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10 CFR 50.4  
10 CFR 50.55a(a)(3)(i)

February 19, 2001

PSLTR: #01-0023

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

Dresden Nuclear Power Station, Units 2 and 3  
Facility Operating License Nos. DPR-19 and DPR-25

NRC Docket Nos. 50-237 and 50-249

Subject: Request for Additional Information Regarding Risk-Informed Inservice  
Inspection Program Relief Request

Reference: (1) Letter from P. Swafford (EGC) to U. S. NRC, "Alternative to the ASME  
Boiler and Pressure Vessel Code Section XI Requirements for Class 1  
and 2 Piping Welds Risk Informed Inservice Inspection Program," dated  
October 18, 2000

(2) Letter from U. S. NRC to O. D. Kingsley (EGC), "Dresden Units 2 and 3 –  
Request for Additional Information Regarding Risk-Informed Inservice  
Inspection Program Relief Request," dated January 18, 2001

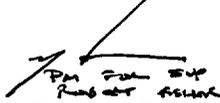
The purpose of this letter is to provide the Dresden Nuclear Power Station (DNPS) response to the Request for Additional Information (RAI), Reference 2, regarding our submittal of Relief Request CR-21 to implement risk informed changes to the inservice inspection (ISI) program provided in Reference 1. The Attachments to this letter contains our response to the RAI.

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Should you have any additional questions regarding this letter, please contact Mr. Dale Ambler, Regulatory Assurance Manager, at (815) 942-2920 extension 3800.

Respectfully,

A handwritten signature in black ink, appearing to read "Preston Swafford", with a horizontal line drawn underneath it.

Preston Swafford  
Site Vice President  
Dresden Nuclear Power Station

- Attachment 1            Response to Request for Additional Information, Alternative to the ASME Boiler and Pressure Vessel Code Section XI Requirements for Class 1 and 2 Piping Welds Risk Informed Inservice Inspection Program
- Attachment 2            Excerpted from Dresden Tier 2 RI-ISI Documentation, Risk Impact of Implementing Risk Informed Inspection Program

cc:    Regional Administrator, Region III  
      NRC Senior Resident Inspector, Dresden Nuclear Power Station

**Attachment 1**

**Response to Request for Additional Information  
Alternative to the ASME Boiler and Pressure Vessel Code Section XI Requirements  
for Class 1 and 2 Piping Welds Risk Informed Inservice Inspection Program**

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?

Exelon Generation Company (EGC) Response:

Columns (a), (b), and (c) in the following table provide a response to each question respectively.

Unit	(a) Third Inspection Interval	(b) Third Inspection Period	(c) <sup>1</sup> Third Inspection Interval Inspections				
			B-F	B-J	C-F-1	C-F-2	Total
Unit 2	3/1/92 to 1/19/03	10/1/99 to 1/19/03	82%	69%	64%	60%	69%
Unit 3	3/1/92 to 10/31/02	11/1/99 to 10/31/02	65%	67%	64%	57%	65%

Notes:

- 1. Cumulative percentage of Code inspections completed for the third inspection interval under the current ASME Section XI ISI program.

NRC Question #2:

The implementation of a Risk Informed Inservice Inspection (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 American Society of Mechanical Engineers (ASME) Code Section XI, Edition and Addenda committed to by the Owner in accordance with 10 CFR Part 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 inspection intervals under Program B should be 16 percent, 50 percent, and 100 percent, respectively, and the maximum examinations credited at the end of the respective periods should be 34 percent, 67 percent, and 100 percent.

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

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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).

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.

#### EGC Response:

Dresden Nuclear Power Station (DNPS) has elected to implement the RI-ISI program during the third inspection period of the third 10-year inservice inspection interval. RI-ISIs and the balance of inspections will remain on the same 10-year inservice interval so that a consistent code of record committed to in accordance with 10 CFR Part 50.55a, "Codes and standards," will be applied. The code required percentages for the first and second inspection periods have already been satisfied by the completion of code required inspections under the current American Society of Mechanical Engineers (ASME) Section XI ISI program. The required percentages for the third inspection period will be satisfied with either the RI-ISIs or the code required inspections. This approach will result in 100% of the RI-ISIs being completed over the third inspection interval by either code required inspections under the current ASME Section XI ISI program or by the RI-ISIs with the minimum and maximum examinations being credited consistent with ASME XI requirements. The following table provides the current distribution of RI-ISI inspections during the third 10-year inservice inspection interval.

Unit	Total	Current ASME Section XI ISI Program			RI-ISIs
		Period 1	Period 2	Period 3	
Unit 2	95	18 (19%)	39 (60%)	8 (65%)	30 (100%)
Unit 3	94	27 (29%)	34 (65%)	0	33 (100%)

#### NRC Question #3:

Will the RI-ISI program be updated every 10 years and submitted to the Nuclear Regulatory Commission (NRC) consistent with the Current ASME XI requirements?

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#### EGC Response:

The RI-ISI program is an alternative to the requirements of ASME Section XI requirements for Class 1 and 2 piping welds implemented through the use of a relief request in accordance with 10 CFR 50.55a(a)(3)(i). Therefore, a relief request for implementation of a RI-ISI program during subsequent 10-year inservice inspection intervals will be submitted concurrent with the update to the latest edition and addenda of the Code every ten years in accordance with 10 CFR 50.55a(g)(4)(ii).

#### NRC Question #4:

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

#### EGC Response:

It is not our intent to resubmit the RI-ISI program to the NRC before the end of a 10-year interval. The RI-ISI program will be maintained as a living program and updated consistent with EPRI TR 112657, "Revised Risk-Informed Inservice Inspection Evaluation Procedure."

Changes that could impact the RI-ISI program include major changes to the Dresden PRA or changes to weld selection. Our Risk Management program requires a review of past applications following a PRA update. This requirement will be applied to the RI-ISI program. If the review determines that a change to the RI-ISI program is required, the change would be performed consistent with the EPRI methodology. Likewise, a change to the welds selected would cause a revision to the RI-ISI program consistent with the EPRI methodology. These changes to the RI-ISI program would not be resubmitted to the NRC.

It should be noted that requirements for RI-ISI program maintenance are being developed by EPRI. The EPRI "Living Program Criteria" document is expected to be published by the end of 2001.

#### NRC Question #5:

Section 3.5, page 9 of 25 states that longitudinal welds are considered subsumed with examination of the associated circumferential weld when the circumferential weld is selected for RI-ISI examination as per Code Case N-524. However, Section 3.6, page 11 of 25 states that Code Case N-524 will be removed from the ISI plan upon approval of the relief request. Please clarify your position regarding Code Case N-524.

#### EGC Response:

Code Case N-524 will no longer be directly applicable to the inspection of Class 1 and 2 welds, and therefore it will be removed from the ISI plan upon approval of the relief request. Code Case N-524 is approved as an alternative to Section XI, Examination Categories B-J, C-F-1, and C-F-2. Upon approval of the RI-ISI relief request, the requirements of

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Examination Categories B-J, C-F-1, and C-F-2 will no longer be applicable to Class 1 and 2 welds; and therefore, Code Case N-524 will no longer be directly applicable. However, DNPS believes that the alternative requirements of the Code Case are still valid under the risk-informed inspection program. To this extent and in accordance with footnote (4) of Code Case N-578-1, Table 1, Examination Category R-A, DNPS will examine those longitudinal welds that intersect the circumferential welds selected under the risk-informed process. For those longitudinal welds intersecting circumferential welds, the portion of the weld within the associated circumferential weld volume will be inspected, and the inspection requirements for the longitudinal weld will be met for both transverse and parallel flaws.

#### NRC Question #6:

Section 3.5, page 9 of 25 states that 13.3 percent of Class 1, butt welded elements, were selected for volumetric examination at Unit 2. This section also states that 5.5 percent of socket welded elements were selected for VT-2 examination. For Unit 2, please specify if any of the socket welded elements are included in the 13.3 percent sample. The corresponding number for Unit 3 are 12.0 percent and 6.7 percent. For Unit 3, please specify if any of the socket welded elements are included in the 12.0 percent sample. The staff has concluded that at least 10 percent of butt welded elements need to be selected for examination to assure adequate safety margins and defense in depth.

#### EGC Response:

For both DNPS Units 2 and 3, the butt welded element and socket welded element populations are mutually exclusive. The calculation of the percent of Class 1 butt welded elements does not include IGSCC and FAC-only welds. The IGSCC and FAC-only welds are removed from the RI-ISI population for element selection (no RI-ISI inspections are selected for these welds). To include them in the population of Class 1 butt welded elements would make the percentage artificially low. The following paragraphs provide a summary of each population and the percentage selected.

The 13.2% (Note: The original RI-ISI relief request submittal stated 13.3%, the actual percentage is 13.2%) of butt welded Class 1 elements selected for inspection at DNPS Unit 2 under the RI-ISI program results from selecting 58 class 1 butt welds out of a RI-ISI population of 438 Class 1 butt welded elements. Socket welds are not included in this number. The 5.5% of socket welds selected at DNPS Unit 2 results from selecting 13 Class 1 socket welds out of a RI-ISI population of 234 Class 1 socket welds. Butt welds are not included in the calculation for the socket weld percentage.

The 12.0% of butt welded Class 1 elements selected for inspection at DNPS Unit 3 under the RI-ISI program results from selecting 54 class 1 butt welds out of a RI-ISI population of 452 Class 1 butt welded elements. Socket welds are not included in this number. The 6.7% of socket welds selected at DNPS Unit 3 results from selecting 17 Class 1 socket welds out of a RI-ISI population of 253 Class 1 socket welds. Butt welds are not included in the calculation for the socket weld percentage.

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#### NRC Question #7:

Page 5 states that, "If no other damage mechanism was identified, the element was removed from the RI-ISI element selection population and retained in the appropriate augmented program."

- a) How many Class 1, Intergranular Stress Corrosion Cracking (IGSCC) Category B through G welds does Dresden Have? How many Class 2, IGSCC Category B through G welds does Dresden have? Have all these welds been "removed from the RI-ISI element selection population?" Are any of these inspections credited as an inspection in the RI-ISI program?
- b) Our understanding of your terminology is that a flow accelerated corrosion (FAC) element is a run of pipe that may contain one or more welds within the element or at the boundaries. Is the entire length of an element in your FAC program always inspected? If there are no other degradation mechanisms in this FAC element, is the population of welds within the element and/or at the boundary of the element "removed from the RI-ISI element selection program?" If there are any welds within and/or at the boundary of this element that are currently being inspected under the Section XI program, what happens to these inspections under RI-ISI program and how are they included in the change in risk calculations?

Does the reported 13 percent and 12 percent of Class 1, butt welded elements inspected include the population of IGSCC Category B through G welds, and the FAC element welds, in the denominator?

#### EGC Response:

- a) DNPS Unit 2 has a total of 175 Class 1 and 26 Class 2 IGSCC Category B through G welds. From the 175 Class 1 welds, 134 welds were removed from the RI-ISI element selection population since no other damage mechanism was identified, and 5 welds are categorized as low risk welds and removed from the RI-ISI element selection population. The remaining 36 Class 1 IGSCC Category B through G welds are included in the RI-ISI element selection population. Of the 36 Class 1 welds remaining in the RI-ISI element selection population, 12 welds are selected under the RI-ISI program, and therefore are credited in both the RI-ISI and IGSCC programs. When inspections are credited under the RI-ISI and IGSCC programs, all inspection requirements for both programs are met. All 26 Class 2 IGSCC Category B through G welds were removed from the RI-ISI element selection population since no other damage mechanism was identified. The Class 1 and 2 welds removed from the RI-ISI program continue to be addressed by the IGSCC program.

DNPS Unit 3 has 35 Class 1 IGSCC welds and 28 Class 2 IGSCC welds. These welds were removed from the population for RI-ISI element selection since no other damage mechanism was identified. Therefore, there is no credit taken for any exams done on

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these welds in the RI-ISI program. These welds continue to be addressed by the IGSCC program.

- b) FAC elements which have no other degradation mechanism are modeled and inspected in accordance with the FAC program. Inspection locations within a FAC element are selected in accordance with the FAC program. The extent of examination for selected inspection points is in accordance with Section 4.7, "Flow Accelerated Corrosion" of the EPRI RI-ISI Topical Report (EPRI TR 112657). Welds identified as having FAC as the only degradation mechanism are removed from the RI-ISI population for element selection. FAC-only welds currently inspected under Section XI will not be selected for inspection under the RI-ISI program, but will continue to be addressed by the FAC program. The FAC-only and IGSCC welds that are not selected for the RI-ISI program are all included in the delta risk calculations. Those examinations eliminated at any of these welds would result in a slight increase in risk for those specific welds and contribute to the overall delta risk that was quantified for the system.

The denominator for the calculation of the percent of Class 1 butt welded elements does not include IGSCC and FAC-only welds. The IGSCC and FAC-only welds are removed from the RI-ISI population for element selection since no RI-ISI inspections are selected for these welds; to include them in the denominator when they are excluded from the numerator would make the percentage artificially low.

#### NRC Question #8:

It is acceptable to credit a weld inspected in the current IGSCC Category A-G program as a RI-ISI program inspection (within certain percentage limits). If a weld that is currently inspected in the IGSCC A-G program but not credited as a Section XI inspection is credited in the RI-ISI program, how is this weld treated in the change in risk estimates? If a weld is currently inspected in the IGSCC A (only) program but not credited as a Section XI inspection is not credited in the RI-ISI program (e.g., the inspection will be discontinued), how is this weld treated in the change in risk estimates?

#### EGC Response:

As described in the response to 7a, the IGSCC Category B through G welds that have no other degradation mechanisms identified are all removed from the RI-ISI population for element selection. These IGSCC only Category B through G welds that are currently inspected under Section XI will not be selected for inspection under RI-ISI. For the delta risk calculations, they are treated the same as the FAC-only welds, as described in the response to 7b. In the RI-ISI evaluation IGSCC Category A welds are not regarded as susceptible to IGSCC and are retained in the RI-ISI population for possible element selection or exclusion (i.e., for low risk category welds) from current examinations. If such welds were being inspected as part of the IGSCC program for Category A and were eliminated from the RI-ISI population, this reduction would have been evaluated for delta risk using the same inputs for other RI-ISI welds that are not susceptible for IGSCC. For example if such a weld were

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found to be susceptible to no ISI amenable damage mechanism, the failure rates and rupture frequencies for design and construction errors would be used.

#### NRC Question #9:

Table 3 on Page 20, lists 324 high risk and 494 medium risk welds. The Electric Power Research Institute (EPRI) methodology calls for inspecting twenty-five percent of high risk and ten percent of medium risk welds. Twenty-five percent of 324 is 81 and ten percent of 494 is 49. Yet Table 5 on page 22, only identifies 42 high risk and 53 medium risk inspection locations? Please clarify this apparent discrepancy. There is a similar apparent discrepancy between Tables 4 and 6 for Unit 2.

#### EGC Response:

Tables 3 and 4 include all the welds in the initial scope of welds assessed by the RI-ISI analysis. Welds having only IGSCC or only FAC as a degradation mechanism are removed from the population for element selection prior to applying the 25% and 10% sampling percentages. DNPS Unit 2 has 42 FAC welds that are High risk, 123 IGSCC welds that are High risk, and 37 IGSCC welds that are Medium risk. That leaves 159 High risk welds and 457 Medium risk welds for element selection under the RI-ISI program. 25% of 159 is 39.75 welds and 42 are selected for inspection. 10% of 457 is 45.7 welds and 53 are selected for inspection. The reason why the number selected is somewhat greater than that directly obtained from these percentages is that the percentages were applied for each EPRI risk matrix category. So if there were welds in both EPRI Risk Category 2 and Category 3 for a given system, the 25% sampling percentage was applied within each Category and the resulting number was rounded up each time. Similarly if there were welds in Categories 4 and 5, the 10% sampling percentage was applied within each Category and the resulting number was rounded up each time. Hence the combined number of exams in the High and Medium Risk Categories was significantly greater than the number that would have been obtained if the broader categories were combined before applying the percentages. We used this conservative application of percentages to provide a margin for error to avoid the need for too many iterations in the element selection/evaluation process.

DNPS 3 also has 42 FAC welds in the High risk category and 8 High risk IGSCC welds and 39 Medium risk IGSCC welds. This leaves 135 High risk welds and 519 Medium risk welds for element selection under RI-ISI. 25% of 135 is 33.75 welds and 36 are selected. 10% of 519 is 51.9 and 58 are selected for inspection. The same procedure of applying the sampling percentages was applied as with Unit 2.

#### NRC Question #10:

Page 6 states that "The potential for synergy between two or more damage mechanisms working on the same location was considered in the estimation of pipe failure rates and rupture frequencies which was reflected in the risk impact assessment." Specifically, how

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was this synergy reflected in the risk impact? Was synergy also reflected in the safety significant categorization and if so how?

EGC Response:

The delta risk assessment for the DNPS Units 2 and 3 RI-ISI evaluations were performed by ERIN Engineering and Research, who co-authored the EPRI RI-ISI Topical Report, EPRI TR-112657, "Revised Risk-Informed Inservice Inspection Evaluation Procedure." They also were the lead author of the supporting reports that developed failure rates and rupture frequencies and the Markov Model to delta risk evaluations in the EPRI method (EPRI TR-111880 and EPRI TR-110161, respectively). Neither the EPRI RI-ISI procedure described in the Topical Report, nor the supporting analysis of failure rates and rupture frequencies performed in EPRI TR-11880, nor any other source of failure rates that we are aware of addresses the situation in which a segment is found to be susceptible to two or more damage mechanisms. The following describes how failure rates and rupture frequencies were impacted by synergy for the conservative assumptions in the delta risk evaluation.

The failure rates and 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 are a function of (conditioned on) the system and combination of damage mechanisms identified for that segment and do not take credit for any pipe inspections. These failure rates and rupture frequencies were applied as follows.

- Conditional core damage probabilities (CCDPs) and conditional large early release probabilities (CLERPs) from the consequence analysis and application of the existing plant specific PRA models are used for all of the delta risk evaluations. Separate calculations were performed for delta CDF and delta LERF for each pipe location in the scope of the RI-ISI evaluation.
- For segments with no assessed damage mechanism, the failure rates and rupture frequencies associated with design and construction errors for the appropriate system category are used.
- For segments with one and only one ISI amenable damage mechanism, the failure rates and rupture frequencies for that mechanism were summed with the rates and frequencies for design and construction errors which could occur at any location. The exception is when the associated damage mechanism is IGSCC or FAC and these mechanisms are covered in an augmented inspection program that is not being changed in the RI-ISI program. Only those mechanisms associated with a change to the inspection in the RI-ISI program are considered. Note that for consistency with the treatment of damage mechanisms in EPRI TR 111880 which used Thermal Fatigue as a general category to include both Thermal Transient (TT) and Thermal Stratification Cycling and Striping (TASCS), these two mechanisms occurring singly or in combination were simply regarded as susceptible to Thermal Fatigue. Hence no synergy between TT and TASCS was assumed.
- For segments with two or more ISI amenable damage mechanisms, the associated failure rates and rupture frequencies for these and design and construction errors are

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summed, with the exception that IGSCC and FAC contributions are not added if the weld is part of the associated augmented inspection program for IGSCC or FAC. These contributions were not added as the associated augmented inspection programs will not change. Only those damage mechanisms whose inspection programs are changed in the RI-ISI program were included. However, when there are two or more damage mechanisms, including IGSCC or FAC, the failure rates and rupture frequencies for the applicable ISI amenable damage mechanisms are increased by a factor of 3 to consider the possible effects of synergy, i.e., to consider the potential that through wall cracks would occur more quickly when two or more mechanisms were present at the same location.

The above treatment was used because the service data upon which the EPRI methodology for damage mechanism assessment was based does not explicitly address multiple damage mechanisms. Two examples serve to better explain the procedure that was followed. If a segment was found to be susceptible to both thermal fatigue (i.e., TT, TASCs or both) and corrosion cracking and the corrosion cracking is not covered in the augmented program for IGSCC (hypothetical case), the failure rates for design and construction errors, thermal fatigue, and stress corrosion cracking from EPRI TR-111880 would be summed and then this result would be multiplied by a factor of 3 for synergy. The rupture frequencies would be determined in the same way: The appropriate contribution to rupture frequency summed and the result multiplied times 3. But if the segment was found susceptible to the same three damage mechanisms and the stress corrosion cracking was covered in the augmented IGSCC program, the stress corrosion cracking contribution would not be included in the failure rate or rupture frequency, but its synergy effects with thermal fatigue would be included by application of the factor of 3.

While the potential for synergy was considered using engineering judgement in the delta risk evaluation as explained above, the assignment of failure potential categories in the application of the EPRI RI-ISI risk matrix was not changed as a result of this consideration of synergy. This judgement was based on insights developed by our contractors in estimating failure rates and rupture frequencies for many different damage mechanisms and system categories in preparation of EPRI TR-111880. Hence if a location was susceptible to say two or more ISI amenable damage mechanism other than FAC, the failure potential category was not increased from Medium to High due to consideration of synergy. The judgement of our contractor team was that a factor of 3 increase in rupture frequency would provide a conservative upper bound on the possible effects of synergy. The assumption in the risk classification matrix in the EPRI methodology was that the difference in frequency between Medium and High failure potential was more than an order of magnitude. In summary, our approach to treatment of synergy effects from two or more damage mechanisms was thought to be both reasonable and beyond the requirements set forth in Regulatory Guide 1.174, Regulatory Guide 1.178, and the EPRI RI-ISI Topical Report.

NRC Question #11:

Please provide references to all the equations that you are using to calculate the change in risk. Please also provide references from which all the input parameters required by the

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equations were developed and justified (except for the conditional core damage and condition large early release probabilities). Please provide specific references, e.g. equation numbers, table numbers, page numbers, and report references.

EGC Response:

The requested information on equations and data sources is provided in the Table below.

<b>Model/Equation</b>	<b>Report Reference</b>	<b>Page, Table, Equation References</b>
Equations for calculating changes in CDF and LERF	EPRI TR-112657	Equation 3-9 on p. 3-86
Equation for calculating CDF and LERF	EPRI TR-110161	Equation 3.40 on p. 3-34
Markov Model used for ISI amenable damage mechanisms	EPRI TR-110161	Figure 3-9 on p. 3-24 Equations 3.26 through 3.38 on pp. 3-24 to 3-27
Definition of inspection effectiveness factor for use in delta risk equation	EPRI TR-110161	$I = \frac{h_{40} \{ \omega_{NEW} \}}{h_{40} \{ \omega_{OLD} \}}$ <p>This is similar to Equation 3.41 on p. 3-37 except that 40 year vs. steady state hazard rates are used. NEW corresponds with RI-ISI and OLD with ASME Sec. XI.</p>
Definition of the flaw inspection repair rate $\omega$	EPRI TR-110161	Equation (3.23) on p. 3-18
Definition of the leak detection repair rate $\mu$	EPRI TR-110161	Equation (3.24) on p. 3-18
Failure rates and rupture frequencies	EPRI TR-111880	Table A-11
Plant specific documentation of all other input data needed to quantify above equations	DNPS Units 2 and 3 RI-ISI Evaluation (Tier 2 Documentation)	Section 7

The justification for all the input parameters used in application of the Markov model to each system in the scope of the RI-ISI for DNPS Units 2 and 3 is provided in Attachment 2, "Excerpted from Dresden Tier 2 RI-ISI Documentation, Risk Impact of Implementing Risk Informed Inspection Program."

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#### NRC Question #12:

It is our understanding that you are calculating an "inspection effectiveness factor" for use in equation 3-9 of EPRI Topical Report (TR) 112657. Please provide the distribution of inspection effectiveness values calculated (clearly identifying the upper and lower bounds) and a discussion on how these values compare with other probability of detection estimates (redefined to the same format).

#### EGC Nuclear Response:

The inspection effectiveness factor is the ratio of the inspected weld rupture frequency to the non-inspected rupture frequency. The EPRI Topical Report in Section 3.7.2 discusses two methods for determining these factors, one based on an application of the Markov model and the other based on an assumption that the factor is proportional to the complement of the probability of detection of the ISI exam, or POD. The POD is the conditional probability of detection of damage in a pipe element, given the existence of a detectable flaw or crack in the pipe element that exceeds the pipe repair criteria. When the effectiveness factor is developed from the Markov model, the following variables impact its numerical value: the POD which may be different whether the exam is done per ASME Section XI or per EPRI RI-ISI examination criteria, the assumed failure rates and rupture frequencies which are taken to be dependent and conditional on the system, pipe size, and applicable ISI amenable damage mechanisms. There are other inputs to the Markov model that are not varied between EPRI and ASME Section XI programs that describe the frequency and effectiveness of pipe leaks when leak before break applies.

A tabulation of all the unique inspection effectiveness factors for all pipe segments evaluated within the scope of the RI-ISI evaluation for DNPS Units 2 and 3 is presented in Table RAI 12-A. For comparison purposes, the corresponding POD values that were used were presented along with their complements that provide the alternative method of computing the inspection effectiveness factor. A plot that compares the two approaches to computing the inspection effectiveness factors is provided in Figure RAI 12-A for the RI-ISI exams.

As seen in these exhibits, there is fairly good agreement between these alternative approaches to estimating the inspection effectiveness factors. When the POD values are around 0.50, the Markov model predicts a somewhat higher level of inspection effectiveness, as reflected in somewhat lower inspection effectiveness factors. For higher POD values, the Markov model predicts a somewhat lower level of inspection effectiveness, as reflected in somewhat higher inspection effectiveness factors. Details documenting the inputs to computing these factors are discussed in response to RAI 11.

The inspection effectiveness factors developed using the Markov model are viewed as a more realistic assessment of inspection effectiveness for several reasons, including:

- The use of the (1-POD) model for inspection effectiveness is simply an assumption and has no real logical or scientific basis, whereas

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### **Response to Request for Additional Information Alternative to the ASME Boiler and Pressure Vessel Code Section XI Requirements for Class 1 and 2 Piping Welds Risk Informed Inservice Inspection Program**

- The Markov model is based on an explicit model of the interactions between degradation phenomena and inspection processes. The results of the Markov model are a function of the POD as well as many other parameters that account for the relative frequency of cracks, leaks, and ruptures, the possibility for leak before break and leak detection and repair prior to rupture, the fraction of the weld that is accessible, the possibility for synergy between different damage mechanisms, the time intervals between inspections, and other factors.

Having stated this, it is noted that in the context of developing order of magnitude estimates of risk impacts, both methods provide comparable results as seen in the presented exhibits.

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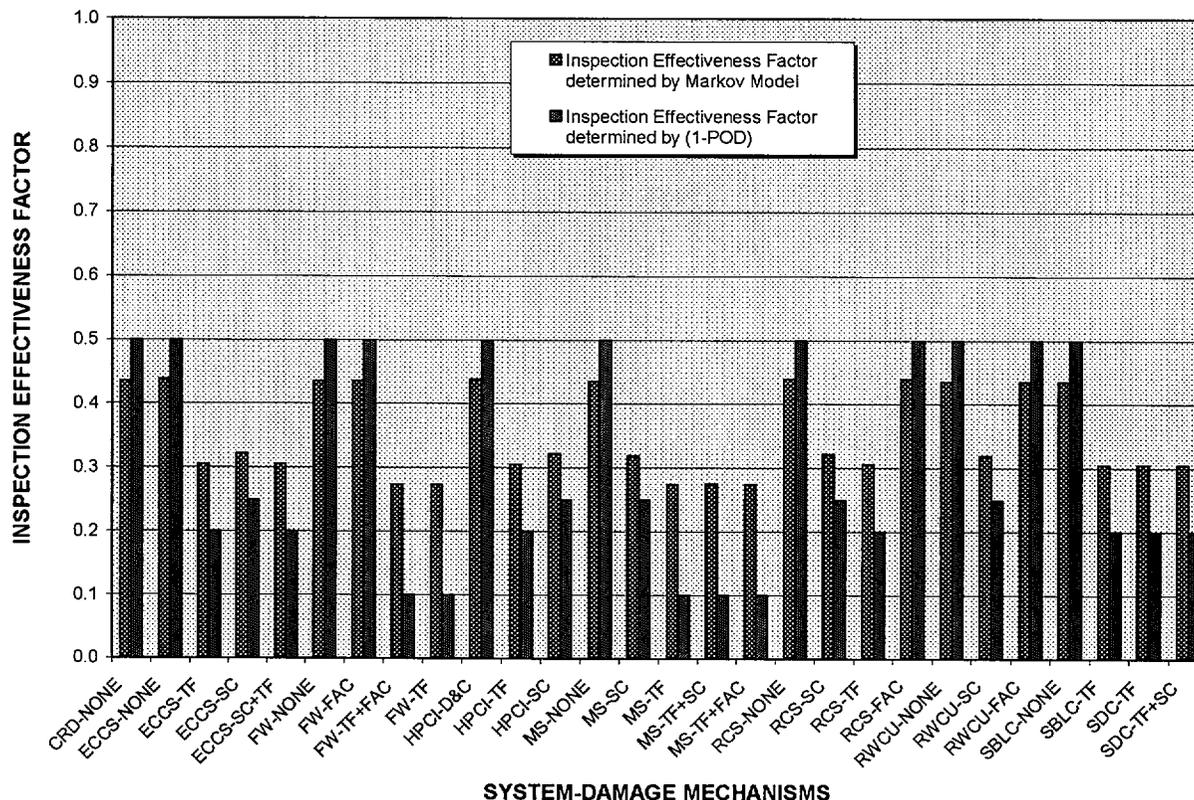
**Table RAI 12-A  
Probability of Detection (POD) and Inspection Effectiveness Factors  
Used for DNPS Units 2 and 3 Delta Risk Evaluations**

System	Damage Mechanism(s)	EPRI RI-ISI Exams			ASME Section XI Exams		
		POD	Inspection Effectiveness Factor per Markov Model	Inspection Effectiveness Factor per (1-POD)	POD	Inspection Effectiveness Factor per Markov Model	Inspection Effectiveness Factor per (1-POD)
CRD	D&C <sup>1</sup>	0.500	0.435	0.500	0.500	0.435	0.500
ECCS	D&C <sup>1</sup>	0.500	0.438	0.500	0.500	0.438	0.500
	TASCS	0.800	0.305	0.200	0.500	0.438	0.500
	TT	0.800	0.305	0.200	0.500	0.438	0.500
	IGSCC	0.750	0.322	0.250	0.500	0.438	0.500
	TASCS, TT	0.800	0.305	0.200	0.500	0.438	0.500
	TASCS, TT, IGSCC	0.800	0.305	0.200	0.500	0.438	0.500
	TT, IGSCC	0.800	0.305	0.200	0.500	0.438	0.500
FW	D&C <sup>1</sup>	0.500	0.435	0.500	0.500	0.435	0.500
	FAC	0.500	0.435	0.500	0.500	0.435	0.500
	TASCS, FAC	0.900	0.273	0.100	0.500	0.436	0.500
	TASCS, TT	0.900	0.273	0.100	0.500	0.436	0.500
	TASCS, TT, FAC	0.900	0.273	0.100	0.500	0.436	0.500
	TT, FAC	0.900	0.273	0.100	0.500	0.436	0.500
HPCI	D&C <sup>1</sup>	0.500	0.438	0.500	0.500	0.438	0.500
	TT	0.800	0.305	0.200	0.500	0.438	0.500
	IGSCC	0.750	0.322	0.250	0.500	0.438	0.500
MS	D&C <sup>1</sup>	0.500	0.435	0.500	0.500	0.435	0.500
	IGSCC	0.750	0.319	0.250	0.500	0.435	0.500
	TASCS	0.900	0.274	0.100	0.500	0.437	0.500
	TT, TASCS	0.900	0.272	0.100	0.500	0.435	0.500
	TASCS, IGSCC	0.900	0.275	0.100	0.500	0.437	0.500
	TASCS, TT, FAC	0.900	0.274	0.100	0.500	0.437	0.500
RCS	D&C <sup>1</sup>	0.500	0.439	0.500	0.500	0.439	0.500
	IGSCC	0.750	0.322	0.250	0.500	0.439	0.500
	TASCS	0.800	0.306	0.200	0.500	0.439	0.500
RWCU	D&C <sup>1</sup>	0.500	0.435	0.500	0.500	0.435	0.500
	IGSCC	0.750	0.319	0.250	0.500	0.435	0.500
	FAC	0.500	0.435	0.500	0.500	0.435	0.500
SBLC	D&C <sup>1</sup>	0.500	0.435	0.500	0.500	0.435	0.500
	TASCS	0.800	0.305	0.200	0.500	0.438	0.500
SDC	TT	0.800	0.305	0.200	0.500	0.438	0.500
	TT, IGSCC	0.800	0.305	0.200	0.500	0.438	0.500
	TT, TASCS	0.800	0.305	0.200	0.500	0.438	0.500
	TASCS, TT, IGSCC	0.800	0.305	0.200	0.500	0.438	0.500

1) Design and construction errors were included for all welds and are shown here only for cases with no other damage mechanism present.

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**Figure RAI 12-A Comparison of Inspection Effectiveness Factors for  
EPRI RI-ISI Exams at DNPS Units 2 and 3**

NRC Question #13:

If results from the bounding evaluations described in the EPRI TR, instead of the Markov calculations are sufficient to illustrate that the suggested change in risk guidelines are not exceeded, you may provide a brief description of these evaluations and the results instead of the information requested in questions 11 and 12.

EGC Response:

A simplified and conservative risk impact calculation, not using the Markov model calculation of pipe break frequency, was performed for DNPS Units 2 and 3. This calculation was performed using the same approach as was implemented for the previously approved relief request for South Texas Project which was performed by ERIN. This was documented in a letter from the U.S. NRC to STP Nuclear Operation Company, "South Texas Project, Units 1 and 2 – Request for Relief from ASME Code Requirements for the Second 10-Year Interval Inservice Inspection Program Based on Risk-Informed Alternative Approach," dated September 11, 2000. The change in risk for a particular system was calculated using the

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following:

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

where

- $\Delta CDF_j$  = Change in CDF for system j  
 $FR_{i,j}$  = Rupture frequency per element for risk segment i of system j  
 $SXI_{i,j}$  = Number of Section XI inspection elements for risk segment i of system j  
 $RISI_{i,j}$  = Number of RI-ISI inspection elements for risk segment i of system j  
 $CCDP_{i,j}$  = Conditional core damage probability given a break in risk segment i of system j

The total change in risk for all systems within the RI-ISI evaluation scope is calculated by summing the changes in risk for each individual system, as follows:

$$\Delta CDF_{TOTAL} = \sum_j \Delta CDF_j \quad (2)$$

Similar calculations were performed using the CLERP (conditional large early release probability) to determine the change in LERF for each system and the total change in LERF due to implementing the RI-ISI program. The risk impact calculations were also performed excluding the low risk category welds from the calculation. Results of these calculations are presented in Tables RAI 13-A and RAI 13-B, for DNPS Unit 2 and Unit 3, respectively. Also shown in Tables RAI 13-A and RAI 13-B are the results of the Markov model calculation of the change in risk, for comparison purposes.

Using this method to calculate the change in risk requires making several assumptions. Those assumptions are as follows:

- Inspections are 100% successful at finding flaws and preventing ruptures.
- Increased probability of detection (POD) due to inspection for cause is not credited.
- Pipe failure rates and rupture frequencies are constant, not age dependent.

## RESULTS

The results of the DNPS 2 risk impact calculation are shown in Table RAI 13-A. Even using the simplified risk impact approach and including all of the welds in the RI-ISI scope including those in Low Risk regions of the EPRI risk matrix, none of the systems came close to the change in CDF criterion of 1.0E-07 per system per year. The largest change in CDF came from the feedwater system, at 1.39E-08. The total change in CDF was 1.59E-08, well below the criterion of risk significance from Regulatory Guide 1.174 of 1.0E-06 for all systems. Similarly, the change in LERF values were all well below the criterion of 1.0E-08 per system per year. Again, the largest change came from the feedwater system, at 1.30E-09. The total change in LERF was 2.50E-09, well below the criterion of risk significance from Regulatory Guide 1.174 1.0E-07 for all systems.

The results of the DNPS 3 risk impact calculation are shown in Table RAI 13-B. Even using

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### Response to Request for Additional Information Alternative to the ASME Boiler and Pressure Vessel Code Section XI Requirements for Class 1 and 2 Piping Welds Risk Informed Inservice Inspection Program

the simplified risk impact approach and including all of the welds in the RI-ISI scope, none of the systems came close to the CDF criterion of  $1.0E-07$  per system per year. The largest change in CDF came from the feedwater system, at  $1.39E-08$ . The total change in CDF was  $1.79E-08$ , well below the criterion of  $1.0E-06$  for all systems. Similarly, the change in LERF values were all well below the criterion of  $1.0E-08$  per system. Again, the largest change came from the feedwater system, at  $1.30E-09$ . The total change in LERF was  $3.60E-09$ , well below the criterion of  $1.0E-07$  per system per year for all systems.

Note that in some cases such as that for the feedwater system, the realistic evaluation predicts a decrease in CDF and LERF, whereas the conservative approach forces an increase. Compared to the more realistic calculation of risk impact using the Markov model, the simplified method produced changes in CDF for a single system as much as a factor of 15 higher than the Markov model results. The largest differences between the simplified approach and the Markov method are observed in the feedwater system and main steam system. These differences are mainly due to a single risk segment in each system with a relatively high CCDP that credited an enhanced POD in the Markov model calculation that is not credited in the simplified approach. The simplified risk impact calculation for other systems results in  $\Delta$ CDFs and  $\Delta$ LERFs that are generally less than a factor of 2 higher than the Markov model results.

In preparation of this RAI response, supplements to the Tier 2 documentation were prepared to document these calculations on a segment by segment basis. In most cases, the conservative values are about a factor of 2 or so higher than the associated realistic values, but in a few cases, the increase is more than an order of magnitude. Nonetheless, the risk acceptance criteria for all analyzed systems at DNPS Units 2 and 3 are still met with a large margin.

These conservative results are regarded as a sensitivity study as they only reflect upper bounds on the expected risk impacts. The results obtained using the Markov model are considered more reasonable and realistic for the following reasons.

- There were many cases in which the effectiveness of the inspection will be increased as a result of the application of the "inspection for cause" principle in which the knowledge of the applicable damage mechanisms and the application of mechanism specific inspection methods provide a reasonable basis to expect enhanced inspection effectiveness. A good example is the case of locations susceptible to thermal fatigue in which the EPRI RI-ISI exams call for an expanded examination volume into the Heat Affected Zone (HAZ) of the weld in comparison with ASME Section XI examination requirements. This expanded volume recommendation is based on insights from service experience that indicate the location of cracks in the areas of welds caused by thermal fatigue. These inspection for cause effects are ignored in the bounding evaluations.
- The conservative calculation assumes that all the change in risk in a given risk segment comes from the net change in the number of exams; which implies that there can be no change from redistributing a fixed number of welds. This does not reflect the true philosophy of risk management as expressed in Regulatory Guide 1.178, Regulatory

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### **Response to Request for Additional Information Alternative to the ASME Boiler and Pressure Vessel Code Section XI Requirements for Class 1 and 2 Piping Welds Risk Informed Inservice Inspection Program**

Guide 1.174, or the EPRI Topical Report regarding the balancing of resources away from areas with marginal risk impact toward areas of more significant risk impact.

- The risk impact of a changing the inspection strategy of a given weld is one of the factors that was considered in the element selection. If that input to the selection is skewed by conservative assumptions that do not uniformly impact across the elements in the program, the goal of an optimized program is not as well supported in comparison with the case where realistic assumptions are used for all the welds in the examination.
- The inspection effectiveness factors obtained using the Markov model provide a more realistic perspective on the benefits of ISI exams. This permits better tradeoffs in balancing the combined influences of removing exams, redistributing exam locations, and enhancing the effectiveness of exams through the inspection for cause principle.
- This approach of performing a realistic risk impact assessment provides a better basis to normalize risks and risk impacts across different risk informed initiatives such as RI-ISI, RI-IST, and risk informed technical specifications, in contrast to limiting the analysis for RI-ISI to a conservative bounding assessment. If one of these applications uses conservative bounding estimates and the remaining ones use realistic treatment, the balancing of resources expected from risk informed regulation is not as well supported as when all applications aspire for a comparable level of realism.

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Table RAI 13-A. Comparison of Risk Impact Results for DNPS Unit 2

DNPS 2 Risk Impact Report*						
System	CDF			LERF		
	Conservative Delta CDF for All Welds	Conservative Delta CDF Excluding Low Risk Welds	Realistic Delta CDF using Markov Model	Conservative Delta LERF for All Welds	Conservative Delta LERF Excluding Low Risk Welds	Realistic Delta LERF using Markov Model
CRD	3.44E-11	0.00E+00	1.94E-11	3.44E-11	0.00E+00	1.94E-11
ECCS	-5.20E-10	-5.42E-10	-4.21E-10	-4.92E-10	-5.14E-10	-3.22E-10
FW	1.39E-08	1.39E-08	9.02E-10	1.30E-09	1.30E-09	2.32E-10
HPCI	5.16E-10	5.06E-10	2.90E-10	3.70E-11	3.62E-11	2.08E-11
MS	9.90E-10	5.21E-10	3.84E-10	7.33E-10	5.21E-10	2.59E-10
RCS	6.15E-10	1.73E-10	3.34E-10	6.30E-10	1.50E-10	3.43E-10
RWCU	6.42E-11	0.00E+00	3.63E-11	2.55E-11	0.00E+00	1.44E-11
SBLC	2.28E-10	2.09E-10	1.25E-10	2.28E-10	2.09E-10	1.25E-10
SDC	6.30E-14	0.00E+00	-3.73E-11	1.80E-14	0.00E+00	-3.55E-11
Total	1.59E-08	1.48E-08	1.63E-09	2.50E-09	1.70E-09	6.57E-10

\* Positive values indicate a risk increase while negative values denote a risk decrease

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Response to Request for Additional Information  
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Table RAI 13-B. Comparison of Risk Impact Results for DNPS Unit 3

DNPS 3 Risk Impact Report*						
System	CDF			LERF		
	Conservative Delta CDF for All Welds	Conservative Delta CDF Excluding Low Risk Welds	Realistic Delta CDF using Markov Model	Conservative Delta LERF for All Welds	Conservative Delta LERF Excluding Low Risk Welds	Realistic Delta LERF using Markov Model
CRD	6.61E-11	0.00E+00	3.74E-11	4.69E-11	0.00E+00	2.65E-11
ECCS	2.28E-10	2.05E-10	9.76E-11	1.16E-10	1.05E-10	4.90E-11
FW	1.39E-08	1.39E-08	9.02E-10	1.30E-09	1.30E-09	2.32E-10
HPCI	2.24E-12	-5.14E-13	1.05E-12	1.09E-12	-1.29E-13	5.60E-13
MS	3.85E-10	-4.84E-11	5.98E-11	1.52E-10	-4.82E-11	-7.19E-11
RCS	2.52E-09	8.81E-11	1.41E-09	1.25E-09	7.34E-11	6.95E-10
RWCU	5.66E-12	0.00E+00	3.20E-12	5.58E-12	0.00E+00	3.15E-12
SBLC	7.57E-10	7.28E-10	4.24E-10	7.56E-10	7.27E-10	4.24E-10
SDC	-2.56E-11	-2.57E-11	-4.37E-11	-2.57E-11	-2.57E-11	-4.26E-11
<b>Total</b>	<b>1.79E-08</b>	<b>1.49E-08</b>	<b>2.89E-09</b>	<b>3.60E-09</b>	<b>2.13E-09</b>	<b>1.32E-09</b>

\* Positive values indicate a risk increase while negative values denote a risk decrease

**Attachment 1**

**Response to Request for Additional Information  
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NRC Question #14:

Please provide a table where the number of Class 1, Class 2, and augmented inspections credited in the RI-ISI program is given for each system.

EGC Response:

The table below summarizes the RI-ISI inspections by risk category, system and piping class. Augmented inspections for FAC and IGSCC are not credited in the RI-ISI program.

Risk Cat.	System	DNPS 2		DNPS 3	
		Class 1	Class 2	Class 1	Class 2
1	FW	6	1	7	
2	CS	5		3	
	ISCO	3		1	
	LPCI	9		4	
	SDC	12		11	
3	MS	6		10	
4	CS	3	1	1	4
	ECCS		6		6
	HPCI	1	7	1	6
	ISCO	1		1	
	LPCI		6		4
	MS	12		11	
	RHV			1	
	RPV	1		1	
	RR	2		9	
	RWCU	2		2	
SBLC	3		4		
5	HPCI		3		3
	ISCO	1		1	
	RPV	1		1	
	RVBD	1			
	SBLC	2		2	
<b>TOTAL</b>		71	24	71	23

## Attachment 2

### Excerpted from Dresden Tier 2 RI-ISI Documentation Risk Impact Of Implementing Risk Informed Inspection Program

#### 7.1 OVERVIEW

This analysis is conducted to support the risk-informed evaluation of piping systems at the Dresden Nuclear Power Station (DNPS), Units 2 and 3. The objectives of the overall evaluation process, as illustrated in Figure 7-1, are to identify risk important piping segments, define the elements that are to be inspected within this risk important piping, evaluate the risk impacts of proposed changes to the inspection program, and identify appropriate inspection methods. As part of determining the risk significance piping, the Risk Impact of Implementing Risk Informed Inspection Program provided in this section (highlighted in Figure 7-1) focuses on evaluating the changes in CDF and LERF associated with the changes that are introduced by the risk informed ISI program.

As seen in Figure 7-1, the risk impact assessment is performed in Step 5 of the overall procedure to evaluate the element selection that was made in the previous step in terms of the impact on the risk of pipe failures. Depending on the results of this evaluation, adjustments may have to be made to the element selection to ensure that risk acceptance criteria are met.

The risk acceptance criteria that have been established for applications of the EPRI method for RI-ISI, as noted in the EPRI Topical Report [7-1, 7-2] are applied on a system by system basis. The risk impacts of proposed changes to the inspection program are considered insignificant, so long as the following criteria are met.

$$\begin{aligned}\Delta\text{CDF}_{\text{SYSTEM}} &< 1\text{E-}7/\text{ reactor-year, and} \\ \Delta\text{LERF}_{\text{SYSTEM}} &< 1\text{E-}8/\text{ reactor-year}\end{aligned}$$

Where:

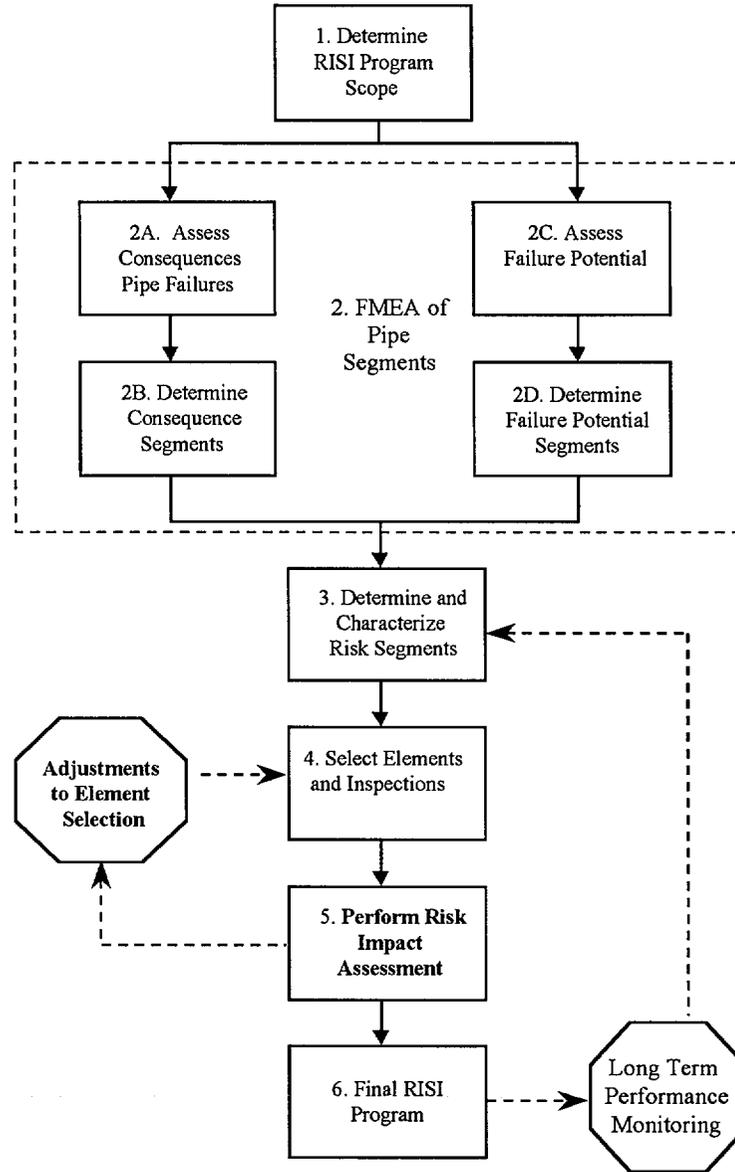
$$\begin{aligned}\Delta\text{CDF}_{\text{SYSTEM}} &= \text{Change in system CDF due to changes in ISI program, and} \\ \Delta\text{LERF}_{\text{SYSTEM}} &= \text{Change in system LERF due to changes in ISI program}\end{aligned}$$

The above limits were set at 10% of the risk significance limits set forth in Regulatory Guide 1.174 [7-17] with the understanding that a "full scope" RI-ISI program would typically involve about 10 piping systems.

The technical approach to risk impact assessment in the EGC RI-ISI evaluation is described in Section 7.2. An important input to this evaluation is the development of piping system failure rates and rupture frequencies, which is documented in Section 7.3. The results of the risk impact evaluation for DNPS Units 2 and 3 piping systems are documented in Sections 7.4 and 7.5, respectively.

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### Excerpted from Dresden Tier 2 RI-ISI Documentation Risk Impact Of Implementing Risk Informed Inspection Program



#### LEGEND



Step Covered in This Section



Step Covered in Another Section

**Figure 7-1**  
**Overview of EPRI RI-ISI Methodology**

## Attachment 2

### Excerpted from Dresden Tier 2 RI-ISI Documentation Risk Impact Of Implementing Risk Informed Inspection Program

#### 7.2 TECHNICAL APPROACH

##### 7.2.1 Qualitative Evaluation of Changes to CDF and LERF

There are exactly three situations in comparing the RI-ISI program for a particular element selection with the existing ISI program that is being changed, that would lead to changes in CDF or LERF. These include:

- Adding elements to inspection program in relative high risk piping segments that were not in the previous inspection program.
- Improving the probability of detection of an inspection by incorporating the “inspection for cause” concept.
- Eliminating an element from the inspection program in relatively low risk piping segments.

The first two of these will result in a decrease in pipe failure frequency, and a corresponding decrease in CDF and LERF for each pipe element that applies. The last one will result in at least a small increase in CDF and LERF for each pipe element that applies. For any element that is not impacted by the change to the ISI program, there is absolutely no change to the CDF and LERF contribution from pipe failures at such element. Hence the net change in CDF and LERF for a system is comprised of the sum of the changes in CDF and LERF over all the elements in which there is a change in the inspection program. Moreover, even though there may be a large net reduction in the number of welds inspected in a given system, the CDF and/or LERF may actually decrease if the magnitude of changes associated with ISI program enhancements in the high risk segments exceeds that of the elements eliminated from the low risk segments.

##### 7.2.2 Model For Estimating Changes In CDF And LERF

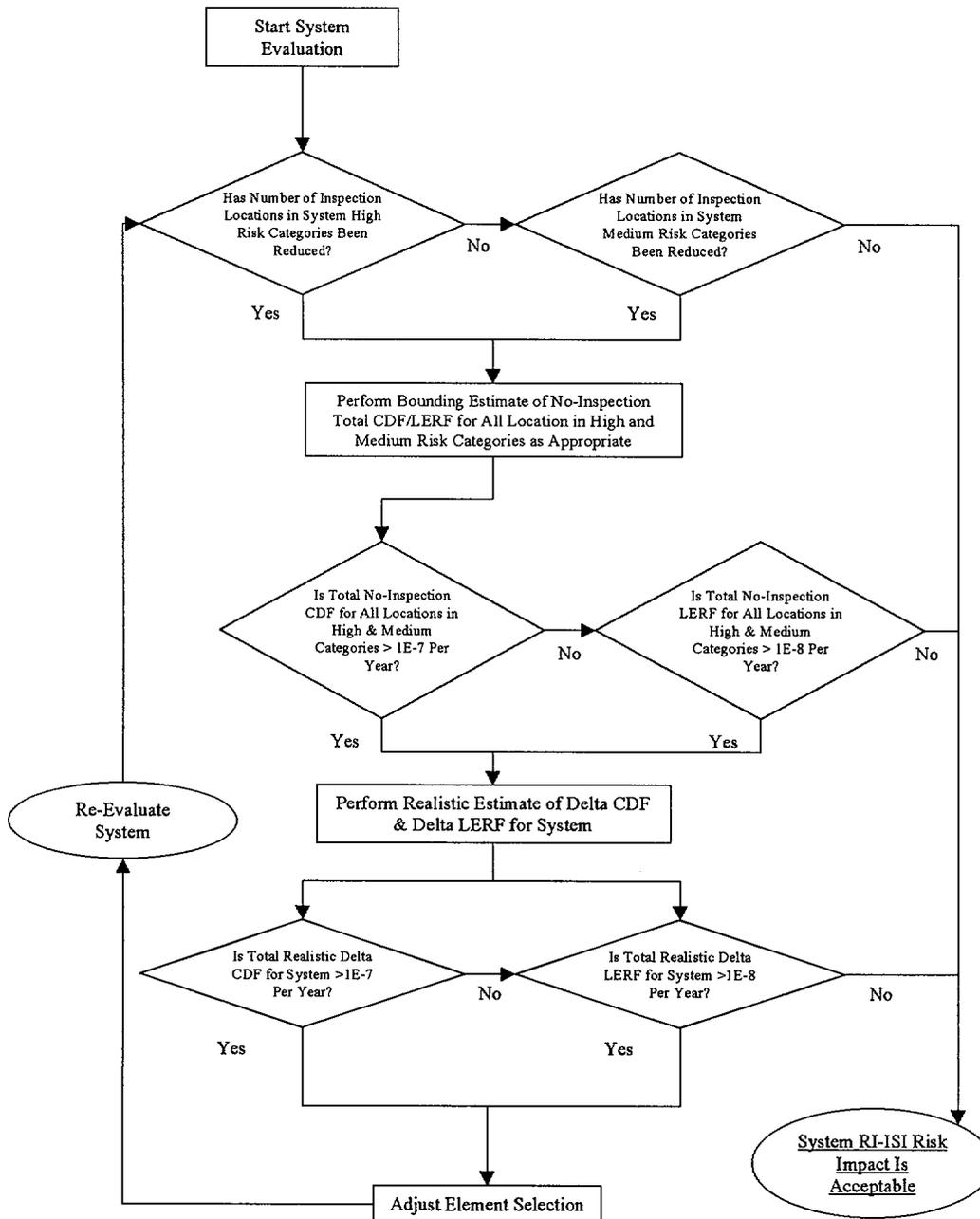
The EPRI approach to RI-ISI calls for risk impact evaluations to be performed using qualitative analyses, bounding quantitative estimates or realistic quantitative estimates as illustrated in Figure 7-2. This flow chart was developed to minimize the amount of work that was needed to evaluate the risk impact by first trying to evaluate based on qualitative and bounding quantitative estimates. The authors of this evaluation have determined that it is better to actually perform realistic quantitative estimates for all pipe elements for the following reasons:

- Full quantification using realistic assumptions will put this application of the PRA on an equal footing with other risk informed programs and will assist EGC in balancing risks and resources across applications
- If any realistic quantitative estimates are needed, the data that is needed for these estimates is a large fraction of the data that is needed for full realistic quantification
- The evaluation of risk impacts is set up using a spreadsheet which, in combination with the previous item, results in a minimal reduction of effort by using a mixture of three methods (i.e., qualitative, bounding quantitative, and realistic quantitative) that is suggested in Figure 7-2.

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### Excerpted from Dresden Tier 2 RI-ISI Documentation Risk Impact Of Implementing Risk Informed Inspection Program

- In future updates of the RI-ISI program, and in the iterations between element selection and risk impact assessment, it is much easier to have the data in place to perform an evaluation in all segments within the scope of the evaluation.



**Figure 7-2**  
**Flow Chart for Evaluation of Risk Impacts [7-1]**

## Attachment 2

### Excerpted from Dresden Tier 2 RI-ISI Documentation Risk Impact Of Implementing Risk Informed Inspection Program

For these reasons, it was decided to perform full realistic risk impact assessments for the entire scope of the formal RI-ISI program, which includes all non-exempt pipe in Class 1 and 2 piping systems. This approach is followed consistently for all EGC plants as well as the qualitative evaluation steps covered in Figure 7-2.

The changes in CDF and LERF associated with changes to the inspection strategy for each system are estimated using the following equations [7-1]:

$$\Delta CDF = \sum_{i=1}^N n_i \lambda_i P_i(R|F) (I_{i,new} - I_{i,old}) CCDP_i \quad (7.1)$$

and

$$\Delta LERF = \sum_{i=1}^N n_i \lambda_i P_i(R|F) (I_{i,new} - I_{i,old}) CLERP_i \quad (7.2)$$

Where:

- $\Delta CDF$  = change in core damage frequency due to changes in inspection strategy for the system
- $\Delta LERF$  = change in large early release frequency due to changes in the inspection strategy for the system
- $i$  = index for risk segment having the same degradation mechanisms and consequence of pipe ruptures
- $N$  = number of risk segments in the system
- $n_i$  = number of elements (welds) in risk segment  $i$
- $\lambda_i$  = failure rate for welds in risk segment  $i$  (including leak and rupture failure modes) assuming no inspections, estimated from service data
- $P_i(R|F)$  = conditional probability of rupture given failure of welds in risk segment  $i$  assuming no inspections, estimated from service data
- $I_{i,new}$  = inspection effectiveness factor for proposed risk informed inspection strategy for risk segment  $i$ , calculated from Markov model
- $I_{i,old}$  = inspection factor for current ASME Section XI based inspection strategy for segment  $i$ , calculated from Markov model
- $CCDP_i$  = conditional core damage probability due to pipe ruptures in risk segment  $i$ , obtained from Consequence Evaluation (Steps 2A and 2B in Figure 7-1).
- $CLERP_i$  = conditional large early release probability due to pipe ruptures in risk segment  $i$ , obtained from Consequence Evaluation (Steps 2A and 2B in Figure 7-1).

## Attachment 2

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#### 7.2.3 Method of Estimating Model Parameters

The input parameters in Equations (7.1) and (7.2) are estimated as indicated in Table 7-1. Weld counts are established from the EGC ISI database in which piping system line numbers have been subdivided into risk segments, i.e., segments with the same degradation mechanism potential and consequence potential. The pipe failure and rupture parameters are estimated using Bayesian failure rate estimation techniques that were specifically developed and approved for use in the EPRI RI-ISI applications [7-3, 7-4, 7-5]. EPRI sponsored a project to develop a specific set of failure rates and rupture frequencies in RI-ISI applications [7-3]. However, to take advantage of more recently available information, a new set of failure rates and rupture frequencies for Westinghouse Class 1 and 2 systems was developed specifically for this evaluation. The technical approach, sources of data, and results from this work are presented in Section 7.3. To estimate the inspection effectiveness factors, the Markov method is used as documented in Section 7.2.4 [7-4]. The Markov method derives equations for the inspection effectiveness factors that are in turn dependent on the same failure rates and rupture frequencies and parameters that describe the inspection and leak detection processes. This method was also approved for use by the NRC in the Safety Evaluation (SE) for RI-ISI applications following the EPRI methodology [7-2, 7-5]. An overview of the Markov model for piping systems is provided in Section 7.2.4 together with documentation of how it was applied to Class 1 and 2 piping systems at DNPS Units 2 and 3.

**Table 7-1  
Method of Quantification of Parameters in Equations (7.1) and (7.2)**

Parameter	Method of Quantification
$\Delta CDF$	Computation of Equation (7.1)
$\Delta LERF$	Computation of Equation (7.2)
$i$	From risk segment definition in Step 3 of Figure 7-1
$N$	From risk segment definition in Step 3 of Figure 7-1
$n_i$	From risk segment definition in Step 3 of Figure 7-1
$\lambda_i$	Estimated from service data as documented in Section 7.3
$P(R F)$	Estimated from service data as documented in Section 7.3
$I_{i,new}$	Markov model solution used to develop equation in terms of parameters that describe degradation and inspection processes as explained in this section
$I_{i,old}$	Markov model solution used to develop equation in terms of parameters that describe degradation and inspection processes as explained in this section
$CCDP_i$	Evaluated in Steps 2A and 2B in Figure 7-1 using plant specific PRA models and the results of the consequence analysis
$CLERP_i$	Evaluated in Steps 2A and 2B in Figure 7-1 using plant specific PRA models and the results of the consequence analysis

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#### 7.2.4 Markov Model for Piping System Reliability

##### 7.2.4.1 Overview of Markov Model

There are several different approaches that have been applied to estimation of pipe failure frequencies. Of course, the most straightforward approach is to simply obtain statistical estimates of pipe element failure rates which is the most common approach to this problem [7-6, 7-7, 7-8]. The primary limitation of a statistical analysis approach is that past historical data reflects some indeterminate impact of previous inspection programs and if we are going to propose changes to these programs, such changes may render the previous failure rate estimates invalid. Another approach is to make use of probabilistic fracture mechanics models to predict crack initiation and growth from existing flaws. Such models reflect our understanding of the physical processes of fracture mechanics but to date have not been fully benchmarked against service experience. To examine an alternative approach and to pursue the objective of keeping the approach practical and useful for utility piping engineers, the concept of Markov models supported by analysis of service experience was pursued.

During a third party review of the original EPRI RI-ISI methodology [7-9], an idea emerged to utilize an established reliability modeling technique, known as the Markovian technique, to address the impact of inspections on pipe rupture frequencies. The objective of this approach is to explicitly model the interactions between degradation mechanisms and the inspection, detection, and repair strategies that can reduce the probability that failures occur or that failures will progress to ruptures. This Markov modeling technique starts with a representation of a piping "system" in a set of discrete and mutually exclusive states. At any instant in time, the system is permitted to change state in accordance with whatever competing processes are appropriate for that plant state. In this application of the Markov model, the states refer to various degrees of piping system degradation or repairs, i.e., the existence of flaws, leaks, or ruptures. The processes that can create a state change are the failure mechanisms operating on the pipe and the processes of inspecting or detecting flaws and leaks, and repair of damage prior to the progression of the failure mechanism to rupture.

The basic form of a Markov Model for pipe failure and inspection processes is presented in Figure 7-3. This model consists of four states of a pipe segment or element (e.g., a weld or section of pipe) reflecting the progressive stages of pipe failure mechanisms: the development of flaws or detectable damage, the occurrence of leaks, and the occurrence of pipe ruptures. As seen in this model, pipe leaks and ruptures are permitted to occur directly from the flaw or leak state, or may also occur in a progression. The model accounts for state dependent failure and rupture processes and two repair processes. Once a flaw occurs, there is an opportunity for inspection and repair to account for the in-service inspection program and other programs that search for signs of degradation prior to the occurrence of pipe failures. When a pipe leak occurs, there is another opportunity for detection and repair prior to the occurrence of a rupture for failure mechanisms that have a "leak before break" characteristic.

The Markov model diagram describes the failure and inspection processes as a discrete state-continuous time problem. It is used to develop a set of differential equations, the solution of which is the time dependent probability of the system occupying each state. For the study of pipe ruptures, state "R" is the failure state of interest. Once the solution is obtained, the hazard

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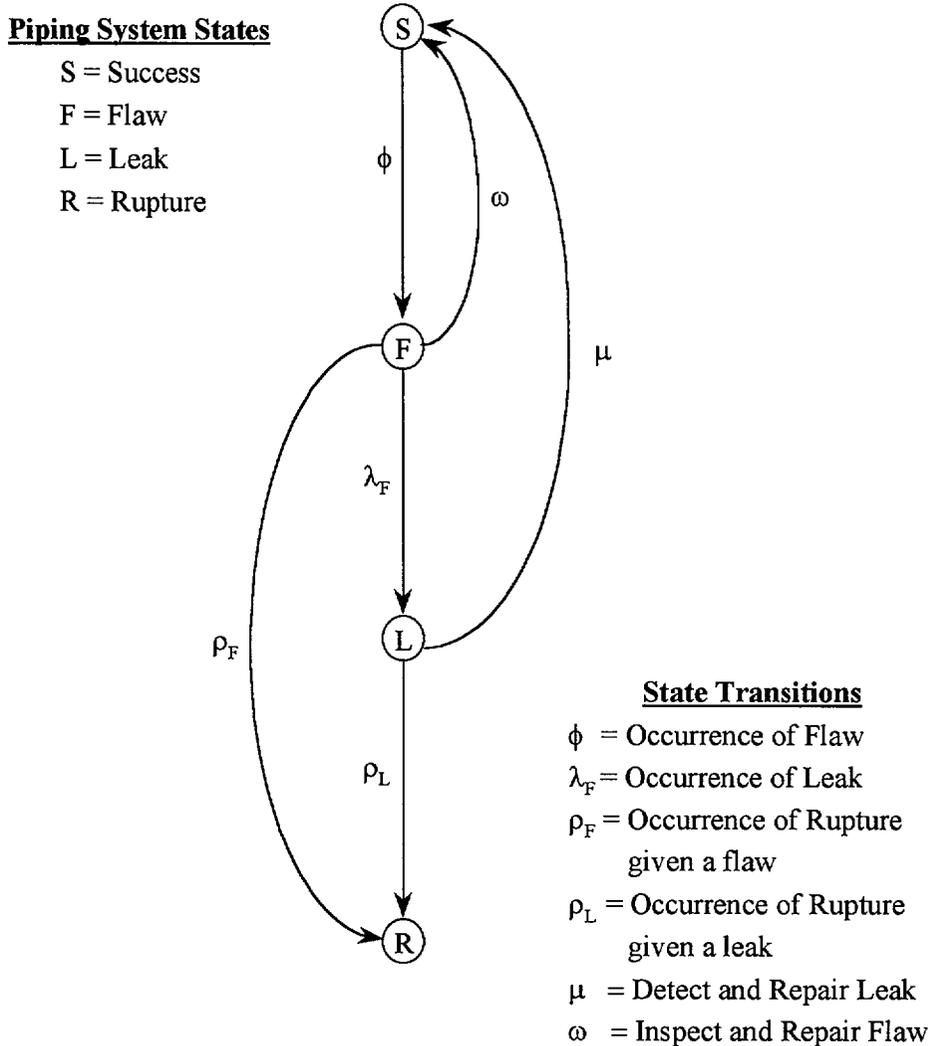
rate of the system can be determined. For this example, the hazard rate corresponds to the time dependent frequency or failure rate for pipe ruptures. The time dependent failure rate for ruptures asymptotically converges to a constant value which is a function only of the parameters of the model. This long term failure rate or hazard rate is the long term pipe rupture frequency that determines the long term risk of pipe ruptures. These parameters are in turn related to the time constants of the underlying processes. The occurrence rates for flaws, leaks, and ruptures are estimated from service data. The occurrence rates for inspections and repairs are estimated based on the characteristics of the inspection process, non-destructive examination (NDE) reliability, time interval of leak detection, and mean time to repair flaws and leaks upon detection. Application of the Markov model can be accomplished based on this steady state hazard, or as a time dependent hazard that varies over the life of the plant.

The Markov models for pipe ruptures are used to set up and solve differential equations for the time dependent state probabilities associated with the model. These equations are based on the assumption that the probability of transition from one state to another is proportional to the transition rates indicated on the diagrams and that there is no memory of how the current state is arrived at. Under the assumption that all the transition rates are constant, the Markov model equations will consist of a set of coupled linear differential equations with constant coefficients. The solution of these differential equations is obtained to compute the time dependent probability that the pipe segment in question is in each state S, F, L, or R. Once these results are obtained, other results such as the system hazard rate that defines the time dependent frequency of pipe ruptures can be developed. This frequency is the form of the result that is needed to support a PSA model of pipe ruptures as initiating events. Details of how this method is developed and solved are provided in Reference [7-4].

Based on insights from service experience, it was decided to use several different models for estimating pipe rupture frequencies depending on the specific failure mechanism. There are several reasons for this. One is that certain mechanisms can be attributed to specific elements of the piping system that are susceptible to failure. These are associated with degradation mechanisms that tend to occur either at specific welds or specific sections of pipe that exhibit the conditions necessary for these failure mechanisms. The applicable damage mechanisms for this type include corrosion, corrosion fatigue, erosion corrosion, erosion-cavitation, stress corrosion cracking, and thermal fatigue. Of these, all except corrosion and erosion corrosion, which do not necessarily occur at welds, tend to occur at or near welds. Hence, estimating pipe rupture failure frequencies in terms of ruptures per susceptible weld or ruptures per susceptible foot of pipe are viable approaches for these failure mechanisms, all of which are damage mechanisms. Another common feature exhibited by these failure mechanisms is that they have demonstrated in the service experience data to show a strong "leak before break" characteristic, i.e., the observed frequency of leak type failure modes is much greater than the rupture type failure mode.

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**Figure 7-3**  
**Markov Model for Pipe Elements With In-service Inspection and Leak Detection**

A summary of the different models being used in the EPRI RI-ISI program is provided in Table 7-2. The model we use to estimate these degradation type failure mechanisms is referred to as Model A which expressed the pipe rupture frequency in terms of a pipe failure rate or frequency and a conditional probability of pipe rupture given failure. The conditional probability of rupture given failure provides a means of quantifying the "leak before break" characteristics of the failure mechanism. In this model, the service data is broken down to support dependence of the rupture and failure parameters on the reactor vendor, system type, and specific damage mechanism. Model A1 supports estimates in terms of ruptures and failures per susceptible foot of pipe per year for corrosion and erosion corrosion, while Model A2 supports estimates of pipe rupture frequency in terms of ruptures and failures per susceptible weld per year. Model A2 is used in Equations (7.1) and (7.2) as all the piping of interest in this evaluation is subject to the

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class of degradation mechanisms that occur in welds. Although there are some piping segments in DNPS Class 1 and 2 systems subjected to erosion corrosion or FAC, the RI-ISI program is not proposing any changes to augmented inspection programs for FAC. Hence, any change in risk for this evaluation will be solely due to weld type degradation mechanisms at inspection locations that may be removed from the ISI program.

**Table 7-2  
Failure Rate Models Used for Different Failure Mechanisms**

Failure Mechanism Class	Failure Mechanism	Failure Rate Basis	Failure Rate Models Employed**
Degradation Mechanisms	Corrosion	Failures/pipe-ft-yr.*	Model A1
	Erosion Corrosion		
	Erosion Cavitation	Failures/weld-yr.*	Model A2
	Thermal Fatigue		
	Stress Corrosion Cracking		
	Corrosion Fatigue		
Design and Construction Defects			
Severe Loading Conditions	Water Hammer	Failures/system-yr	Models B and C
	Over-pressurization		Model B
	Frozen Pipes		
	Vibrational Fatigue		

\* Failure rates applicable only to welds and section of pipe found susceptible to specified damage mechanism

\*\*Model A  $Freq\{Rupture\} = Freq\{Failure\} \times Prob\{Rupture | Failure\}$  failure and rupture data used in Bayes update of Generic Priors

\*\*Model B  $Freq\{Rupture\}$  developed directly from rupture data and used in Bayes update of Generic Prior

\*\*Model C  $Freq\{Rupture\} = Freq\{Water Hammer\} \times Prob\{Rupture | Water hammer\}$  used in Bayes update of Generic Priors

The remaining failure mechanisms that have been identified are described as loading conditions and include water hammer, over-pressurization, frozen pipes, and vibration fatigue and are not amenable to in-service inspection as a means of failure prevention. Design and construction defects occur at welds and are amenable to ISI in the sense that such errors can be found during NDE type inspections. These loading conditions occur randomly and have the potential for failure or rupture anywhere in a system. Another aspect of the severe loading type failure mechanisms is that at the plant level they exhibit a weak "leak before break characteristic." For these mechanisms, we use rupture data directly to estimate rupture frequencies, and the unit of measurement that is sensible for these are ruptures for system year for different system groups and specific loading conditions. We refer to this approach as Model B.

A third model was developed to support the particular loading condition of water hammer. While

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Model B can be used to obtain a kind of average frequency of pipe ruptures due to water hammer, the available data on this mechanism (Stone and Webster, 1992 [7-11]) supports a more specialized model. This is known as Model C in which pipe ruptures from water hammer are expressed in terms of the frequency of water hammer events, obtained from a special database, and the conditional probability of pipe rupture given a water hammer event.

Models B and C are not used in this evaluation because there is no impact of inspection program changes on these failure mechanisms. While these models are relevant to the task of estimating the total failure rates and rupture frequencies of pipes due to all failure mechanisms, they are not relevant to evaluating changes in failure rates and rupture frequencies. Returning to Equations (7.1) and (7.2), only model A2 is relevant to determination of the inspection effectiveness factor.

#### 7.2.4.2 Use of the Markov Model to Calculate Inspection Effectiveness Factor

With reference to Equations (7.1) and (7.2) 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:

$$I_{i,new} = \frac{h_{40}\{RISI\}}{h_{40}\{noinsp\}} \quad (7.3)$$

$$I_{i,old} = \frac{h_{40}\{SecXI\}}{h_{40}\{noinsp\}} \quad (7.4)$$

Where:

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

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

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

The solutions to the Markov model for time dependent hazard rates are developed in Reference [7-4]. These solutions are developed in terms of closed form analytic solutions that have been applied to EGC systems in Microsoft Excel spreadsheets. Independent reviews have been performed by EdF, the University of Maryland (discussed in Reference [7-4]), and Los Alamos National Laboratory [7-5]. The hazard rates are a function of time and of the parameters of the Markov model presented in Figure 7-3. The quantification of these parameters is discussed in the section below.

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#### 7.2.4.3 Estimation of Markov Model Parameters

As seen in Figure 7-3, there are six parameters that are associated with the Markov model, an occurrence rate for detectable flaws,  $\phi$ , a failure rate for leaks given the existence of a flaw,  $\lambda_F$ , two rupture frequencies including one from the initial state of a flaw  $\rho_F$ , and one from the initial state of a leak,  $\rho_L$ , a repair rate for detectable flaws,  $\omega$ , and a repair rate for leaks,  $\mu$ .

The latter two parameters dealing with repair are further developed by the following simple models.

$$\omega = \frac{P_{FI} P_{FD}}{(T_{FI} + T_R)} \quad (7.5)$$

Where:

$P_{FI}$  = probability that a piping element with a flaw will be inspected per inspection interval. This parameter has a value of 0 if it is not in the inspection program, and 1 if it is in the inspection program.

$P_{FD}$  = probability that a flaw will be detected given this element is inspected. This is the reliability of the inspection program and is equivalent to the term used by NDE experts, "Probability of detection (POD)". This probability is conditioned on the occurrence of one or more detectable flaws in the segment according to the assumptions of the model. Also note that

$T_{FI}$  = mean time between inspections for flaws, (inspection interval)

$T_R$  = mean time to repair once detected. There is an assumption that any significant flaw that is detected will be repaired. Depending on the location of the weld to be repaired, the weld repair could take on the order of several days to a week. However, since this term is always combined with  $T_{FI}$ , and  $T_{FI}$  is 10 years, in practice the results are insensitive to assumptions regarding  $T_R$

Similarly, estimates of the repair rate for leaks can be estimated according to:

$$\mu = \frac{P_{LD}}{(T_{LI} + T_R)} \quad (7.6)$$

Where:

$P_{LD}$  = probability that the leak in the element will be detected per leak inspection or detection period

$T_{LI}$  = mean time between inspections for leaks. For pipes containing radioactive fluid such as the RCS, the time interval between leaks can be essentially instantaneous if the

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leak is picked up by radiation alarms, to as long as the time period between leak tests performed on the system. All ASME Class 1, 2, and 3 piping must be tested for leaks at least once per refueling outage.

$T_R$  = as defined above but for full power applications, this time should be the minimum of the actual repair time and the time associated with any LCO if the leak rate exceeds technical specification requirements.

Now we have developed the root input parameters of the Markov model, which if quantified will enable us to quantify the inspection effectiveness factors. A summary of the root input parameters of the Markov model and the general strategy for estimation of each one is presented in Table 7-3. The specific basis for estimation of each of these parameters for DNPS Class 1 and 2 systems is provided in Section 7.2.4.4 below.

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**Table 7-3  
Strategy for Estimation of Markov Model Parameters**

Symbol	Parameter Definition	Strategy For Estimation
$\phi$	Occurrence rate of a flaw	Data from results of NDE inspections and service data with cracks; for selected damage mechanisms normally estimated in terms of a multiple of the total failure rate using the argument that there must be at least one flaw to produce a damage mechanism related leak or rupture.
$\lambda_F$	Occurrence rate of a leak from a flaw state	Estimated in terms of failure rates conditioned on the susceptibility for the indicated damage mechanism according to the EPRI damage mechanism evaluation criteria. It is assumed that if the element is considered susceptible to a damage mechanism according to the EPRI criteria that there is at least one detectable flaw in the element. Different failure rates are estimated for different systems and damage mechanisms.
$\rho_F$	Occurrence rate of a rupture from a flaw state	Estimated in terms of rupture frequencies conditioned on the susceptibility for the indicated damage mechanism according to the EPRI damage mechanism evaluation criteria. Different failure rates for different systems and damage mechanisms. It is assumed that if the element is considered susceptible to a damage mechanism according to the EPRI criteria that there is at least one detectable flaw in the element.
$\rho_L$	Occurrence rate of a rupture from a leak state	This rupture rate occurs during an advanced state of degradation and is normally estimated in terms of the frequency of severe loading conditions such as a water hammer event or overpressure event.
$\omega$	Inspection and repair rate of a flaw state	Model of Equation (7.5) and estimates of $P_i$ , $P_{FD}$ , $T_{FI}$ , $T_R$ as estimated below.
$\mu$	Detection and repair of a leak state	Model of Equation (7.6) and estimates of $P_{LD}$ , $T_{LD}$ , $T_R$ as estimated below.
$P_{FI}$	Probability per inspection interval that the pipe element will be inspected	Set to 1 if the element is included in the inspection program, and 0 if not..
$P_{FD}$	Probability per inspection that an existing flaw will be detected	Estimate based on NDE reliability performance data and difficulty and accessibility of inspection for particular element based on engineering judgement.
$P_{LD}$	Probability per detection interval that an existing leak will be detected	Estimate based on system, presence of leak detection systems, technical specifications, and locations and accessibility of element based on engineering judgement.
$T_{FI}$	Flaw inspection interval, mean time between in service inspections	Normally 10 years for ASME Section XI or RI-ISI piping systems.
$T_{LD}$	Leak detection interval, mean time between leak detections	Estimate based on method of leak detection; ranges from immediate to frequency of routine inspections for leaks, normally set to the maximum permitted by Section XI leak testing requirements once per refueling outage which corresponds to ASME Class 1 systems.
$T_R$	Mean time to repair the piping element given detection of a critical flaw or leak	Estimate of time to tag out, isolate, prepare, repair, leak test and tag in service; if to be conditioned for at power, can be no longer than technical specification limit for operating with element tagged out of service; normally set to a value of 200 hours.

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#### 7.2.4.3 Estimation of Markov Model Parameters for DNPS Class 1 and 2 Systems

##### Failure Rates and Rupture Frequencies from the flaw state ( $\lambda, \rho_F$ )

The Markov Model was applied to each system in the scope of the formal RI-ISI evaluation for DNPS Units 2 and 3. These systems include those portions of the following systems that contain ASME Class 1 or 2 piping elements. For the purpose of organizing the risk impact evaluation, the systems within the scope of the RI-ISI evaluation were placed into the 9 system groups indicated in the table below. These system groups were defined to implement the intent of the EPRI risk significance thresholds that are based on the assumption that a full plant evaluation would have on the order of 10 equivalent piping systems and in consideration of the similarities in piping design, materials and operating conditions.

System Category	Description	Systems Included
CRD	Control Rod Drive Injection	CRD, Scram Discharge Volume (CRDSD)
ECCS	Emergency Core Cooling Systems	Core Spray (CS); Low Pressure Coolant Injection (LPCI); Reactor Head Spray (RHS,RHSP)
FW	Main Feedwater	FW
HPCI	High Pressure Coolant Injection	HPCI
MS	Main Steam	MS, Isolation Condenser (ISCOCR, ISCOSS)
RCS	Reactor Coolant System	Reactor Recirculation (RR); Jet Pump Instrument Taps (JPIA,JPIB); Reactor Vessel Level Instrument Taps (LVLA,LVLB,UVLA,UVLB)
RWCU	Reactor Water Cleanup	RWCU
SLBC	Standby Liquid Control	SBLC
SDC	Shutdown Cooling	SDC

Each of the above system groups was evaluated for impacts in terms of both CDF and LERF. Separate Markov model spreadsheets were set up to evaluate CDF and LERF impacts for all the piping segments within each system group. Key inputs and results for these spreadsheets are provided in Attachments 7A and 7B for DNPS Units 2 and 3, respectively.

The results of the degradation mechanism evaluation presented in Section 4 have found that the piping elements in the above systems have the following possibilities for degradation:

- No degradation mechanism potential
- Thermal fatigue potential (TF, includes TT and TASCS)

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- Stress Corrosion Cracking (SC, includes IGSCC, PWSCC, TGSCC)
- Erosion-corrosion (E/C, includes FAC)
- Erosion cavitation (E-C)
- Corrosion (MIC, Pitting)
- Combinations of two or more of the above mechanisms

The failure rates and rupture frequencies used in the Markov model and in Equations (7.1) and (7.2) have to account for each combination of system and damage mechanism possibility, after excluding those associated with augmented inspection programs that are not being changed in the RI-ISI program. This excludes erosion-corrosion, and corrosion. Hence the failure rates and rupture frequencies that are needed must account for each of the above systems, and elements subject to no degradation, TF, SC, E-C, and combinations of these mechanisms.

The failure rates and rupture frequencies used for each system and damage mechanism combination are summarized in Table 7-4. These failure rates and rupture frequencies are developed from service data, the simple models described in Table 7-2, and the Bayes estimation methodology that was developed in Reference [7-3] and approved by the NRC for use in RI-ISI applications in Reference [7-2]. The failure rates and rupture frequencies for data sets broken down by reactor vendor, system group, and failure mechanisms in Reference [7-3] were used for the delta risk evaluation for DNPS systems. Reference [7-3] was developed to support the NRC review of the EPRI RI-ISI methodology and supporting research to confirm that the EPRI method would result in acceptable risk impacts for the EPRI pilot studies as documented in Reference [7-4]. The failure rates, conditional rupture probabilities, and rupture frequencies developed for GE Boiling Water Reactor piping systems in Reference [7-3] and their applicability to the DNPS Unit 2 and 3 system groups are listed in Table 7-4. Note that these failure rates and rupture frequencies are conditional on the susceptibility to the indicated damage mechanisms, except for design and construction errors, which have the potential at each location. The use of conditional failure rates and rupture frequencies is appropriate for this application because each element has already been subjected to an evaluation for the potential for damage mechanisms.

A key assumption that is made in application of the failure rates and rupture frequencies to the Markov model is that the conditional failure rates and rupture frequencies given susceptibility to a damage mechanism, which is the basis for the numerical estimates, equals the conditional parameter estimates given the existence of a flaw or crack that exceeds the ASME Section XI repair criteria. This is a reasonable assumption because it is necessary to be susceptible to a damage mechanism to have a flaw or crack by that damage mechanism.

As discussed in Section 4, most pipe segments are found to susceptible to no active damage mechanisms. For these segments, the failure rates and rupture frequencies for design and construction errors are used as the only failure mechanism found from the service data that could be identified in a pipe inspection that was not otherwise known to be subject to a damage mechanism. Note that there are other failure mechanisms that would apply to such locations such as water hammer, vibration fatigue, and others but such mechanisms are not amenable to in-service inspections. Only those mechanisms that could be identified in a pipe inspection are appropriate for inclusion in this evaluation, because these are the only mechanisms that could be affected by a change in the inspection program.

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For pipe segments that are found to be susceptible to one ISI amenable damage mechanism such as thermal fatigue, stress corrosion cracking, or erosion-cavitation, the failure rates and rupture frequencies for these elements are determined by combining the contributions in Table 7-4 from the applicable damage mechanism with those from design and construction errors. This is done since all inspection locations are susceptible to design and construction errors. Hence in this case the failure rates and rupture frequencies are determined by summing the contributions from one ISI amenable damage mechanism and design and construction errors.

For pipe segments that are found to be susceptible to two or more ISI amenable damage mechanisms, the following rules are used: The total failure rate for the element is determined by summing the failure rates of each applicable damage mechanism plus the contribution from design and construction errors and then the resulting sum is multiplied by a factor of 3 to account for the possibility of synergy between the damage mechanisms. For DNPS, there were a small number of segments in the FW, ECCS, MS, and SDC system groups that were found to be susceptible to 2 or more ISI amenable damage mechanisms. The vast majority of the evaluated segments were found to be susceptible to no ISI amenable damage mechanism, and the remaining ones were only susceptible to one ISI amenable damage mechanism. The factor of 3 determined via engineering judgment accounts for the possibility that two or more damage mechanisms might influence the propagation of the same flaw or crack. This is viewed as a conservative assumption because no such factor should be applied if the damage mechanisms really act independently.

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**Table 7-4  
Mean Failure Rates, Conditional Rupture Probabilities, and Rupture Frequencies  
Used in DNPS Risk Impact Assessment**

Damage Mechanism	Parameter*	EPRI TR-111880 System Group** [7-3]							
		RCS	SIR	CS	RAS	AUXC	FWC	ST	FPS
Thermal Fatigue (TF)	$\lambda_f$	[The data in this table was removed and can be found in Reference [7-3]]							
	P(R F)								
	$\rho_F$								
Stress Corrosion Cracking (SC)	$\lambda_f$								
	P(R F)								
	$\rho_F$								
Erosion-Cavitation (E-C)	$\lambda_f$								
	P(R F)								
	$\rho_F$								
Design Construction Defects (DC)	$\lambda_f$								
	P(R F)								
	$\rho_F$								

- Failure rates,  $\lambda_f$ , and rupture frequencies,  $\rho_F$ , given in units of events/weld-year, conditional rupture probabilities, P(R|F) are dimensionless

\*\* Definition of System Groups:

RCS	Reactor Coolant System	Used for DNPS RCS System Group
SIR	Safety Injection and Recirculation	Used for DNPS ECCS, HPCI, and SDC System Groups
RAS	Reactor Auxiliary System	Used for DNPS RWCU and SBLC System Groups
AUXC	Auxiliary Cooling Systems	Not Used for DNPS
FWC	Feedwater and Condensate	Used for DNPS FW System
ST	Steam Systems	Used for DNPS MS System Group
FPS	Fire Protection Systems	Not Used for DNPS

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As noted in Section 6 on Element Selection, the RI-ISI program does not impact current augmented programs for corrosion, FAC, or IGSCC. So if a segment is susceptible to a damage mechanism covered in an augmented program, such mechanisms are not included in the failure rates and rupture frequency development described above. However the possibility of synergy between FAC and another ISI amenable damage mechanism is accounted for as follows. There were some segments in the Feedwater systems and other systems that were found to be susceptible to both FAC and another damage. Since the failure rates and rupture frequencies for thermal fatigue were not amenable to resolution into those with thermal fatigue implications and since there is judged to be some potential for synergy, the failure rates and rupture frequencies for thermal fatigue and design and construction errors were combined and then multiplied by a factor of 3. We do not include the FAC contributions as there are no changes to the FAC program, only to the inspection programs for other ISI amenable damage mechanisms. The factor of 3 increase is judgmentally assigned to account for the possibility that the failure rate for thermal fatigue could be higher than that inferred from the service data due to wall thinning that could occur in the area of the welds that are subject to thermal fatigue.

The other potential situation of multiple degradation mechanisms that was considered was the case where one of the damage mechanisms is corrosion that is covered in the augmented programs for MIC. However, for DNPS, there were no systems within the RI-ISI scope that were found to be susceptible to this damage mechanism.

#### Rupture Frequency from the Leak State ( $\rho_L$ )

The Markov model of Figure 7-3 includes the possibility that a leaking pipe element will remain undetected such that degradation may continue until the damage increases to the point of a rupture. The probability of this occurrence is reduced by the occurrence of leak inspections with a relatively high probability of leak detection and repair of the pipe. This would be an advanced stage of pipe aging, such that the frequency of pipe rupture is expected to be much greater than the case permitted by the Markov model in which ruptures occur from the initial state of a flaw or crack in which case the extent of degradation and aging is less advanced. It is difficult to estimate this parameter as there are no data on pipe ruptures in which it is known that the pipe element was leaking previously.

However, there is another consideration that needs to be addressed in the estimation of this parameter and that is the fact that in such an advanced state of degradation, a pipe element would be much more susceptible to pipe rupture due to the combination of this degradation and a severe loading condition. Detailed analysis of service data performed by ERIN for EPRI [7-10] has shown that piping failures have resulted from the following severe loading conditions:

- Water hammer
- Overpressurization
- Frozen pipes

Of these severe loading conditions, water hammer events are by far the most likely. Overpressurization is very unlikely because of the ASME requirements to protect piping in Class 1 and 2 systems with safety and relief valves and the likelihood of challenging pipes beyond the design basis of the relief valves is very small. Frozen pipes are only credible at certain U.S.

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sites in the wintertime in areas that are exposed to the outside atmosphere which is not very likely with Class 1 and 2 systems. On the other hand, water hammer events have occurred in practically all Class 1 and 2 systems including the reactor coolant system (pressurizer spray lines, for example). The rupture frequency given the initial state of a leak,  $\rho_L$ , is conservatively estimated to be equal to the frequency of water hammer events that occur in piping systems. A study performed by Stone and Webster for EPRI [7-11] collected data on reported water hammer events in U.S. commercial nuclear plants through 1991. In this study, a total of 283 water hammer events were reported over a period of about 1,200 reactor years of experience. Using an estimate of 12 piping systems per plant which is consistent with the estimate provided in Reference [7-3], the following point estimate of  $\rho_L$  is obtained:

$$\rho_L = \frac{283}{(12)(1200)} = 1.97 \times 10^{-2} / \text{system} - \text{year} \quad (7.7)$$

#### Frequency of Flaws ( $\phi$ )

In the Markov model, flaws are defined as degradation that has progressed to the point of meeting the repair criteria in Section XI of the ASME code because once the flaw state of the Markov model is occupied, the model assumes that the element will be repaired if the flaw is detected.

Estimates of the frequency of flaws are determined from the same service data that is used to develop the failure rates and rupture frequencies. For evaluation of EGC BWR plants, estimates of the frequency of flaws are developed as a multiple of the pipe failure rate for two different cases. One case is BWR Class 1 piping susceptible to IGSCC which is generally recognized as having a relatively large number of flaws per observed leak. The second case is other Class 1 and 2 piping subject to damage mechanisms other than IGSCC. The former is developed from service data in BWR reactor coolant systems [7-20] and [7-21]. The latter is developed from PWR experience that was analyzed in the EGC RI-ISI evaluations for Braidwood and Byron [7-22] and [7-23]. The service data used in development of failure rates, rupture frequencies, and flaw occurrence rates is based on the world-wide operating experience of BWR plants from 1970 through 1998 and PWR plants from 1970 through March 2000. The BWR data is documented in Reference [7-20] and the PWR data is summarized in Attachment 7A of Reference [7-21]. A query from this database was made to capture events in the database that cover BWR Class 1 piping in the RCS for IGSCC, and PWR Class 1 and 2 piping systems for the CVCS, RCS, RHRS, and SIS systems covering a full range of damage mechanisms. The BWR survey for pipe failures due to IGSCC revealed a total of 212 cracks, 109 leaks, and no ruptures in RCS piping covered in the database. The results for the PWR survey are summarized in Table 7-5. This data includes cracks, pinhole leaks, leaks, and evidence that over this data set there have been no reported pipe ruptures, which are defined as failures with leak flow rates in excess of 50gpm.

In Reference [7-3], this failure data together with other information is used to estimate failure rates and rupture frequencies for different systems and degradation mechanisms in GE BWR plants like DNPS Units 2 and 3. We note that each leak or rupture that is found in the database that resulted from a particular degradation mechanism must have resulted from a flaw that progressed to the state of the rupture. Hence, there is at least one flaw for each of the

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observed leaks and ruptures in the database. In addition, there are additional events denoted in this table in which cracks of a sufficient size to cause the need for repair of the pipe occurred and were repaired before an opportunity to fail occurred. There may have been additional flaws undetected that were created by this same experience, due to the fact that most welds are not ever inspected, and even when welds are inspected, the NDE process may have overlooked some flaws. Based on this reasoning, we take the view that the flaw occurrence rate is best estimated as a multiple of the rate of occurrence of failures, which include both leaks and ruptures.

**Table 7-5**  
**PWR Piping Failures by System and Degradation Mechanism**  
(Taken from Attachment 7A of Reference [7-21])

System Group	Pipe Diameter	Degradation Mechanism*	No. of Events	Failure Mode			
				Crack	Pinhole Leak	Leak	Rupture
CVCS	≥ 2 inch	D&C	3	0	0	3	0
		SC	16	1	2	13	0
		TF	10	0	1	9	0
RCS**	≥ 1-1/2 inch	D&C	7	1	0	6	0
		SC	13	2	0	11	0
		TF	1	0	0	1	0
RHRS	≥ 2 inch	D&C	12	5	1	6	0
		SCC	24	11	9	4	0
		TF	5	3	0	2	0
SIS	≥ 2 inch	D&C	1	0	0	1	0
		SCC	46	14	7	25	0
		TF	10	5	0	5	0
		ALL	149	43	20	86	0

**\*NOTES:**

D&C = Design and construction defects

SC = Stress corrosion cracking

TF = Thermal fatigue (Thermal transients, cycling, striping)

\*\* Note that there are no reported degradations affecting cold leg, hot leg, or crossover leg piping in U.S. Westinghouse PWR plants. All the reported cracks and leaks occurred in 1½ to 2½" diameter piping.

In general, a point estimate of the frequency of pipe failure events,  $\lambda$ , is given by the following expression:

$$\lambda = \frac{n_F}{NT} \quad (7.8)$$

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Where:

- $n_F$  = the number of failure events including both leaks and ruptures in the service data
- $T$  = the total time over which failure events were collected
- $N$  = the number of components that provided the observed pipe failures

A point estimate of the total frequency of flaws (cracks and leaks),  $\phi$ , is given by the following expression:

$$\phi = \frac{n_C}{N \cdot T \cdot f \cdot P_{FD}} + \frac{n_F}{NT} = \frac{n_C}{N \cdot T \cdot f \cdot P_{FD}} + \lambda \quad (7.9)$$

Where:

- $n_C$  = the number of crack events
- $f$  = the fraction of welds inspected for flaws
- $P_{FD}$  = the probability that an expected weld will find an existing flaw

The other variables in Equation (7.9) are as defined above. In this equation we account for the observed cracks in the database and the fact that only a fraction of the welds in the database are inspected for this condition and that those found are subject to a finite NDE reliability. This equation also reflects the fact that each failure in the database has an additional crack that produced the failure, whose exposure parameter is the entire population of welds at risk for failure. This is based on the insight that nearly all failures are found not from NDE inspections but from independent observations. This is an important observation because all the population of welds in the surveyed data are at risk for failure observation, but only a small fraction are at risk for the observation of cracks which can only be found from NDE inspections.

If we now take the ratio of  $\phi$  to  $\lambda$ , we get an expression for the factor by which to multiply the pipe failure rate to obtain the flaw rate:

$$R_{C/F} = \frac{\phi}{\lambda} = \frac{n_C}{n_F \cdot f \cdot P_{FD}} + 1 \quad (7.10)$$

Where:

- $R_{C/F}$  = Number of cracks or flaws per pipe failure:

Point Estimates of  $R_{C/F}$  for different data sets in Table 7-5 are presented in Table 7-6. The fraction of welds inspected listed in this table is estimated as follows. The current ASME Section XI requirements are to inspect 25% of the Class 1 welds and 7.5% of the Class 2 welds and these inspection requirements call for the same welds to be inspected each inspection

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interval. When cracks or significant flaws are found, the ASME code requires that an expanded search be made, however the frequency of flaws and failures is so rare that this requirement adds very few additional inspections. Using data from Braidwood Units 1 and 2 on the number of Class 1 and Class 2 welds of 3109 and 3448, respectively, which is assumed to be representative of the relative populations of Class 1 and 2 welds in this PWR service data, the following estimate of the parameter  $f$  is obtained:

$$f = \frac{3109(.25) + 3448(.075)}{(3109 + 3448)} = .157 \quad \text{for PWR Class 1 and 2 piping} \quad (7.11)$$

$$f = .25 \quad \text{for BWR Class 1 piping}$$

**Table 7-6**  
**Estimates of the Crack to Leak Ratio for Various Damage Mechanisms in PWR Plants**

Parameter	Damage Mechanism					
	PWR-SC	PWR-D&C	PWR-TF	PWR-Non-SC	BWR RCS-SC	
Number of Cracks, $n_c$	28	6	8	16	212	
Number of Failures, $n_F$	71	17	18	50	109	
Fraction of welds inspected, $f$	0.157	0.157	0.157	0.157	0.25	
$R_{C/F}$	$P_{FD}=.50$	6.01	5.48	6.64	6.08	16.56
	$P_{FD}=.75$	4.34	3.99	4.76	4.39	11.37
	$P_{FD}=.90$	3.78	3.49	4.14	3.82	9.19

Hence, even though the observed number of cracks in Table 7-4 for PWRs is only 40% of the observed number of failures, the flaw occurrence rates are actually much higher than the weld failure rates. Also the BWR data for IGSCC in Class 1 piping show that the underlying crack to leak ratios for this case are nearly 3 times as high as for Non SC mechanisms in PWRs. The estimates for the ratio of flaws to total pipe failures obtained in Table 7-6 reflect the different degrees to which pipe welds are exposed to the class of events that have been reported. The evidence for the observed crack frequency is based on an exposed weld population that is only about 16% of the exposed weld population for pipe failures in PWR Class 1 and 2 systems, and 25% for BWR Class 1 systems, as only the inspected welds are available to produce this evidence. This fact combined with the additional implicit crack that must have existed prior to each of the pipe leak events, creates an underlying failure rate for cracks that is at least 4 times higher than the underlying failure rate for leaks and ruptures for non-SC mechanisms and more than 9 times higher for IGSCC in BWR Class 1 systems. For DNPS Units 2 and 3, the crack to leak ratio that was used is 9.19 for IGSCC piping that is part of the IGSCC augmented program, and 4.28 for other damage mechanisms in all piping within the scope of the RI-ISI program. Even though the non-SC number was derived from PWR piping systems, the differences between PWRs and BWRs outside the RCS piping is not regarded as significant. In addition, the BWR IGSCC ratio is based on a POD value of 0.90 versus the 0.75 value used for the other damage mechanisms due to the heightened focus on IGSCC that was brought about by the augmented program for IGSCC.

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#### Probability per inspection interval that the pipe element will be inspected ( $P_{FI}$ )

As noted in Table 7-3, this parameter is set to 1.0 if the element is selected for inspection and 0.0 if it is not. Since the Markov model is evaluated separately for the Section XI and RI-ISI programs, this parameter is set to the appropriate value for each weld for each of the inspection programs.

#### Probability per inspection that an existing flaw will be detected ( $P_{FD}$ )

These values are set based on engineering judgment to reflect the probability that an inspected weld with a crack or flaw that exceeds the critical flaw size will be detected in each in-service inspection. The values used in the DNPS risk impact assessment for different situations are listed in Table 7-7. Each value is scaled by a factor  $F_A$  which is the fraction of the weld that is accessible. This value is normally 1.0 as the accessibility of the weld is one of the factors that is taken into account in the element selection process as discussed more fully in Section 6. The value of  $F_A$  is assumed to be 1.0 for all DNPS welds, since there is no information available at this time about limited accessibility. As experience in performing the RI-ISI inspections is accumulated, these values are subject to change and should be updated in the next RI-ISI program update. It is emphasized however, that as explained more fully in Reference [7-4], the Markov results are not sensitive to small variations in this parameter in the range of 0.7 to 1.0. For welds in the existing ISI program, the factor  $F_A$  is conservatively set to 1.0 which will tend to overstate the importance of the existing inspections.

#### Probability per detection interval that an existing leak will be detected ( $P_{LD}$ ) and Leak Detection Interval ( $T_{LD}$ )

According to the ASME Code Section XI, all Class 1 piping systems must be inspected for leaks by performing a system leak test and observing for leaks at least once per refueling cycle. For Class 2 piping, the requirement is to perform these leak tests once per ISI inspection period. In between these leak tests there are other opportunities to identify leaks via routine plant walkdowns and other test and maintenance activities on the piping systems that occur much more frequently than the ASME Section XI imposed leak tests. The following default values are used for all segments in this evaluation:

$$P_{LD} = .90 \quad (7.12)$$

$$T_{LD} = 1.5 \text{ years} \quad (7.13)$$

The same values are used for both Class 1 and 2 segments and were not varied between the Section XI and RI-ISI evaluation cases. Since the Markov model results are not sensitive to variations in this parameter and because the parameter does not differentiate between the ASME Section XI and RI-ISI programs, it was not necessary to develop segment dependent inputs for this parameter.

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**Table 7-7**  
**Estimation of the Probability of Detection of Inspected Elements with Flaws,  $P_{FD}$**

Applicability	Assumed value of $P_{FD}$	Basis
EPRI RI-ISI of Element in Carbon Steel pipe subject to thermal fatigue	$P_{FD} = 0.90 \cdot F_A$ ; $F_A$ = fraction of element that is accessible to inspection	EPRI RI-ISI procedure calls for expanded inspection zone for elements susceptible to TF, assumption used in NRC reviewed Markov applications [7-2] and [7-4]
EPRI RI-ISI of element in Stainless steel pipe subject to thermal fatigue	$P_{FD} = 0.80 \cdot F_A$ ; $F_A$ = fraction of element that is accessible to inspection	Carbon steel value reduced slightly to reflect insights from EPRI NDE qualification program [7-14]
EPRI RI-ISI of element subject to other damage mechanism subject to inservice inspection	$P_{FD} = 0.75 \cdot F_A$ ; $F_A$ = fraction of element that is accessible to inspection	Inspection for cause principle expected to pick up most flaws above critical size but no expanded volumes as in TF
EPRI RI-ISI of element subject to design and construction errors only	$P_{FD} = 0.50 \cdot F_A$ ; $F_A$ = fraction of element that is accessible to inspection	Since there is no inspection for cause principle to apply, high confidence in detection cannot be assured
Section XI ISI of element due to (unknown) damage mechanism	$P_{FD} = 0.50 \cdot F_A$ ; $F_A$ = fraction of element that is accessible to inspection	Since there is no inspection for cause principle to apply, high confidence in detection cannot be assured

These values are considered to be conservative for the following reasons. Some leaks from the RCS, FW, and MS systems will be instantaneously alarmed in the control room due to high radiation levels from the release of circulating coolant activity. Other leaks will be picked up in operator walk-arounds that occur either hourly or once per shift according to the procedures. Still other leaks will be detected rather promptly via sump alarms. Hence only some leaks need to wait for the system leak test to become visible to the plant personnel. While leaks in some locations may be difficult to detect, most leaks will be identified well within the 1.5 years assumed.

#### Flaw inspection interval, mean time between in service inspections ( $T_{FI}$ )

This parameter is fixed in the ASME Code Section XI to once per 10 years. The risk informed procedure used in this evaluation proposes no change in this inspection interval. If in the future a different inspection interval is selected, this parameter can easily be changed in the Markov model calculations.

#### Mean time to repair the piping element given detection of a critical flaw or leak ( $T_R$ )

This parameter is set to 200 hours which translates into a little over 8 days. This is a conservative value for most situations, but as discussed in Reference [7-4] the results of the Markov model are not sensitive to this parameter in the slightest. Increasing this value to 1000 hours would not change the results appreciably since this term must be compared to the mean time between pipe failures which in a given pipe location is typically thousands of years.

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#### 7.3 RESULTS FOR DNPS UNIT 2

The purpose of this section is to present the results of the risk impact assessment that was obtained for each of the piping segments within the scope of the RI-ISI evaluation, which includes all piping that is currently within the scope of the ASME Section XI inspection program for ASME Class 1 and 2 piping.

The procedure for performing this evaluation in the EPRI topical Report was presented earlier in this section as Figure 7-2. This flow chart describes how the risk evaluation is performed on a system by system basis. This flow chart provides the option of performing the analysis using a combination of qualitative arguments, bounding analyses, and realistic analyses. In this evaluation, a realistic analysis is performed for all piping segments. The premise of Figure 7-2 is that the work to perform the assessment could be minimized by avoiding the need for realistic analysis to those segments in High or Medium risk locations in the risk matrix and for which there is a net reduction in the number of welds to be inspected in the RI-ISI program. The authors of this evaluation have determined that once the data is developed to realistically quantify these segments, there is only a small additional effort to realistically quantify all the piping segments. Hence, realistic estimates of CDF and LERF changes due to changes in the RI-ISI program relative to the existing ASME Section XI program were obtained for all pipe segments in the scope of the RI-ISI evaluation. This scope covers Class 1 and 2 piping within the current Section XI inspection program scope in the following systems at DNPS Units 2 and 3:

- CRD System, including the Scram Discharge Volume (CRDSD)
- ECCS Systems, including CS, LPCI, RHS, and RHSP
- FW System
- HPCI System
- MS System, including isolation condenser (ISCOCR, ISCOSS)
- RCS System including Reactor Recirculation (RR), Reactor Head Vent (RHV), Jet Pump (JPIA, JPIB), and RPV level instrument nozzles (LVLA, LVLB, UVLA, UVLB)
- RWCU System
- SBLC System
- SDC System

Summaries and system by system results for CDF and LERF impacts are provided in the subsections below. The calculations were performed using an Excel spreadsheet that executes the solutions of the Markov model based on specified inputs that were described in the first part of this section. [7-15]

##### 7.3.1 Summary of Results by System

A tabulation of the results by system for DNPS Unit 2 is provided in Tables 7-8 and 7-9. The former table includes results for the pipe failure frequencies, pipe rupture frequencies, and

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changes to these frequencies associated with three inspection strategies: maintaining the current Section XI based program, incorporating the Risk Informed program that is described in this report, and the case of removing in-service inspection entirely. The results for system CDF and system  $\Delta$ CDF are plotted in Figures 7-4 and 7-5, respectively, and those for  $\Delta$ LERF are in Figure 7-6. The following points summarize the overall results of this evaluation for DNPS Unit 2:

- Each system at DNPS Unit 2 was found to meet the acceptance criteria for changes in CDF and LERF by a large margin. These acceptance criteria are less than  $1\text{E-}7$  per year per system for increases in CDF and less than  $1\text{E-}8$  per year per system for increases in LERF.
- The acceptance criteria for changes in CDF and LERF are also met for the extreme case of total elimination of the ISI exams currently performed under Section XI, with the understanding that system leak tests mandated by Section XI would still be performed. This is included as a sensitivity case as NRC will not accept total elimination of Section XI exams as noted in Regulatory Guide 1.178 [7-16].
- The results of the risk impact assessment showed small changes in pipe failure frequency for each system due to the changes in the proposed inspection program where failure frequency includes both leaks and ruptures. These included small decreases in failure frequency for the FW system and small increases in the remaining systems with a net increase of about  $3.1\text{E-}05$  per reactor year across all Class 1 and 2 systems.
- The results showed acceptably small increases in pipe rupture frequency for all systems except for FW, which exhibited a small decrease, with a net increase in rupture frequency of about  $8.4\text{E-}05$  per reactor year across all Class 1 and 2 systems.
- The results for the predicted change in CDF and LERF due to implementation of the risk informed program showed small decreases in both of these risk metrics for the ECCS and SDC systems, and small increases for the remaining systems. The total change in CDF and LERF due to the proposed ISI changes in all the Class 1 and 2 systems combined was found to be about  $3.14\text{E-}9$  and  $7.57\text{E-}10$ , respectively, which are well within the guidelines for risk significance set forth in Regulatory Guide 1.174 and the EPRI Topical Report.

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Table 7-8  
 Impact of RI-ISI and No Inspections on System Piping Failure and Rupture Frequency for DNPS Unit 2 Systems

System	System Failure Frequency Events/Reactor-Year			Δ Failure Frequency Events/Reactor-Year		System Rupture Frequency Events/Reactor-Year			Δ Rupture Frequency Events/Reactor-Year	
	Section XI	RI-ISI	No Inspection	RI-ISI	No Inspection	Section XI	RI-ISI	No Inspection	RI-ISI	No Inspection
CRD	9.50E-05	9.56E-05	9.58E-05	6.33E-07	7.92E-07	5.43E-05	5.63E-05	5.68E-05	1.94E-06	2.43E-06
ECCS	1.43E-04	1.43E-04	1.45E-04	4.33E-07	1.93E-06	5.58E-05	5.72E-05	6.03E-05	1.34E-06	4.42E-06
FW	2.57E-04	2.55E-04	2.64E-04	-2.20E-06	6.71E-06	1.27E-04	1.25E-04	1.47E-04	-2.17E-06	1.92E-05
HPCI	3.85E-05	3.86E-05	3.89E-05	1.16E-07	4.18E-07	1.53E-05	1.56E-05	1.62E-05	3.03E-07	9.51E-07
MS	3.69E-04	3.70E-04	3.80E-04	6.73E-07	1.04E-05	1.78E-04	1.85E-04	2.08E-04	7.18E-06	2.98E-05
RCS	1.88E-03	1.91E-03	1.92E-03	2.86E-05	3.47E-05	7.20E-04	7.87E-04	8.00E-04	6.71E-05	7.98E-05
RWCU	7.29E-05	7.37E-05	7.40E-05	7.92E-07	1.11E-06	4.05E-05	4.29E-05	4.39E-05	2.43E-06	3.40E-06
SBLC	1.21E-04	1.23E-04	1.26E-04	2.03E-06	4.17E-06	5.14E-05	5.73E-05	6.20E-05	5.82E-06	1.06E-05
SDC	2.42E-05	2.42E-05	2.49E-05	1.68E-08	6.88E-07	8.95E-06	9.20E-06	1.05E-05	2.49E-07	1.59E-06
Total	3.00E-03	3.04E-03	3.07E-03	3.11E-05	6.09E-05	1.25E-03	1.34E-03	1.40E-03	8.42E-05	1.52E-04

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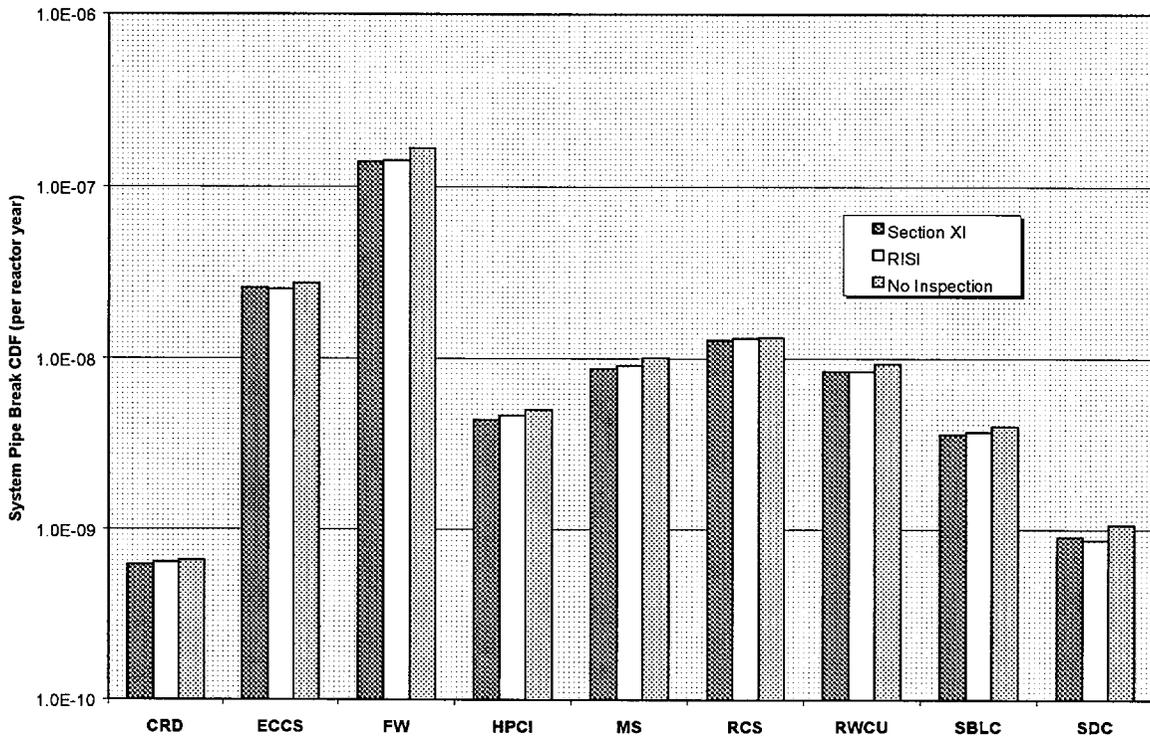
Excerpted from Dresden Tier 2 RI-ISI Documentation  
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Table 7-9  
 Impact of RI-ISI and No Inspections on CDF and LERF Due to Pipe Ruptures for DNPS Unit 2 Systems

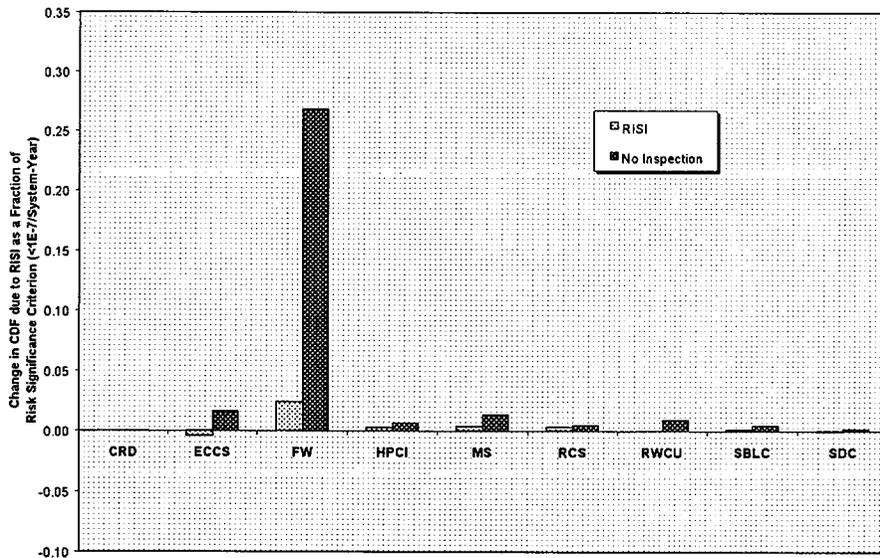
System	System CDF Events/Reactor-Year			Δ CDF Events/Reactor-Year			Δ LERF Event/Reactor-Year		
	Section XI	RI-ISI	No Inspection	RI-ISI	No Inspection	Acceptance Criterion	RI-ISI	No Inspection	Acceptance Criterion
CRD	6.23E-10	6.42E-10	6.60E-10	1.94E-11	3.74E-11	<1.00E-07	1.94E-11	2.65E-11	<1.00E-08
ECCS	2.59E-08	2.55E-08	2.76E-08	-4.21E-10	1.62E-09	<1.00E-07	-3.22E-10	9.50E-10	<1.00E-08
FW	1.39E-07	1.42E-07	1.66E-07	2.41E-09	2.68E-08	<1.00E-07	3.33E-10	2.12E-09	<1.00E-08
HPCI	4.37E-09	4.66E-09	5.02E-09	2.90E-10	6.51E-10	<1.00E-07	2.08E-11	1.81E-10	<1.00E-08
MS	8.74E-09	9.13E-09	1.01E-08	3.84E-10	1.33E-09	<1.00E-07	2.59E-10	1.02E-09	<1.00E-08
RCS	1.28E-08	1.32E-08	1.33E-08	3.34E-10	4.85E-10	<1.00E-07	3.43E-10	4.33E-10	<1.00E-08
RWCU	8.44E-09	8.47E-09	9.35E-09	3.63E-11	9.11E-10	<1.00E-07	1.44E-11	8.89E-10	<1.00E-08
SBLC	3.60E-09	3.72E-09	4.03E-09	1.25E-10	4.36E-10	<1.00E-07	1.25E-10	4.35E-10	<1.00E-08
SDC	9.06E-10	8.69E-10	1.06E-09	-3.73E-11	1.58E-10	<1.00E-07	-3.55E-11	1.50E-10	<1.00E-08
Total	2.05E-07	2.08E-07	2.37E-07	3.14E-09	3.24E-08	<1.00E-06	7.57E-10	6.20E-09	<1.00E-07

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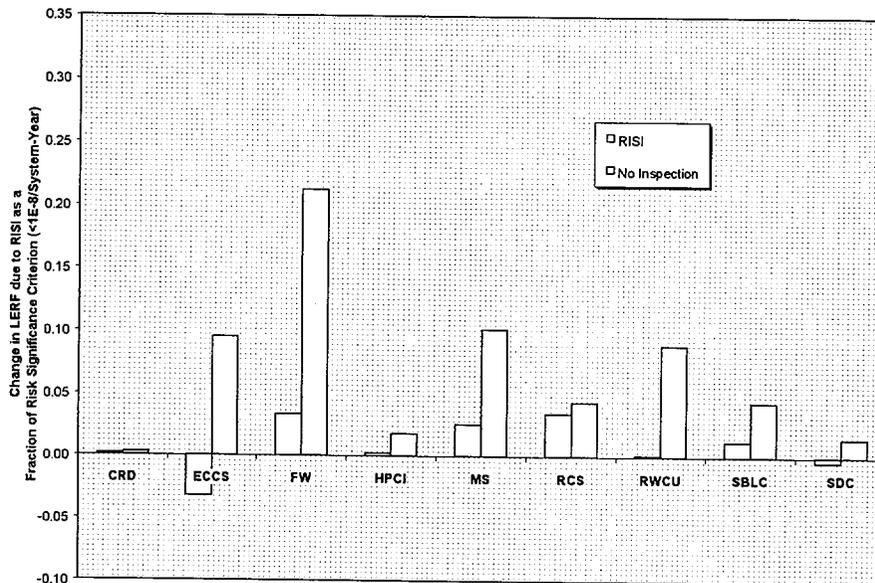
**Figure 7-4**  
Pipe Rupture CDF Results for DNPS Unit 2 Systems



**Figure 7-5**  
Change in Pipe Rupture CDF for DNPS Unit 2 Systems as a Fraction of EPRI Risk Significance Criterion

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**Figure 7-6**  
Change in Pipe Rupture LERF for DNPS Unit 2 Systems as a Fraction of EPRI Risk Significance Criterion

The changes in CDF and LERF due to inspection program changes are the net result of changes that are made to each inspection location from three sources: If a weld is added to the inspection program, the CDF and LERF contributions decrease due to a reduced pipe rupture frequency at that location in comparison to the uninspected case. If a weld is retained in the inspection program, the CDF and LERF contributions decrease due a reduced pipe rupture frequency associated with an inspection for cause principle. In the risk informed program, the applicable damage mechanisms are identified and this is expected to result in a reduced pipe rupture frequency due to an increased reliability of the NDE inspection (referred to by NDE specialists as "probability of detection"). If an inspected weld is removed from the risk informed program, the CDF and LERF contributions from the weld will increase due to an increased pipe rupture frequency. A summary of the net numbers of inspections with the Section XI program and with the proposed RI-ISI program by system and risk category is provided in Table 7-10. For the SDC system which had no net change in the number of inspections in the High or Medium risk segments, there was still a net decrease in CDF and LERF due to selection of more risk informed locations in comparison with the Section XI exam locations. This was also the case with ECCS which had a net decrease of 5 exams in High risk segments and an increase of 3 exams in Medium risk segments, yet experienced a net decrease in CDF and LERF due to a more risk informed selection. For the remaining systems, there were net decreases in exams in these risk regions of the matrix but the resulting increases in CDF and LERF were very small and a minute fraction of the EPRI RI-ISI risk significance thresholds. This explains the relative contributions that each system makes to the change in CDF in Figure 7-5 and to the change in LERF in Figure 7-6.

## Attachment 2

### Excerpted from Dresden Tier 2 RI-ISI Documentation Risk Impact Of Implementing Risk Informed Inspection Program

**Table 7-10  
Number of Inspections by Risk Category for DDNPS Unit 2**

System	High Risk						Medium Risk				Low Risk		All Risk Categories	
	Category 1		Category 2		Category 3		Category 4		Category 5		Category 6 or 7			
	Sec. XI	RI-ISI	Sec. XI	RI-ISI	Sec. XI	RI-ISI	Sec. XI	RI-ISI	Sec. XI	RI-ISI	Sec. XI	RI-ISI	Sec. XI	RI-ISI
CRD <sup>1</sup>							1	1			4	0	5	1
ECCS <sup>2</sup>			19	14			13	16	0	0	39	0	71	30
FW	8	7			0	0							8	7
HPCI							11	8	2	3	6	0	19	11
MS <sup>3</sup>			3	3	5	6	32	13	1	1	19	0	60	23
RHV									0	0	18	0	18	0
RCS <sup>4</sup>			0	0			1	2	6	2	19	0	26	4
RWCU	0	0	0	0	0	0	4	2			3	0	7	2
SBLC							4	3	5	2	4	0	13	5
SDC			12	12							5	0	17	12
<b>TOTAL</b>	<b>8</b>	<b>7</b>	<b>34</b>	<b>29</b>	<b>5</b>	<b>6</b>	<b>66</b>	<b>45</b>	<b>14</b>	<b>8</b>	<b>117</b>	<b>0</b>	<b>244</b>	<b>95</b>

1. Includes scram discharge volume (CRDSD).
2. Includes CS, LPCI, RHS, and RHSP.
3. Includes isolation condenser (ISCOCR, ISCOSS).
4. Includes jet pump and RPV level instrument nozzles (JPIA, JPIB, LVLA, LVLB, UVLA, UVLB).

NOTE: This table provides a comparison of the RI-ISI element selection to the original ASME Section XI program. The total number of inspections is significantly lower for the RI-ISI program. Some RI-ISI inspection locations are new when compared to the Section XI program, i.e., they were previously not addressed.

## Attachment 2

### Excerpted from Dresden Tier 2 RI-ISI Documentation Risk Impact Of Implementing Risk Informed Inspection Program

#### 7.3.2 CDF Results for DNPS Unit 2 Systems

The details of the CDF and LERF results for DNPS Unit 2 are provided in Attachment 7A. For each system, there are three tables provided – one with the label “-input” for the Markov Model input data, one with the label “-qual” for the results of the qualitative risk impact assessment, and a third with the label “-quant” with the results of the quantitative evaluation. System codes described in the following sections are used to identify the system category for the evaluation. These tables fully document the inputs and results from the Markov model evaluation of each piping segment within the scope of the RI-ISI evaluation for DNPS Unit 2.

##### 7.3.2.1 CDF and LERF Results for CRD System

The Markov Model input data for the DNPS Unit 2 CRD System is presented in Table DN2-CRD-input. The qualitative risk analysis for this system is provided in Table DN2-CRD-qual, and the quantitative assessment of pipe rupture frequency, CDF, change in CDF ( $\Delta$ CDF), LERF, and change in LERF ( $\Delta$ LERF) are provided in Table DN2-CRD-quant. These quantitative and qualitative analyses are provided on a segment by segment basis and cover three cases: current Section XI based ISI program, RI-ISI program based on the element selection described in this report, and the case of total elimination of inspections.

##### 7.3.2.2 CDF and LERF Results for ECCS Systems

The Markov Model input data for the DNPS Unit 2 ECCS System is presented in Table DN2-ECCS-input. The qualitative risk analysis for this system is provided in Table DN2-ECCS-qual, and the quantitative assessment of pipe rupture frequency, CDF, change in CDF ( $\Delta$ CDF), LERF, and change in LERF ( $\Delta$ LERF) are provided in Table DN2-ECCS-quant. These quantitative and qualitative analyses are provided on a segment by segment basis and cover three cases: current Section XI based ISI program, RI-ISI program based on the element selection described in this report, and the case of total elimination of inspections.

##### 7.3.2.3 CDF and LERF Results for FW System

The Markov Model input data for the DNPS Unit 2 FW System is presented in Table DN2-FW-input. The qualitative risk analysis for this system is provided in Table DN2-FW-qual, and the quantitative assessment of pipe rupture frequency, CDF, change in CDF ( $\Delta$ CDF), LERF, and change in LERF ( $\Delta$ LERF) are provided in Table DN2-FW-quant. These quantitative and qualitative analyses are provided on a segment by segment basis and cover three cases: current Section XI based ISI program, RI-ISI program based on the element selection described in this report, and the case of total elimination of inspections.

##### 7.3.2.4 CDF and LERF Results for HPCI System

The Markov Model input data for the DNPS Unit 2 HPCI System is presented in Table DN2-HPCI-input. The qualitative risk analysis for this system is provided in Table DN2-HPCI-qual, and the quantitative assessment of pipe rupture frequency, CDF, change in CDF ( $\Delta$ CDF), LERF, and change in LERF ( $\Delta$ LERF) are provided in Table DN2-HPCI-quant. These quantitative and qualitative analyses are provided on a segment by segment basis and cover

## Attachment 2

### Excerpted from Dresden Tier 2 RI-ISI Documentation Risk Impact Of Implementing Risk Informed Inspection Program

three cases: current Section XI based ISI program, RI-ISI program based on the element selection described in this report, and the case of total elimination of inspections.

#### 7.3.2.5 CDF and LERF Results for MS System

The Markov Model input data for the DNPS Unit 2 MS System is presented in Table DN2-MS-input. The qualitative risk analysis for this system is provided in Table DN2-MS-qual, and the quantitative assessment of pipe rupture frequency, CDF, change in CDF ( $\Delta$ CDF), LERF, and change in LERF ( $\Delta$ LERF) are provided in Table DN2-MS-quant. These quantitative and qualitative analyses are provided on a segment by segment basis and cover three cases: current Section XI based ISI program, RI-ISI program based on the element selection described in this report, and the case of total elimination of inspections.

#### 7.3.2.6 CDF and LERF Results for RCS System

The Markov Model input data for the DNPS Unit 2 RCS System is presented in Table DN2-RCS-input. The qualitative risk analysis for this system is provided in Table DN2-RCS-qual, and the quantitative assessment of pipe rupture frequency, CDF, change in CDF ( $\Delta$ CDF), LERF, and change in LERF ( $\Delta$ LERF) are provided in Table DN2-RCS-quant. These quantitative and qualitative analyses are provided on a segment by segment basis and cover three cases: current Section XI based ISI program, RI-ISI program based on the element selection described in this report, and the case of total elimination of inspections.

#### 7.3.2.7 CDF and LERF Results for RWCU System

The Markov Model input data for the DNPS Unit 2 RWCU System is presented in Table DN2-RWCU-input. The qualitative risk analysis for this system is provided in Table DN2-RWCU-qual, and the quantitative assessment of pipe rupture frequency, CDF, change in CDF ( $\Delta$ CDF), LERF, and change in LERF ( $\Delta$ LERF) are provided in Table DN2-RWCU-quant. These quantitative and qualitative analyses are provided on a segment by segment basis and cover three cases: current Section XI based ISI program, RI-ISI program based on the element selection described in this report, and the case of total elimination of inspections.

#### 7.3.2.8 CDF and LERF Results for SBLC System

The Markov Model input data for the DNPS Unit 2 SBLC System is presented in Table DN2-SBLC-input. The qualitative risk analysis for this system is provided in Table DN2-SBLC-qual, and the quantitative assessment of pipe rupture frequency, CDF, change in CDF ( $\Delta$ CDF), LERF, and change in LERF ( $\Delta$ LERF) are provided in Table DN2-SBLC-quant. These quantitative and qualitative analyses are provided on a segment by segment basis and cover three cases: current Section XI based ISI program, RI-ISI program based on the element selection described in this report, and the case of total elimination of inspections.

#### 7.3.2.9 CDF and LERF Results for SDC System

The Markov Model input data for the DNPS Unit 2 SDC System is presented in Table DN2-SDC-input. The qualitative risk analysis for this system is provided in Table DN2-SDC-qual, and

## Attachment 2

### Excerpted from Dresden Tier 2 RI-ISI Documentation Risk Impact Of Implementing Risk Informed Inspection Program

the quantitative assessment of pipe rupture frequency, CDF, change in CDF ( $\Delta$ CDF), LERF, and change in LERF ( $\Delta$ LERF) are provided in Table DN2-SDC-quant. These quantitative and qualitative analyses are provided on a segment by segment basis and cover three cases: current Section XI based ISI program, RI-ISI program based on the element selection described in this report, and the case of total elimination of inspections.

#### 7.4 RESULTS FOR DNPS UNIT 3

The purpose of this section is to present the results of the risk impact assessment that was obtained for each of the piping segments within the scope of the RI-ISI evaluation for DNPS Unit 3, which includes all piping that is currently within the scope of the ASME Section XI inspection program for ASME Class 1 and 2 piping.

The scope of the systems covered for Unit 3 is the same as that delineated in Section 7.4 for Unit 2. The differences in the evaluation for Unit 3 stem from the following sources:

- There are some small differences in the weld populations in Class 1 and 2 differences that are delineated in Sections 3 (Consequence Analysis) and 4 (Degradation Mechanism Assessment).
- The current Section XI exam locations in Unit 3 are not always the same as the corresponding exam locations in Unit 2. This leads to many small differences in the element selections for the RI-ISI program in each unit.

Hence, even though the consequence and degradation results in a given pipe segment are the same for both units in nearly all cases, there are many small changes in the delta risk evaluation that dictate the need for a separate risk impact assessment for each unit. The results for Unit 3 are presented in the following sections.

##### 7.4.1 Summary of Results by System

A tabulation of the results by system for DNPS Unit 3 is provided in Tables 7-11 and 7-12. The former table includes results for the pipe failure frequencies, pipe rupture frequencies, and changes to these frequencies associated with three inspection strategies: maintaining the current Section XI based program, incorporating the Risk Informed program that is described in this report, and the case of removing in-service inspection entirely. The CDF results are presented in Figure 7-7. The results for system  $\Delta$ CDF and  $\Delta$ LERF are plotted in Figures 7-8 and 7-9, respectively. The following points summarize the overall results of this evaluation for DNPS Unit 3 which are essentially the same results as obtained for Unit 2:

- Each system at DNPS Unit 3 was found to meet the acceptance criteria for changes in CDF and LERF by a large margin. These acceptance criteria are less than 1E-7 per year per system for increases in CDF and less than 1E-8 per year per system for increases in LERF.

The acceptance criteria for changes in CDF and LERF are also met for the extreme case of total elimination of the ISI exams currently performed under Section XI, with the understanding that system leak tests mandated by Section XI would still be performed.

## Attachment 2

### Excerpted from Dresden Tier 2 RI-ISI Documentation Risk Impact Of Implementing Risk Informed Inspection Program

This is included as a sensitivity case as NRC will not accept total elimination of Section XI exams as noted in Regulatory Guide 1.178 [7-16].

- The results of the risk impact assessment showed small changes in pipe failure frequency for each system due to the changes in the proposed inspection program where failure frequency includes both leaks and ruptures. These included small decreases in failure frequency for the MS, and FW systems, and small increases in the remaining systems with a net increase of about  $3.7E-05$  per reactor year across all Class 1 and 2 systems.
- The results also showed small changes in pipe rupture frequency for each system due to the changes in the proposed inspection program where failure frequency includes both leaks and ruptures. These included a small decrease in rupture frequency for the FW system, and small increases in the remaining systems with a net increase of about  $9.8E-05$  per reactor year across all Class 1 and 2 systems.
- The results for the predicted change in CDF and LERF due to implementation of the risk informed program showed a small decrease in both of these risk metrics for all the SDC system, a small decrease in LERF for the MS system, and small increases in CDF and LERF for the remaining systems. The total change in CDF and LERF due to the proposed ISI changes in all the Class 1 and 2 systems combined was found to be net decreases of about  $2.9E-9$  and  $1.3E-9$ , per year respectively, which are well within the guidelines for risk significance set forth in Regulatory Guide 1.174 and the EPRI RI-ISI Topical Report.

**Attachment 2**

**Excerpted from Dresden Tier 2 RI-ISI Documentation  
Risk Impact Of Implementing Risk Informed Inspection Program**

**Table 7-11  
Impact of RI-ISI and No Inspections on System Piping Failure and Rupture Frequency for DNPS Unit 3 Systems**

System	System Failure Frequency Events/Reactor-Year			ΔFailure Frequency Events/Reactor-Year		System Rupture Frequency Events/Reactor-Year			ΔRupture Frequency Events/Reactor-Year	
	Section XI	RI-ISI	No Insp.	RI-ISI	No Insp.	Section XI	RI-ISI	No Insp.	RI-ISI	No Insp.
CRD	7.59E-05	7.67E-05	7.69E-05	7.92E-07	9.50E-07	4.27E-05	4.51E-05	4.56E-05	2.43E-06	2.92E-06
ECCS	1.36E-04	1.36E-04	1.37E-04	9.50E-07	1.58E-06	5.34E-05	5.56E-05	5.70E-05	2.26E-06	3.59E-06
FW	2.74E-04	2.72E-04	2.82E-04	-2.46E-06	7.21E-06	1.38E-04	1.35E-04	1.59E-04	-2.67E-06	2.10E-05
HPCI	3.84E-05	3.85E-05	3.88E-05	1.16E-07	3.97E-07	1.53E-05	1.56E-05	1.62E-05	3.03E-07	9.04E-07
MS	4.34E-04	4.34E-04	4.45E-04	-3.12E-07	1.05E-05	2.12E-04	2.17E-04	2.42E-04	5.26E-06	3.01E-05
RCS	1.28E-03	1.31E-03	1.32E-03	3.42E-05	4.22E-05	4.54E-04	5.33E-04	5.50E-04	7.86E-05	9.62E-05
RWCU	5.97E-05	6.06E-05	6.09E-05	9.50E-07	1.27E-06	3.22E-05	3.51E-05	3.61E-05	2.92E-06	3.89E-06
SBLC	1.75E-04	1.78E-04	1.80E-04	2.83E-06	5.12E-06	7.70E-05	8.52E-05	9.05E-05	8.25E-06	1.35E-05
SDC	1.86E-05	1.86E-05	1.91E-05	8.12E-09	5.00E-07	6.94E-06	7.11E-06	8.10E-06	1.73E-07	1.16E-06
TOTAL	2.49E-03	2.53E-03	2.56E-03	3.70E-05	6.97E-05	1.03E-03	1.13E-03	1.20E-03	9.75E-05	1.73E-04

Attachment 2

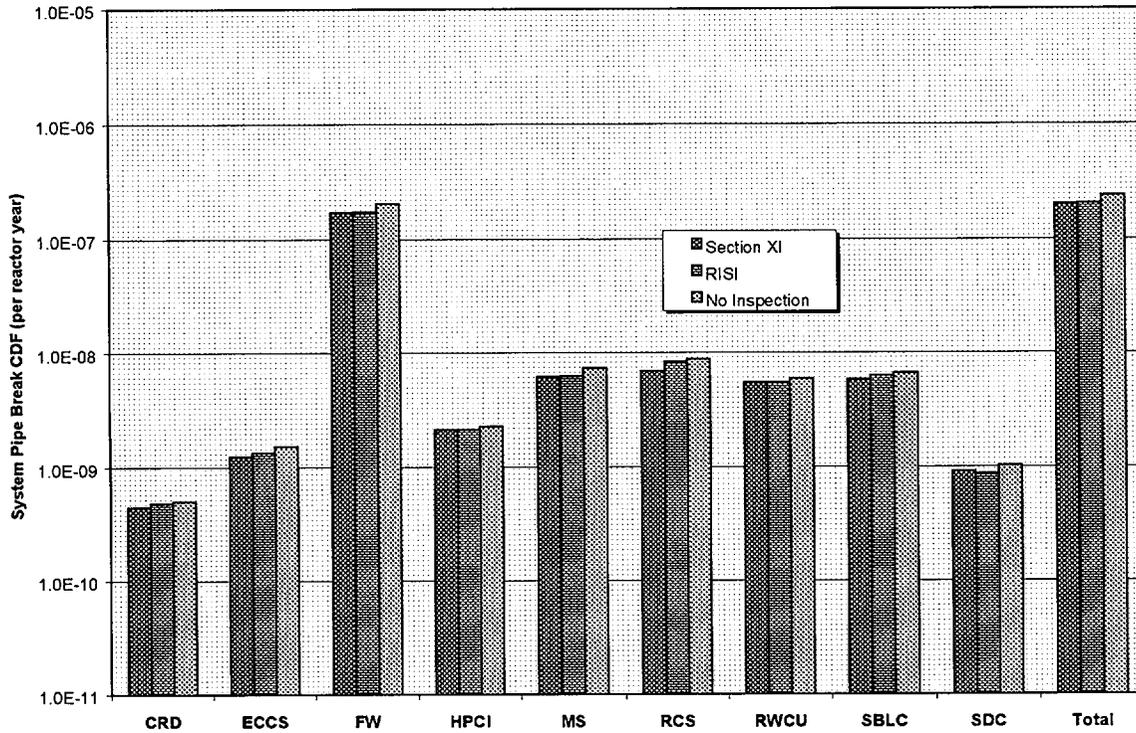
Excerpted from Dresden Tier 2 RI-ISI Documentation  
 Risk Impact Of Implementing Risk Informed Inspection Program

Table 7-12  
 Impact of RI-ISI and No Inspections on CDF and LERF Due to Pipe Ruptures for DNPS Unit 3 Systems

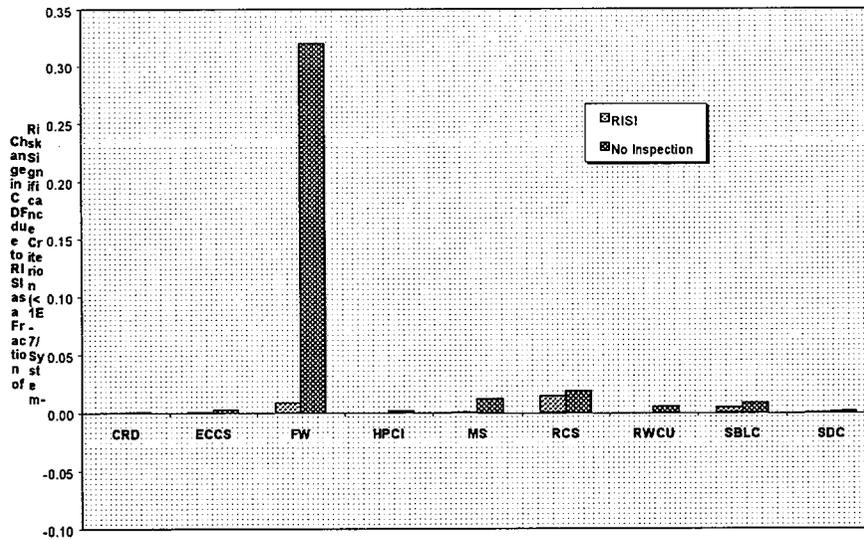
System	System CDF Events/Reactor-Year			ΔCDF Events/Reactor-Year			ΔLERF Events/Reactor-Year		
	Section XI	RI-ISI	No Inspection	RI-ISI	No Inspection	Acceptance Criterion	RI-ISI	No Inspection	Acceptance Criterion
CRD	4.47E-10	4.84E-10	5.02E-10	3.74E-11	5.53E-11	<1.00E-07	2.65E-11	3.35E-11	<1.00E-08
ECCS	1.24E-09	1.34E-09	1.53E-09	9.76E-11	2.87E-10	<1.00E-07	4.90E-11	1.51E-10	<1.00E-08
FW	1.71E-07	1.72E-07	2.03E-07	9.02E-10	3.20E-08	<1.00E-07	2.32E-10	2.47E-09	<1.00E-08
HPCI	2.12E-09	2.12E-09	2.29E-09	1.05E-12	1.63E-10	<1.00E-07	5.60E-13	1.60E-11	<1.00E-08
MS	6.21E-09	6.27E-09	7.42E-09	5.98E-11	1.20E-09	<1.00E-07	-7.19E-11	9.88E-10	<1.00E-08
RCS	6.95E-09	8.36E-09	8.85E-09	1.41E-09	1.90E-09	<1.00E-07	6.95E-10	9.02E-10	<1.00E-08
RWCU	5.51E-09	5.51E-09	5.97E-09	3.20E-12	4.59E-10	<1.00E-07	3.15E-12	4.48E-10	<1.00E-08
SBLC	5.87E-09	6.30E-09	6.70E-09	4.24E-10	8.31E-10	<1.00E-07	4.24E-10	8.31E-10	<1.00E-08
SDC	9.17E-10	8.73E-10	1.03E-09	-4.37E-11	1.09E-10	<1.00E-07	-4.26E-11	1.04E-10	<1.00E-08
TOTAL	2.00E-07	2.03E-07	2.37E-07	2.89E-09	3.70E-08	<1.00E-06	1.32E-09	5.95E-09	<1.00E-07

## Attachment 2

### Excerpted from Dresden Tier 2 RI-ISI Documentation Risk Impact Of Implementing Risk Informed Inspection Program



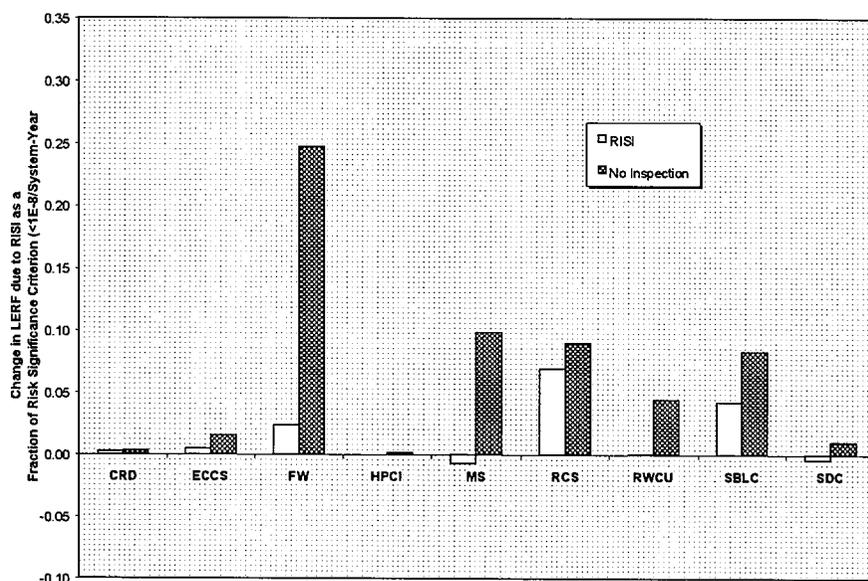
**Figure 7-7**  
Pipe Rupture CDF for DNPS Unit 3 Systems for RI-ISI, Section XI ISI, and No ISI



**Figure 7-8**  
Change in Pipe Rupture CDF for DNPS Unit 3 Systems as a Fraction of EPRI Risk Significance Criterion

## Attachment 2

### Excerpted from Dresden Tier 2 RI-ISI Documentation Risk Impact Of Implementing Risk Informed Inspection Program



**Figure 7-9**  
Change in Pipe Rupture LERF for DNPS Unit 3 Systems as a Fraction of EPRI Risk Significance Criterion

The changes in CDF and LERF due to inspection program changes are the net result of changes that are made to each inspection location from three sources: If a weld is added to the inspection program, the CDF and LERF contributions decrease due to a reduced pipe rupture frequency at that location in comparison to the uninspected case. If a weld is retained in the inspection program, the CDF and LERF contributions decrease due a reduced pipe rupture frequency associated with an inspection for cause principle. If an inspected weld is removed from the risk informed program, the CDF and LERF contributions from the weld will increase due to an increased pipe rupture frequency. The net change in CDF and LERF from all three sources reflects the net change from all three sources of risk change. There were many examples of segment in which the number of exams is reduced but the probability of detection of the retained exams was increased due to the inspection for cause principle. In the risk informed program, the applicable damage mechanisms are identified and this is expected to result in a reduced pipe rupture frequency due to an increased reliability of the NDE inspection (referred to by NDE specialists as "probability of detection"). A summary of the net numbers of inspections with the Section XI program and with the proposed RI-ISI program by system and risk category is provided in Table 7-13. The changes indicated in this table in the High and Medium Risk regions of the EPRI RI-ISI risk matrix helps to explain the relative contributions that each system makes to the change in CDF in Tables 7-11 and 7-12.

#### 7.4.2 CDF and LERF Results for DNPS Unit 3 Systems

The CDF and LERF results for DNPS Unit 3 are fully detailed in Attachment 7B.

**Attachment 2**

**Excerpted from Dresden Tier 2 RI-ISI Documentation  
Risk Impact Of Implementing Risk Informed Inspection Program**

**Table 7-13  
Number of Inspections by Risk Category for DNPS Unit 3**

System	High Risk						Medium Risk				Low Risk		All Risk Categories	
	Category 1		Category 2		Category 3		Category 4		Category 5		Category 6 or 7		Sec. XI	RI-ISI
	Sec. XI	RI-ISI	Sec. XI	RI-ISI	Sec. XI	RI-ISI	Sec. XI	RI-ISI	Sec. XI	RI-ISI	Sec. XI	RI-ISI		
CRD <sup>1</sup>							2	1			4	0	6	1
ECCS <sup>2</sup>			11	7			18	15	0	0	41	0	70	22
FW	9	7			0	0							9	7
HPCI			0	0			6	7	2	3	10	0	18	10
MS <sup>3</sup>			1	1	8	10	28	12	2	1	15	0	54	24
RHV							1	1	0	0	19	0	20	1
RCS <sup>4</sup>							34	9	3	1	13	0	50	10
RWCU	0	0			0	0	2	2			6	0	8	2
SBLC							8	4	5	2	6	0	19	6
SDC			10	11							6	0	16	11
<b>TOTAL</b>	<b>9</b>	<b>7</b>	<b>22</b>	<b>19</b>	<b>8</b>	<b>10</b>	<b>99</b>	<b>51</b>	<b>12</b>	<b>7</b>	<b>120</b>	<b>0</b>	<b>270</b>	<b>94</b>

1. Includes scram discharge volume (CRDSD).
2. Includes CS, LPCI, RHS, and RHSP.
3. Includes isolation condenser (ISCOCR, ISCOSS).
4. Includes Reactor Recirculation (RR), jet pump (JPIA, JPIB), RPV level instrument nozzles (LVLA, LVLB, UVLA, UVLB).

## Attachment 2

### Excerpted from Dresden Tier 2 RI-ISI Documentation Risk Impact Of Implementing Risk Informed Inspection Program

#### 7.4.2.1 CDF and LERF Results for CRD System

The Markov Model input data for the DNPS Unit 3 CRD System is presented in Table DN3-CRD-input. The qualitative risk analysis for this system is provided in Table DN3-CRD-qual, and the quantitative assessment of pipe rupture frequency, CDF, change in CDF ( $\Delta$ CDF), LERF, and change in LERF ( $\Delta$ LERF) are provided in Table DN3-CRD-quant. These quantitative and qualitative analyses are provided on a segment by segment basis and cover three cases: current Section XI based ISI program, RI-ISI program based on the element selection described in this report, and the case of total elimination of inspections.

#### 7.4.2.2 CDF and LERF Results for ECCS Systems

The Markov Model input data for the DNPS Unit 3 ECCS System is presented in Table DN3-ECCS-input. The qualitative risk analysis for this system is provided in Table DN3-ECCS-qual, and the quantitative assessment of pipe rupture frequency, CDF, change in CDF ( $\Delta$ CDF), LERF, and change in LERF ( $\Delta$ LERF) are provided in Table DN3-ECCS-quant. These quantitative and qualitative analyses are provided on a segment by segment basis and cover three cases: current Section XI based ISI program, RI-ISI program based on the element selection described in this report, and the case of total elimination of inspections.

#### 7.4.2.3 CDF and LERF Results for FW System

The Markov Model input data for the DNPS Unit 3 FW System is presented in Table DN3-FW-input. The qualitative risk analysis for this system is provided in Table DN3-FW-qual, and the quantitative assessment of pipe rupture frequency, CDF, change in CDF ( $\Delta$ CDF), LERF, and change in LERF ( $\Delta$ LERF) are provided in Table DN3-FW-quant. These quantitative and qualitative analyses are provided on a segment by segment basis and cover three cases: current Section XI based ISI program, RI-ISI program based on the element selection described in this report, and the case of total elimination of inspections.

#### 7.4.2.4 CDF and LERF Results for HPCI System

The Markov Model input data for the DNPS Unit 3 HPCI System is presented in Table DN3-HPCI-input. The qualitative risk analysis for this system is provided in Table DN3-HPCI-qual, and the quantitative assessment of pipe rupture frequency, CDF, change in CDF ( $\Delta$ CDF), LERF, and change in LERF ( $\Delta$ LERF) are provided in Table DN3-HPCI-quant. These quantitative and qualitative analyses are provided on a segment by segment basis and cover three cases: current Section XI based ISI program, RI-ISI program based on the element selection described in this report, and the case of total elimination of inspections.

#### 7.4.2.5 CDF and LERF Results for MS System

The Markov Model input data for the DNPS Unit 3 MS System is presented in Table DN3-MS-input. The qualitative risk analysis for this system is provided in Table DN3-MS-qual, and the quantitative assessment of pipe rupture frequency, CDF, change in CDF ( $\Delta$ CDF), LERF, and change in LERF ( $\Delta$ LERF) are provided in Table DN3-MS-quant. These quantitative and qualitative analyses are provided on a segment by segment basis and cover three cases:

## Attachment 2

### Excerpted from Dresden Tier 2 RI-ISI Documentation Risk Impact Of Implementing Risk Informed Inspection Program

current Section XI based ISI program, RI-ISI program based on the element selection described in this report, and the case of total elimination of inspections.

#### 7.4.2.6 CDF and LERF Results for RCS System

The Markov Model input data for the DNPS Unit 3 RCS System is presented in Table DN3-RCS-input. The qualitative risk analysis for this system is provided in Table DN3-RCS-qual, and the quantitative assessment of pipe rupture frequency, CDF, change in CDF ( $\Delta$ CDF), LERF, and change in LERF ( $\Delta$ LERF) are provided in Table DN3-RCS-quant. These quantitative and qualitative analyses are provided on a segment by segment basis and cover three cases: current Section XI based ISI program, RI-ISI program based on the element selection described in this report, and the case of total elimination of inspections.

#### 7.4.2.7 CDF and LERF Results for RWCU System

The Markov Model input data for the DNPS Unit 3 RWCU System is presented in Table DN3-RWCU-input. The qualitative risk analysis for this system is provided in Table DN3-RWCU-qual, and the quantitative assessment of pipe rupture frequency, CDF, change in CDF ( $\Delta$ CDF), LERF, and change in LERF ( $\Delta$ LERF) are provided in Table DN3-RWCU-quant. These quantitative and qualitative analyses are provided on a segment by segment basis and cover three cases: current Section XI based ISI program, RI-ISI program based on the element selection described in this report, and the case of total elimination of inspections.

#### 7.4.2.8 CDF and LERF Results for SBLC System

The Markov Model input data for the DNPS Unit 3 SBLC System is presented in Table DN3-SBLC-input. The qualitative risk analysis for this system is provided in Table DN3-SBLC-qual, and the quantitative assessment of pipe rupture frequency, CDF, change in CDF ( $\Delta$ CDF), LERF, and change in LERF ( $\Delta$ LERF) are provided in Table DN3-SBLC-quant. These quantitative and qualitative analyses are provided on a segment by segment basis and cover three cases: current Section XI based ISI program, RI-ISI program based on the element selection described in this report, and the case of total elimination of inspections.

#### 7.4.2.9 CDF and LERF Results for SDC System

The Markov Model input data for the DNPS Unit 3 SDC System is presented in Table DN3-SDC-input. The qualitative risk analysis for this system is provided in Table DN3-SDC-qual, and the quantitative assessment of pipe rupture frequency, CDF, change in CDF ( $\Delta$ CDF), LERF, and change in LERF ( $\Delta$ LERF) are provided in Table DN3-SDC-quant. These quantitative and qualitative analyses are provided on a segment by segment basis and cover three cases: current Section XI based ISI program, RI-ISI program based on the element selection described in this report, and the case of total elimination of inspections.

## Attachment 2

### Excerpted from Dresden Tier 2 RI-ISI Documentation Risk Impact Of Implementing Risk Informed Inspection Program

#### 7.5 REFERENCES

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- [7-12] ANO-2 Code Case N-578 Application Submittals, Letter #2CAN099706 dated September 30, 1997 and Letter #2CAN039808 dated March 31, 1998.
- [7-13] STPNOC, "Risk Informed Inservice Inspection Program Plan – South Texas Project Units 1 and 2", December 1999.
- [7-14] Personal communication with Steve Gosselin, Ron Gamble and Jack Cole, April 19,

## Attachment 2

### Excerpted from Dresden Tier 2 RI-ISI Documentation Risk Impact Of Implementing Risk Informed Inspection Program

2000.

[7-15] Markov model spreadsheet files (ERIN Proprietary Information):

#### DNPS Unit 2

CLERP CRD (DN2).xls	897,536	06-12-00 3:03p
CLERP ECCS (DN2).xls	1,329,152	06-12-00 3:02p
CLERP FW (DN2).xls	925,184	06-23-00
	12:19p	
CLERP HPCI (DN2).xls	944,640	06-12-00 3:00p
CLERP MS (DN2).xls	961,536	06-12-00 2:59p
CLERP RCS (DN2).xls	991,744	06-12-00 2:58p
CLERP RWCU (DN2).xls	898,560	06-23-00
	11:07a	
CLERP SBLC (DN2).xls	852,480	06-12-00 3:01p
CLERP SDC (DN2).xls	917,504	06-12-00 3:06p
CRD (DN2).xls	873,984	06-13-00 9:11a
ECCS (DN2).xls	1,419,776	06-13-00 9:14a
FW (DN2).xls	1,008,128	07-06-00 3:06p
HPCI (DN2).xls	939,520	06-13-00 9:21a
MS (DN2).xls	964,096	06-13-00 9:24a
RCS (DN2).xls	1,227,776	06-13-00 9:27a
RWCU (DN2).xls	877,056	07-06-00 2:18p
SBLC (DN2).xls	865,792	06-13-00 9:34a
SDC (DN2).xls	900,096	06-14-00 9:27a

#### DNPS Unit 3

CLERP CRD (DN3).xls	899,072	06-12-00 4:29p
CLERP ECCS (DN3).xls	1,156,608	06-12-00 4:35p
CLERP FW (DN3).xls	923,648	06-23-00 2:17p
CLERP HPCI (DN3).xls	945,152	06-12-00 4:24p
CLERP MS (DN3).xls	958,976	06-12-00 4:14p
CLERP RCS (DN3).xls	993,280	06-12-00 4:08p
CLERP RWCU (DN3).xls	898,048	06-23-00 12:58p
CLERP SBLC (DN3).xls	851,968	06-12-00 3:56p
CLERP SDC (DN3).xls	911,872	06-13-00 10:38a
CRD (DN3).xls	876,544	07-06-00 2:39p
ECCS (DN3).xls	1,748,480	07-06-00 2:52p
FW (DN3).xls	1,009,664	07-06-00 3:02p
HPCI (DN3).xls	937,984	07-06-00 3:19p
MS (DN3).xls	1,020,928	07-06-00 3:26p
RCS (DN3).xls	1,230,336	07-06-00 3:29p
RWCU (DN3).xls	874,496	07-06-00 3:33p
SBLC (DN3).xls	827,392	07-06-00 3:35p
SDC (DN3).xls	893,440	07-06-00 3:38p

## Attachment 2

### Excerpted from Dresden Tier 2 RI-ISI Documentation Risk Impact Of Implementing Risk Informed Inspection Program

- [7-16] USNRC Regulatory Guide 1.178 for Trial Use, "An Approach for Plant Specific Risk-informed Decision Making: Inservice Inspection for Piping," May 1998.
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- [7-23] Fleming, K.N. et al., EGC Risk Informed Inservice Inspection Program, Byron Units 1 and 2 Final Report (Draft), Volumes 1-3, May 2000.

**Attachment 2**

**Excerpted from Dresden Tier 2 RI-ISI Documentation  
Risk Impact Of Implementing Risk Informed Inspection Program**

**ATTACHMENT 7A**

**DNPS Unit 2 Markov Model Spreadsheets**

**[Tables for Unit 2 Feedwater System included in this attachment to the RAI response. Remaining tables for all remaining systems for Units 2 and 3 can be found in Tier 2 Documentation]**

## Attachment 2

### Excerpted from Dresden Tier 2 RI-ISI Documentation Risk Impact Of Implementing Risk Informed Inspection Program

**Table Dn2-FW-Input  
Markov Model Input Data for DNPS Unit 2 Feedwater System**

<b>SYSTEM NAME</b>	Feedwater and Condensate Systems (FWC)
<b>REACTOR VENDOR</b>	General Electric
<b>PLANT NAME AND REACTOR UNIT</b>	Dresden Reactor Unit 2

MARKOV PARAMETERS				
ARIABLE NAM	VALUE*	UNITS	DESCRIPTION	REFERENCE
RF_TF	1.69E-07	/Weld-Yr.	Rupture Frequency - Thermal Fatigue	Section 7.3 RISI Report
RF_SC	0.00E+00	/Weld-Yr.	Rupture Frequency - Stress Corrosion Cracking	Section 7.3 RISI Report
RF_DC	1.38E-06	/Weld-Yr.	Rupture Frequency - Design and Construction Defects	Section 7.3 RISI Report
RF_EC	9.30E-08	/Weld-Yr.	Rupture Frequency - Erosion Cavitation	Section 7.3 RISI Report
FR_TF	4.82E-06	/Weld-Yr.	Failure Frequency - Thermal Fatigue	Section 7.3 RISI Report
FR_SC	0.00E+00	/Weld-Yr.	Failure Frequency - Stress Corrosion Cracking	Section 7.3 RISI Report
FR_DC	7.08E-06	/Weld-Yr.	Failure Frequency - Design and Construction Defects	Section 7.3 RISI Report
FR_EC	2.84E-05	/Weld-Yr.	Failure Frequency - Erosion Cavitation	Section 7.3 RISI Report
PSUBLI1	<b>0.9</b>		Probability of Successful Leak Detection	Section 7.3 RISI Report
TSUBLI1	<b>1.5</b>		Leak Detection Interval	Section 7.3 RISI Report
PHI_FACTOR	<b>4.39</b>		Flaw/Leak Ratio - Non SC mechanisms	Section 7.3 RISI Report
SCPHI_FACTO	<b>4.34</b>		Flaw/Leak Ratio - Stress Corrosion Cracking	Section 7.3 RISI Report
TSUBFI1	<b>10</b>	Yr.	ISI Inspection Interval	ASME SEC XI
TSUBR1	<b>0.0228</b>	Yr	Time to repair damaged pipe, normally 200 hours converted to years	ASSUMPTION
RHO_L1	<b>2.00E-02</b>	/Weld-Yr.	Rupture Frequency given leakage in progress	Section 7.3 RISI Report
POD_SECXI	<b>0.50</b>		Probability of Flaw detection given inspected - Section XI exams	Section 7.3 RISI Report
POD_EPRI_TF	<b>0.90</b>		Probability of Flaw detection given inspected - EPRI TF	Section 7.3 RISI Report
POD_EPRI_SC	<b>0.75</b>		Probability of Flaw detection given inspected - EPRI SC	Section 7.3 RISI Report
POD_EPRI_D	<b>0.50</b>		Probability of Flaw detection given inspected - EPRI DC	Section 7.3 RISI Report
POD_EPRI_EC	<b>0.75</b>		Probability of Flaw detection given inspected - EPRI EC	Section 7.3 RISI Report
RS1	<b>0.1</b>		Conditional probability of medium LOCA given rupture in medium or l	ASSUMPTION
RS2	<b>0.01</b>		Conditional probability of large LOCA given rupture in large pipe	ASSUMPTION
CCDP_LLOCA	2.73E-04		Conditional probability of core damage given a Large LOCA	Section 4 RISI Report
CCDP_MLOCA	3.13E-04		Conditional probability of core damage given a Medium LOCA	Section 4 RISI Report
CCDP_SLOCA	5.58E-06		Conditional probability of core damage given a Small LOCA	Section 4 RISI Report
T	<b>40</b>	YR	Time for calculating hazard rate, usually 40 years	EPRI TR 110161
FAC-FACTOR	<b>3</b>		Factor increase in FR and RF due to FAC synergy**	ASSUMPTION

\* DEFAULT VALUES IN BOLD FONT

\*\* Segments 4,6,7,8,10,12,14,16 have 2 or more damage mechanisms; FR and RF increased by factor of 3 (Notes Provided)  
Segments 1,2,3,6,9,11,13,15: Failure rates and rupture frequencies multiplied by a factor of 3 due to FAC synergy.

**Attachment 2**

**Excerpted from Dresden Tier 2 RI-ISI Documentation  
Risk Impact Of Implementing Risk Informed Inspection Program**

Table Dn2-FW-Qual Qualitative Risk Evaluation of Pipe Segments in DNPS Unit 2 Feedwater System									
No.	Segment ID	No. Welds	Sec XI Exams	RBISI Exams	Max LOCA	Modified CDDP for Delta Risk	Damage Mechanisms	Qualitative Risk Analysis	
								EPRI Matrix	RI-ISI Risk Impact
1	3204A-18-A	9	0	0	MLOCA	3.69E-05	FAC	High [Cat. 3]	No risk Impact
2	3204A-18-B	1	0	0	NONE	6.00E-06	FAC	High [Cat. 3]	No risk Impact
3	3204A-18-C	3	0	0	NONE	1.00E-04	FAC	High [Cat. 3]	No risk Impact
4	3204B-18-A	3	0	1	NONE	1.00E-04	TASCS,FAC	High [Cat. 3]	Risk decrease due to added exams
5	3204B-18-B	8	0	0	NONE	3.00E-03	FAC	High [Cat. 1]	No risk Impact
6	3204B-18-C	2	1	0	NONE	3.00E-03	TASCS,FAC	High [Cat. 1]	Risk Increase due to reduced exams
7	3204B-18-D	1	0	0	NONE	1.00E-04	TASCS,FAC	High [Cat. 3]	No risk Impact
8	3204B-18-E	3	1	0	NONE	1.00E-04	TASCS,FAC	High [Cat. 3]	Risk Increase due to reduced exams
9	3204C-12-A	2	0	0	MLOCA	3.69E-05	FAC	High [Cat. 3]	No risk Impact
10	3204C-12-B	4	1	1	MLOCA	3.69E-05	TT,FAC	High [Cat. 3]	Risk decrease due to enhanced POD
11	3204D-12-A	2	0	0	MLOCA	3.69E-05	FAC	High [Cat. 3]	No risk Impact
12	3204D-12-B	4	2	2	MLOCA	3.69E-05	TT,FAC	High [Cat. 3]	Risk decrease due to enhanced POD
13	3204E-12-A	2	0	0	NONE	3.00E-03	FAC	High [Cat. 1]	No risk Impact
14	3204E-12-B	4	2	2	NONE	3.00E-03	TT,FAC	High [Cat. 1]	Risk decrease due to enhanced POD
15	3204F-12-A	4	0	0	NONE	3.00E-03	FAC	High [Cat. 1]	No risk Impact
16	3204F-12-B	4	1	1	NONE	3.00E-03	TT,FAC	High [Cat. 1]	Risk decrease due to enhanced POD

Attachment 2

Excerpted from Dresden Tier 2 RI-ISI Documentation  
 Risk Impact Of Implementing Risk Informed Inspection Program

Table Dn2-FW-Quant Quantitative Risk Evaluation of Pipe Segments in DNPS Unit 2 Feedwater System															
No.	Segment ID	Segment Rupture Frequency (per yr.)			Segment CDF (per yr.)			Segment ΔCDF (per yr.)		Segment LERF (per yr.)			Segment ΔLERF (per yr.)		
		Sec XI	EPRI	No Insp.	Sec XI	EPRI	No Insp.	EPRI	No Insp.	Sec XI	EPRI	No Insp.	EPRI	No Insp.	
1	3204A-18-A	1.24E-05	1.24E-05	1.24E-05	1.37E-09	1.37E-09	1.37E-09	0.00E+00	0.00E+00	5.40E-10	5.40E-10	5.40E-10	0.00E+00	0.00E+00	
2	3204A-18-B	1.38E-06	1.38E-06	1.38E-06	2.48E-11	2.48E-11	2.48E-11	0.00E+00	0.00E+00	4.14E-12	4.14E-12	4.14E-12	0.00E+00	0.00E+00	
3	3204A-18-C	4.14E-06	4.14E-06	4.14E-06	1.24E-09	1.24E-09	1.24E-09	0.00E+00	0.00E+00	1.24E-09	1.24E-09	1.24E-09	0.00E+00	0.00E+00	
4	3204B-18-A	1.39E-05	1.06E-05	1.39E-05	1.39E-09	1.06E-09	1.39E-09	-3.38E-10	0.00E+00	2.79E-10	2.11E-10	2.79E-10	-6.75E-11	0.00E+00	
5	3204B-18-B	1.10E-05	1.10E-05	1.10E-05	9.94E-08	9.94E-08	9.94E-08	0.00E+00	0.00E+00	6.62E-09	6.62E-09	6.62E-09	0.00E+00	0.00E+00	
6	3204B-18-C	6.67E-06	9.29E-06	9.29E-06	2.00E-08	2.79E-08	2.79E-08	7.86E-09	7.86E-09	1.33E-09	1.86E-09	1.86E-09	5.24E-10	5.24E-10	
7	3204B-18-D	4.65E-06	4.65E-06	4.65E-06	4.65E-10	4.65E-10	4.65E-10	0.00E+00	0.00E+00	9.29E-11	9.29E-11	9.29E-11	0.00E+00	0.00E+00	
8	3204B-18-E	1.13E-05	1.39E-05	1.39E-05	1.13E-09	1.39E-09	1.39E-09	2.62E-10	2.62E-10	1.13E-09	1.39E-09	1.39E-09	2.62E-10	2.62E-10	
9	3204C-12-A	2.76E-06	2.76E-06	2.76E-06	3.05E-10	3.05E-10	3.05E-10	0.00E+00	0.00E+00	1.20E-10	1.20E-10	1.20E-10	0.00E+00	0.00E+00	
10	3204C-12-B	1.60E-05	1.52E-05	1.86E-05	5.89E-10	5.61E-10	6.86E-10	-2.79E-11	9.67E-11	2.32E-10	2.21E-10	2.70E-10	-1.10E-11	3.80E-11	
11	3204D-12-A	2.76E-06	2.76E-06	2.76E-06	3.05E-10	3.05E-10	3.05E-10	0.00E+00	0.00E+00	1.20E-10	1.20E-10	1.20E-10	0.00E+00	0.00E+00	
12	3204D-12-B	1.33E-05	1.18E-05	1.86E-05	4.92E-10	4.36E-10	6.86E-10	-5.58E-11	1.93E-10	1.94E-10	1.72E-10	2.70E-10	-2.19E-11	7.60E-11	
13	3204E-12-A	2.76E-06	2.76E-06	2.76E-06	2.48E-08	2.48E-08	2.48E-08	0.00E+00	0.00E+00	1.66E-09	1.66E-09	1.66E-09	0.00E+00	0.00E+00	
14	3204E-12-B	1.33E-05	1.18E-05	1.86E-05	4.00E-08	3.55E-08	5.58E-08	-4.54E-09	1.57E-08	2.67E-09	2.37E-09	3.72E-09	-3.02E-10	1.05E-09	
15	3204F-12-A	5.52E-06	5.52E-06	5.52E-06	4.97E-08	4.97E-08	4.97E-08	0.00E+00	0.00E+00	3.31E-09	3.31E-09	3.31E-09	0.00E+00	0.00E+00	
16	3204F-12-B	5.33E-06	5.07E-06	6.20E-06	4.79E-08	4.56E-08	5.58E-08	-2.27E-09	7.86E-09	3.19E-09	3.04E-09	3.72E-09	-1.51E-10	5.24E-10	
		1.27E-04	1.25E-04	1.47E-04	2.89E-07	2.90E-07	3.21E-07	9.02E-10	3.20E-08	2.27E-08	2.30E-08	2.52E-08	2.32E-10	2.47E-09	