to consider the possible effects of synergy.

Energy Northwest Response to Question 8a (Continued)

From EPRI TR-112657 Section 3.7.2, "Quantitative Evaluation of Risk Impact," the change in CDF at a particular weld can be estimated by equations 3-9 & 3-10.

$$\Delta CDF = (I_r - I_s) \cdot Fo_j \cdot CCDP \qquad \text{(equation 3-9)}$$

$$I_s = 1 - POD_r \qquad \text{(equation 3-10)}$$

$$I_s = 1 - POD_s$$

Where,

I_r and I_s are inspection effectiveness factors (i.e., probability of non-detection) for RI-ISI and Section XI program inspections respectively.

By combining equations, the change in CDF at a weld in terms of POD can be written as:

$$\Delta CDF = (POD_s - POD_r) \cdot Fo_j \cdot CCDP$$

Therefore, the change in CDF in pipe segment "j" is:

$$\Delta CDF_j = (SXIISI_j \cdot POD_s - RIISI_j \cdot POD_r) \cdot Fo_j \cdot CCDP_j$$

Where,

ΔCDF_j = Change in Core Damage Frequency for segment "j"

$CCDP_j$ = Conditional Core Damage Probability for segment "j"

Fo_j = Rupture frequency per weld (w/o FAC) for segment "j"

$SXIISI_j$ = Number of ISI inspections performed in segment "j" under the current ASME Section XI ISI program

$RIISI_j$ = Number of ISI inspections in segment "j" proposed for the RI-ISI program

Energy Northwest Response to Question 8a (Continued)

Since we assume that the NDE performance is the same and 100% effective for both inspection programs (i.e., $POD_s = POD_r = 1$), the change in CDF for pipe segment "j" becomes:

$$\Delta CDF_j = (SXIISI_j - RIIISI_j) \cdot Fo_j \cdot CCDP_j \qquad \text{and} \qquad Fo_j = C_j \cdot (\lambda_{D\&C} + \Sigma\lambda_i)_j$$

Where,

C_j = Synergy Factor

$C_j = 1$ if $i \leq 1$ or $C_j = 3$ if $i \geq 2$ and i = Number of damage mechanisms

$\lambda_{D\&C}$ = Rupture frequency per weld for design and construction defects in segment "j" from TR-111880, Table A-11

λ_i = Rupture frequency per weld for degradation mechanism "i" in segment "j" from TR-111880 Table A-11

With regard to the first RRC entry in Table 3.8-1, this entry consists of a single pipe segment with FAC. The segment has a total of 20 welds. In addition to the FAC program examinations, 8 welds are inspected under the current Section XI program. These Section XI weld inspections are independent of and in addition to the base material wall thinning exams performed under the FAC program.

Segment j	Risk Cat j	CCDPj	CLERP j	DM	SXIISI j	RIISI j	Delta ISI Exams
RCS-39	1	7.18E-4	3.27E-5	FAC	8	0	-8

The rupture frequency for this segment is based on the rupture frequency per weld for design and construction defects associated with the RCS system group in Table A-11 of TR-111880. Since FAC is present, a correction for synergy effects is applied and $C_{39} = 3$.

$$Fo_{39} = C_{39} \cdot (\lambda_{D\&C}) = 3 \cdot (2.23 \times 10^{-6}) = 6.69 \times 10^{-6}$$

$$\Delta CDF = CCDP_{39} \cdot Fo_{39} \cdot (SXIISI_{39} - RIISI_{39})$$

$$\Delta CDF = 7.18 \times 10^{-4} \cdot 6.69 \times 10^{-6} \cdot (8 - 0)$$

$$\Delta CDF = 3.843 \times 10^{-8}$$

NRC Question 8b

The HELB augmented inspection program is not directed toward a specific degradation mechanism. How do you consider/credit HELB in the evaluation? For example, in Table 3.8-1, in the first entry for the MS system, it is stated that there is a reduction of 6 inspections. How many welds are in this segment and how many welds are currently being inspected in this segment under the ISI program? How many welds are currently being inspected in this segment under the HELB program and how many will be inspected under the HELB program after the RI-ISI program is in place? What is the failure frequency used to estimate the change in risk due to the decrease of 6 inspections and from what Table and line is it taken from in EPRI TR-111880? How is the HELB program addressed in the ΔCDF and ΔLERF calculations?

Energy Northwest Response to Question 8b

The HELB augmented inspection program is not discussed in EPRI TR-112657 and is not addressed in the NRC SER on this topic. Therefore, discussion of this program will be removed and our report resubmitted.

NRC Question 8c

In Table 3.8-1, in the second entry for the RRC system, which is susceptible to the IGSCC damage mechanism, there are 14 Section XI Exams and 36 RI-ISI Exams identified, resulting in a delta inspection of +22. How many welds are in this segment, how many welds are currently being inspected in this segment under the IGSCC program, and how many welds are currently inspected in this segment outside the IGSCC program under the current ISI program? Likewise, after the RI-ISI program is in place, how many welds will be inspected in this segment under the IGSCC program and how many welds will be inspected in this segment outside the IGSCC program under the RI-ISI program? What is the failure frequency used to estimate the change in risk due to the increase in inspections and from what Table and line is it taken from in EPRI TR-111880?

Energy Northwest Response to Question 8c

All the current Section XI inspection locations in Class 1 piping susceptible to IGSCC will continue to be inspected as part of the plant's GL 88-01 program. With regard to the second entry of the RRC system in Table 3.8-1, there are 120 welds in these Risk Category 2 segments. Fifty (50) of these welds are currently being examined under the GL 88-01 program. Of these 50 welds, 14 of these exams are also used to satisfy the existing Section XI requirements. These 14 Section XI exams are performed at weld locations that are also inspected under the IGSCC program. None of the current Section XI exams are being performed at locations that are outside the IGSCC inspection scope. Under the RI-ISI program the existing GL 88-01 program is unchanged. Thirty-six (36) of these welds will be credited under the RI-ISI program. The example in 8a above details the use of EPRI TR 111880 to estimate failure frequencies.

NRC Question 9

The submittal states that a conservative quantitative evaluation was performed using the rupture frequencies from Table A-11 in EPRI TR-111880. As stated at the end of Chapter 1 and in the discussion of "Enhancements to the Methodology" section of Chapter 2 of EPRI TR-111880, the data provided in EPRI TR-111880 is only for the pipe sizes that are greater than or equal to two inches. Do any of the evaluated segments have a pipe size less than two inches and, if so, what reference was used to derive the rupture frequencies for these segments?

Energy Northwest Response to Question 9

The SLC system contains two welds that are 1-1/2 inches NPS. These welds were not previously in the ISI population as can be seen in Table 3.8-1. Thus there is no change in qualitative risk impact. See question 5 above for additional information on the basis for not selecting these welds under the proposed RI-ISI.

NRC Question 10

Water hammer is identified as a damage mechanism in Table 3.8-1 for RCIC, but this mechanism is not shown in Table 3.3-1. The submittal does not appear to address any of the loading degradation types (e.g., vibration fatigue, water hammer, overpressure) identified in EPRI TR-111880. Are any of the evaluated segments susceptible to these loading degradation types and, if so, how are these mechanisms addressed or why are they not addressed?

Energy Northwest Response to Question 10

As described in EPRI TR-112657, loading degradation types such as vibration fatigue, water hammer (WH), and overpressure are damage mechanisms that are not amenable to periodic inservice inspection, and for the purpose of the degradation mechanism assessment for implementation of risk-informed inspection programs these mechanisms are identified by service experience.

In accordance with EPRI TR-112657, any segment with an identified degradation mechanism would have a higher failure rate if the pipe segment was also subject to one of the above loads. Thus, a segment with a degradation mechanism classified as producing a moderate break potential would be assigned a high break potential if it is subject to one of the above loads.

A vibration-monitoring program was performed on the plant piping for vibration amplitude as part of the original plant startup and in conjunction with plant power uprate. Those two vibration-monitoring programs found no ASME Class 1 large bore piping that was subject to vibration fatigue. Review of subsequent plant service experience on the Class 1 large bore piping confirmed the absence of any abnormal vibration fatigue loads. Thus, vibration fatigue

Energy Northwest Response to Question 10 (Continued)

Water hammer was identified as a damage mechanism for a portion of the RCIC system. That line also is included in the plant FAC program, thus its failure potential category is high in accordance with the guidelines specified in EPRI TR-112657. Water hammer was not included in the listing of Table 3.3-1 "Degradation Mechanism Assessment Summary" because it is a loading condition rather than a degradation mechanism as specified in the title. This is consistent with the James A. Fitzpatrick Nuclear Power Plant submittal.

NRC Question 11

Section 3.8 states that the evaluation of risk impacts considered the possible effects of synergy between different damage mechanisms for segments found to be susceptible to two or more ISI amenable damage mechanisms. How was this synergy addressed?

Energy Northwest Response to Question 11

Please refer to our response to Question 8a with regard to how we addressed synergy in the delta risk evaluation.

NRC Question 11a

Could consideration of multiple degradation mechanisms result in a higher assigned rupture potential and how were failure frequencies derived for segments with multiple degradation mechanisms? For example, in Table 3.8-1, the third entry for the RHR system includes TASCS, TT, and IGSCC mechanisms in one segment. What is the rationale for assigning a medium rupture potential? What is the failure frequency used to estimate the change in risk due to the decrease of 2 inspections and from what Table and line is it taken from in EPRI TR-111880?

Energy Northwest Response to Question 11a

The third entry for the RHR system in Table 3.8-1 consists of two pipe segments, each with TF (TT & TASCS) and IGSCC degradation mechanisms. A "Medium" rupture potential was assigned to these pipe segments in accordance with Section 3.4.2.5, Table 3-18 of EPRI TR-112657. The rupture frequency calculation and corresponding change in CDF for this entry is shown below. This example demonstrates how various degradation mechanisms and the corresponding rupture frequencies from Table A-11 of TR-111880 are included in the delta risk evaluation.

Segment j	Risk Cat j	CCDPj	CLERP j	DM	SXIISI j	RIISI j	Delta ISI Exams
RHR-15	2	1.59E-4	4.5E-8	TF/IGSCC	3	0	-3
RHR-18	2	6.49E-4	4.5E-8	TF/IGSCC	1	2	+1
				TOTAL	4	2	-2

$$Fo_{15} = C_{15} \cdot (\lambda_{D\&C} + \lambda_{TF} + \lambda_{IGSCC}) = 3 \cdot \left(8.44 \times 10^{-8} + 4.41 \times 10^{-8} + 7.76 \times 10^{-6}\right) = 2.37 \times 10^{-5}$$

$$Fo_{18} = C_{18} \cdot (\lambda_{D\&C} + \lambda_{TF} + \lambda_{IGSCC}) = 3 \cdot \left(8.44 \times 10^{-8} + 4.41 \times 10^{-8} + 7.76 \times 10^{-6}\right) = 2.37 \times 10^{-5}$$

$$\Delta CDF = \Delta CDF_{15} + \Delta CDF_{18}$$

$$\Delta CDF = CCDP_{15} \cdot Fo_{15} \cdot (SXIISI_{15} - RIISI_{15}) + CCDP_{18} \cdot Fo_{18} \cdot (SXIISI_{18} - RIISI_{18})$$

$$\Delta CDF = \left[1.59 \times 10^{-4} \cdot 2.37 \times 10^{-5} \cdot 3\right] + \left[6.49 \times 10^{-4} \cdot 2.37 \times 10^{-5} \cdot (-1)\right]$$

$$\Delta CDF = -4.1 \times 10^{-9}$$

NRC Question 11b

Were design and construction defects, which are stated in the "Modeling Approach to Estimation of Pipe Rupture Frequencies" section of Chapter 2 of EPRI TR-111880 as being amenable to ISI, considered synergistically with other damage mechanisms? Why were loading degradation types that are not amenable to ISI not considered when combined with damage mechanisms that are ISI amenable? For example, in Table 3.8-1, the first two entries include FAC and WH mechanisms. What is the failure frequency used to estimate the change in risk due to the decrease of all 18 inspections (11 from the high segment and 7 from the medium segment) and from what Table and line is it taken from in EPRI TR-111880?

Energy Northwest Response to Question 11b

The contribution to rupture frequencies from design and construction (D&C) defects was included in all pipe segments. Loading degradation mechanisms that are not amenable to ISI were considered and combined with degradation mechanisms amenable to ISI in accordance with EPRI TR-112657 as described in our response to question 10.

With regard to the first two RCIC system segment entries in Table 3.8-1 of the submittal, the degradation and loading mechanisms assigned to these segments were FAC and WH. Since no changes are made to the current FAC inspection program, and since no other degradation mechanisms were identified, the risk change evaluation for these pipe segments is concerned with the impact that the reduction in ISI inspections have on the rupture frequency due to D&C defects. Since FAC is present in these lines, it is assumed that there exists a synergistic effect from FAC on the rupture potential from D&C. Therefore, for each of these pipe segments, the D&C rupture frequency per weld was obtained from the "SIR" system group column in Table A-11 (8.44E-8 per yr-weld) and increased by a factor of 3 to obtain a rupture frequency corrected for synergy of 2.53E-7 per yr-weld.

NRC Question 12

Section 3.3 of EPRI TR-112657 Rev. B-A requires the consideration of external events (e.g., seismic events) and operation modes outside the scope of the PRA (e.g., shutdown) in the categorization of segments. Were external events and operation modes outside the scope of the PRA systematically considered and was the plant multidisciplinary review team involved in this evaluation?

Energy Northwest Response to Question 12

We have conducted a qualitative assessment on consequence evaluation for shutdown operations and a semi-quantitative assessment for external events. The objective was to determine whether a higher consequence should be assigned and to gain a level of confidence that the consequence ranking during shutdown operation and external events would not be more limiting than the consequence ranking during at-power operation.

Our Individual Plant Examination for External Events was used to determine whether fire and/or seismic events could affect the consequence ranking. Because consequence segments that were ranked as "High" during at-power operation covered the resulting consequence during shutdown operation and external events, only consequence segments that were ranked as "Medium or "Low" during at-power operation were reexamined. The studies showed that fire and seismic events would not impact the consequence ranking. Likewise, the studies showed that there was no impact to the consequence ranking when the shutdown mode was considered. In summary, the consequence assessment (categorization) documented in the RI-ISI program include the potential impacts during at-power and shutdown operations and external events (fires and seismic). These conclusions will be included in the revised RI-ISI program submittal.

In accordance with the guidance provided in Section 3.6.5 of EPRI TR-112657, a multi-discipline review team was established to support the RI-ISI application. This team consisted of representatives from the groups that performed the consequence and degradation analysis and senior plant personnel knowledgeable in the following disciplines:

- ISI program requirements and implementation
- Piping system design and operation
- PSA
- Piping/material degradation mechanisms
- Piping stress analysis and fracture mechanics
- Nuclear industry service experience
- Non-destructive examination
- Health physics
- Scaffolding, insulation, and craft requirements
- EPRI RI-ISI Methodology

Energy Northwest Response to Question 12 (Continued)

The objectives of this team were to make sure that all relevant factors had been considered in the consequence assessment, degradation mechanism assessment, risk-ranking processes, and to address any comments resulting from the review. One additional objective was to select elements for inspection and to focus the selection on those elements that are most likely to experience inservice problems.

The role of the multi-discipline review team was expanded and selected members of the team were involved in the review of all steps of the RI-ISI application (i.e., failure modes and effects analysis, consequence, degradation mechanism, and risk impact assessments).

NRC Question 13

In accordance with EPRI TR-112657 Rev. B-A Section 3.6.5.1, does the RI-ISI program include inspections for all the degradation mechanism types identified?

Energy Northwest Response to Question 13

Yes, each high risk RI-ISI inspection location will be inspected for all degradation mechanism types existing at that location.