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

March 22, 2001

LICENSEE: Exelon Generation Company, LLC

FACILITIES: Braidwood Station, Units 1 and 2
Byron Station, Units 1 and 2
Dresden Nuclear Power Station, Units 2 and 3
Quad Cities Nuclear Power Station, Units 1 and 2

SUBJECT: SUMMARY OF FEBRUARY 27, 2001, MEETING WITH EXELON GENERATION COMPANY, LLC TO DISCUSS THE RISK-INFORMED INSERVICE INSPECTION RELIEF REQUESTS; BRAIDWOOD STATION, UNITS 1 AND 2, BYRON STATION, UNITS 1 AND 2, DRESDEN STATION, UNITS 2 AND 3, AND QUAD CITIES STATION, UNITS 1 AND 2 (TAC NOS. MB0507, MB0508, MB0567, MB0568, MB0362, MB0363, MB0721, AND MB0722)

On February 27, 2001, the U.S. Nuclear Regulatory Commission (NRC) staff met with members of the Exelon Generation Company, LLC (Exelon, the licensee) staff and their contractors to discuss the risk informed inservice inspection (RI-ISI) relief requests that have been submitted for the subject sites. A list of those attending the meeting is provided as Enclosure 1. Enclosure 2 is a copy of the licensees handout used during the meeting.

By letters dated October 16, 2000 (Braidwood), October 18, 2000 (Dresden), November 17, 2000 (Byron), and November 30, 2000 (Quad Cities), the licensee requested approval of a proposed alternative to the existing ASME Boiler and Pressure Vessel Code, Section XI requirements for the selection and examination of Class 1 and 2 piping welds. The requests used the methodology incorporated in Electric Power Research Institute Topical Report 112657, Rev. B-A, "Revised Risk-Informed Inservice Inspection Evaluation Procedure," which was approved by the NRC in its letter of October 28, 1999. Prior to the meeting, the staff requested additional information regarding the Dresden submittal in its letter of January 18, 2001. The licensee's response was submitted on February 19, 2001. The staff developed questions regarding the Braidwood submittal (Enclosure 3). The purpose of the meeting was to discuss the questions in Enclosure 3, and Exelon's February 19, 2001, letter RAI response.

With three exceptions, all questions were satisfactorily answered. The answers will be documented in the licensee's response to the questions in Enclosure 3. No revision to the answers in the February 19, 2001, letter is required. The three open issues are:

1. The EPRI methodology for development of RI ISI programs which was approved by the staff incorporated a data base developed under sponsorship of EPRI, and a methodology to generate failure parameters from the data base. The EPRI data base development guidelines and the failure parameter estimation methodology were reviewed by the staff coincident with the methodology. As part of the development of the relief requests for Byron and Braidwood, the licensee used a new and modified methodology to estimate the failure parameters. Consequently, the staff will

discuss the issue internally to develop a position on the applicability of the new data. The Dresden and Quad Cities submittals used the original EPRI data, so the issue does not apply to their requests.

2. The staff will review its Safety Evaluation, the information supplied in the Dresden RAI response specifying the equations used, and the supporting technical report from its contractor of the EPRI topical report with regard to the broad applicability of the Markov model.

3. The staff will assess the acceptability of the licensee's proposal to remove from the RI-ISI inspection location selection, welds which are currently included in the augmented inspection program. For example, welds included in the existing service water surveillance program are not selected for inspection in the RI-ISI program. A related issue is how discontinued inspection on welds within the augmented programs will be included in the change in risk calculations.

George F. Dick, Jr.

George F. Dick, Jr., Project Manager, Section 2
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. STN 50-454, STN 50-455,
STN 50-456, STN 50-457, 50-237, 50-249,
50-254, and 50-265

Enclosures:
1. Meeting Attendees
2. Handout
3. Braidwood Discussion Questions

cc w/encls: See next page

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/RA/
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EXELON RISK INFORMED INSERVICE INSPECTION MEETING

FEBRUARY 27, 2001

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TREATMENT OF RISK IMPACT ASSESSMENTS IN EXELON'S RI-ISI EVALUATIONS

By
Dr. William E. Burchill
Mr. Karl N. Fleming

Presentation to
U.S. Nuclear Regulatory Commission
February 27, 2001

Exelon

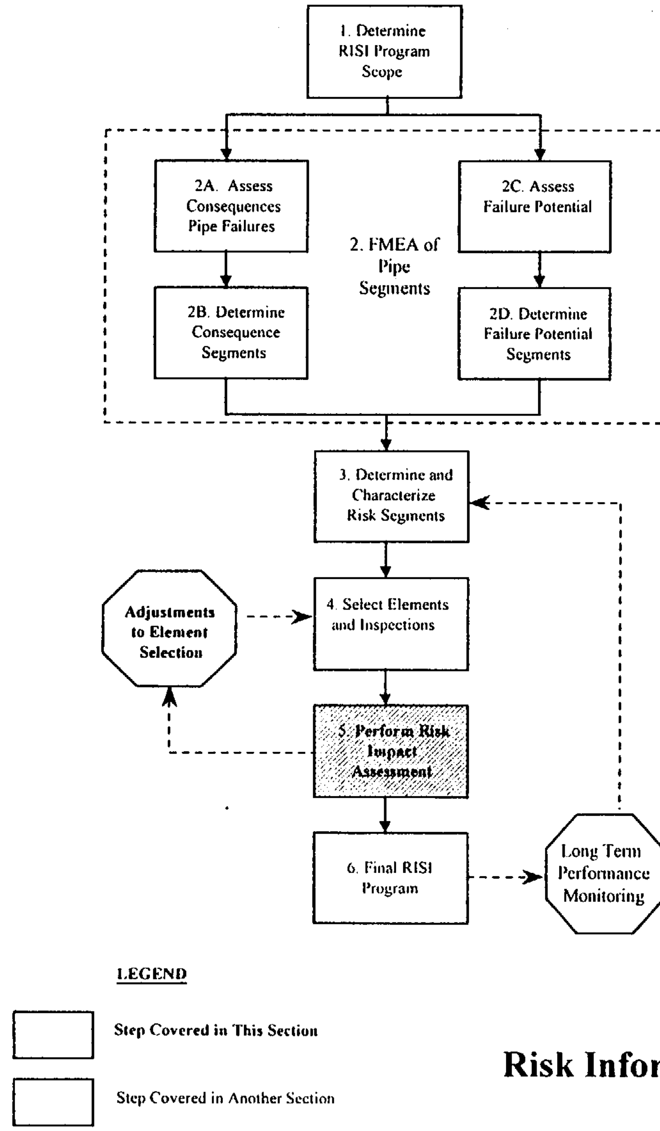
ENCLOSURE 2

OBJECTIVES

- Clarify Exelon approach to risk impact assessment
  - Takes no exceptions to EPRI Topical Report
  - Applies EPRI methodology which has not been reviewed by NRC in previous submittals
- Clarify application of risk impact methodology
- Explain update of pipe failure data
- Present conclusions on Exelon submittals
- Discuss RAI responses

EXELON RI-ISI
RISK IMPACT ASSESSMENT
METHODOLOGY

EPRI RI-ISI METHODOLOGY



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EVOLUTION OF EPRI RI-ISI RISK IMPACT ASSESSMENT

- Original EPRI RI-ISI method used by pilot plants did not include step for risk impacts of ISI program changes
- Risk impacts addressed in each pilot via RAI responses on a case by case basis
- EPRI added an explicit step to perform risk impact assessment in Topical Report TR-112657, Rev. B
- NRC SER approved the EPRI risk impact methodology
- NRC has approved only 2 relief requests since the SER prior to the Exelon RI-ISI submittals

EPRI TR PROCEDURE FOR RISK IMPACT ASSESSMENT

- Requires qualitative evaluation of risk impacts in each risk segment
  - addition of exams
  - redistribution of exams
  - removal of exams
  - enhancements to the inspection effectiveness ("inspection for cause")
- Concludes risk impact of changes in Low Risk segments is insignificant based on bounding estimates in EPRI TR
- Concludes a quantitative risk impact assessment is needed only when net exams removed from Medium or High risk segment
- Presents 3 alternate methods for quantitative risk impact assessment

EPRI METHODS FOR QUANTITATIVE ASSESSMENTS

(Ref. EPRI TR 112657, Sect. 3.7.2)

- Conservative bounding estimates
  - 3 Options for Pipe Rupture Frequencies (DM Category)
    - 1E-4 (High), 1E-5(Medium), 1E-6(Low)
    - Other defensible source of frequencies (EPRI TR-111880)
  - No credit for inspection effectiveness
- Two options for realistically estimating inspection effectiveness
  - Simplified Method (1-POD)
  - Markov Model

EXELON
RISK IMPACT ASSESSMENT

- Performed qualitative risk impact assessments for all risk segments
- Calculated realistic quantitative risk impacts for all segments using the Markov inspection effectiveness model
- Prepared sensitivity studies to illustrate realistic inspection effectiveness compared to no inspection effectiveness
- Prepared sensitivity studies using each of the EPRI approved methods on all segments
- Conformed to the requirements of EPRI TR, NRC SER, and NRC RGs 1.174 and 1.178

EPRI METHODOLOGY ILLUSTRATED IN RI-ISI SUBMITTALS

- Previous submittals included
  - qualitative analysis of all segments
  - bounding analysis of High and Medium Risk Segments
  - realistic analysis of selected segments using the (1-POD) model
  - realistic analysis of all segments using the Markov model
  - estimates of rupture frequencies from EPRI TR 102266 and EPRI TR 111880
- Exelon submittal and RAI responses include:
  - qualitative analysis of all segments
  - realistic analysis of all segments using the Markov method; comparison of inspection effectiveness factors with (1-POD) method
  - bounding analysis of all segments as a sensitivity study
  - estimates of failure rates and rupture frequencies from EPRI TR 111880 for BWRs
  - updates of failure rates and rupture frequencies using SKI-PIPE for PWRs

APPLICATION OF MARKOV MODEL IN EXELON RI-ISI

EXELON APPLIED MARKOV BASED ON NRC APPROVAL

- NRC SER states staff "adopts the analysis of the Markov model" and "finds the [Markov] model can be used as a basis for the estimation of pipe rupture frequencies instead of the bounding pipe failure frequencies"
- NRC contractor reviews of EPRI methodology endorsed use of Markov

RISK IMPACT ASSESSMENT

- Changes in risk in a pipe segment arise from changes in pipe rupture frequency due to changes in ISI moving from Section XI to RI-ISI program:
  - New exams may be added to the segment
  - Some exams may be removed from the segment
  - Effectiveness of exams may be improved due to "inspection for cause" principle
  - ISI program has no impact on CCDP and CLERP

- The change in pipe rupture frequency is estimated in terms of a baseline rupture frequency and changes in the Inspection Effectiveness Factor
  - Inspection Effectiveness Factor is Ratio of inspected weld rupture frequency to the uninspected weld rupture frequency

- The change in risk due to ISI changes at a weld is the change in pipe rupture frequency at the weld times the CCDP for ΔCDF or CLERP for the ΔLERF

QUANTIFICATION OF RISK MODEL

Parameter	Method of Quantification
ΔCDF	Computation of Eq. (3-9) in TR-112657; Eq. (3.40) in TR-110161
$\Delta LERF$	Same Equations as CDF with CLERP instead of CCDP
i	From risk segment definition in Step 3 of RISI Procedure
N	From risk segment definition in Step 3 of RISI Procedure
n_i	From risk segment definition in Step 3 of RISI Procedure
λ_i	Estimated from service data in using methodology of TR-111880
$P_i\langle R \mid F \rangle$	Estimated from service data in using methodology of TR-111880
$I_{i,new}$	Markov model solution used to develop equation in terms of parameters that describe degradation and inspection processes as explained in TR-110161 applied to RISI program
$I_{i,old}$	Markov model solution used to develop equation in terms of parameters that describe degradation and inspection processes as explained in TR-110161 applied to Section XI program
$CCDP_i$	Evaluated in Steps 2A and 2B in RISI Procedure using plant specific PRA models and the results of the consequence analysis
$CLERP_i$	Evaluated in Steps 2A and 2B in RISI Procedure using plant specific PRA models and the results of the consequence analysis

MARKOV MODEL FOR PIPING RELIABILITY ANALYSIS

- Is an established method for time dependent reliability of repairable components
- Was used in this application to model influence of ISI exams and leak detection as strategies to repair pipe degradation to prevent ruptures
- Uses a set of four pipe states (ok, cracked, leaking, ruptured) and transition rates to model time dependent state transitions
- Produces output similar to probabilistic fracture mechanics codes
  - Time dependent state probabilities
  - Time dependent rupture frequencies (hazard rates)
  - Inspection effectiveness factors

APPLICATION OF NRC REVIEWED EQUATIONS

Model/Equation	Report Reference	Page, Table, Equation References
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) though (3.38) on pp. 3-24 to 3-27
Definition of Inspection effectiveness Factor for use in delta risk equation	EPRI TR-110161	$I = \frac{h_{40}\{\omega_{NEW}\}}{h_{40}\{\omega_{OLD}\}}$ 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.
Definition of the flaw inspection repair rate, ω	EPRI TR-110161	Equation (3.23) on p. 3-18
Definition of the leak detection repair rate, μ	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

MODELING IMPACT OF INSPECTION (ISI)

- The benefits of ISI are modeled by the transition in the Markov model from the flaw state to the success state to reflect the opportunity to detect flaws or cracks via ISI exams before they propagate to pipe leaks or ruptures and to repair the damaged pipe.
- Estimation of ω: the repair rate for flaws

$$\omega = \frac{P_{FI}\,P_{FD}}{\left(T_I + T_R\right)}$$

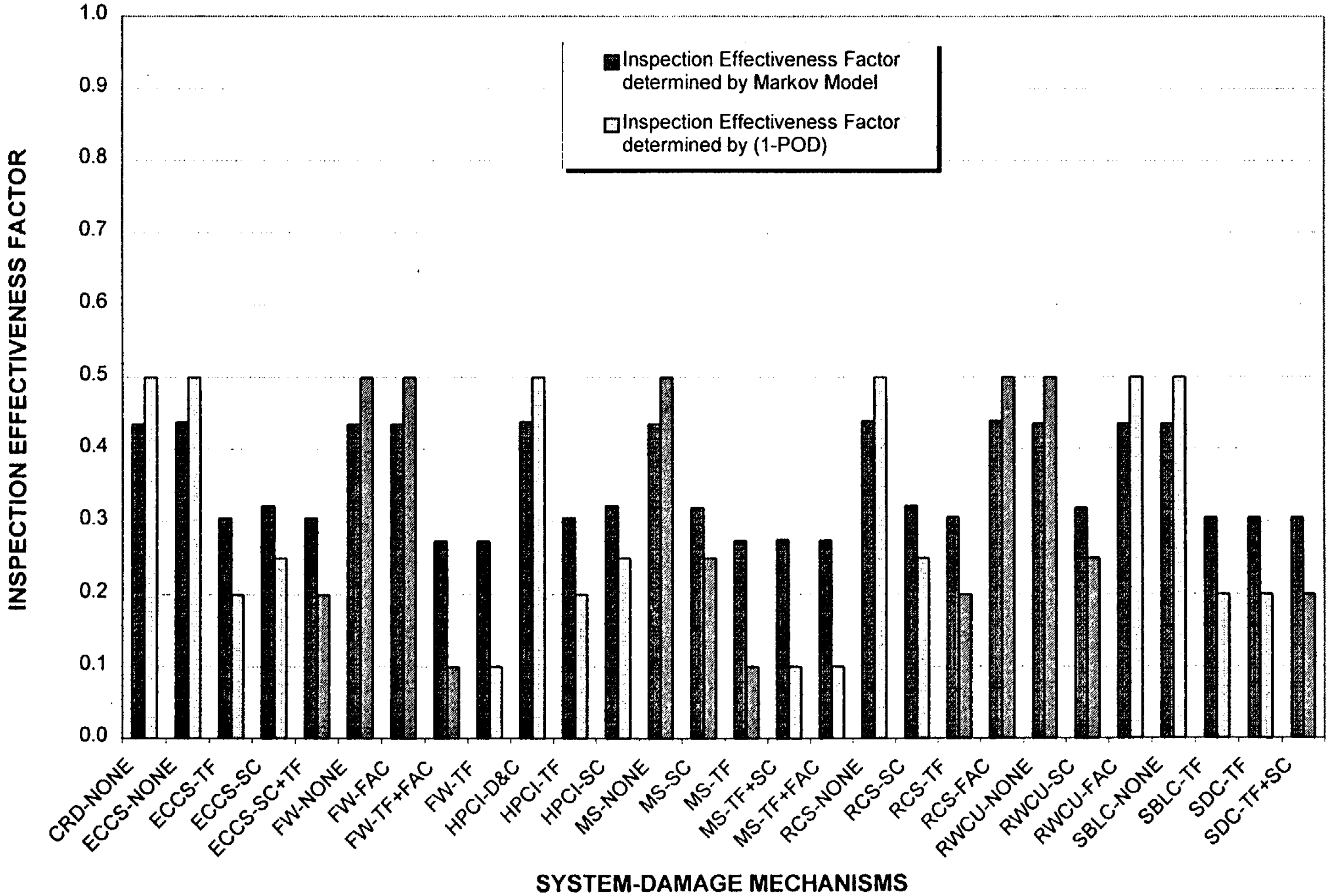
where:

- P_{FI} = 1 if the weld is inspected; 0 if it is not inspected
- P_{FD}= probability that flaw is detected given inspection ("POD")
- T_I = mean time between inspections (e.g. 10 years per ASME Section XI)
- T_R = mean time to repair the damaged pipe after detection in ISI exam

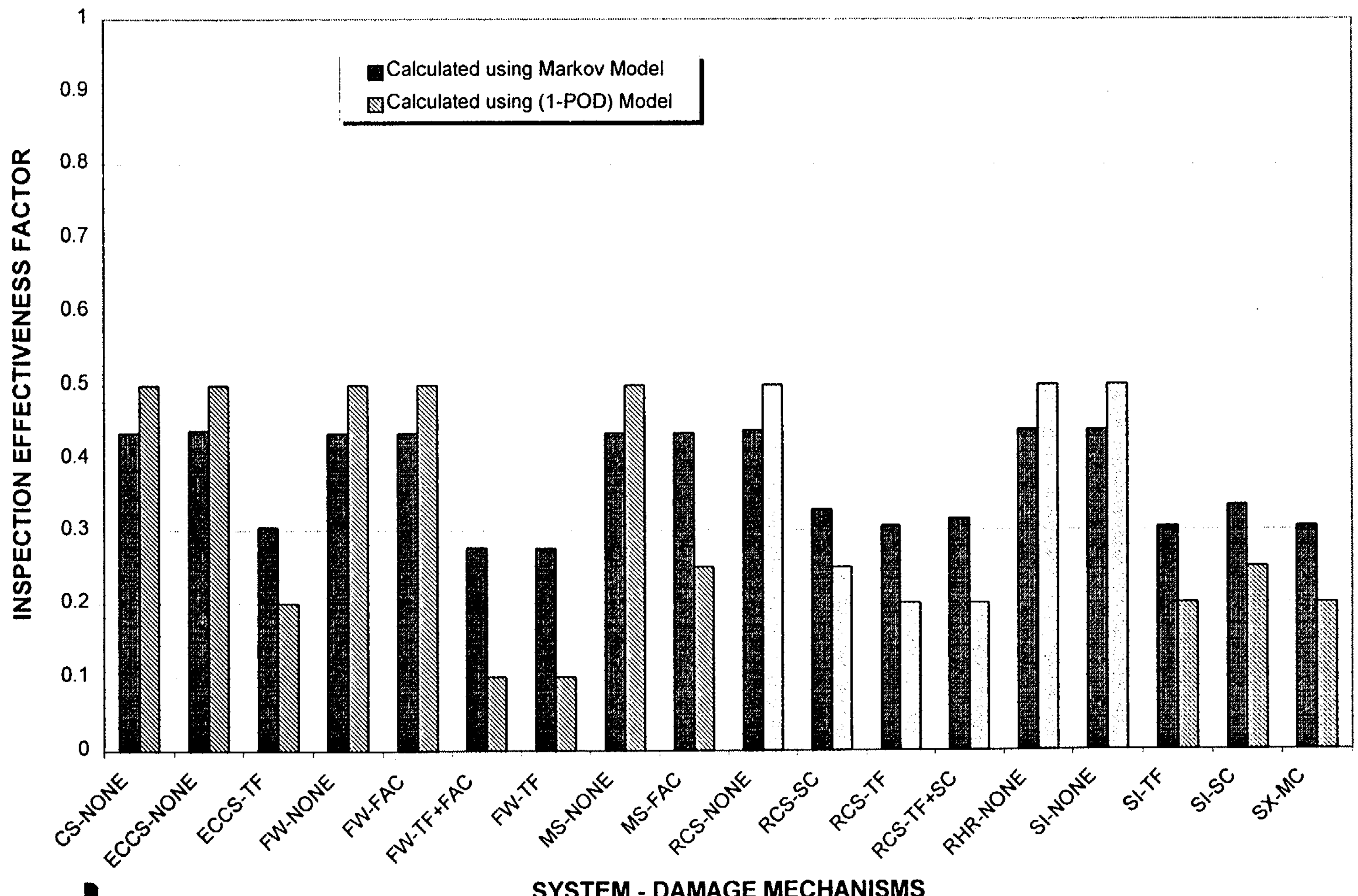
ESTIMATION OF MARKOV MODEL INPUT PARAMETERS

- Applied to each weld in scope of RISI program separately
- Crack and Leak Failure Rates and Rupture Frequency inputs
  - Estimated from service data and Bayes update methodology of EPRI TR-111880; updated for selected PWR systems and damage mechanisms; modified to account for damage mechanism synergy
- Inspection Repair Rates
  - Simple model from EPRI TR-110161 and estimates of POD modified for ISI accessibility; ISI inspection intervals, repair time
- Leak Detection Repair Rates
  - Simple model from EPRI TR-110161; estimates of detection probabilities, inspection intervals and repair time; not varied between RISI and Section XI cases

COMPARISON OF INSPECTION EFFECTIVENESS FACTORS FOR BWRS



COMPARISON OF INSPECTION EFFECTIVENESS FACTORS FOR PWRS



Exelon

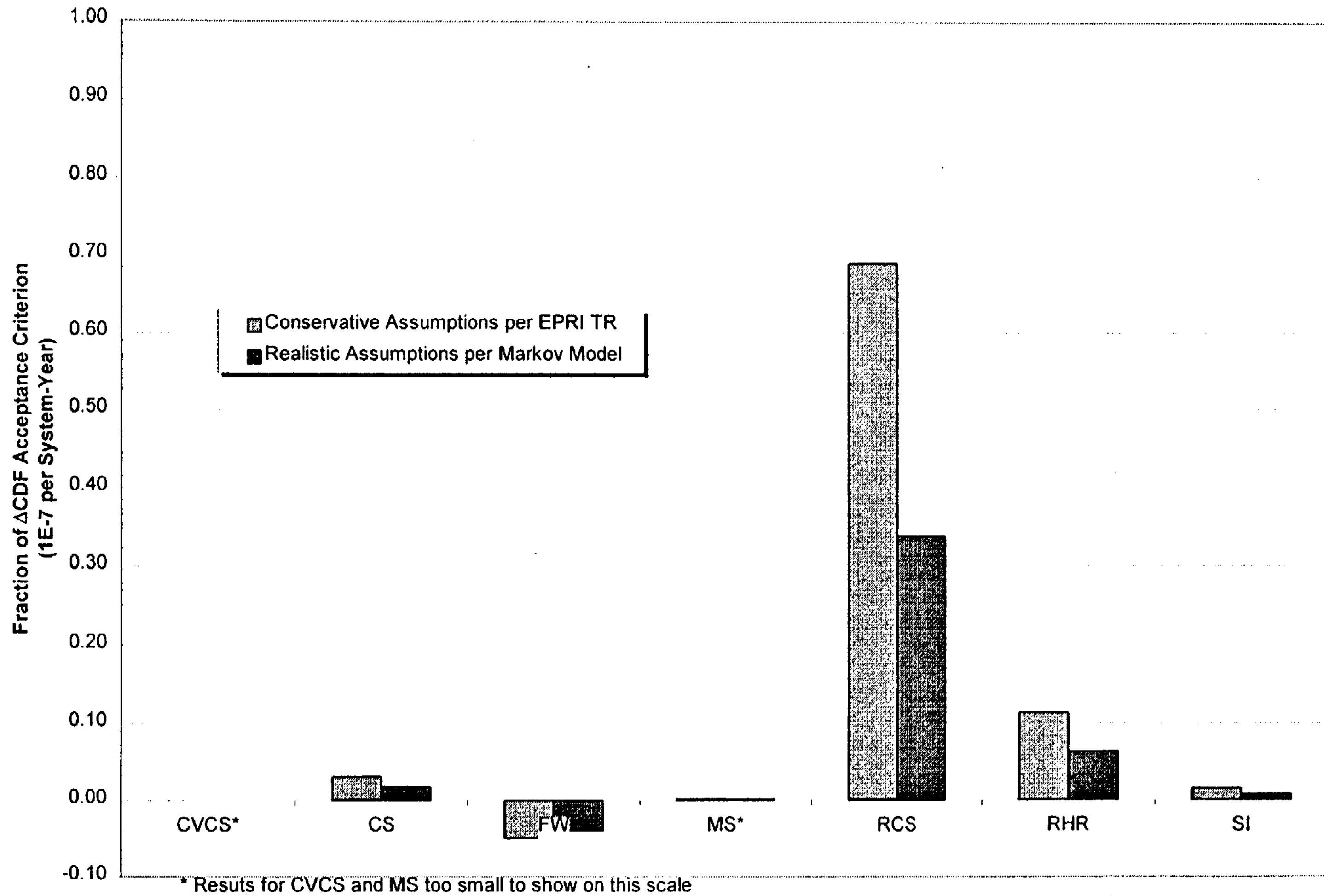
APPLICATION OF MARKOV MODEL TO EXELON RI-ISI

- Equations and methodology are identical to those described in EPRI reports reviewed by NRC
- Differences in application vs. PWR pilot plant RCS example
  - Explicitly considered crack and leak ratios
  - Updated failure rates and rupture frequencies for PWRs
  - Modified PODs to reflect limited accessibility
  - Applied Markov models to predict change in both leak and rupture frequencies
  - Applied to both delta CDF and delta LERF

BOUNDING SENSITIVITY ANALYSIS

- Used values from realistic analysis for:
  - Failure rates and rupture frequencies
  - CCDPs and CLERPs
- Took no credit for inspection effectiveness changes in RISI
- Credited added and redistributed welds; all risk change comes from net welds added or removed from each segment
- Evaluated all risk segments and case with High and Medium welds only

COMPARISON OF BOUNDING AND REALISTIC RISK IMPACTS BRAIDWOOD 1



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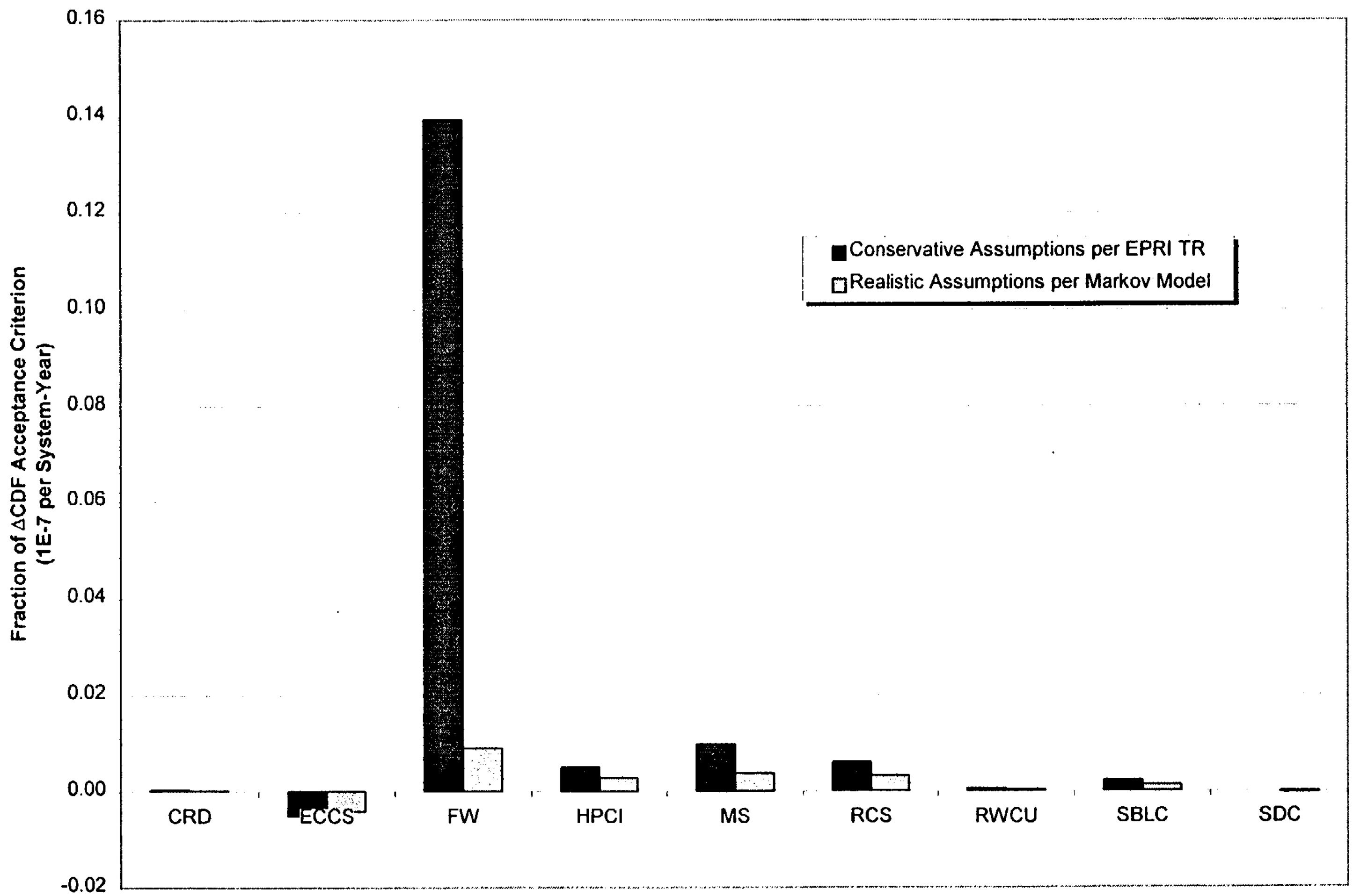
COMPARISON OF BOUNDING AND REALISTIC RISK IMPACTS DRESDEN 2



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CONCLUSIONS FROM SENSITIVITY STUDY

- Risk acceptance criteria are met for all Exelon plants and systems in RI-ISI scope
  - large margins for all BWR and most PWR systems
  - small margins for PWR RCS system
- Realistic estimates provide more reasonable basis for RI-ISI evaluation
  - reasonable to expect risk reductions from inspection for cause especially for thermal fatigue susceptible segments
  - bounding analysis overstates risk importance of ISI on mitigating pipe rupture frequencies
  - risk impacts from low risk segments not necessarily dominated by high risk segments
  - enhanced consistency with other risk informed applications such as technical specification changes; enhanced capability to balance resources and risk across different risk informed applications

UPDATE OF PIPE FAILURE DATA

Exelon

ESTIMATION OF FAILURE PROCESSES

- Estimate flaw failure rates from results of NDE inspections in pipe database; must have at least one flaw for each failure
- Estimate leak and rupture failure rates from service experience, failure rate models for different system types, and failure mechanisms
- Determine distinct rupture failure rates depending on the presence of a flaw or leak to model effects of aging
- Apply Bayes' theorem to incorporate available generic and system-failure mechanism specific experience in full quantification of uncertainties

Exelon

Sources of Data

- Piping Reliability Databases
  - EPRI-97 Database
    - Based on reports by Bush, Chockie, Jamali, Fleming, et al
  - SKI-PIPE 98 Database
    - Worldwide Piping Reliability Database by Lydell
    - Basis for OECD (Office of Economic Cooperation and Development) International Piping Reliability Database

ESTIMATION OF FAILURE RATES AND RUPTURE FREQUENCIES

- For BWRs used values direct from EPRI TR-111880 Table A-11
- For PWRs used same Bayes' methodology as EPRI TR-111880 but updated to reflect:
  - More complete and more accurate and traceable account of pipe failures
  - Improved estimates of weld populations from completed RI-ISI submittals
  - Improved estimates of fractions of population susceptible to different damage mechanisms from completed RI-ISI submittals

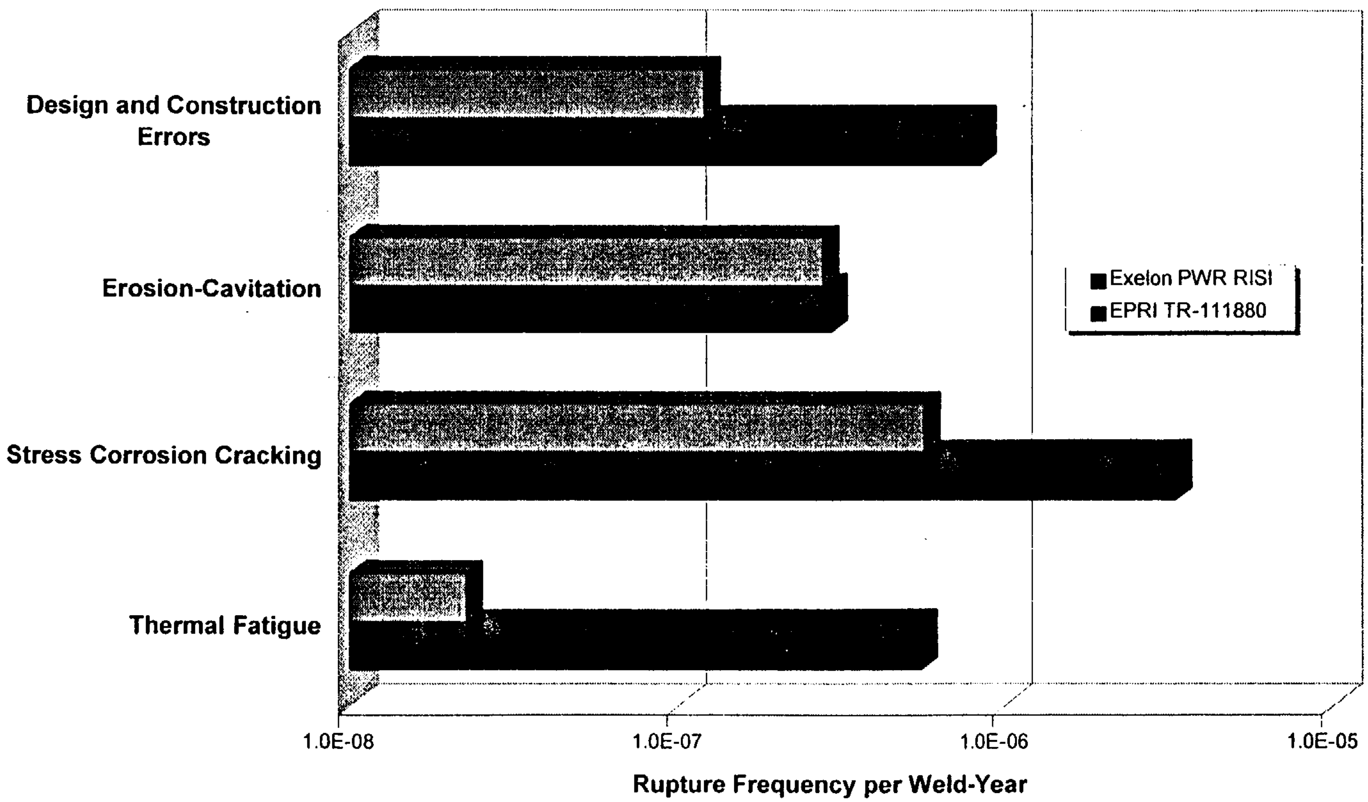
CONSIDERATIONS IN THE SELECTION OF FAILURE RATES FOR RISI

- EPRI and SKI-PIPE databases were compared following publication of EPRI TR-111880 to address weld overlay issue for BWR VIP 75
- SKI-PIPE is a superior data source for developing failure rates and rupture frequencies
- PWR data was updated with SKI-PIPE because it provides much more data available to estimate PWR weld populations and DM susceptibility fractions
- BWR data was not updated because risk assessment showed significant margins using EPRI TR-111880 data and expected changes would have been to reduce failure rates and rupture frequencies for key mechanisms

COMPARISON OF PWR FAILURE RATE ESTIMATES

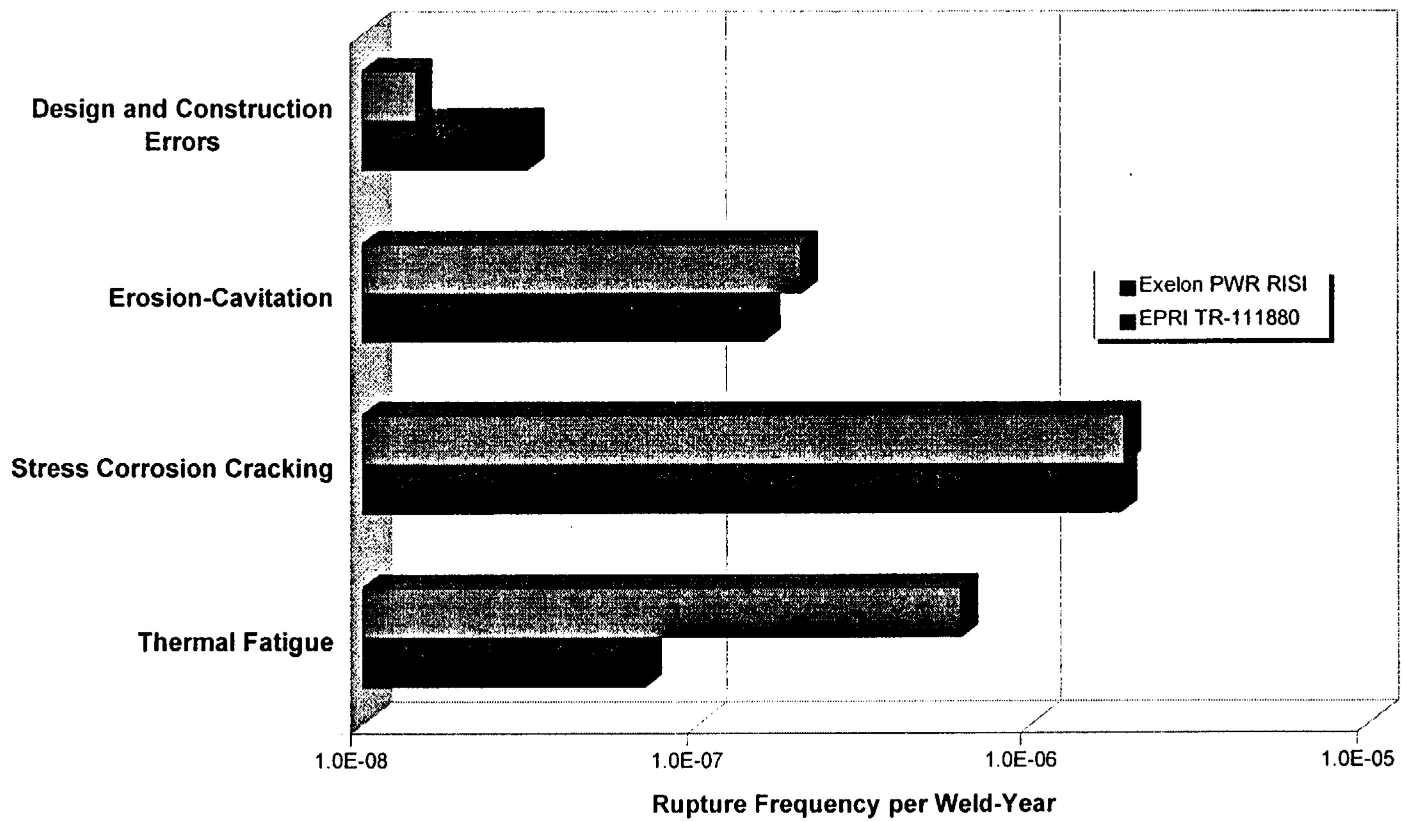
Failure Rate Parameters		EPRI TR 111880	Exelon PWR RISI
Failure Data Source		SKI 96:20 as modified in EPRI TR-111880 with U.S. experience through 1995	SKI-PIPE Database described in SKI 98:30 and updated through May 2000;
Westinghouse PWR reactor years experience		905 U.S. only	2,234 U.S, Europe and Japan
Number of ISI amenable Failures in Westinghouse PWR Class 1 and 2 piping	Leaks	16	55
	Ruptures	0	0
Weld Population estimates per plant	RCS	409	364
	SI	1,211 for entire SIR system group	1,520
	RHR		420
	CVC S		744
Total Class 1 and 2 component exposure estimate		1.47×10^6 weld-years	6.81×10^6 weld-years
Plant data available to support weld population and damage mechanism susceptibility fraction estimates		ANO-2	ANO-2, STP-1, STP-2, Bw-1, Bw-2, By-1, By-2
Bayes Update Methodology		As described in EPRI TR-11880	Same procedure with refinements to take advantage of better data; use of Beliczey-Schulz correlation to anchor priors for conditional rupture probabilities

COMPARISON OF RUPTURE FREQUENCIES FOR RCS



Exelon

COMPARISION OF RUPTURE FREQUENCIES FOR SI SYSTEM



CONCLUSIONS FROM UPDATE OF FAILURE DATA

- EPRI TR-111880 provides reasonable and somewhat conservative basis to support BWR delta risk evaluation
- SKI-PIPE provides more realistic estimates for PWR Class 1 and 2 systems
  - Rupture frequencies for PWR RCS thermal fatigue and design and construction errors are significantly lower than EPRI TR 111880
  - Rupture frequencies for PWR SI thermal fatigue are significantly higher than EPRI TR 111880
  - Updated PWR estimates are based on enhanced estimates of weld populations and damage mechanism susceptibility fractions and correct some classification inconsistencies

OVERALL CONCLUSIONS

- Submittal takes no exceptions to EPRI TR requirements for risk impact evaluation
- Submittal realistically quantifies risk impact in accordance with the EPRI TR
- RAI response shows (1-POD) model and Markov model produce comparable inspection effectiveness factors
- RAI response shows risk impact acceptance criteria are met even with conservative bounding risk impact estimates
- Submittal provides improved estimates of failure rates and rupture frequencies for PWRs
- The submittal and RAI responses support timely NRC review

BRAIDWOOD UNITS 1 AND 2 RI-ISI
Discussion Points

1. In accordance with the guidance provided in RGs 1.174 and 1.178, an engineering analysis of the proposed changes is required using a combination of traditional engineering analysis and supporting insights from the probabilistic risk assessment (PRA). The purpose of the traditional engineering analysis is to ensure that the impact of the proposed ISI changes is consistent with the principles of defense-in-depth. Based on the staff's experience with the review of RI-ISI submittals, the percentage of volumetric inspection of ASME Class 1 welds has ranged from about 7% to 12%. In cases where the original proposal was for less than 10% volumetric inspection of these welds, the staff has been requesting that the sample obtained by the risk-informed process be increased to obtain a 10 % level of inspection sample by selecting elements for inspection to obtain a distribution of inspections among various systems including considerations of various potential degradation mechanisms. This request is based on the staff conclusion that a minimum of 10% volumetric inspection sample of ASME Class 1 welds is needed for the staff to find that an acceptable level of defense-in-depth is being provided. The Braidwood submittal states that 8.9% of the Class 1 welds for Unit 1 will be volumetrically inspected. Please clarify numbers of total category B-F and B-J welds, and numbers of butt welds performing volumetric inspection in each category in the RI-ISI program to ensure that a minimum of 10% stated above is met.

2. Please clarify the following:

   a) In the second page of the transmittal letter, the licensee provided the "start" and "end" dates of the ISI periods. For Period 2 in both units, the year in the start dates are marked 2001. However, the years for the end dates of Period 1 are 2002. Please clarify.

   b) In attachment 1, on page 2 of 4, item c for all dissimilar metal welds in the category B-J, the licensee should indicate that these dissimilar welds include those not covered by the B-F as indicated in the Note (c) of the ASME Code Table IWB-2500-1 for category B-J.

   c) In attachment 1, on page 2 of 4, the licensee discusses the Table IWC 2500-1 requirements for category C-F-1. However, similar discussions for C-F-2 are missing in the submittal RR 12R-39, Revision 0. Please explain.

   d) Is there any recognizable plant experience on piping failures at Braidwood?

   e) What is the minimum pipe diameter included in the RI-ISI evaluation and program?

   f) Both Tables 5 and 6 included the Risk Category 4 in the High Risk columns. Should these be under Medium Risk columns?

3. In accordance with the Section 3.2.3 of the SER to the EPRI topical report, a pipe segment susceptible to a degradation other than flow accelerated corrosion (FAC) and which also has the potential for water hammer receives high pipe failure potential. The licensee has not identified water hammer as a potential degradation mechanism for selected pipe segments. Clarify if any of the selected system welds are susceptible to water hammer and

any other aging mechanism than FAC.

4. Please provide a reference to the version of the PRA used to support the RI-ISI submittal. Please also provide the CDF and the LERF estimates from the PRA version used to support the RI-ISI submittal.

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

6. 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." Does "removed from the RI-ISI element selection population" mean that all welds within a medium ranked segment that is included in the FAC program, for example, are excluded from the required 10% and that discontinued Section XI inspections within the segment will not be included in the change is risk calculations? If not please explain what removed from the population means. Does the reported 8.9% and 10.1% of Class 1 butt welded elements inspected include the population of Class 1 HELB and the FAC element welds in the denominator?

7. The licensee has included the essential service water system (SX) within the scope of the RI-ISI program but chose not to subsume the service water inspection program. The licensee has also included the containment purge system (VQ) within the scope of your RI-ISI. Neither SX nor VQ appear in the tables identifying inspection locations selected for RI-ISI. Were there any segments in SX or VQ that had a medium or a high consequence ranking? How many Section XI inspections are currently being performed in VQ and SX?

8. In the note to Table 4 regarding Unit 2 the licensee indicates that the difference in the distribution of welds in the different risk categories is due primarily to the Unit 1's steam generators (SGs) being replaced whereas Unit 2's SGs has not been replaced. Please explain how the replacement of the SGs could cause such a large reduction in the number of Unit 1's Category 3 FW (108) and Category 4 RC (23) locations as compared to Unit 2. Additionally, the total number of welds in the systems seems to vary substantially between the two units. For example Unit 1 has 104 less FW and 27 less RC welds than Unit 2, but 65 more SI welds than Unit 2. Do these difference in total welds reflect actual physical difference between the piping systems in the two units?

9. Page 12 of the submittal discusses a "separate Markov calculation" for the change in LERF for lines connected to the RC that continue outside containment. Normally such lines have an inboard and an outboard isolation valve. A rupture outside containment and failure of the inboard isolation valve will result in an unisolatable LOCA outside of containment. Is this the scenario that is being addressed here? If this is not the scenario, please provide an example to illustrate the scenario. The methodology in EPRI TR-112657 includes a semi-quantitative technique for this situation in Table 3-14. Alternatively, the probability of the inboard isolation valve failing can be factored into the CLERP probability. If the

licensee's methodology deviates from the EPRI TR-112657 for unisolatable LOCAs, please provide a comparison of the licensee's method with the accepted method.

10. If the calculations are performed using the data from the Tables in TR-111880 instead of the updated failure rates, the licensee may identify the Tables used instead of responding to question 11 on the Bayesian update. If results from the bounding evaluations described in the EPRI TR-112657 instead of the Markov calculations are sufficient to illustrate that the suggested change in risk guidelines are not exceeded, please provide a brief description of these evaluations and the results instead of responding to the questions 12 and 13 on the Markov calculations.

11. In Section 3.7 on pages 11, reference is made to the use of updated failure rates and rupture frequencies. The EPRI report, "Piping System Failure rates and Rupture Frequencies for Use in Risk Informed In-Service Inspection Applications," TR-111880 was completed in September, 1999. A copy of the report was submitted to the NRC in support of a RI-ISI relief request at another nuclear power plant. A draft version of the report was submitted to the NRC during the review of the EPRI Topical report TR-112657. As indicated by its tittle, EPRI TR-111880 contains tables of vendor and system specific parameter values to be used to support RI-ISI applications. The evaluation documented in TR-111880 was performed by a team sponsored by EPRI. This team developed required plant characteristics, evaluated individual failure events collected from plant operating experience, interpreted the observed experience, characterized and grouped the observed experienced, and calculated a specific set of suggested failure parameters. EPRI TR-111880 states that the values provided in the Tables includes about 905 years of operating experience for Westinghouse reactors. The licensee's submittal states that the updated parameters include 1000 years of experience. As illustrated in the following Table RAI-11, examination of your new parameters reveals a difference in the grouping of the systems and large differences in the parameter values. These differences do not appear consistent with an increases of 10% in the years of experience for rare events such as failures and ruptures. The differences appear to indicate differences in the judgements interpreting, and the subsequent manipulations of the experience data. The staff will need to fully understand any differences in the evaluation of the experience data, and the justification for these differences, to accept the plant specific data as an improved set of parameters that need to be used instead of the industry data to support the change in risk calculations in the Braidwood submittal.

a) Please describe how the failure rates were updated; that is, were the rates in the table updated or were the original calculations performed with the new data? Please provide a reference to the equations' numbers in TR-111880 or TR-110161 that were used in the update.

b) EPRI TR-111880 reported that, "[t]o provide the best possible estimates of pipe failure rates, rupture rates for each failure mechanism are calculated for eight different system groups, for each type of reactor vendors." In the table in the Braidwood submittal, the safety injection (SI) system and the residual heat removal (RH) system (both originally in the RAS group) are individually listed. In EPRI TR-111880 the two systems are assigned the same parameters, but there are very large differences between the

systems' parameters in the submittal. The desire to balance resolution with available data reflects the Bayesian update procedure where, as less and less data is available, the result of the update become more and more dependent on the initial judgements and less and less on the experience data. Please characterize the quantity of data available for each system's update and the impact that data has on the priors developed from judgement. Please justify the development of finer groupings and explain why this finer grouping is applicable to the analysis supporting this submittal but was not applicable for the TR-111880 calculations.

c) What is the range of dates used in the update and how many additional reactor years are in the update?

d) The Braidwood submittal states that the updated failure parameters reflect estimates of " weld population exposure" that were not available when TR-111880 was developed. Was it only exposure information that was collected or were any failures observed and also used to update the parameters? What information sources was used to estimate the extra years of exposure and to identify any failures that might have occurred during these years? Are these the same information sources that were used to develop the original estimates in EPRI TR-111880?

e) Please explain why all systems were not "updated," but rather, some (i.e., CS, SX, FAC, and ST) used the existing values from EPRI TR-111880.

f) Although not illustrated in the Table, the staff notes that the probability of rupture given a failure (P(R/F)) has been changed, in some cases, by almost a factor of 5 reduction. This change implies that there has been additional data collected on the number of observed ruptures and flaws. Both events are infrequent. Is this parameter being calculated as described in EPRI TR-110161? Please describe and summarize the experience data that was used to calculate the change in this parameter.

g) Please explain why there is such a wide variation in the magnitude of the changes when the same calender time was used for the update of all the parameters.

Table RAI-11

The entries give the factor change for the rupture failure frequencies between EPRI TR-111880 and Table 7 in the submittal. A "9 X reduction" means that the failure frequency in the submittal is 9 times smaller than in EPRI TR-11880

Damage mechanism	RC System	SI* System	CAC** System	RH* System
T.F.	25 X reduction	9 X increase	10 X reduction	6 X increase
SC	6 X reduction	negligible change	5 X reduction	5 X reduction
E.C.	2 X reduction	negligible change	10 X increase	60 X increase
DC	7 X reduction	2 X reduction	70 X reduction	4 X increase

*SIR in TR-111880
**RAS in TR-111880

12. Please provide references to all the equations that describe the Markov calculation that the licensee is using to calculate the change in risk. For example Equation 6.1 in TR-110161 refers to multiple failure sizes and multiple conditional core damage probabilities for each segment. Is the licensee using this equation? Please give the values of all the input parameters required by the equations and also provide a references from which the input parameters were developed and justified (except for the conditional core damage, condition large early release probabilities, and weld failure rates). For example, if the licensee is using Equations 3.23 and 3.24 in TR-110161, what values are being used for the parameters? Please provide specific references, e.g. equation numbers, table numbers, page numbers, and report references.

13. It is the staff's understanding that the Markov calculations include calculating an "inspection effectiveness factor" for use in equation 3-9 of EPRI-TR 112657. Please provide the distribution of inspection effectiveness values calculated and a discussion on how these values compare with the direct use of the probability of detection estimates.