January 26, 2006

WOG-06-25

WCAP-16168-NP, Rev. 1
Project Number 694

U.S. Nuclear Regulatory Commission
Document Control Desk and Chief Financial Officer
Washington, DC 20555-0001

Attention: Chief, Information Management Branch,
Division of Program Management

Subject: Westinghouse Owners Group
Transmittal of WCAP-16168-NP Rev. 1, "Risk-Informed Extension of Reactor Vessel In-Service Inspection Interval" (MUHP-5097/5098/5099, Tasks 2008/2059)

This letter transmits four (4) copies of WCAP-16168-NP Rev. 1, entitled "Risk-Informed Extension of Reactor Vessel In-Service Inspection Interval," dated January 2006. The Westinghouse Owners Group (WOG) is submitting WCAP-16168-NP Rev. 1 in accordance with the Nuclear Regulatory Commission (NRC) licensing topical report program for review and acceptance for referencing in licensing actions. WCAP-16168-NP Rev. 1, provides the justification to extend the current inspection interval requirements of ASME Section XI Table IWB-2500-1 Category B-A reactor vessel seam welds, Category B-D reactor vessel nozzle and nozzle inner radius welds, and Category B-J welds at the reactor vessel nozzle from 10 years to 20 years.

The current inspection requirements for reactor vessel pressure-containing welds were originally required by the 1989 Edition of American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI, as supplemented by NRC Regulatory Guide 1.150. The manner in which these inspections are conducted has been augmented by Appendix VIII of Section XI, through the 1996 Addenda, as implemented by the NRC in an amendment to 10CFR50.55a effective November 22, 1999.

Specific pilot studies have been performed on the Westinghouse, Combustion Engineering, and Babcock and Wilcox reactor vessel and NSSS designs. The results show that the change in risk associated with eliminating all inspections after the initial 10-year in-service inspection satisfies the guidelines specified in Regulatory Guide 1.174 for an acceptable change in risk. This applies for both a 40 year operating license and a 20 year license extension to a 60 year operating license.

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Representatives of the Westinghouse Owners Group met with the NRC on October 11, 2005 to discuss revised technical results, a revision to WCAP-16168-NP and the proposed schedule for the WCAP re-submittal. This meeting is summarized in NRC meeting summary, ADAMS accession number ML052910148, dated October 26, 2005, "Summary of Meeting held on October 11, 2005, with the Westinghouse Owners Group to discuss resubmittal of WCAP-16168-NP, 'Risk-Informed Extension of Reactor Vessel Inservice Inspection Interval'."

The revised WCAP contains the following changes:

- Inclusion of revised PTS transients
- FAVOR results based on revised PTS transients
- Incorporation of a Babcock and Wilcox pilot plant evaluation
- Incorporation of non-pilot lead plant examples
- Additional text providing clarification on how the input and methodology from the recent NRC PTS Risk Reevaluation are used in the topical report

Furthermore, the WOG has reviewed NRR's comments to RES on the technical basis for the PTS rulemaking to revise 10 CFR 50.61, which were included in the October 11, 2005 meeting summary (ADAMS accession number ML052910148), and it is our assessment that these comments do not impact the WCAP.

The WOG requests that a fee waiver be considered for the NRC review of WCAP-16168-NP Rev. 1 pursuant to the provisions of 10 CFR 170.11. This report is generically applicable to all domestic PWR designs and supports NRC generic regulatory improvements (ASME Boiler and Pressure Vessel Code, Section XI/Regulatory Guide 1.150.). Specifically, this topical report provides the technical background requested by the NRC Staff in their comments on ASME Code Case N-691. Furthermore, the inspection interval extension discussed in this report will reduce regulatory burden and allow for more appropriate allocation of industry inspection resources.

During the fee waiver decision period, the WOG would like the NRC Staff to review WCAP-16168-NP, Rev. 1. The WOG will assume the responsibility of the NRC review fees accrued if the fee waiver is not granted.

Consistent with the Office of Nuclear Reactor Regulation, Office Instruction LIC-500, "Processing Request for Reviews of Topical Reports," the WOG requests that the NRC provide target dates for any Request(s) for Additional Information and for issuance of the Safety Evaluation for WCAP-16168-NP Rev 1.

Correspondence related to this transmittal and invoices associated with the review of WCAP-16168-NP Rev. 1 should be addressed to:

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If you require further information, please contact Mr. Jim Molkenthin in the Owners Group Program Management Office at 860-731-6727.

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WCAP-16168-NP
Revision 1

January 2006

Risk-Informed Extension of the Reactor Vessel In-Service Inspection Interval

Westinghouse Owners Group

CEOG Task 2008, 2059
WOG MUHP-5097, 5098, 5099
WOG PA-MSC-0119, 0120

Westinghouse

WCAP-16168-NP Revision 1

Risk-Informed Extension of the Reactor Vessel In-Service Inspection Interval

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January 2006

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This work was performed for the Westinghouse Owners Group under WOG Project MUHP-5097, MUHP-5098, MUHP-5099, WOG Project Authorization MSC-0119, MSC-0120 and CEOG Task 2008, 2059.

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Westinghouse Owners Group
Member Participation* for WOG Project MUHP-5097, MUHP-5098, MUHP-5099, and CEOG Task 2059.

Utility Member	Plant Site(s)	Participant	
		Yes	No
AmerenUE	Callaway (W)	X	
American Electric Power	D.C. Cook 1&2 (W)	X	
Arizona Public Service	Palo Verde Unit 1, 2, & 3 (CE)	X	
Constellation Energy Group	Calvert Cliffs 1 & 2 (CE)	X	
Constellation Energy Group	Ginna (W)	X	
Dominion Connecticut	Millstone 2 (CE)	X	
Dominion Connecticut	Millstone 3 (W)	X	
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Duke Energy	Catawba 1 & 2, McGuire 1 & 2 (W)	X	
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Florida Power & Light Group	St. Lucie 1 & 2 (CE)		X
Florida Power & Light Group	Turkey Point 3 & 4, Seabrook (W)	X	
Nuclear Management Company	Prairie Island 1 & 2, Point Beach 1 & 2 (W)	X	
Nuclear Management Company	Palisades (CE)	X	
Omaha Public Power District	Fort Calhoun (CE)		X
Pacific Gas & Electric	Diablo Canyon 1 & 2 (W)	X	
Progress Energy	Robinson 2, Shearon Harris (W)	X	
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Southern Nuclear Operating Co.	Farley 1 & 2, Vogtle 1 & 2 (W)	X	
Tennessee Valley Authority	Sequoyah 1 & 2, Watts Bar (W)	X	
TXU Power	Comanche Peak 1 & 2 (W)	X	
Wolf Creek Nuclear Operating Co.	Wolf Creek (W)	X	

* Project participants as of the date the final deliverable was completed. On occasion, additional members will join a project. Please contact the WOG Program Management Office to verify participation before sending this document to participants not listed above.

Westinghouse Owners Group
International Member Participation* for WOG Project MUHP-5097, MUHP-5098, MUHP-5099, and CEOG Task 2059.

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		Yes	No
British Energy	Sizewell B	X	
Electrabel (Belgian Utilities)	Doel 1, 2 & 4, Tihange 1 & 3	X	
Kansai Electric Co., LTD	Mihama 1, Ohi 1 & 2, Takahama 1 (W)	X	
Korea Hydro & Nuclear Power Corp.	Kori 1, 2, 3 & 4 Yonggwang 1 & 2 (W)	X	
Korea Hydro & Nuclear Power Corp.	Yonggwang 3, 4, 5 & 6 Ulchin 3, 4 & 5 (CE)	X	
Nuklearna Electrarna KRSKO	Krsko (W)	X	
Nordostschweizerische Kraftwerke AG (NOK)	Beznau 1 & 2 (W)	X	
Ringhals AB	Ringhals 2, 3 & 4 (W)	X	
Spanish Utilities	Asco 1 & 2, Vandellos 2, Almaraz 1 & 2 (W)	X	
Taiwan Power Co.	Maanshan 1 & 2 (W)	X	
Electricite de France	54 Units	X	

* This is a list of participants in this project as of the date the final deliverable was completed. On occasion, additional members will join a project. Please contact the WOG Program Management Office to verify participation before sending documents to participants not listed above.

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LIST OF ACRONYMS AND ABBREVIATIONS

ADV	Atmospheric dump valve
AFW	Auxiliary feedwater
ART	Adjusted reference temperature
ASME	American Society of Mechanical Engineers
B&PV	Boiler and Pressure Vessel
B&W	Babcock & Wilcox
BV1	Beaver Valley Unit 1
CCDP	Conditional core damage probability
CDF	Core damage frequency
CE	Combustion Engineering
ECT	Eddy current examination
EFPY	Effective full-power year
EOL	End of life
EPRI	Electric Power Research Institute
FENOC	FirstEnergy Nuclear Operating Company
FCG	Fatigue crack growth
FP	Failure probability
FSAR	Final Safety Analysis Report
GQA	Graded quality assurance
HPI	High-pressure injection
HUCD	Heat-up and cool-down transient
HZP	Hot-zero power
IEF	Initiating event frequency
IGSCC	Intergranular stress corrosion cracking
ID	Inner diameter
ISI	In-service inspection
IST	In-service testing
LBLOCA	Large-break loss-of-coolant accident
LERF	Large early release frequency
LOCA	Loss-of-coolant accident
MBLOCA	Medium-break loss-of-coolant accident
MSIV	Main steam isolation valve
MSLB	Main steam line break
MT	Magnetic particle examination
NDE	Non-destructive examination
NMC	Nuclear Management Company
NRC	Nuclear Regulatory Commission
NSSS	Nuclear Steam Supply System
OC1	Oconee Unit 1
OD	Outer diameter
ORNL	Oak Ridge National Laboratory
PFM	Probabilistic fracture mechanics
PNNL	Pacific Northwest National Laboratory
POD	Probability of detection

LIST OF ACRONYMS AND ABBREVIATIONS (cont.)

PRA	Probabilistic risk assessment
PT	Liquid penetrant examination
PTS	Pressurized thermal shock
PVRUF	Pressurized Vessel Research User Facility
PWR	Pressurized water reactor
QA	Quality Assurance
RAI	NRC Request for Additional Information
RCP	Reactor coolant pump
RCS	Reactor Coolant System
RG	NRC Regulatory Guide
RI-ISI	Risk-informed ISI
RPV	Reactor pressure vessel
RT_{NDT}	Reference nil-ductility transition temperature
RV	Reactor vessel
RV ISI	Reactor Vessel In-service Inspection
RVID	Reactor vessel integrity database
SBLOCA	Small-break loss-of-coolant accident
SER	NRC Safety Evaluation Report
SG	Steam generator
SRP	Standard Review Plan
SRRA	Structural Reliability and Risk Assessment
SRV	Safety and relief valve
SSC	Structures, systems, and components
TH	Thermal hydraulics
TWCF	Through Wall Cracking Frequency
UT	Ultrasonic examination
VT	Visual examination
WOG	Westinghouse Owners Group

EXECUTIVE SUMMARY

The current requirements for the inspection of reactor vessel pressure-containing welds have been in effect since the 1989 Edition of *American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code*, Section XI, as supplemented by Nuclear Regulatory Commission (NRC) Regulatory Guide 1.150. The manner in which these examinations are conducted has recently been augmented by Appendix VIII of Section XI, 1996 Addenda, as implemented by the NRC in an amendment to 10CFR50.55a effective November 22, 1999. The industry has expended significant cost and man-rem exposure that have shown no service-induced flaws in the reactor vessel (RV) for ASME Section XI Category B-A, B-D, or B-J RV welds.

The objective of the methodology discussed in this report is to provide the technical basis for decreasing the frequency of inspection by extending the Section XI Inspection interval from the current 10 years to 20 years for ASME Section XI Category B-A, B-D, and B-J RV nozzle welds. Specific pilot studies have been performed on the Westinghouse, Combustion Engineering, and Babcock and Wilcox reactor vessel and NSSS designs. The results show that the change in risk associated with eliminating all inspections after the initial 10-year in-service inspection satisfies the guidelines specified in Regulatory Guide 1.174 for an acceptable change in risk for large early release frequency (LERF).

This conclusion is applicable to all Westinghouse, Combustion Engineering, and Babcock and Wilcox reactor vessel designs given that the applicable individual plant parameters are bounded by the critical parameters identified in Appendix A.

1 INTRODUCTION

The current requirements for the inspection of reactor vessel (RV) pressure containing welds have been in effect since the 1989 Edition of *American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code*, Section XI [1], as supplemented by the U.S. Nuclear Regulatory Commission (NRC) Regulatory Guide (RG) 1.150 [2]. The manner in which these examinations are conducted has been augmented by Appendix VIII of Section XI, 1996 Addenda, as implemented by NRC in an amendment to 10CFR50.55a effective November 22, 1999 [3]. The industry has expended significant cost and man-rem exposure by performing the required examinations that have shown no service-induced flaws in the RV for ASME Section XI Category B-A, B-D, or B-J RV nozzle welds. The current code criteria for the selection of examination areas and the frequency of examinations is not be an effective way to expend inspection resources.

The objective of this study was to verify that a reduction in frequency of volumetric examination of the RV full-penetration welds could be accomplished with an acceptably small change in risk. The methodology used to justify this reduction involved an evaluation of the change in risk associated with extending the 10-year in-service inspection (ISI) interval for three pilot plant bounding cases based on the calculated difference in the frequency of RV failure. RV failure was defined for this study to be the extension of a crack all the way through the RV wall. The difference in frequency of RV failure was evaluated using RG 1.174 [4] to determine if the values met the specified regulatory guidelines. The intent was that licensees can then use the results of this bounding assessment to demonstrate that their RV and plant are bounded by the generic analysis, thereby justifying a plant-specific extension in the RV weld inspection interval.

This study followed the approach specified in ASME Code Case N-691 [5], which provides guidelines for using risk-informed insights to increase the inspection interval for pressurized water reactor (PWR) vessel welds.

2 BACKGROUND

The original objective of the ASME B&PV Code, Section XI [1] ISI program was to assess the condition of pressure-containing components in nuclear power plants to ensure continued safe operation. If non-destructive examination (NDE) found indications that exceeded the allowable standards, examinations were extended to additional welds in components in the same examination category. If NDE found indications that exceeded the acceptance standards in those welds, then the examinations were extended further to similar welds in similar components, etc.

With respect to the method defined in this report, 100 percent of the present examination areas will be retained. The methodology is limited to justification of a reduction in the frequency of examination, i.e., increasing the time interval between inspections.

The original examination interval of 10 years was based on "wear-out" rate experience in the pre-nuclear utility and petrochemical process industries. As with some other Section XI ISI requirements, with no indications being found in the vessel welds under evaluation in this report, these inspections are decreasing in value with increasing industry experience to rely upon. The U.S. NRC has granted a number of exemptions to inspections for other areas and components (e.g., piping [6], reactor coolant pump motor flywheels [7], etc.) based on experience and man-rem reductions. This has been attributed to the combined design, fabrication, examination, and Quality Assurance (QA) rigor of the nuclear codes, and more careful control of plant operating parameters by the utilities.

A critical component of the justification of the interval extension is a fracture mechanics evaluation of the reactor vessel, which shows that flaws, if they do exist, would not grow to a critical size if the inspection interval is increased to more than 10 years. This can be demonstrated by selecting critical areas of the reactor vessel for the evaluation such as, the beltline, flange, and outlet nozzle regions. These locations are known to be areas of primary concern and are currently considered in ASME Section III, Appendix G [6] evaluations for protection against nonductile failure of the reactor vessel. As part of this study, a deterministic fracture mechanics evaluation of limiting locations in a typical geometry for a RV identified that the beltline region was the critical location with respect to the potential for growth of fatigue cracks. Fatigue crack growth is recognized as the primary degradation mechanism in the carbon and low alloy steel components in PWR Nuclear Steam Supply System (NSSS), that could contribute to any potential growth of existing flaws in the component base materials and weld metals.

Fatigue can be defined as repeated exposure to cyclic loading resulting from a variety of operating conditions or events (e.g., heatups, cooldowns, reactor trips). Design basis documents provided descriptions of the conditions that would contribute to cyclic fatigue. This information was used to identify and define the frequency of occurrence for each of the events that was considered when determining the potential for fatigue crack growth.

A technical consideration critical to success was the application of risk-informed assessment techniques to substantiate the deterministic fracture mechanics flaw growth evaluation. Risk assessment techniques provided a means to quantify and calculate cumulative results from contributing mechanisms and uncertainties associated with the critical parameters. A probabilistic fracture mechanics (PFM) methodology was used to consider the distributions and uncertainties in flaw numbers, flaw sizes, fluence, material properties, crack growth rate, stresses, and the effectiveness of inspections. The PFM methodology was also used to calculate the change in the frequency of RV failure due to a change in inspection interval. This change in RV failure frequency was used to evaluate the viability of such an inspection interval change. Recognized guidelines for evaluating the change in failure frequencies are provided in RG 1.174 [4] and the NRC risk assessment developed in conjunction with the current pressurized thermal shock (PTS) evaluations [8].

Significant work is on-going in the nuclear industry to investigate the impacts from PTS or "off-normal" plant transients that may be outside the current design basis. These transients are commonly understood to present the most severe challenge to RV structural integrity. The NRC effort to address PTS has identified FirstEnergy Nuclear Operating Company's (FENOC's) Beaver Valley Unit 1 (BV1), Nuclear Management Company's (NMC's) Palisades, and Duke Energy's Oconee Unit 1 (OC1) as the representative plants based on geometry and embrittlement for the Westinghouse, Combustion Engineering (CE), and Babcock and Wilcox (B&W) PWR designs. These are the primary PWR manufacturers in the U.S. and were evaluated by the NRC and Oak Ridge National Laboratory (ORNL) as part of the NRC PTS Risk Study [8].

This report summarizes the results from an evaluation of the extension of the inspection of ASME Section XI [1] Examination Category B-A and B-D welds in the reactor pressure vessel (RPV) and Category B-J welds to the RV nozzle from the current requirement of every 10 years to an extension of 20 years. It demonstrates that for the pilot plant reactor vessel geometry and fabrication history, any potential change in risk when the inspection interval is extended meets the change in risk evaluation guidelines defined in RG 1.174 [4]. The evaluation documented in this report considers FENOC's BV1 as the Westinghouse pilot plant. NMC's Palisades Plant and Duke Energy's OC1 are the respective Combustion Engineering (CE) and Babcock and Wilcox (B&W) pilot plants for this evaluation. To apply the results of this report to non-pilot plants, it must be shown, using the tables contained in Appendix A that the pilot plant evaluations for the respective design bound the non-pilot plant.

The following paragraphs address the current Section XI ISI requirements for PWR RV welds under consideration for the proposed extension. The following topics are included:

1. Reactor Vessel In-Service Inspection (RV ISI)
2. Location-specific ISI data from participating plants
3. The man-rem exposure and other costs of RV weld inspection
4. Generic RV weld experience at various plants
5. Development of the ISI interval extension methodology
6. Pilot plants
7. Safety impact

2.1 REACTOR VESSEL IN-SERVICE INSPECTION

Since its beginning, ASME B&PV Code, Section XI [1] has required inspections of weld areas of reactor vessels and other pressure-containing nuclear system components. The selection of inspection locations was based on areas known to have high-service factors and additional areas to provide a representative sampling for the condition of pressure-containing nuclear system components. While weld and adjoining areas were specified, it was recognized that the volumetric examination of the weld and adjoining base material would result in a significant degree of examination of the base metal.

Examination Volumes

Initially, for longitudinal and circumferential welds in a reactor vessel shell, Section XI required examination of 10 percent of the length of longitudinal welds, and 5 percent of the length of circumferential welds. Welds receiving exposure in excess of specified neutron fluence would require an inspection of 50 percent of the length. The 1977 Edition of Section XI increased the examination of RV welds from 5 or 10 percent of the length to 100 percent, with all welds examined in the first 10-year interval. Subsequent intervals required 100 percent examination of specified circumferential and longitudinal welds. The 1989 Edition of Section XI [1] extended the examination to include all welds.

There has been no report of structural failure or leakage from any full-penetration weld being addressed in this report in a PWR RV shell, globally. In volumetric examinations of these welds in ISIs performed in accordance with the requirements of Section XI (and RG 1.150 [2]), flaws identified in the original construction have been detected and were acceptable under Section XI requirements. These flaws have been monitored and to date, no growth has been identified. There has been no evidence of in-service flaw initiation in these welds.

Examination Approaches

The preceding discussion of RV welds addresses the Category B-A, RV seam welds of Table IWB-2500-1 of Section XI. Category B-D, RV nozzle welds and nozzle inner radius, and Category B-J RV nozzle-to-piping welds are also included in this evaluation.

The ultrasonic examinations (UTs) of these RV welds, as of the 1996 Addenda of Section XI, were conducted in accordance with Appendix I, I-2110. This Addenda requires Appendix VIII inspections for:

- Shell and head welds excluding flange welds
- Nozzle-to-vessel welds
- Nozzle inside radius region

Precedent for Change

There have been a number of revisions (often by ASME Code Case) to the Section XI ISI program that have eliminated or reduced the extent of examinations and tests based on successful operating experience and analytical evaluation. Examples of ASME Code Cases applicable to the RV and its piping connections include:

N-481 [9] Associated with cast austenitic pump casings. This was the first example of substituting an analysis plus a visual examination (VT) for a volumetric examination, for a Class 1 component.

N-560 [10] Permits a reduction in the examination of Class 1 Category B-J piping welds from 25 to 10 percent, provided a specified risk-importance ranking selection process is followed. This was a substantive reduction of an established Class 1 examination.

N-577 [11] N-578 [12]	Provide requirements for risk-informed ISI of Class 1, 2, and 3 piping. The cases provide different methods to achieve the same objective. This was the first use of the plant probabilistic risk assessment (PRA). Both methods have received extensive implementation in the U.S. and in several other countries in Europe and Asia.
N-613 [13]	Reduces the examination volume of Category B-D nozzle welds in adjacent material from 1/2 shell thickness to 1/2 inch. This permits a significant reduction in qualification and scanning time.
N-552 [14]	Permits computational modeling for the qualification of nozzle inner radius examination techniques, in lieu of qualification on a multitude of configurations.
N-610 [15]	Permits a K_{IR} curve in Appendix G, in lieu of a K_{IA} curve. Indirectly, this is beneficial to the pressure-temperature limit curve during plant startup.

Not all of the changes in Section XI, due to operating considerations, have led to a relaxation in inspection or evaluation requirements.

Over the past 10 years, there have also been a number of changes (often by code case) to the Section XI ISI program that have increased the extent of examinations and tests based on operating experience and analytical evaluation. The following examples of ASME Code Cases are limited to those applicable to the RV and its piping connections.

N-409 [16]	Introduced procedure and personnel qualification requirements for UT of intergranular stress corrosion cracking (IGSCC) in austenitic piping welds, a precursor to Appendix VIII, UT performance demonstration requirements.
N-512 [17]	Provided requirements for the assessment of RVs with low upper shelf Charpy impact energy levels.
N-557 [18]	Introduced requirements for in-place dry annealing of a PWR RV.

2.2 LOCATION-SPECIFIC ISI DATA FROM PARTICIPATING PLANTS

While it is known that the number of flaws found in RPV welds is very small, it is important to relate their number to the number of welds that have been examined over the past 30 years with no evidence of the development of service-induced flaws.

To develop location-specific ISI data from nuclear plants, ISI data on the RV weld categories noted above were gathered in a survey [19]. This information focused on service-induced flaws. It did not address the detection of original fabrication flaws, unless the flaws had grown due to service conditions. The response to this survey is summarized in Table 2-1.

Table 2-1 Summary of Survey Results on RV ISI Findings [19]

No. of Plants	Total Years of Service Prior to Survey	ASME Weld Category / Item	No. of Welds in Category	Welds with No Flaws	Welds with Flaws	Means of Detection[1]	Cause of Flaw/Failure
14	301	**B-A**					
		Shell, B1.10	112	112	0		
		Head. B1.20	105	105	0		
		Shell-to-flange. B1.30	16	16	0		One plant reported 3 indications that may be just scratches.
		Head-to-flange, B1.40	16	16	0		One plant reported 3 indications that may be just scratches.
		B-D					
		Nozzle-to-shell, B3.90	102	102	0		
		Nozzle inside radius B3.100	102	102	0		
		B-F					
		Dissimilar metal, B.5.10	84	84	0		
		B5.30	32	32	0		
		B-J					
		Piping, B9.10	64	64	0		
		B-K					
		Welded attach, B10.10	4	4	0		
		B-N					

Table 2-1 Summary of Survey Results on ISI Findings [19] (cont.)

No. of Plants	Total Years of Service Prior to Survey	ASME Weld Category	No. of Welds in Category	Welds with No Flaws	Welds with Flaws	Means of Detection[1]	Cause of Flaw/Failure
		Vessel interior, B13.10	34	34	0		
		Interior attach.-beltline, B13.50	6	6	0		
		Other interior attach., B13.60	53	53	0	VT-3, UT, ECT	One plant reported crack arrest holes drilled in core barrel.
		Core support struct., B13.70	41	5	0		

Note 1: VT = Visual Inspection, UT = Ultrasonic Inspection, ECT = Eddy Current Inspection

2.3 EXPOSURE AND COST REDUCTION

Data was gathered on CE and Westinghouse plants related to the cost of a typical RV ISI outage, as well as the cost of the exposure affecting the involved personnel [19]. The objective of this effort was to investigate the exposure and financial aspects of the RV ISI. The results of the survey were tabulated based on the probability of a life extension program (60 years), and the potential savings were calculated with regards to a proposed extension of the RV ISI interval to 20 years. The radiation exposure cost is contingent on the utility and is typically $15,000 to $20,000 per man-rem. A summary of the results is presented in Table 2-2.

Table 2-2 Savings on the Proposed Extension of RV ISI Interval from 10-Years to 20-Years (Per Plant) [19]

Probability of 20-Year Life Extension (%)		0%	50%	100%
Cost of Typical RV ISI Outage, $	min	506,410	759,615	1,012,820
	max	7,680,000	9,600,000	11,520,000
	average	3,878,521	5,391,656	7,115,317
Dose of Exposure, Man-rems	min	0.2	0.4	0.6
	max	6.5	9.75	13.0
	average	1.66	2.32	2.98
Cost of Dose of Exposure, $	min	2,492	4,984	7,476
	max	65,000	97,500	130,000
	average	20,611	28,856	37,101

As shown in Table 2-2, the savings associated with even the most conservative assumption, i.e., no life extension program (40 years) for any of the surveyed plants, are significant. The extension of the RV ISI interval to 20 years will save every unit an average of $3,878,521 for the cost of the outage, and 1.66 man-rems of exposure.

The saving values associated with the less conservative assumption of the guaranteed life extension program (60 years) for any of the surveyed plants are considerably higher. The extension of the RV ISI interval to 20 years will save every unit an average of $7,115,317 for the cost of outage, and 2.98 man-rems of exposure. The critical path outage time for RV inspections is approximately 3 ½ days. While this data was gathered for Westinghouse and CE designed plants, the savings for B&W designed plants are expected to be similar.

2.4 GENERIC REACTOR VESSEL WELD EXPERIENCE AT VARIOUS PLANTS

Section XI ISI requirements developed in the early 1970s were based on the detection of fatigue cracking in primary welds. This has not been substantiated by subsequent operating experience. Fatigue cracking in primary welds has not been a problem. Random sampling for the assessment of condition of pressure-containing components has not been effective; when leakage and other deterioration have been identified, it has been by examinations other than the Section XI ISI NDE.

Primary system failures/leakage have almost always been associated with dissimilar metal welds or control rod drive, bottom mounted instrumentation, or vent connections of the RV and its head. The latter connections are all partial penetration welds. They were not included in the survey, since the current effort does not propose to recommend changes to their present ISI interval requirements. Their examinations are not contingent on the removal of the reactor internals and the use of the RV inspection tool. Category B-F dissimilar metal welds, Category B-K welded attachments, and Category B-N interior attachment and support welds were not included in the inspection interval extension.

In many plants, the most highly stressed reactor vessel weld is the weld between the closure head flange and the dome. There have been no reports of degradation of this joint. This joint ranks quite low in its contribution to cumulative risk determined through typical PFM methods. Calculations [20] have shown that flaw growth due to fatigue would be extremely small, so that even pre-existing flaws that clearly exceed the acceptance standards would not be subject to measurable growth.

2.5 DEVELOPMENT OF ISI INTERVAL EXTENSION METHODOLOGY

The ISI interval extension methodology is primarily based on a risk analysis, including a PFM analysis of the effect of different inspection intervals on the frequency of reactor vessel failure due to postulated PTS transients. Reactor vessel failure is defined for the purposes of this study as the point which a crack has extended all the way through the RV wall. The likelihood of reactor vessel failure is postulated to increase with increasing time of operation due to the growth of pre-existing fabrication flaws by fatigue in combination with a decrease in reactor vessel toughness due to irradiation. Credible, postulated PTS transients that could potentially lead to reactor vessel failure must be considered to occur at the worst time in the life of the plant. The PFM methodology allows the consideration of distributions and uncertainties in flaw number and size, fluence, material properties, crack growth rate, stresses, and the effectiveness of inspections. The PFM approach leads to a conditional reactor vessel failure frequency due to a given loading condition and a prescribed inspection interval. All locations of interest in the reactor vessel can be addressed in a similar way or, as in the case of this study, a bounding approach can be used to minimize the areas receiving a detailed evaluation.

A feasibility study was performed [19] that showed that this fracture mechanics and risk methodology can be used to calculate the change in the frequency of reactor vessel failure due to a change in inspection interval and to evaluate the acceptability of the associated change in risk. The impact on plant safety from the change in risk presented in this study was based on the standards for risk-informed assessment as defined by RG 1.174 [4].

3 PILOT PLANT SUMMARY

The risk evaluations summarized in this report utilized the same pilot plants as used in the NRC PTS Risk Re-evaluation effort [8]. The NRC effort to address PTS risk identified FirstEnergy Nuclear Operating Company's (FENOC's) Beaver Valley Unit 1 (BV1), Nuclear Management Company's (NMC's) Palisades, and Duke Energy's Oconee Unit 1 (OC1) as the pilot plants. These pilot plant applications also used fleet-specific design transient data for the Combustion Engineering (CE) and Westinghouse designs. A typical generic heatup/cooldown transient was used for the Babcock & Wilcox (B&W) study. A study was also performed to determine the bounding location from among the applicable weld locations on a typical PWR reactor vessel. The results of all of these investigations are included in the following sections.

3.1 BOUNDING LOCATION

The focus of the evaluations for reactor vessel inspection interval extension was on the beltline of the RV. To confirm that the beltline location represented the bounding location for the reactor vessel, all locations currently required for examination in the reactor pressure vessel (RPV) needed to be identified and considered. The beltline weld locations were found to be the bounding locations primarily due to irradiation induced change in the fracture toughness. This was consistent with the location assumptions used to support the NRC PTS Risk Study [8]. Table 3-1 summarizes the current ISI requirements for RPV inspection as identified in Table IWB-2500-1 of the ASME B&PV Code, Section XI [1]. While this table identifies all welds with Section XI inspection requirements, this report only addresses the ISI interval extension of the Category B-A, B-D, and B-J welds.

Table 3-1	ASME Section XI [1] ISI Requirements for RPVs (ASME Section XI, Table IWB-2500-1)		
	Item No.	**RPV Location**	**Examination Requirement**
		Pressure Retaining Welds in Reactor Vessel	
B-A	B1.10	Shell Welds	Volumetric
B-A	B1.11	Circumferential	Volumetric
B-A	B1.12	Longitudinal	Volumetric
B-A	B1.20	Head Welds	Volumetric
B-A	B1.21	Circumferential	Volumetric
B-A	B1.22	Meridional	Volumetric
B-A	B1.30	Shell-to-Flange Weld	Volumetric
B-A	B1.40	Head-to-Flange Weld	Surface and Volumetric
B-A	B1.50	Repair Welds	Volumetric
B-A	B1.51	Beltline Region	Volumetric
		Full Penetration Welded Nozzles in Vessels	
B-D	B3.90	RPV Nozzle-to-Vessel Welds	Volumetric
B-D	B3.100	RPV Nozzle Inside Radius Section	Volumetric
		Pressure Retaining Dissimilar Metal Welds in Vessel Nozzles	
B-F	B5.10	RPV Nozzle-to-Safe End Butt Welds, NPS 4 or Larger	Surface and Volumetric
B-F	B5.20	RPV Nozzle-to-Safe End Butt Welds, Less Than NPS 4	Surface
B-F	B5.30	RPV Nozzle-to-Safe End Socket Welds	Surface
		Pressure Retaining Welds in Piping	
B-J	B9.10	NPS 4 or Larger	Surface and Volumetric
B-J	B9.11	Circumferential Welds	Surface and Volumetric
		Welded Attachments for Vessels, Piping, Pumps and Valves	
B-K	B10.10	Welded Attachments	Surface
		Interior of Reactor Vessel	
B-N-1	B13.10	Vessel Interior	Visual, VT-3
		Welded Core Support Structures and Interior Attachments to Reactor Vessels	
B-N-2	B13.50	Interior Attachments within Beltline Region	Visual, VT-1
B-N-2	B13.60	Interior Attachments Beyond Beltline Region	Visual, VT-3
		Removable Core Support Structures	
B-N-3	B13.70	Core Support Structure	Visual, VT-3

To confirm that the beltline was the limiting location, an assessment was performed using deterministic fracture mechanics that considered the following:

- Existence of 10-percent through-wall initial flaw
- In-service fatigue crack growth of the flaw due to normal plant operating transients
- 40 EFPY embrittlement throughout plant life
- Peak reactor vessel ID fluence assumed regardless of flaw depth, i.e., maximum embrittlement
- Design basis heat-up and cool-down transients
  - 500 cycles/40 years for CE NSSS
  - 200 cycles/40 years for Westinghouse NSSS
- 7 Weld Locations
  - Closure Head to Flange
  - Upper Shell to Flange
  - Lower Shell Transition
  - Bottom Head to Shell
  - Beltline
  - Inlet Nozzle to Safe End
  - Outlet Nozzle to Safe End

The study evaluated the effect of various ISI intervals by comparing the change in margins on ASME Code allowable flaw sizes for the respective locations. This approach was preceded by considering 3 iterative steps:

1. Select the first inspection interval, I1, based on the growth of the assumed initial flaw to a fraction of the tolerable flaw size.

2. Perform the inspection. If no defects larger than the assumed flaw size are found, the second inspection interval, I2, is the same as the first.

3. Continue subsequent inspections until actual flaws are detected that require repair or augmented inspections.

The results of the study are summarized in Figures 3-1 and 3-2. Inspection intervals were based on 10-, 20-, 30-, or 40-year inspection intervals over a 40-year plant life. Each reactor vessel location was evaluated by calculating the amount of crack extension that would occur due to fatigue crack growth over a 10-year period of operation. Each crack length was then evaluated for the maximum applied K_I from a transient. The ratio of the maximum allowable K_I, per the ASME Section XI [1] Appendix A criteria, to the maximum K_I applied, was used as a measure of the margin a flaw in a given location has to the acceptance criteria. Note that in Figure 3-1 the margins on the acceptance standard are greater than 1, except for the beltline region axial and circumferential flaws. This indicates that all of the flaw sizes in other locations are acceptable with varying degrees of margin. The margin less than one for the beltline locations is an indication that the assumed initial flaw size of 10-percent throughwall was greater than the acceptable flaw size. The other feature to note in Figures 3-1 and 3-2 is that, for each subsequent 10-year period that was evaluated, there was an insignificant change in the degree of margin for all of the locations. This observation was simply a reflection of the fact that the increments of fatigue crack growth of the flaws were so small that the applied K_I values were not changing. Therefore, the ratios of the applied to allowable K_I did not change.

These results confirmed that the beltline was the limiting location and that the change in fatigue crack growth increment for RPV flaws was insignificant relative to the inspection interval. While a specific number of design basis heat-up and cool-down transients was not analyzed for B&W designs in this bounding location assessment, it is reasonable to expect that the conclusions of this assessment would also be applicable to B&W plants due to similarities in the RV and NSSS designs.

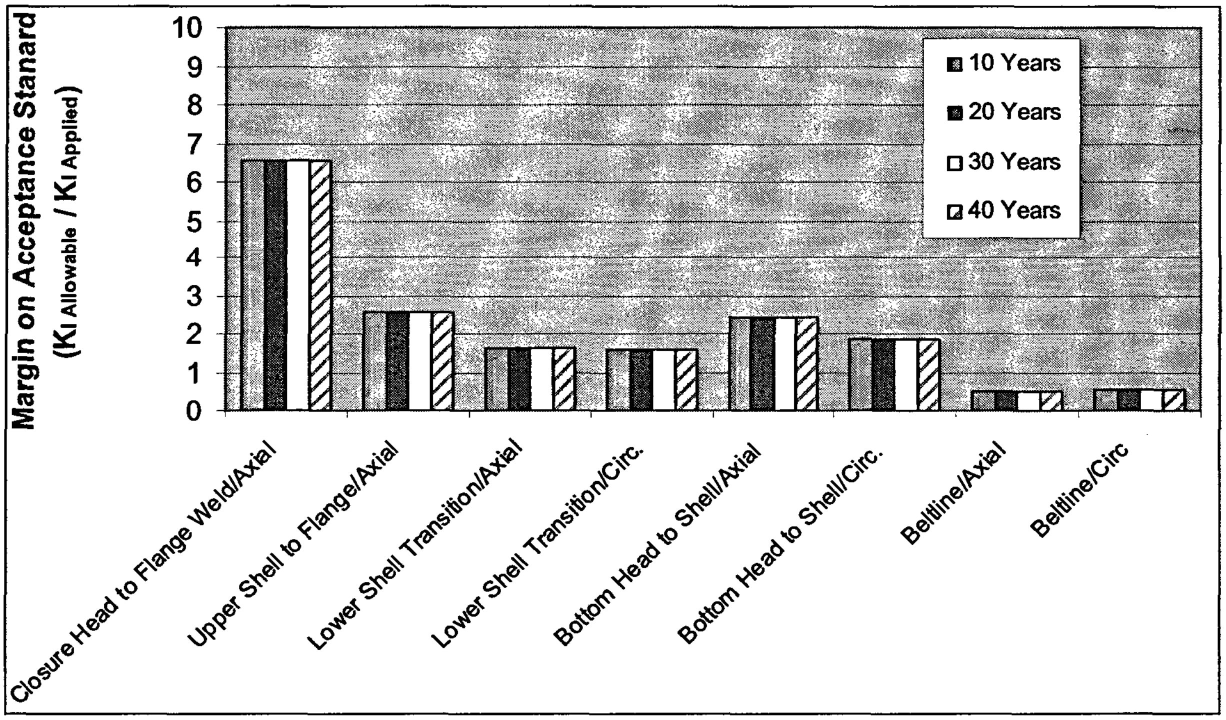


Figure 3-1 Comparison to Acceptance Criteria – Minimum Margins Code Allowable

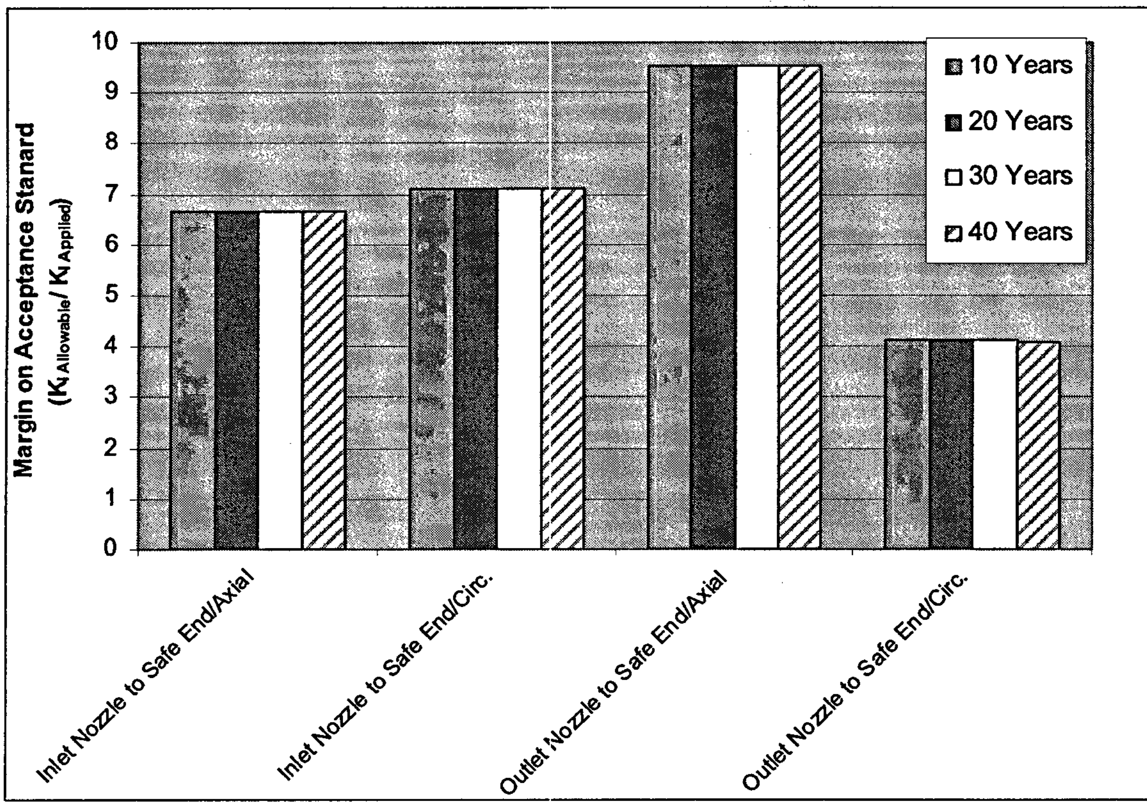


Figure 3-2 Comparison to Acceptance Criteria – Minimum Margins Code Allowable

3.2 BASIS FOR RISK DETERMINATION

As indicated in ASME Code Case N-691 [5], the application of risk-informed insights from PFM and risk analyses can be used to justify an increase from 10 to 20 years in the requirements of Section XI, IWB-2412 for the inspection interval for the examination of Category B-A and B-D welds in PWR reactor vessels, and Category B-J welds to the RV nozzles. The guidelines in Regulatory Guide 1.174 provide the basis for an acceptable change in risk resulting from an extension in inspection interval. As the basis for determining the change in risk, the inputs to the RV PFM and risk analyses included the following:

Accident Transients and Frequency

ASME Code Case N-691 [5] states that it is necessary to define a complete set of accident transients that can be postulated to realistically result in RV failure and their frequencies of occurrence. As previously mentioned, PTS events are viewed as providing the greatest challenge to PWR RPV structural integrity. For this reason, the pilot plant applications in this report used the PTS transients and frequencies from the NRC PTS Risk Study [8]. As part of the NRC study, probabilistic risk assessment (PRA) models were developed for each of the pilot plants using plant specific information [21, 22, 23]. These PRA models included an event-tree analysis that defined both the sequences of events that are likely to produce a PTS challenge to RPV structural integrity and the frequency with which such events can be expected to occur. The typical sequence of concern was cool-down and depressurization due to the initiating event, followed by repressurization due to high-pressure safety injection or charging. Historically, a small-break loss-of-coolant accident (SBLOCA) with low decay heat has been the sequence identified as a major contributor to PTS risk. However, other events considered included a large break in the main steam line upstream of the main steam isolation valves, a double-ended main steam line break (MSLB) upstream of the main steam isolation valves (MSIVs), small steam line break downstream of the MSIVs, and excessive feedwater flow, all with the reactor coolant pump (RCP) shutdown and multiple failures of the operator to take remedial action.

The PTS Risk Study utilized the plant specific PRA models to determine the possible sequences which could result in a PTS event for each of the pilot plants. Due to the large number of sequences which were identified, it was necessary to group (i.e., bin) sequences with like characteristics into representative transients that could later be analyzed using thermal-hydraulic codes. This resulted in 178 binned sequences for OC1, 118 for BV1, and 65 for Palisades. Thermal-hydraulic analyses were performed for each of these bins (i.e., representative transients) to develop time histories of temperature, pressure, and heat transfer coefficients [24]. These histories were then input into the PFM analysis to determine conditional probability of reactor vessel failure for each transient. From this analysis, it was determined that only a portion of the transients contribute to the total risk of RPV failure, while the remainder have an insignificant or zero contribution. The transients which were identified to be contributors to PTS risk were then used for the PFM analysis in the PTS study and for the pilot plant studies in this report. Consistent with the PTS Risk Study, 61 transients were analyzed for BV1, 30 for Palisades, and 55 for OC1 in this study on the impact of extending the RV ISI interval. Details of the transients are provided in Appendix D for BV1, Appendix H for Palisades, and Appendix L for OC1.

As part of the NRC PTS Risk Reevaluation Program, a study was performed to determine the applicability of the pilot plant detailed analyses to the remainder of the domestic PWR fleet. This "Generalization" Study [25] examined the results from the three detailed pilot plant studies (BV1, Palisades, and OC1) and identified a set of plant design and operational features considered to be important in determining whether or not certain types of overcooling scenarios are significant contributors to PTS. These features were then analyzed for five additional plants and compared to the features of the pilot plants. These five plants included the following:

- Salem Unit 1 (Westinghouse 4-loop plant comparable to Beaver Valley Unit 1)
- TMI Unit 1 (B&W plant comparable to Oconee Unit 1)
- Fort Calhoun (CE plant comparable to Palisades)
- Diablo Canyon (Westinghouse 4-loop plant comparable to Beaver Valley Unit 1)
- Sequoyah Unit 1 (Westinghouse 4-loop plant comparable to Beaver Valley Unit 1)

They were chosen for the generalization study on the basis of:

- having a high reference temperature metric (RT_{PTS}), which reflects their potential sensitivity to PTS,
- further demonstrating the applicability of the pilot plant analyses to the remainder of the fleet for the nuclear steam supply system (NSSS) vendors, and
- including plants having different limiting materials (i.e., welds, plates, and forgings).

It was determined in the generalization study that there were no differences in plant features that from a PRA, thermal hydraulic, and PFM standpoint would be expected to cause significant differences in the through wall cracking frequencies due to the postulated PTS scenarios. It was further concluded through the generalization study that the pilot plant results at a comparable embrittlement level could be applied to the remainder of the domestic PWR fleet.

Operational Transients and Cycles

ASME Code Case N-691 [5] states that the operational transients that contribute to fatigue crack growth and the number of cycles occurring each year must be identified. Typically, the start-up (heat-up) and shut-down (cool-down) events are the dominant loading conditions as seen in ASME Code Section XI, Non-Mandatory Appendix A [1] calculations for fatigue crack growth of an existing flaw.

For the purpose of the pilot plant studies in this report, an 80-year life for fatigue crack growth was used. This 80-year life envelopes plants seeking to obtain license extensions to 60 years and provides an additional margin of conservatism. The design basis transients for the pilot plants were reviewed and it was determined that the greatest contributor to fatigue crack growth for the pilot plants is heat-up and cool-down. Each transient represents a full heat-up and cool-down cycle between atmospheric pressure at room temperature and full-system pressure at 100-percent power operating temperature, and thus envelopes many transients with a smaller range of conditions. For the pilot plant evaluations, 7 heat-up and cool-down cycles per year were used for Westinghouse plants (BV1) and 13 cycles were used for CE plants (Palisades) to bound all the design basis transients for the respective PWR plant designs in each fleet. Based upon available information, 12 cycles were used for Babcock and Wilcox plants. For any B&W plant using the results of this WCAP to extend the reactor vessel ISI interval from 10 to 20 years, including the pilot plant (OC1), the fatigue crack growth for 12 heatup/cooldown transients per year will have to be verified to bound the fatigue crack growth for all design basis transients.

It is important to note that most plants' operational histories indicate that they will not reach this number of design transients by end of life (EOL) (80 years). However, this calculation was performed as a bounding analysis and the number of design transients was used rather than the number of operational transients so that plants with operational histories different than those of the pilot plants would be enveloped.

Initial Flaw Distribution

ASME Code Case N-691 [5] requires credible flaw distributions for a PWR reactor vessel. Significant work by Pacific Northwest National Laboratory (PNNL) and the NRC was performed to more completely specify the initial flaw size distributions and their densities for input into the NRC PTS Risk Study [8]. This work focused on making detailed destructive and non-destructive measurements of fabrication flaws in nuclear grade RPV welds and plates. Whenever possible, this experimental evidence was used exclusively or given the greatest "weight" in establishing the flaw distributions. In cases where experimental evidence was not sufficient, physical models and expert opinion were used to supplement the experimental evidence in establishing the flaw distributions. For the NRC PTS Risk Study, flaw distributions were developed for embedded flaws in welds, plates (includes forgings), and inner surface breaking flaws.

The weld flaw distribution was based on the highest densities of the Shoreham reactor vessel and the largest sizes of the PVRUF vessel. The embedded flaws are distributed evenly through the thickness of the weld. Flaws are postulated only in the same orientation as the weld. The flaw distribution represents a blended combination of weld types with 2% of the welds assumed to be repair welds, which have the largest flaw sizes.

Empirical evidence to support a plate flaw distribution is much more limited than that for welds. For this reason, the density for flaws of depths less than 6mm is 10% of that for weld flaws, while the density for flaws of depth above 6mm is 2.5% of that for weld flaws. Half of the simulated flaws are assumed to be axially oriented while the other half are assumed to be circumferentially oriented.

For weld and plate flaws, the pilot plant studies for the RV ISI interval extension study used the flaw distributions from the NRC PTS Risk Study directly. These densities are input into the FAVOR Code PFM analyses as flaw density files, P.dat (plate-embedded flaws) and W.dat (weld-embedded flaws). This is discussed further in the "PFM Computer Tool and Methodology" section.

The inner-diameter of the RPV is clad with a thin layer of stainless steel. Lack of inter-run fusion can occur between adjacent weld beads, resulting in circumferentially oriented cracks (the cladding in the RV is deposited circumferentially). However, none of the cracks discovered in the PNNL studies had broken through the cladding layer on the inside surface of the RV. Therefore, for the NRC PTS Risk Study [8], the BV1 and Palisades evaluations used multi-pass cladding with no surface breaking flaws. Multi-layer cladding is assumed to have no surface breaking flaws due to the small likelihood of two flaws aligning in two different weld layers. The OC1 pilot evaluation used an assumed surface flaw completely through the cladding with a density of 1/1000th of the embedded flaws through the vessel wall.

For this investigation on the impact of extending the RV ISI interval it is important to consider the effects of fatigue crack growth. Due to the fact that embedded flaws do not grow significantly due to fatigue, for the pilot plant studies, the presence of surface breaking flaws with an initial flaw depth equal to the cladding thickness was postulated. Therefore, for the pilot plant evaluations to bound all the plants of the same design, single-pass cladding was conservatively assumed. The initial flaw size and distribution was input into a fatigue crack growth and ISI analysis to determine a surface flaw density file after any inspections (ISI). Surface flaw density files were created two simulate two cases. The first case simulated inspections performed on a 10 year interval as currently required by the ASME Code. The second case simulated a single inspection performed after the first 10 years of operation with no subsequent inspection. These surface breaking flaw density files are then input into the PFM analysis as surface breaking flaw density file S.dat. The methodology for determining the flaw depth and density included in this file is described in the section on PFM and Computer Tool Methodology. Cladding details for the pilot plants are identified in Appendices B, F, and J.

Fluence Distribution

ASME Code Case N-691 [5] requires that the fluence distribution versus operating time, both axial and azimuthal, be based on plant-specific or bounding data for the current operating time and extrapolated as applicable to the end of the current 40 year license or for license renewal to 60 years.

For the pilot plant evaluations in this report, the input fluence distributions were taken directly from the NRC PTS Risk Study [8]. For the NRC PTS Risk Study a series of neutron transport calculations were performed to determine the neutron fluence on the inner-wall of the pilot plant RPVs. The modeling procedures were based on the guidance contained in NRC Reg. Guide 1.190. The models incorporated pilot plant specific geometry and operating data. The fluence for E>1MeV was calculated as a function of the azimuthal and axial location in the inner reactor vessel wall. The fluence was extrapolated from the current state point to various effective full-power years (EFPYs) assuming a linear extrapolation of the most recent operating cycles.

The fluences used in the RV ISI interval extension evaluations were for 60 EFPY for BV1 and Palisades and for fluences at 500 EFPY for OC1 to envelope license extension. 500 EFPY were used for OC1 rather than 60 EFPY because it is recognized that it is not the most embrittled RV in the B&W fleet. The use of 500 EFPY for OC1 should bound the embrittlement of the most highly embrittled RV in the B&W fleet when evaluated against the parameters identified in Appendix A. Representative fluence maps for BV1, Palisades, and OC1 at 32 EFPY, can be found in Appendices B, F, and J, respectively. While the magnitude of the fluence on these maps correspond to 32 EFPY rather than the 60 EFPY and 500 EFPY used in the pilot plant evaluations, the contour of the fluence relative to the reactor vessel weld layout still applies.

Material Fracture Toughness

ASME Code Case N-691 [5] states that the material fracture toughness of the limiting beltline plates and weld materials need to be based on the following plant-specific data:

- Physical and mechanical properties of the base metal, clad, and welds (e.g., copper and nickel content) and their uncertainties.
- Initial reference nil-ductility transition temperature (RT_{NDT}), including uncertainty
- ΔRT_{NDT} due to radiation embrittlement ,versus time and depth, including uncertainty
- Fracture toughness versus time and depth, including uncertainty

These reactor vessel material properties for the BV1, Palisades, and OC1 pilot plants evaluated in this report are identified in Appendices B, F, and J, respectively.

Embrittlement due to irradiation in RPV steels occurs due to matrix hardening and age hardening [8]. Based on the physical insights into these hardening mechanisms a relationship between material composition, irradiation-condition variables, and measurable quantities such as yield strength increase, Charpy-transition-temperature shift, and toughness-transition-temperature shift was established for the NRC PTS Risk Study [8]. Furthermore, a quantitative relationship was developed from the database of Charpy shift values generated in domestic reactor surveillance programs. The Eason and Wright irradiation shift model was developed by fitting this data. This model is used in the FAVOR Code for the NRC PTS Risk Study [8] and the RV ISI interval extension pilot plant studies to calculate the shift and irradiated reference temperature as a function of time.

The results of the significant work at ORNL, the NRC, and within industry to more completely specify the distribution on fracture toughness and its uncertainty for the NRC PTS Risk Study [8] are included in the FAVOR Code which is used for the pilot plant studies for RV ISI interval extension. The FAVOR Code includes fracture toughness models which are based on extended databases of empirically obtained K_{Ic} and K_{Ia} data points and include the effects of the statistical bias for direct measurement of fracture toughness (Master Curve Method). Furthermore, the FAVOR Code [26] uses the latest correlation on irradiated upper shelf fracture toughness.

It should be noted that along with the inspection of a weld, there is a specified amount of base metal inspected. In the FAVOR Code evaluation, if a flaw is placed within a weld that is adjacent to a more highly embrittled plate, the flaw is assigned the embrittlement characteristics of the plate rather than the weld and is assumed to fracture and propagate in the direction of the plate.

The NRC has proposed that through wall cracking frequency (TWCF) can be correlated to the embrittlement index (reference temperature) of the reactor vessel components. The correlation for determining plant specific TWCF based on the plant specific data mentioned can be found in Reference 27. This correlation takes into consideration the contribution to TWCF for each of the most limiting plate, axial weld, and circumferential welds. These individual TWCF contributions are then weighted based on experimental pilot plant data and summed to determine a total reactor vessel TWCF. For application to other plant reactor vessels, the plant specific TWCF must be equal to or less than the values used for the applicable pilot plants evaluated in this report (see Appendix A) at 60 EFPY.

Crack Growth Rate Correlation

ASME Code Case N-691 [5] requires that the basic physical models for fatigue crack growth due to operational transients (e.g., heat-ups, cool-downs, normal plant operating changes, and reactor trips) including the effects of uncertainties, be used for the PFM analysis. Also used are the basic physical models for crack growth during these transient events (i.e., the change in applied stress intensity and the corresponding change in flaw size) for the surface breaking flaws and their uncertainties.

The pilot-plant studies in this report included a probabilistic representation of the fatigue crack growth correlation for ferritic materials in water that was consistent with the previous and current models contained in Appendix A of the ASME Code, Section XI [1]. These correlations represented the behavior of the ferritic reactor vessel materials for all domestic PWRs. This probabilistic representation was consistent with that used by the NRC-supported pc-PRAISE code [28] and the NRC-approved SRRA tool for piping-risk informed ISI [29].

Cladding and Residual Stresses

ASME Code Case N-691 [5] requires that the residual stress distribution in welds and the cladding stress and its temperature dependence due to differential thermal expansion be considered. For the pilot plant studies for RV ISI interval extension, the residual stress distribution through the wall was taken from the NRC PTS Risk Study [8] and is described in the FAVOR Code Theory Manual [26]. This distribution is shown in Figure 3-3. The stress profile was determined for the NRC PTS Risk Study thorugh experiments in which a radial slot was cut in a longitudinal weld in a shell segment from an actual RPV and the deformation of the slot was measured after cutting. Finite element analysis was used to determine the residual stress profile from the measured deformations. The cladding stress used in the pilot plant studies was taken from the NRC PTS Risk Study. The cladding temperature dependence due to differential thermal expansion was based on a stress free temperature of 468°F, which is consistent with that used in the NRC PTS Risk Study [8].

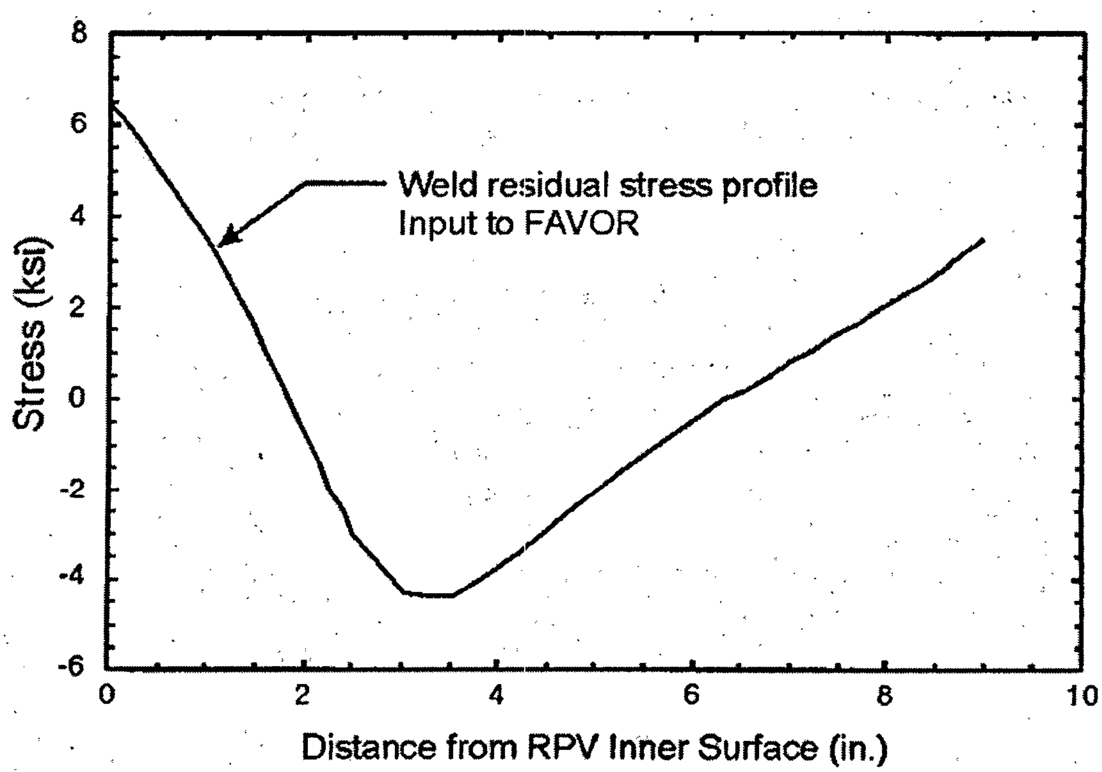


Figure 3-3 Weld Stress Profile

Effectiveness of ISI

The essential requirement for an effective volumetric examination in ASME Code Case N-691 [5] is that it be conducted in accordance with Section XI Appendix VIII [1] or RG 1.150 [2].

The following effects also need to be considered along with the change in ISI interval:

- Extent of inspection (percent coverage)
- Probability of detection (POD) with flaw size
- Repair criterion for removing flaws from service

The POD should correlate to the respective examination method for the RV weld of interest.

The basis for the probability of flaw detection used in the pilot plant studies for the RV ISI interval extension was taken from studies performed at the EPRI NDE Center on the detection and sizing qualification of ISIs on the RV beltline welds [30]. Figure 3-4 shows the probability of detection with respect to flaw size used in the pilot studies in this report.

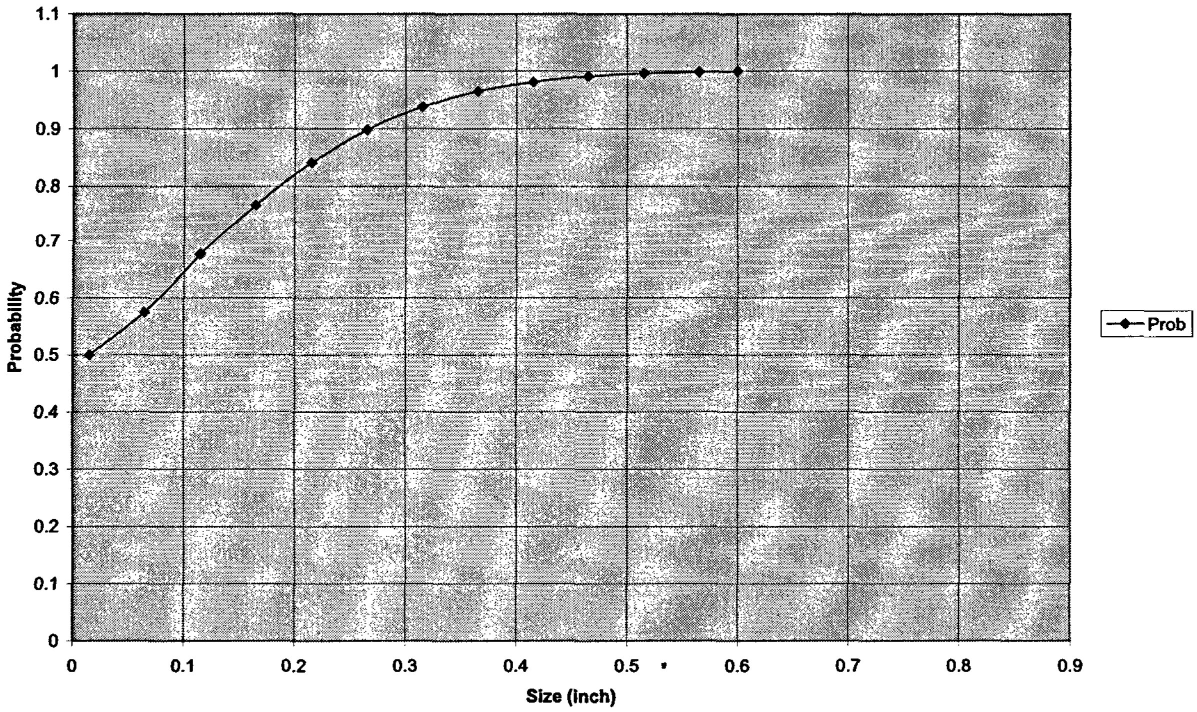


Figure 3-4 ISI Detection Probability

For the pilot plant evaluations, examinations were assumed to be conducted in accordance with Section XI Appendix VIII [1], so that Figure 3-4 could be used. Flaws that were detected were assumed to be repaired with the repaired area returned to a flaw-free condition. If the quality of inspection is not as good as assumed (e.g. ISI per Regulatory Guide 1.150) or the quality of the repair is less than 100 percent, then the result would be fewer flaws found and fewer flaws removed during repair, resulting in less difference in risk from one inspection interval to another. Therefore, the pilot plant studies conservatively calculated a larger potential difference in risk by maximizing the benefits of inspection.

Impact of Other ASME Code Cases on RPV Inspection

While no ASME Code Cases have been found that directly overlap the actions included in ASME Code Case N-691 [5], there are related ASME Code Cases and "problem areas" that may affect implementation of the Code Case. ASME Code Cases that concern reactor vessel inspections but do not affect the applicability of the Code Case are identified in the following:

ASME Code Case N-697 [31] addresses Examination Requirements for PWR Control Rod Drive and In-Core Instrumentation Housing Welds. It adds requirements for examination of in-core instrumentation housing welds greater than 2" Nominal Pipe Size to Examination Category B-O. If these UT or surface examinations of the housing weld inner surface were conducted from inside the RPV, they could result in examination intervals incompatible with effective implementation of N-691 [5]. However, these welds are not inspected from inside the RPV and, therefore, there is no impact.

A top priority in Section XI is to work with the Material Reliability Program Alloy 600 Issue Task Group to identify and incorporate changes needed in the examination of affected partial penetration and dissimilar metal welds. This could result in incompatible examination intervals for Examination Category B-F welds to reactor vessel nozzles, and dissimilar metal welds in Examination Category B-J not covered by Category B-F. A possible approach for some plants, where access permits, would be to examine these welds from the pipe outer diameter (OD) at alternate 10-year intervals, and from the inner diameter (ID) during the Case N-691 [5] examinations.

ASME Code Case N-700 [32] addresses Examination Category B-K, surface examination of welded attachments. It permits examination of a single welded reactor vessel attachment each inspection interval.

ASME Code Case N-648-1 [33] permits a VT-1 visual examination of a reactor vessel nozzle inner radius in lieu of a volumetric examination. Applicability of this Code Case would not be affected by the increased examination interval.

ASME Code Case N-624 [34] provides for modification of the sequence of successive examinations. The increased examination interval would be applicable.

ASME Code Case N-623 [35] permits deferral to the end of the interval of shell-to-flange and head-to-flange welds of a reactor vessel. The methodology of Case N-691 [5] would not be affected by application of this Code Case.

ASME Code Case N-615 [36] permits ultrasonic examination as a surface examination method for Category B-F and B-J piping welds of 4" Nominal Pipe Size and larger. It would be compatible with the increased examination interval.

ASME Code Case N-613-1 [37] reduces the nozzle weld examination volume of Examination Category B-D. It would be compatible with the increased examination interval.

ASME Code Case N-598 [38] provides alternatives to the required percentages of examinations each inspection period. ASME Code Case N-691 [5] would increase the length of the inspection period but would not affect the percentage requirements.

Impacts on Risk-Informed Piping ISI Programs

If the Category B-J piping welds to the RPV nozzles are included in a piping risk-informed inspection program, the impact on the piping program due to the extension in inspection interval must be evaluated per the requirements of ASME Code Case N-691 [5]. It must be determined whether extending the inspection interval for the Category B-J welds included in the risk-informed piping program will be negatively impact (e.g., change the risk ranking of the piping segments) the piping program. If the program is negatively impacted, changes must be made to the program to address the impact.

For the pilot plant evaluations in this report, BV1 does not have Category B-J welds to the RV and the RI-ISI for piping program is not impacted. For Palisades, the Category B-J welds to the RV were included in the RI-ISI piping program, but were selected as defense-in-depth examinations and not credited in the delta risk evaluation. Therefore, extending the inspection interval for these welds would not impact the RI-ISI program. OC1 does not have a RI-ISI for piping program. Therefore, the extension of the inspection interval is not a concern for the OC1 Category B-J welds in this regard.

Probabilistic Fracture Mechanics Computer Tool and Methodology

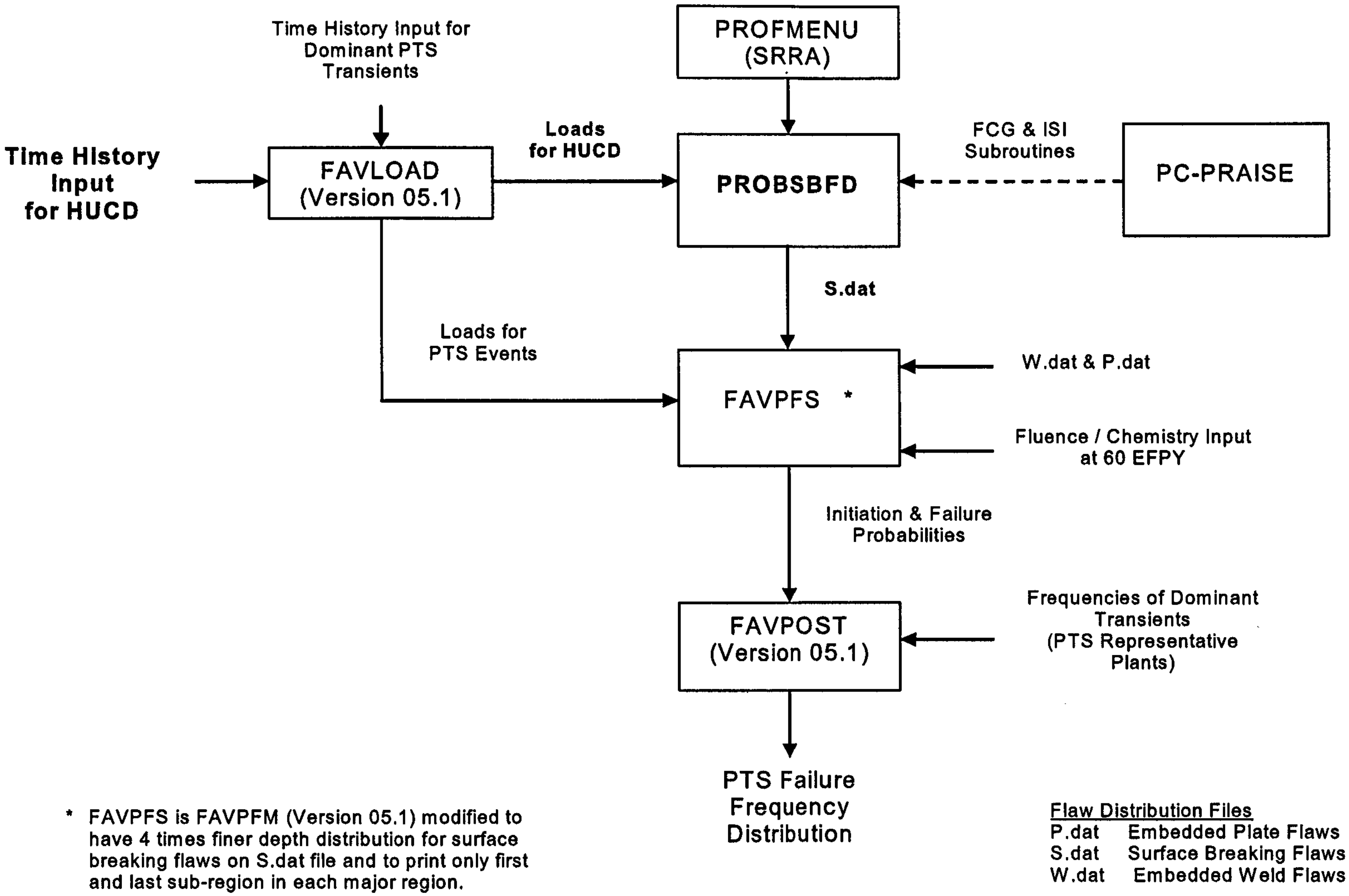
For the pilot-plant applications of the PFM methodology, the failure frequency distributions for all postulated flaws in the RV were calculated using the latest version (05.1) of the FAVOR code [26]. The Fracture Analysis of Vessels – Oak Ridge (FAVOR) computer program was developed as part of the NRC PTS Risk Study [8]. It is a program that performs a probabilistic analysis of a nuclear reactor pressure vessel when subjected to events in which the reactor pressure vessel wall is exposed to time-varying thermal-hydraulic boundary conditions.

To run the FAVOR code, 3 modules (FAVLOAD, FAVPFM and FAVPOST) and various input files were required as shown in Figure 3-5. In the NRC PTS Risk Study [8], the effects of fatigue crack growth and ISI were not considered. However, to perform the risk evaluation for changing the inspection interval from 10 to 20 years, these effects were quantified. Program PROBSBFD (Probabilistic Surface Breaking Flaw Density) was developed to include these effects by modifying the surface-breaking flaw input file to FAVOR (S.dat) as shown in Figure 3-5.

The first module in FAVOR is the load module, FAVLOAD, where the thermal-hydraulic time histories are input for the dominant PTS transients. For each PTS transient, deterministic calculations are performed to produce a load-definition input file for FAVPFM (FAVPFS is also used in this analysis). These load-definition files include time-dependent, through-wall temperature profiles, through-wall circumferential and axial stress profiles, and stress-intensity factors for a range of axially and circumferentially oriented embedded and inner surface-breaking flaw geometries (both infinite and finite-length).

The FAVPFS module in Figure 3-5 is a modification of the FAVPFM module, which is the second module contained in the FAVOR code that was used in the NRC PTS risk study. The modification allows FAVPFS to have a 4 times finer depth distribution for surface breaking flaws in S.dat. The modification also reduces the output by printing only the first and last sub-region in each major region. The FAVPFS FAVOR module uses the input flaw distributions (e.g., S.dat, W.dat, and P.dat), the loads for the PTS events from the FAVLOAD module and fluence/chemistry input data at 60 EFPY (effective full-power years) to calculate the initiation and failure probabilities for each PTS transient.

The FAVPOST post-processor is the third module in FAVOR. It combines the distributions of initiating frequencies for the dominant PTS transients with the results of the PFM analysis (performed with the FAVPFS module) to generate probability distributions for the frequencies of reactor vessel crack initiation and reactor vessel failure. This module also generates statistical information on these distributions and the distributions for the conditional probabilities of reactor vessel crack initiation and failure for each PTS transient included in the risk analysis.



* FAVPFS is FAVPFM (Version 05.1) modified to have 4 times finer depth distribution for surface breaking flaws on S.dat file and to print only first and last sub-region in each major region.

Figure 3-5 Software and Data Flow for Pilot Plant Analyses

The PROBSBFD code was specifically developed for the RV ISI interval extension project and verified in accordance with the Westinghouse Quality Assurance requirements. This program utilizes the Westinghouse Structural Reliability and Risk Assessment (SRRA) library program, which provides standard input and output, including probabilistic analysis capabilities (e.g., random number generation and importance sampling). PROBSBFD was used to develop 1000 random surface breaking flaw distributions that fed into the FAVPFS module via an input file (S.dat is the default name). The loads were determined using the FAVLOAD module, for the input with time histories of temperature, pressure, and heat transfer characteristics for the operational transients (e.g., heat-up and cool-down) that could grow the initial flaws by means of fatigue. The applied stress intensity factor (K) at various times and various depths through the reactor vessel wall were taken directly from the FAVLOAD output file and input into PROBSBFD (FAVLOADS.dat for PROBSBFD).

The beneficial effects of ISI were modeled in the same way as in the NRC's probabilistic analysis code pc-PRAISE [28] and the SRRA Code [29] used with the WOG/ASME piping risk-informed in-service inspection (RI-ISI) program. Specifically, only the flaws not detected during an ISI exam, at 10 years for example, remained. For example, if the probability of detection for the first inspection was 90 percent, then the flaw density was effectively multiplied by 10 percent for input to the next iteration. The effects of subsequent inspections, where the probability of detection was increased because the flaw was bigger (see Figure 3-4), could be either cumulative or independent.

For each of the 1000 simulations performed by PROBSBFD, the initial flaw depth and density were defined. Four aspect ratios, 2, 6, 10, and infinite, were considered. For each time-step and flaw-aspect ratio, the effects of ISI, the stress intensity factors, and the random crack growth were calculated. After all the time steps were completed, the distribution of flaw densities by depth and aspect ratio were written to a surface-breaking, flaw-distribution input file for FAVPFS, which was in the same format as the default S.dat file (see Figure 3-5).

3.3 RESULTS FOR THE WESTINGHOUSE PILOT PLANT: BV1

Reactor vessel failure frequencies were calculated for BV1 for two cases corresponding to the two surface flaw density files discussed in the section on "Initial Flaw Distribution". These cases were referred to as "ISI Every 10 Years" and "10-year ISI Only". As the names imply, the "ISI Every 10 Years" case simulates the current ASME Code required inspections while the "10-year ISI Only" case simulates a discontinuation of inspections after the first 10-year ISI. Statistically, the difference between the mean failure frequencies for the "ISI Every 10 Years" case and the "10-year ISI Only" case is insignificant. This is due to the fact that the difference between the mean values is less than the standard error for each of the cases. However, to calculate a change in risk for comparison to regulatory guidelines, a change in failure frequency was conservatively calculated based on the difference between an "Upper Bound" and a "Lower Bound." The Lower Bound was determined by subtracting 2 times the standard error as reported by FAVPOST from the mean value of the "ISI Every 10 Years" case. The Upper Bound was determined by adding 2 times the standard error as reported by FAVPOST to the mean value of the "10-Year ISI Only" case.

Elimination of ISI after the first 10-year ISI for the BV1 RPV results in a difference in failure (through-wall flaw) frequency of less than 3E-09. A summary table of the results of the evaluation are included in Table 3-2. The results reflect the maximum statistically calculated value for the potential change in risk at a number of reactor vessel simulations at which the Monte Carlo statistical analysis has reached a stable solution. The difference between the Upper Bound and Lower Bound represents the bounding difference between the 10-year inspection interval currently applicable under ASME criteria and elimination of all future inspections following an inspection within the first 10 years of operation.

This change in failure frequency is acceptable per the regulatory guidance discussed in Section 4.1. Transient input was based on design basis transients and the transients used in the NRC PTS Risk Study [7]. The input data included consideration of postulated life extension to 60 EFPY. The FAVPOST outputs for the cases presented in Table 3-2 are presented in Appendix E. As previously mentioned in Section 3.1, BV1 does not have a Category B-J weld to the reactor vessel and the RI-ISI piping program is not impacted by the extended inspection interval.

Table 3-2 BV1 Reactor Vessel Failure Frequency Results	
10-Year ISI Only (Mean Value / Standard Error)	5.04E-09 / 4.83E-10
Upper Bound Value	6.01E-09
ISI Every 10 Years (Mean Value / Standard Error)	4.10E-09 / 2.89E-10
Lower Bound Value	3.52E-09
Bounding Difference in Risk	2.49E-09

The mean effects of fatigue crack growth and ISI on the surface breaking flaw density for 1000 simulations are shown in Figures 3-6 and 3-7. These figures plot the flaw density as a function of the flaw depth for the cases of one initial 10-year ISI, a 10-year ISI interval, and a 20-year ISI interval. These plots display the results for the 10-to-1 and infinite aspect ratio sizes. The PROBSBFD outputs used to generate these plots are included in Appendix C. The crack growth and density reduction due to ISI would both be reduced for the flaw length-to-depth aspect ratios of 2-to-1 and 6-to-1 also considered in the pilot plant study.

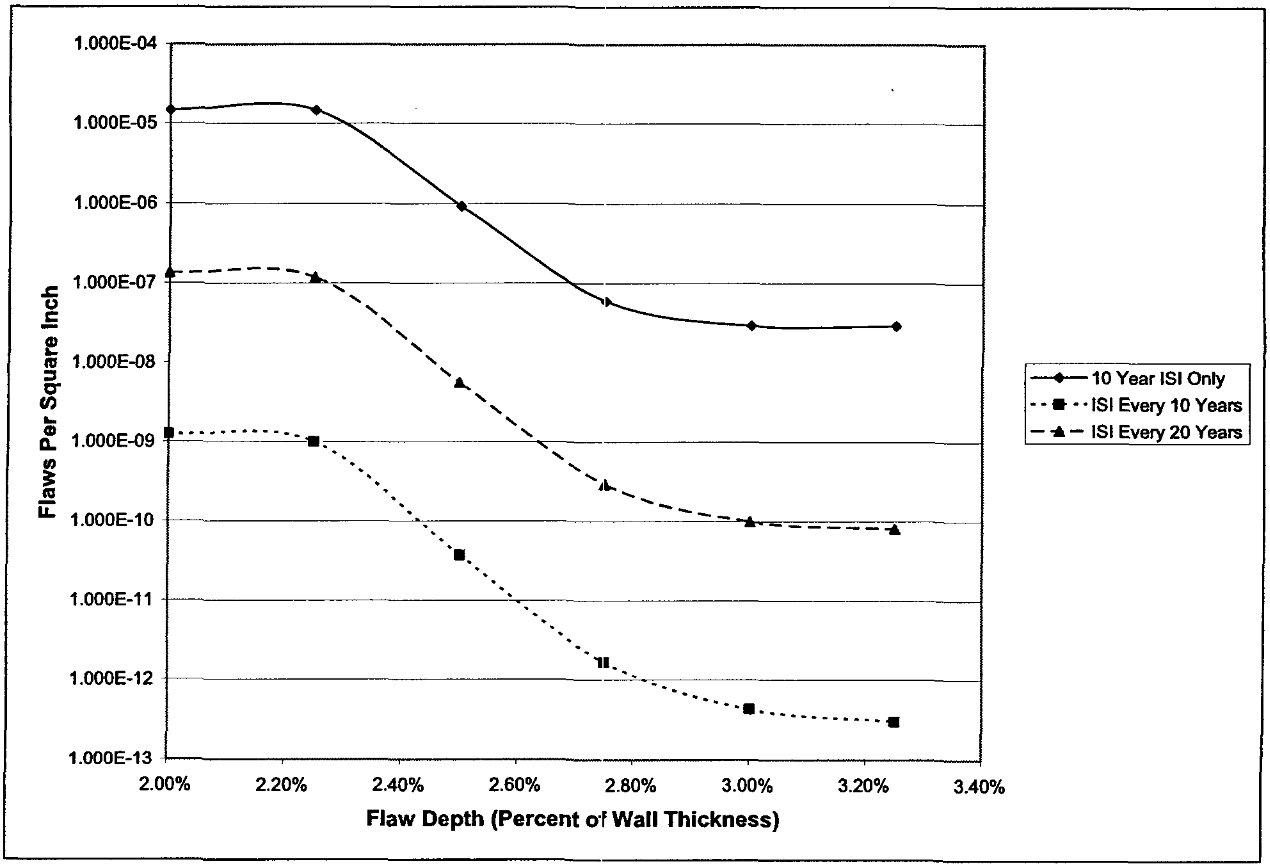


Figure 3-6 Growth of Flaws with an Aspect Ratio of 10 for BV1

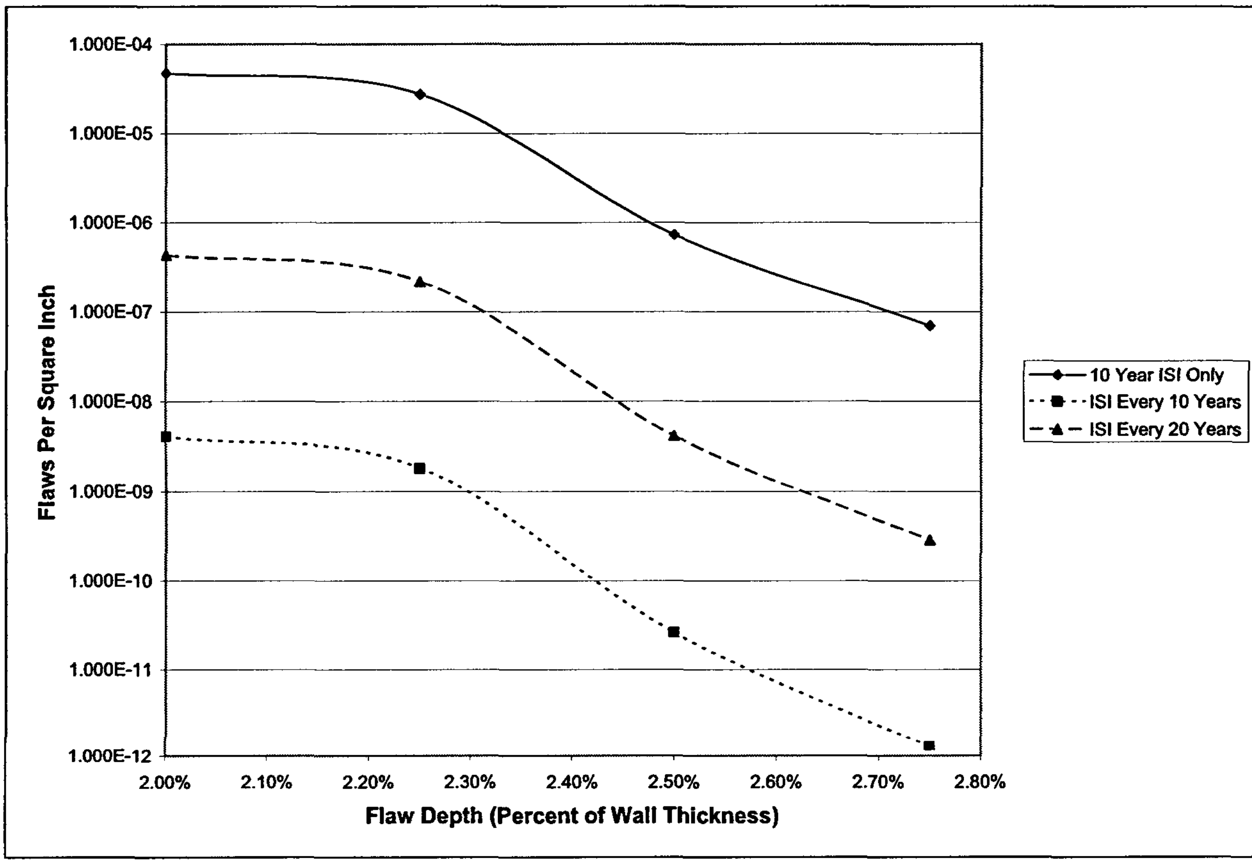


Figure 3-7 Growth of Flaws with an Infinite Aspect Ratio for BV1

3.4 RESULTS FOR THE COMBUSTION ENGINEERING PILOT PLANT: PALISADES

Reactor vessel failure frequencies were calculated for Palisades for two cases corresponding to the two surface flaw density files discussed in the section on "Initial Flaw Distribution". These cases were referred to as "ISI Every 10 Years" and "10-year ISI Only". As the names imply, the "ISI Every 10 Years" case simulates the current ASME Code required inspections while the "10-year ISI Only" case simulates a discontinuation of inspections after the first 10-year ISI. While the failure frequency for the "ISI Every 10 Years" case is higher than the "10-Year ISI Only" case, statistically, the difference between the mean failure frequencies for the "ISI Every 10 Years" case and the "10-year ISI Only" case is insignificant. This is due to the fact that the difference between the mean values is less than the standard error for each of the cases. However, to calculate a change in risk for comparison to regulatory guidelines, a bounding change in failure frequency was calculated based on the difference between an "Upper Bound" and a "Lower Bound." The Lower Bound was determined by subtracting 2 times the standard error as reported by FAVPOST from the mean value of the "ISI Every 10 Years" case. The Upper Bound was determined by adding 2 times the standard error as reported by FAVPOST to the mean value of the "10-Year ISI Only" case.

Elimination of ISI after the first 10-year ISI for the Palisades RPV results in a bounding difference in failure (through-wall flaw) frequency of less than 5E-09. A summary table of the results of the evaluation are included in Table 3-3. The results reflect the maximum statistically calculated value for the potential change in risk at a number of reactor vessel simulations at which the Monte Carlo statistical analysis has reached a stable solution. The difference between the Upper Bound and Lower Bound represents the bounding difference between the 10-year inspection interval currently applicable under ASME criteria and elimination of all future inspections following an inspection within the first 10 years of operation.

This change in failure frequency is acceptable per the regulatory guidance discussed in Section 4.1. Transient input was based on design basis transients and the transients used in the NRC PTS Risk Study. The input data included consideration of postulated life extension to 60 EFPY. The FAVPOST outputs for the cases presented in Table 3-3 are presented in Appendix I. As previously mentioned in Section 3.1, the Category B-J welds were included in the Palisades RI-ISI piping program, but were selected as defense-in-depth examinations and not credited in the delta risk evaluation. Therefore, extending the inspection interval for these welds would not impact the RI-ISI program.

Table 3-3 Palisades Reactor Vessel Failure Frequency Results	
10-Year ISI Only (Mean Value / Standard Error)	1.54E-08 / 1.62E-09
Upper Bound Value	1.86E-08
ISI Every 10 Years (Mean Value / Standard Error)	1.67E-08 / 1.23 E-09
Lower Bound Value	1.42E-08
Bounding Difference in Risk	4.40E-09

The mean effects of fatigue crack growth and ISI on the surface breaking flaw density for 1000 simulations are shown in Figures 3-8 and 3-9. These figures plot the flaw density as a function of the flaw depth for the cases of 1 initial 10-year ISI, a 10-year ISI interval, and a 20-year ISI interval. These plots display the results for the of 10-to-1 and infinite aspect ratio sizes. The PROBSBFD outputs used to generate these plots are included in Appendix G. The crack growth and density reduction due to ISI would both be reduced for the flaw length-to-depth aspect ratios of 2-to-1 and 6-to-1 also considered in the pilot plant study.

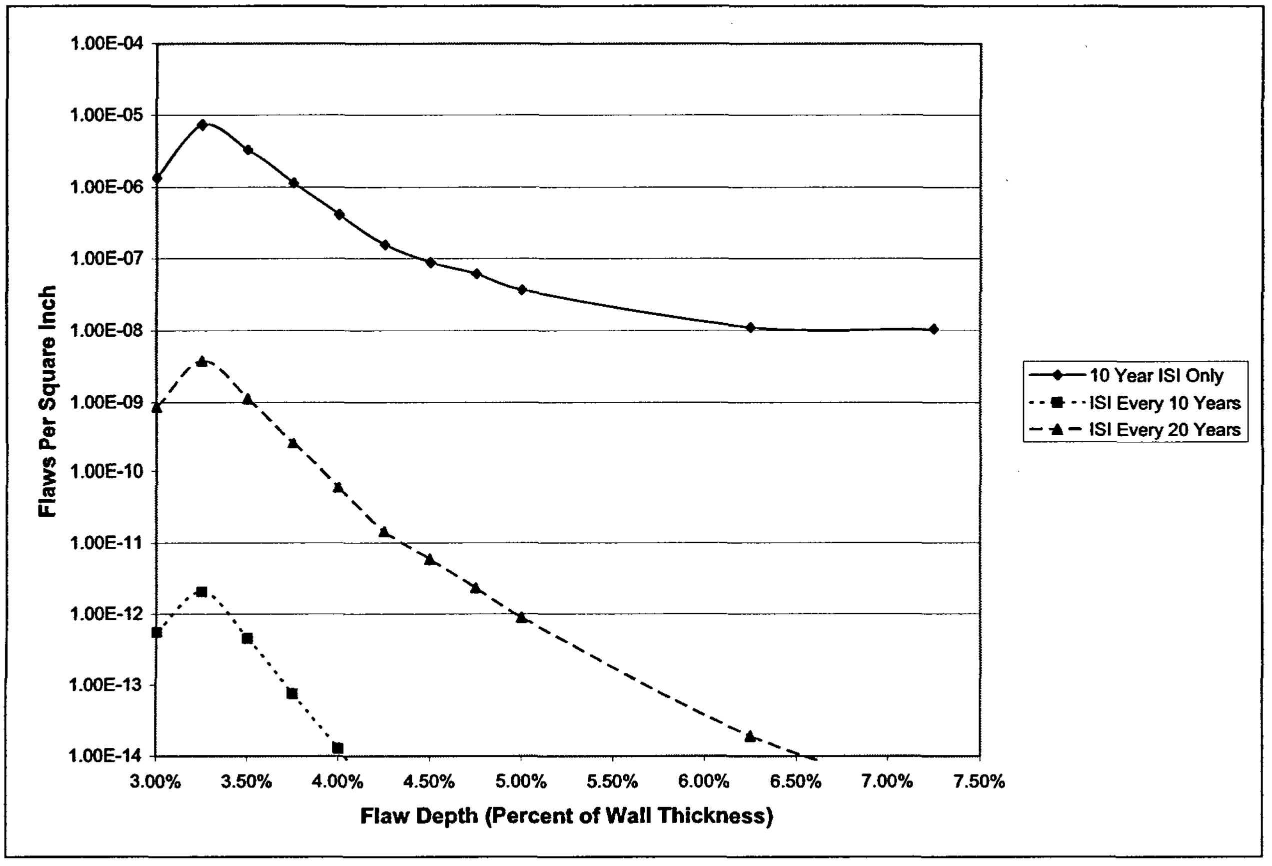


Figure 3-8 Growth of Flaws with an Aspect Ratio of 10 for Palisades

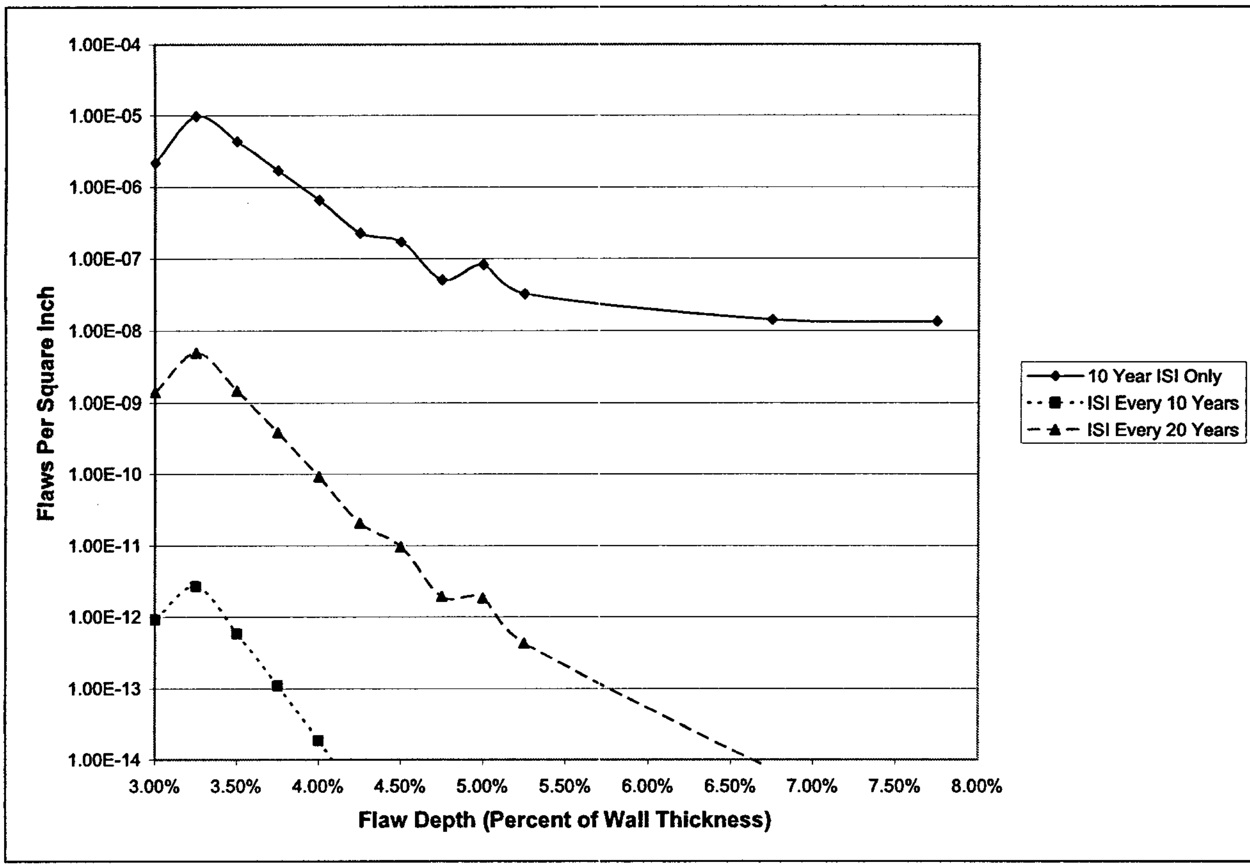


Figure 3-9 Growth of Flaws with an Infinite Aspect Ratio for Palisades

3.5 RESULTS FOR THE BABCOCK AND WILCOX PILOT PLANT: OC1

Reactor vessel failure frequencies were calculated for OC1 for two cases corresponding to the two surface flaw density files discussed in the section on "Initial Flaw Distribution". These cases were referred to as "ISI Every 10 Years" and "10-year ISI Only". As the names imply, the "ISI Every 10 Years" case simulates the current ASME Code required inspections while the "10-year ISI Only" case simulates a discontinuation of inspections after the first 10-year ISI. While the failure frequency for the "ISI Every 10 Years" case is higher than the "10-Year ISI Only" case, statistically, the difference between the mean failure frequencies for the "ISI Every 10 Years" case and the "10-year ISI Only" case is insignificant. This is due to the fact that the difference between the mean values is less than the standard error for each of the cases. However, to calculate a change in risk for comparison to regulatory guidelines, a bounding change in failure frequency was calculated based on the difference between an "Upper Bound" and a "Lower Bound." The Lower Bound was determined by subtracting 2 times the standard error as reported by FAVPOST from the mean value of the "ISI Every 10 Years" case. The Upper Bound was determined by adding 2 times the standard error as reported by FAVPOST to the mean value of the "10-Year ISI Only" case.

Elimination of ISI after the first 10-year ISI for the OC1 RPV results in a difference in failure (through-wall flaw) frequency of 8E-10. A summary table of the results of the evaluation are included in Table 3-4. The results reflect the maximum statistically calculated value for the potential change in risk at a number of reactor vessel simulations at which the Monte Carlo statistical analysis has reached a stable solution. The difference between the Upper Bound and Lower Bound represents the bounding difference between the 10-year inspection interval currently applicable under ASME criteria and elimination of all future inspections following an inspection within the first 10 years of operation.

This change in failure frequency is acceptable per the regulatory guidance discussed in Section 4.1. Transient input was based on design basis transients and the transients used in the NRC PTS Risk Study. The input data included consideration of postulated life extension to 60 EFPY. The FAVPOST outputs for the cases presented in Table 3-4 are presented in Appendix M. As previously mentioned in Section 3.1, the Category B-J welds were not inspected as part of OC1's RI-ISI piping program and are therefore not impacted by the extended inspection interval.

Table 3-4 OC1 Reactor Vessel Failure Frequency Results	
10-Year ISI Only (Mean Value / Standard Error)	2.06E-09 / 2.71E-10
Upper Bound Value	2.60E-09
ISI Every 10 Years (Mean Value / Standard Error)	2.18E-09 / 1.87E-10
Lower Bound Value	1.81E-09
Bounding Difference in Risk	7.96E-10

The mean effects of fatigue crack growth and ISI on the surface breaking flaw density for 1000 simulations are shown in Figures 3-10 and 3-11. These figures plot the flaw density as a function of the flaw depth for the cases of 1 initial 10-year ISI, a 10-year ISI interval, and a 20-year ISI interval. These plots display the results for the 10-to-1 and infinite aspect ratio sizes. The PROBSBFD outputs used to generate these plots are included in Appendix K. The crack growth and density reduction due to ISI would both be reduced for the flaw length-to-depth aspect ratios of 2-to-1 and 6-to-1 also considered in the pilot plant study.

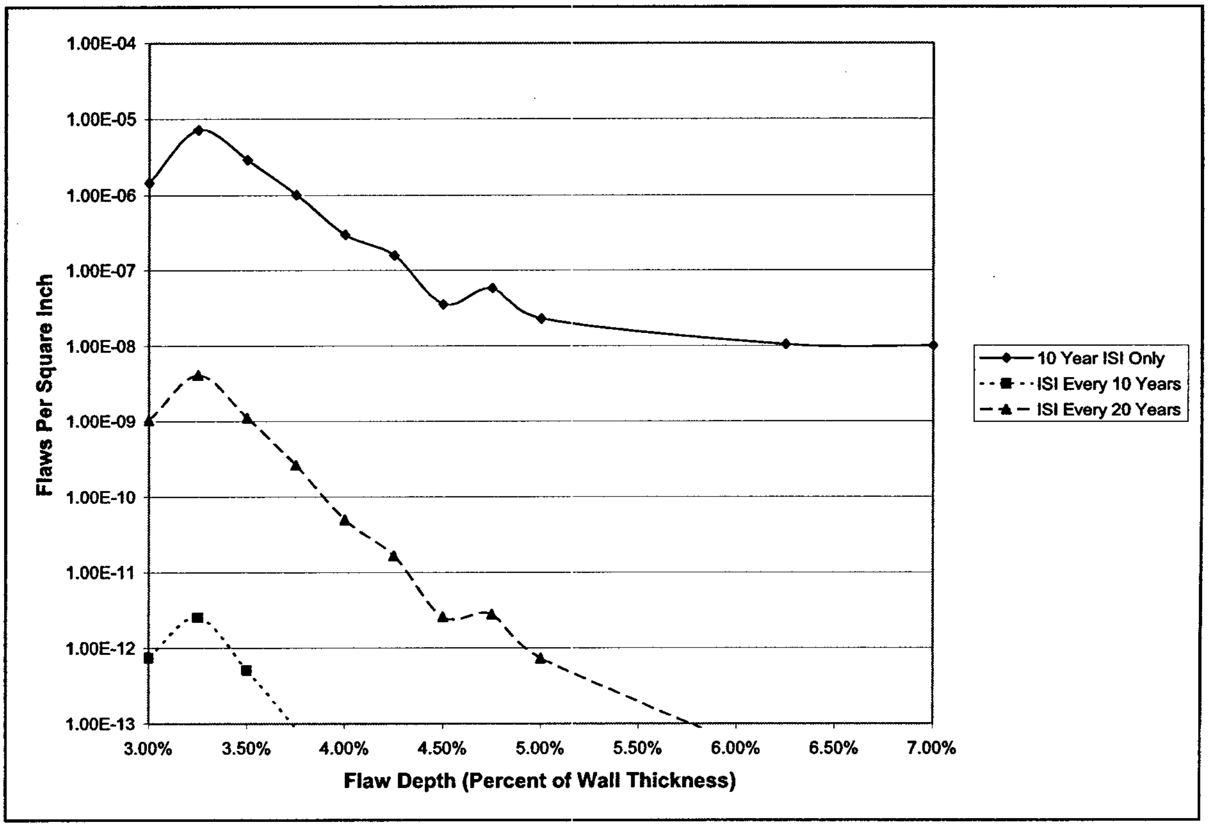


Figure 3-10 Growth of Flaws with an Aspect Ratio of 10 for OC1

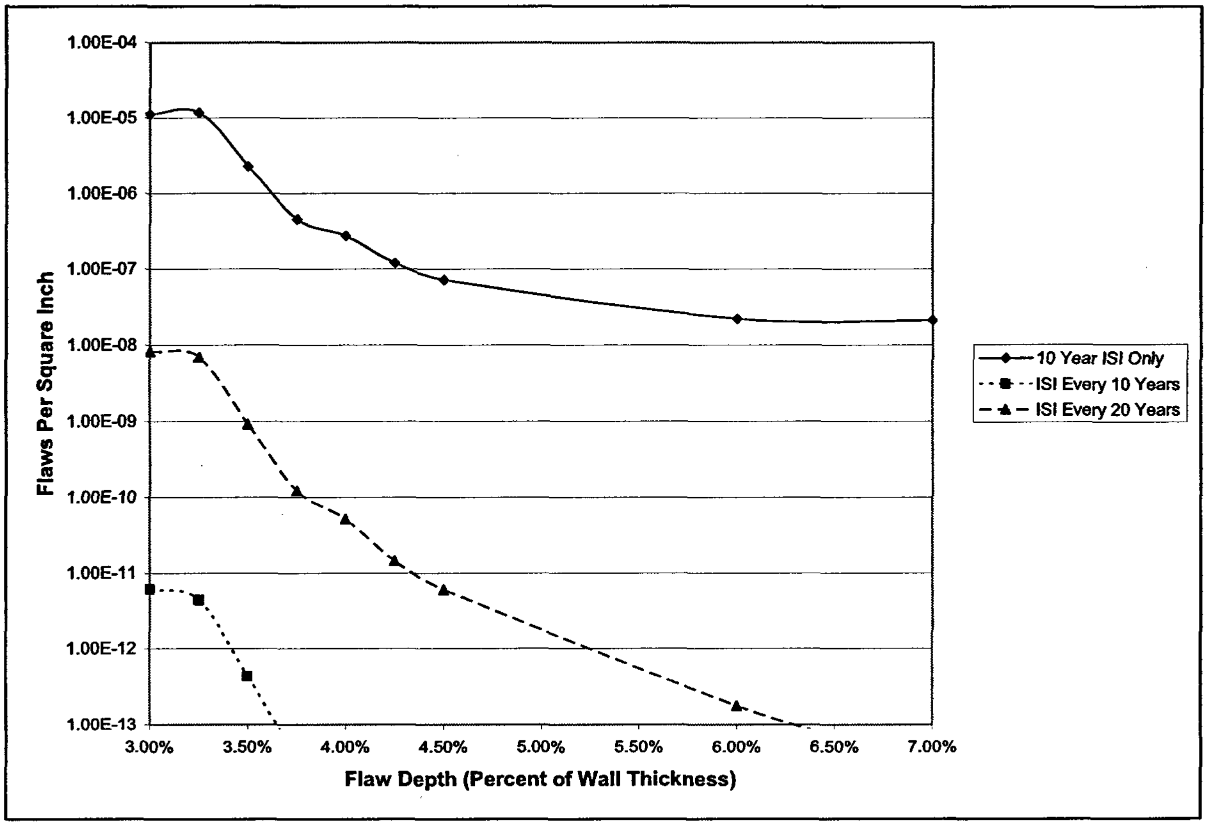


Figure 3-11 Growth of Flaws with an Infinite Aspect Ratio for OC1

4 RISK ASSESSMENT

The quantitative risk assessment discussed below shows that extending the inspection interval from 10 to a maximum of 20 years has an acceptably small impact on risk (core damage frequency [CDF] and large early release frequency [LERF]), i.e., that it is within the bounds of RG 1.174 [4]. A discussion on the requirements of RG 1.174 is included.

4.1 RISK-INFORMED REGULATORY GUIDE 1.174 METHODOLOGY

The NRC has developed a risk-informed regulatory framework. The NRC definition of risk-informed regulation is: "insights derived from probabilistic risk assessments are used in combination with deterministic system and engineering analysis to focus licensee and regulatory attention on issues commensurate with their importance to safety."

The NRC issued RG 1.174, *An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Current Licensing Basis* [4]. In addition, the NRC issued application-specific RGs and Standard Review Plans (SRPs):

- RG-1.175 [39] and SRP Chapter 3.9.7, related to in-service testing (IST) programs
- RG-1.176 [40], related to Graded Quality Assurance (GQA) programs
- RG-1.177 [41] and SRP Chapter 16.1, related to Technical Specifications
- RG-1.178 [42] and SRP-3.9.8, related to ISI of piping programs

These RG and SRP chapters provide guidance in their respective application-specific subject areas to reactor licensees and the NRC staff regarding the submittal and review of risk-informed proposals that would change the licensing basis for a power reactor facility.

Regulatory Guide 1.174 Basic Steps

The approach described in RG 1.174 was used in each of the application-specific RGs/SRPs, and has 4 basic steps as shown in Figure 4-1. The four basic steps are discussed below.

<u>Step 1: Define the Proposed Change</u>

This element includes identifying:

1. Those aspects of the plant's licensing bases that may be affected by the change.

2. All systems, structures, and components (SSCs), procedures, and activities that are covered by the change and consider the original reasons for inclusion of each program requirement.

3. Any engineering studies, methods, codes, applicable plant-specific and industry data and operational experience, PRA findings, and research and analysis results relevant to the proposed change.

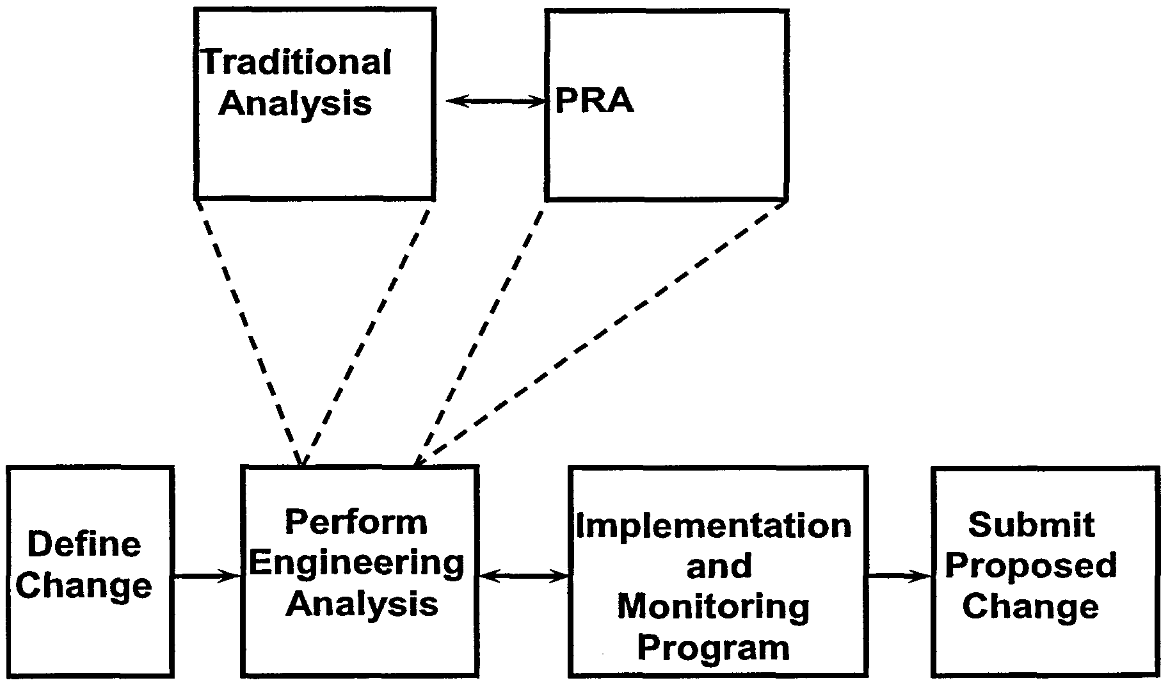


Figure 4-1 Basic Steps in (Principal Elements of) Risk-Informed, Plant-Specific Decision Making (from NRC RG 1.174)

Step 2: Perform Engineering Analysis

This element includes performing the evaluation to show that the fundamental safety principles on which the plant design was based are not compromised (defense-in-depth attributes are maintained) and that sufficient safety margins are maintained. The engineering analysis includes both traditional deterministic analysis and probabilistic risk assessment (PRA). The evaluation of risk impact should also assess the expected change in CDF and LERF, including a treatment of uncertainties. The results from the traditional analysis and the PRA must be considered in an integrated manner when making a decision.

Step 3: Define Implementation and Monitoring Program

This element's goal is to assess SSC performance under the proposed change by establishing performance monitoring strategies to confirm assumptions and analyses that were conducted to justify the change. This is to ensure that no unexpected adverse safety degradation occurs because of the changes. Decisions concerning implementation of changes should be made in light of the uncertainty associated with the results of the evaluation. A monitoring program should have measurable parameters, objective criteria, and parameters that provide an early indication of problems before becoming a safety concern. In addition, the monitoring program should include a cause determination and corrective action plan.

Step 4: Submit Proposed Change

This element includes:

1. Carefully reviewing the proposed change in order to determine the appropriate form of the change request.

2. Assuring that information required by the relevant regulation(s) in support of the request is developed.

3. Preparing and submitting the request in accordance with relevant procedural requirements.

Regulatory Guide 1.174 Fundamental Safety Principles

Five fundamental safety principles are described that each application for a change must meet. These are shown in Figure 4-2, and are discussed below.

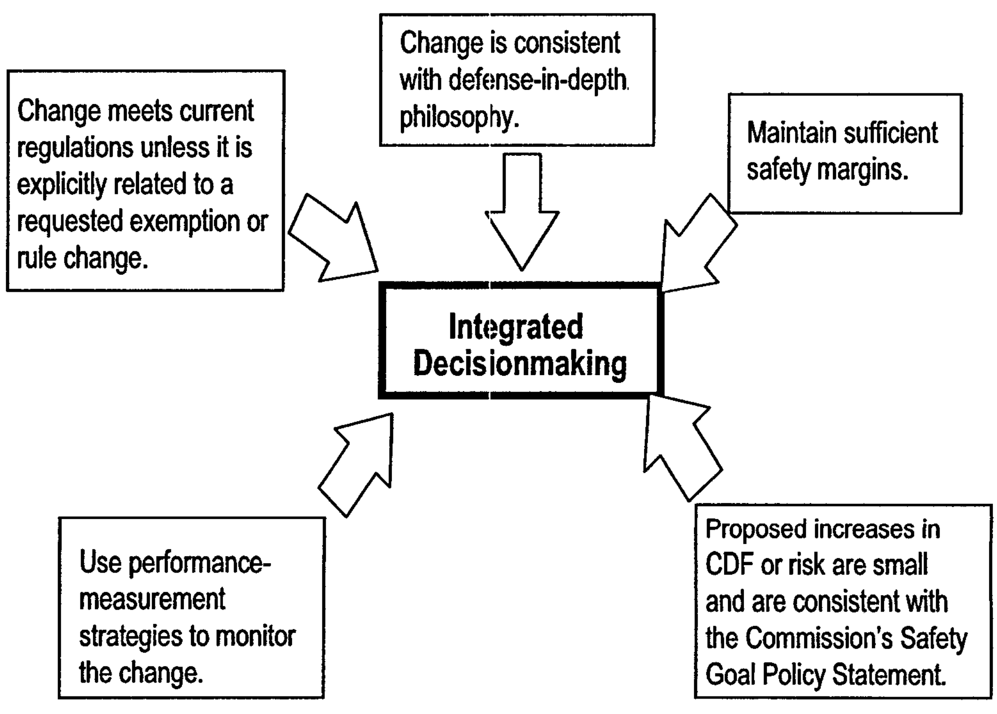


Figure 4-2 Principles of Risk-Informed Regulation (from NRC RG 1.174)

Principle 1: Change meets current regulations unless it is explicitly related to a requested exemption or rule change.

The proposed change is evaluated against the current regulations (including the general design criteria) to either identify where changes are proposed to the current regulations (e.g., Technical Specification, license conditions, and FSAR), or where additional information may be required to meet the current regulations.

Principle 2: Change is consistent with defense-in-depth philosophy.

Defense-in-depth has traditionally been applied in reactor design and operation to provide a multiple means to accomplish safety functions and prevent the release of radioactive material. As defined in RG 1.174 [4], defense-in-depth is maintained by assuring that:

- A reasonable balance among prevention of core damage, prevention of containment failure, and consequence mitigation is preserved.
- Over-reliance on programmatic activities to compensate for weaknesses in plant design is avoided.
- System redundancy, independence, and diversity are preserved commensurate with the expected frequency and consequences to the system (e.g., no risk outliers).
- Defenses against potential common cause failures are preserved and the potential for introduction of new common cause failure mechanisms is assessed.
- Independence of barriers is not degraded (the barriers are identified as the fuel cladding, reactor coolant pressure boundary, and containment structure).
- Defenses against human errors are preserved.

Defense-in-depth philosophy is not expected to change unless:

- A significant increase in the existing challenges to the integrity of the barriers occurs.
- The probability of failure of each barrier changes significantly.
- New or additional failure dependencies are introduced that increase the likelihood of failure compared to the existing conditions.
- The overall redundancy and diversity in the barriers changes.

Principle 3: Maintain sufficient safety margins.

Safety margins must also be maintained. As described in RG 1.174, sufficient safety margins are maintained by assuring that:

- Codes and standards, or alternatives proposed for use by the NRC, are met.
- Safety analysis acceptance criteria in the licensing basis (e.g., FSARs, supporting analyses) are met, or proposed revisions provide sufficient margin to account for analysis and data uncertainty.

Principle 4: Proposed increases in CDF or risk are small and are consistent with the Commission's Safety Goal Policy Statement.

To evaluate the proposed change with regard to a possible increase in risk, the risk assessment should be of sufficient quality to evaluate the change. The expected change in CDF and LERF are evaluated to address this principle. An assessment of the uncertainties associated with the evaluation is conducted. Additional qualitative assessments are also performed.

There are two acceptance guidelines, one for CDF and one for LERF, both of which should be used.

The guidelines for CDF are:

- If the application can be clearly shown to result in a decrease in CDF, the change will be considered to have satisfied the relevant principle of risk-informed regulation with respect to CDF.
- When the calculated increase in CDF is very small, which is taken as being less than 10^{-6} per reactor year, the change will be considered regardless of whether there is a calculation of the total CDF.
- When the calculated increase in CDF is in the range of 10^{-6} per reactor year to 10^{-5} per reactor year, applications will be considered only if it can be reasonably shown that the total CDF is less than 10^{-4} per reactor year.
- Applications that result in increases to CDF above 10^{-5} per reactor year would not normally be considered.

The guidelines for LERF are:

- If the application can be clearly shown to result in a decrease in LERF, the change will be considered to have satisfied the relevant principle of risk-informed regulation with respect to LERF.
- When the calculated increase in LERF is very small, which is taken as being less than 10^{-7} per reactor year, the change will be considered regardless of whether there is a calculation of the total LERF.
- When the calculated increase in LERF is in the range of 10^{-7} per reactor year to 10^{-6} per reactor year, applications will be considered only if it can be reasonably shown that the total LERF is less than 10^{-5} per reactor year.
- Applications that result in increases to LERF above 10^{-6} per reactor year would not normally be considered.

These guidelines are intended to provide assurance that proposed increases in CDF and LERF are small and are consistent with the intent of the Commission's Safety Goal Policy Statement.

Principle 5: Use performance-measurement strategies to monitor the change.

Performance-based implementation and monitoring strategies are also addressed as part of the key elements of the evaluation as described previously.

Risk-Acceptance Criteria for Analysis

For the purposes of this bounding analysis of the risk impact of the proposed change in RV inspection frequency, the following criteria are applied with respect to Principle 4 (small change in risk):

- Change in CDF < 1×10^{-6} per reactor year
- Change in LERF < 1×10^{-7} per reactor year

These values are selected so that the proposed change may be later considered on a plant-specific basis regardless of the plant's baseline CDF and LERF.

To conservatively simplify these acceptance criteria, it will be assumed that through-wall crack growth is equivalent to reactor vessel failure, and that reactor vessel failure results in both core damage and a large early release. It is also conservatively assumed that the conditional probability of a large early release given core damage is 1.0 (See Section 4.3).

Therefore, the simplified conservative/bounding acceptance criterion becomes:

Change in CDF	=	Change in LERF	=	Increase in frequency of through-wall crack growth due to increase in inspection interval	<	1×10^{-7} per reactor year

4.2 FAILURE MODES AND EFFECTS

Failure Modes

The failure mode of concern was thermal fatigue crack growth due typical plant operation. The growth of an existing undetected fabrication flaw in the RV base metal, cladding, or weld metal was assumed to reach a critical size that would lead to reactor vessel through-wall fracture if a PTS-type transient would occur.

Failure Effects

A through-wall flaw failure of the RV was assumed to result in core damage and a large early release.

4.3 CORE DAMAGE RISK EVALUATION

The objective of the risk assessment was to evaluate the core damage risk from the extension of the examination of the RV relative to other plant risk contributors through a qualitative and quantitative evaluation.

NRC RG 1.174 [4] provided the basis for this evaluation as well as the acceptance guidelines to make a change to the current licensing basis.

Risk was defined as the combination of likelihood of an event and severity of consequences of an event. Therefore, the following two questions were addressed:

- What was the likelihood of the event?
- What would the consequences be?

The following sections describe the likelihood and postulated consequences. The likelihood and consequences were then combined in the risk calculation and the results of the evaluation are presented in this report.

What is the Likelihood of the Event?

The likelihood of the event was addressed by identifying the plant transients or operational events that might lead to failure of the RV, and estimating the frequency of these events.

What are the Consequences?

The consequences were defined in terms of the CDF and LERF risk metrics.

For this evaluation, the conditional core damage probability given the failure of the RV was assumed to be 1.0 (no credit for safety system actuation to mitigate the consequences of the failure). Since this was intended as a bounding assessment, it was also conservatively assumed that the conditional probability of a large early release given core damage for this scenario is 1.0 (i.e., no credit for consequence mitigation via the containment and related systems). Note that this was a simplifying assumption, and a specific mechanism for LERF was not implied or defined here.

Risk Calculation

For this evaluation, the CDF and LERF were calculated by:

$$CDF = LERF = IE * CPF$$

where:

CDF	=	Core damage frequency from a failure (events per year)
LERF	=	Large early release frequency from a failure (events per year)
IE	=	Initiating event frequency (in events per year)
CPF	=	Conditional probability of reactor vessel failure

The transient initiating frequency distributions were identified in the NRC PTS Risk Study [7] and are included in Appendices D, H, and L for the pilot plants. The probability of failure was calculated by the FAVPFS module of FAVOR. The FAVPOST module of FAVOR combined the transient initiating frequency distribution with the reactor vessel conditional failure probability distribution to determine a reactor vessel failure frequency distribution for each transient. From these failure frequency distributions, FAVPOST determined a mean reactor vessel failure frequency. In addition to this mean failure frequency a standard error was reported. To account for uncertainties, Upper and Lower Bounds are determined. The Upper Bound was determined by adding 2 times the standard error from the "10-Year ISI-Only" case. The Lower Bound was determined by subtracting 2 times the standard error from the "ISI Every 10 Years" case. The change in reactor vessel failure frequency was determined by subtracting the Lower Bound from the Upper Bound. The mean reactor vessel failure frequencies, Upper and Lower Bounds, and change in failure frequency are given in Sections 3.2 and 3.3. As previously stated, reactor vessel failure results in core damage which results in large early release. Therefore, the large early release frequencies were equal to the reactor vessel failure frequencies. The large early release frequencies, Upper and Lower Bounds, and change in large early release frequency are summarized in Table 4-1, based on FAVOR 05.1 evaluations.

Table 4-1 Large Early Release Frequencies

	BV1 (per year)	Palisades (per year)	OC1 (per year)
10-Year ISI Only	5.04E-09	1.54E-08	2.06E-09
Upper Bound	6.01E-09	1.86E-08	2.60E-09
ISI Every 10 Years	4.10E-09	1.67E-08	2.18E-09
Lower Bound	3.52E-09	1.423E-09	1.81E-09
Bounding Change in Large Early Release Frequency	2.49E-09	4.40E-09	7.96E-10

Risk Results and Conclusions

The analysis described above demonstrates that changes in CDF and LERF do not exceed the NRC's RG-1.174 [4] acceptance guidelines for a small change in CDF and LERF ($<10^{-6}$ per year for CDF, $<10^{7}$ per year for LERF).

As part of this evaluation, the key principles identified in RG-1.174 were reviewed and the responses based on the evaluation are provided in Table 4-2.

This evaluation concluded that extension of the RV in-service examination from 10 to 20 years would not be expected to result in an unacceptable increase in risk. Given this outcome, and the fact that other key principles listed in RG-1.174 continue to be met, the proposed change in inspection interval from 10 to 20 years is acceptable.

Table 4-2 Evaluation with Respect to Regulatory Guide 1.174 [4] Key Principles

Key Principles	Evaluation Response
Change meets current regulations unless it is explicitly related to a requested exemption or rule change.	Change to current RG 1.150 [2] requirements is proposed.
Change is consistent with defense-in-depth philosophy.	Potential for failure of the RV is acceptably small during normal or accident conditions, and does not threaten plant barriers. See the discussion below for additional information on defense in depth.
Maintain sufficient safety margins.	No safety analysis margins are changed.
Proposed increases in CDF or risk are small and are consistent with the Commission's Safety Goal Policy Statement.	Proposed increase in risk is estimated to be acceptably small.
Use performance-measurement strategies to monitor the change.	NDE examinations still conducted, but on less frequent basis not to exceed 20 years. Other indications of potential degradation of RV are available (e.g., foreign experience and periodic testing with visual examinations)

Defense-in-Depth

While the results presented in this report demonstrate that the contribution of eliminating future inspections after the initial 10 year ISI meets prescribed regulatory criteria for assessing risk, the proposed course of action is to extend the inspection interval requirements from 10 to 20 years while not eliminating any portion of the current inspection requirements. This provides additional margin for defense-in-depth and contributes directly toward maintaining plant safety.

Extending the RV ISI interval does not imply that generic degradation mechanisms will be ignored for 20 years. (With the number of PWR nuclear power plants in operation in the U.S. and globally, a sampling of plants inevitably undergo examinations in a given year.) This provides for early detection of any potential emerging generic degradation mechanisms, and would permit the industry to react with more frequent examinations if needed.

In addition, it must be recognized that all reactor coolant pressure boundary failures occurring to date have been identified as a result of leakage, and were discovered by visual examination. The proposed RV ISI interval extension does not alter the visual examination interval. The reactor vessel would undergo, as a minimum, the Section XI Examination Category B-P pressure tests and visual examinations conducted at the end of each refueling before plant start-up, as well as leak tests with visual examinations that precede each start-up following maintenance or repair activities.

5 CONCLUSIONS

Based on the results of this analysis, it is concluded that:

1. The beltline is the most limiting region for the evaluation of risk.

2. RV inspections performed to date have not detected any significant flaws.

3. Crack extension due to fatigue crack growth during service is small.

4. The man-rem exposure can be reduced by extending the inspection interval.

5. The failure frequencies for PWR RVs due to the dominant PTS transients are well below 10^{-7} per year.

6. The change in risk meets the RG 1.174 [4] acceptance guidelines for a small change in LERF.

7. The increase in the RV ISI interval from 10 to 20 years satisfies all the RG 1.174 criteria, including other considerations, such as defense-in-depth.

Based on the above conclusions, the ASME Section XI [1] 10-year inspection interval for examination categories B-A and B-D welds in PWR RVs and category B-J welds to RV nozzles, can be extended to 20 years. In-service inspection intervals of 20 years for FENOC's Beaver Valley Unit 1, NMC's Palisades, and Duke Energy's Oconee Unit 1 are acceptable for implementation. The methodology in WCAP-16168-NP Revision 1 is applicable to plants other than the pilot plants by confirming the applicability of the parameters in Appendix A on a plant specific basis. Since the 10 year inspection interval is required by Section XI, IWB-2412, as codified in 10 CFR 50.55a, an exemption request must be submitted and approved by the NRC to extend the inspection interval to 20 years, unless 10 CFR 50.55a is amended to incorporate ASME Code Case N-691.

6 REFERENCES

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APPENDIX A
BOUNDING PARAMETER CHECKLIST

WCAP-16168-NP Revision 1 describes the methodology used to demonstrate the feasibility of extending the reactor vessel inspection interval required by the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code Section XI, as supplemented by Nuclear Regulatory Commission (NRC) Regulatory Guide 1.150. This methodology was used to perform risk analysis for pilot plants representing the Westinghouse and Combustion Engineering designs. It is an extension of work done as part of the NRC PTS Risk Study. Table A-1 identifies critical parameters to be used to determine if the pilot plant evaluations documented in this report bound a plant specific application. If the plant-specific parameter is not bounded by the pilot plant analysis, additional evaluations or sensitivity studies may be required to support the use of the pilot plant risk studies. Additional information relative to plant specific reactor vessel inspection is to be provided in Table A-2. Examples of plant specific use of these tables for Wolf Creek and Waterford 3 are contained in Appendices A-1 and A-2 respectively.

Table A-1 Critical Parameters for the Application of the Bounding Analysis

Parameter	Pilot Plant Basis	Plant Specific Basis	Additional Evaluation Required? (Y/N)
Dominant PTS Transients in the NRC PTS Risk Study are applicable			
Through Wall Cracking Frequency (TWCF)			
Frequency and Severity of Design Basis Transients			
Cladding Layers (Single/Multiple)			

Table A-2 Additional Information Pertaining to the Reactor Vessel Inspection

Inspection methodology:	
Number of past inspections:	
Number of indications found:	
Proposed inspection schedule for balance of plant life:	

APPENDIX A-1
WOLF CREEK PLANT IMPLEMENTATION EXAMPLE

Table 1	**Critical Parameters for Application of Bounding Analysis**		
Parameter	**Pilot Plant Basis**	**Plant Specific Basis**	**Additional Evaluation Required? (Y/N)**
Dominant Pressurized Thermal Shock (PTS) Transients in the NRC PTS Risk Study are applicable	NRC PTS Risk Study (Reference 8)	PTS Generalization Study (Reference 25)	No
Through Wall Cracking Frequency	4.67E-09 Events per year	2.15E-12 Events per year	No
Frequency and Severity of Design Basis Transients	WCAP-16168-NP Revision 1: Bounded by 7 cooldowns/yr	Bounded by 7 cooldowns/yr	No
Cladding Layers (Single/Multiple)	Single	Single (assumed)	No

Table 2	**Additional Information Pertaining to Reactor Vessel Inspection**
Inspection methodology:	Past inspections have been performed to Regulatory Guide 1.150. Inspections performed during RF13 and RF 14 were also performed to ASME Section XI Appendix VIII.
Number of past inspections:	- Category B-A welds (reactor vessel): 2 inspections, RF8 – Spring 1996 and RF14 – Spring 2005 with the exception of weld RV-101-121 which was also inspected in RF2 – Spring 1987 and RF10 – Spring 1999 - Category B-A welds (closure head): 2 inspections, Interval 1 examinations in RF1 – Fall 1986, RF4 – Spring 1990, and RF6 – Spring 1993. Interval 2 examinations were performed in RF9 – Fall 1997, RF11 – Fall 2000, and RF13 – Fall 2003. 2 welds were examined each outage. - Category B-D welds (outlet nozzles): 3 inspections RF3 – Fall 1988, RF8 – Spring 1996, RF14 – Spring 2005 - Category B-D welds (inlet nozzles): 2 inspections, RF8 – Spring 1996, RF14 – Spring 2005
Number of indications found:	Zero reportable indications have been found to date. Any recordable indications have been acceptable per ASME Section XI IWB-3500
Proposed inspection schedule for balance of plant life:	Third inservice inspection currently scheduled for 2015. The third inservice inspection is proposed to be performed in 2025. The fourth inservice inspection interval is proposed to be performed in 2045.

APPENDIX A-2
WATERFORD 3 PLANT IMPLEMENTATION EXAMPLE

Table 1 Critical Parameters for Application of Bounding Analysis

Parameter	**Pilot Plant Basis**	**Plant Specific Basis**	**Additional Evaluation Required? (Y/N)**
Dominant Pressurized Thermal Shock (PTS) Transients in the NRC PTS Risk Study are applicable .	NRC PTS Risk Re-Evaluation (Reference 8)	PTS Generalization Study (Reference 25)	No
Through Wall Cracking Frequency	6.42E-09 Events per year	5.19E-13 Events per year	No
Frequency and Severity of Design Basis Transients	WCAP-16168-NP Revision 1: Bounded by 13 heatup/cooldowns/yr	Bounded by 13 heatup/cooldowns/yr	No
Cladding Layers (Single/Multiple)	Single	Single	No

Table 2 Additional Information Pertaining to Reactor Vessel Inspection

Inspection methodology:	Past inspections have been performed to Regulatory Guide 1.150
Number of past inspections:	- Category B-A welds (reactor vessel): 1 inspection – 1995, with the exception of weld 01-020 which was also inspected in 1988. - Category B-A welds (closure head): 4 inspections with 3 welds inspected 1986, 3 welds inspected 1989, 1 weld inspected 1994, 3 welds inspected 2000 - Category B-D welds (outlet nozzles): 2 inspections – 1988 and 1995, with the exception of weld 01-021 which was also inspected in 1989. - Category B-D welds (inlet nozzles): 1 inspection – 1995
Number of indications found:	Zero reportable indications have been found to date. Any recordable indications have been acceptable per ASME Section XI IWB-3500
Proposed inspection schedule for balance of plant life:	Second inservice inspection currently scheduled for Spring 2008. The second inservice inspection is proposed to be performed in 2018. The third inservice inspection is proposed to be performed in 2038.

APPENDIX B
INPUTS FOR THE BEAVER VALLEY UNIT 1 PILOT PLANT EVALUATION

A summary of the NDE inspection history based on Regulatory Guide 1.150 and pertinent input data for BV1 is as follows:

1. Number of ISIs performed (relative to initial pre-service and 10–year interval inspections) for full penetration Category B-A, B-D, and B-J reactor vessel welds assuming all of the candidate welds were inspected: 2 (covering all welds of the specified categories).

2. The inspections performed covered: A total of 34 items. 15 Category B-A items had coverage of <90%. 1 Category B-A item had coverage > 90% but <100%. 6 Category B-A items had coverage of 100%. 6 Category B-D items had coverage of 90% and 6 had coverage of 100%.

3. Number of indications found during the most recent inservice inspection: 42
   This number includes consideration of the following additional information.

   a. Indications found that were reportable: 0
   b. Indications found that were within acceptable limits: 42
   c. Indications/anomalies currently being monitored: 0

4. Full penetration relief requests for the RV were submitted and accepted by the NRC for 15 items.

5. Fluence distribution at inside surface of RV beltline until end of life (EOL): see Figure B-1 taken from the NRC PTS Risk Study [7], Figure 4.2.

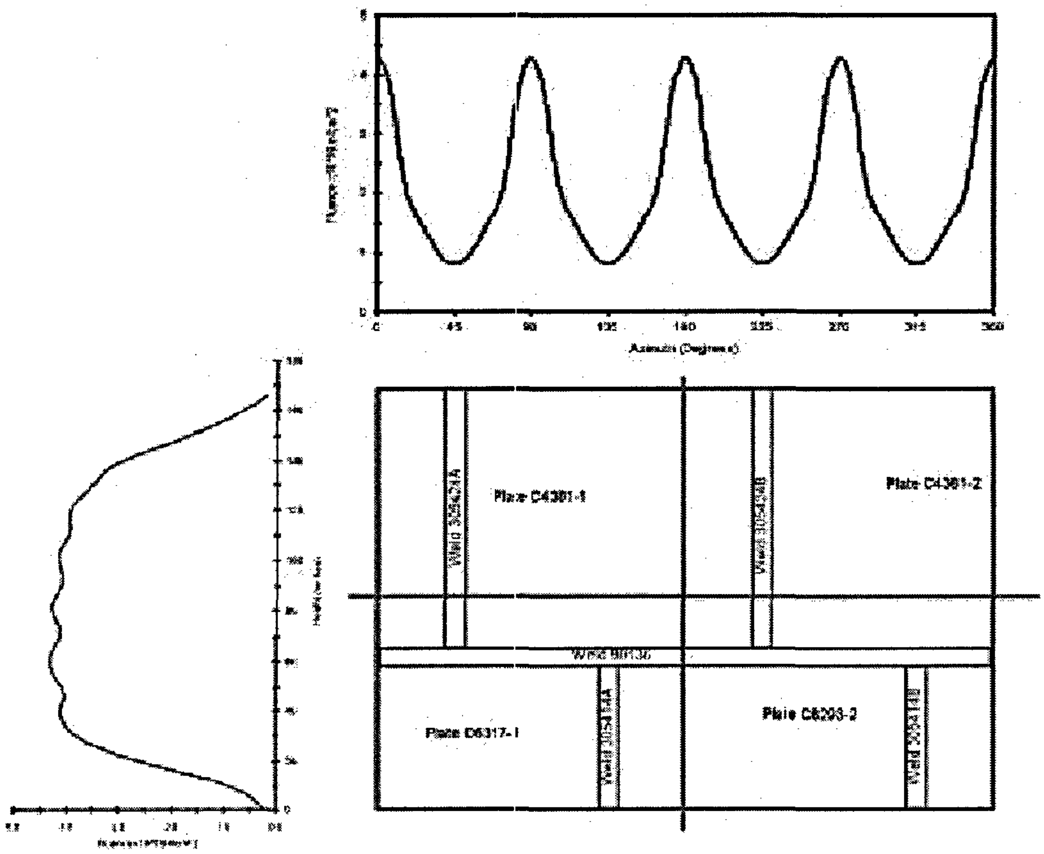

Figure B-1 Rollout Diagram of Beltline Materials and Representative Fluence Maps for BV1

6. Reactor vessel cladding details:

   a. Thickness: 0.156 inches

   b. Material properties (assumed to be independent of temperature):

      1) Thermal conductivity (Btu/hr-ft-°F), K=10.0
      2) Specific heat (Btu/LBM-F),C=0.120
      3) Density (LBM/ft^3).RHO=489.00
      4) Young's Modulus of Elasticity (KSI), E=22800
      5) Thermal expansion coefficient (°F^{-1}), ALPHA=0.00000945
      6) Poisson's Ratio, V=0.3

   c. Material including copper and nickel content: Material properties assigned to clad flaws are that of the underlying material be it base metal or weld. These properties are identified in Table B-1. This is consistent with the NRC PTS Risk Study [7].

   d. Material property uncertainties:

      1) Bead width: 1 inch – bead widths vary for all plants. Based on the NRC PTS Risk Study [7], a nominal dimension of 1 inch is selected for all analyses because this

parameter is not expected to influence significantly the predicted reactor vessel failure probabilities.

2) Truncation limit: Cladding thickness rounded to the next 1/100th of the total reactor vessel thickness to be consistent with the NRC PTS Risk Study [7].

3) Surface flaw depth: 0.161 inch

4) All cladding flaws are surface-breaking. Only flaws in cladding that would influence brittle fracture of the reactor vessel are brittle. This is consistent with the NRC PTS Risk Study [7].

e. Additional cladding properties are identified in Table B-2.

7. Base metal:

a. Wall thickness: 7.875 inches

b. Material properties (assumed to be independent of temperature):

1) Thermal conductivity (Btu/hr-ft-°F), K=24.0
2) Specific heat (Btu/LBM-°F),C=0.120
3) Density (LBM/ft^3).RHO=489.00
4) Young's Modulus of Elasticity (KSI), E=28000
5) Thermal expansion coefficient (°F^{-1}), ALPHA=0.00000777
6) Poisson's Ratio, V=0.3
7) Other material properties are identified in Table B-1

Table B-1 BV1-Specific Material Values Drawn from the RVID (see Ref. 7, Table 4.1)

Major Material Region Description				Cu [wt%]	Ni [wt%]	P [wt%]	Un-Irradiated RT_{NDT}		RTPTS @60 EFPY
#	Type	Heat	Location				[°F]	Method	
1	Axial Weld	305414A	Lower	0.337	0.609	0.012	- 56	Generic	230.4
2	Axial Weld	305414B	Lower	0.337	0.609	0.012	- 56	Generic	230.4
3	Axial Weld	305424A	Upper	0.273	0.629	0.013	- 56	Generic	217.8
4	Axial Weld	305424B	Upper	0.273	0.629	0.013	- 56	Generic	217.8
5	Circ Weld	90136	Intermediate	0.269	0.070	0.013	- 56	Generic	159.1
6	Plate	C6317-1	Lower	0.200	0.540	0.010	27	MTEB 5-2	296.6
7	Plate	C6293-2	Lower	0.140	0.570	0.015	20	MTEB 5-2	275.7
8	Plate	C4381-2	Upper	0.140	0.620	0.015	73	MTEB 5-2	332.9
9	Plate	C4381-1	Upper	0.140	0.620	0.015	43	MTEB 5-2	302.9

8. Weld metal details: Details of information used in addressing weld-specific information are taken directly from the NRC PTS Risk Study [7], Table 4.2. Summaries are reproduced as Table B-2.

Table B-2 Summary of Reactor Vessel-Specific Inputs for Flaw Distribution

Variable		Oconee	Beaver Valley	Palisades	Calvert Cliffs	Notes
Inner Radius (to cladding)	[in]	85.5	78.5	86	86	Vessel specific info
Base Metal Thickness	[in]	8.438	7.875	8.5	8.675	Vessel specific info
Total Wall Thickness	[in]	8.626	8.031	8.75	8.988	Vessel specific info

Variable			Oconee	Beaver Valley	Palisades	Calvert Cliffs	Notes
SAW Weld	Volume fraction	[%]	97%				100% - SMAW% - REPAIR%
	Thru-Wall Bead Thickness	[in]	0.1875	0.1875	0.1875	0.1875	All plants report plant specific dimensions of 3/16-in.
	Truncation Limit	[in]	1				Judgment. Approx. 2X the size of the largest non-repair flaw observed in PVRUF & Shoreham.
	Buried or Surface	--	All flaws are buried				Observation
	Orientation	--	Circ flaws in circ welds, axial flaws in axial welds.				Observation: Virtually all of the weld flaws in PVRUF & Shoreham were aligned with the welding direction because they were lack of sidewall fusion defects.
	Density basis	--	Shoreham density				Highest of observations
	Aspect ratio basis	--	Shoreham & PVRUF observations				Statistically similar distributions from Shoreham and PVRUF were combined to provide more robust estimates, when based on judgment the amount data were limited and/or insufficient to identify different trends for aspect ratios for flaws in the two vessels.
	Depth basis	--	Shoreham & PVRUF observations				Statistically similar distributions combined to provide more robust estimates

Table B-2 Summary of Reactor Vessel-Specific Inputs for Flaw Distribution (cont.)

Variable			Oconee	Beaver Valley	Palisades	Calvert Cliffs	Notes
SMAW Weld	Volume fraction	[%]	1%				Upper bound to all plant specific info provided by Steve Byrne (Westinghouse – Windsor).
	Thru-Wall Bead Thickness	[in]	0.21	0.20	0.22	0.25	Oconee is generic value based on average of all plants specific values (including Shoreham & PVRUF data). Other values are plant specific as reported by Steve Byrne.
	Truncation Limit	[in]	1				Judgment. Approx. 2X the size of the largest non-repair flaw observed in PVRUF & Shoreham.
	Buried or Surface	--	All flaws are buried				Observation
	Orientation	--	Circ flaws in circ welds, axial flaws in axial welds.				Observation: Virtually all of the weld flaws in PVRUF & Shoreham were aligned with the welding direction because they were lack of sidewall fusion defects.
	Density basis	--	Shoreham density				Highest of observations
	Aspect ratio basis	--	Shoreham & PVRUF observations				Statistically similar distributions from Shoreham and PVRUF were combined to provide more robust estimates, when based on judgment the amount data were limited and/or insufficient to identify different trends for aspect ratios for flaws in the two vessels.
	Depth basis	--	Shoreham & PVRUF observations				Statistically similar distributions combined to provide more robust estimates

Table B-2 Summary of Reactor Vessel-Specific Inputs for Flaw Distribution (cont.)

Variable			Oconee	Beaver Valley	Palisades	Calvert Cliffs	Notes
Repair Weld	Volume fraction	[%]	2%				Judgment. A rounded integral percentage that exceeds the repaired volume observed for Shoreham and for PVRUF, which was 1.5%.
	Thru-Wall Bead Thickness	[in]	0.14				Generic value: As observed in PVRUF and Shoreham by PNNL.
	Truncation Limit	[in]	2				Judgment. Approx. 2X the largest repair flaw found in PVRUF & Shoreham. Also based on maximum expected width of repair cavity.
	Buried or Surface	--	All flaws are buried				Observation
	Orientation	--	Circ flaws in circ welds, axial flaws in axial welds.				The repair flaws had complex shapes and orientations that were not aligned with either the axial or circumferential welds; for consistency with the available treatments of flaws by the FAVOR code, a common treatment of orientations was adopted for flaws in SAW/SMAW and repair welds.
	Density basis	--	Shoreham density				Highest of observations
	Aspect ratio basis	--	Shoreham & PVRUF observations				Statistically similar distributions from Shoreham and PVRUF were combined to provide more robust estimates, when based on judgment the amount data were limited and/or insufficient to identify different trends for aspect ratios for flaws in the two vessels.
	Depth basis	--	Shoreham & PVRUF observations				Statistically similar distributions combined to provide more robust estimates

Table B-2 Summary of Reactor Vessel-Specific Inputs for Flaw Distribution (cont.)

Variable			Oconee	Beaver Valley	Palisades	Calvert Cliffs	Notes
Cladding	Actual Thickness	[in]	0.188	0.156	0.25	0.313	Vessel specific info
	# of Layers	[#]	1	2	2	2	Vessel specific info
	Bead Width	[in]	1				Bead widths of 1 to 5-in. characteristic of machine deposited cladding. Bead widths down to ½-in. can occur over welds. Nominal dimension of 1-in. selected for all analyses because this parameter is not expected to influence significantly the predicted vessel failure probabilities. May need to refine this estimate later, *particularly for Oconee* who reported a 5-in bead width.
	Truncation Limit	[in]	Actual clad thickness rounded to the nearest 1/100th of the total vessel wall thickness				Judgment & computational convenience
	Surface flaw depth in FAVOR	[in]	0.259	0.161	0.263	0.360	
	Buried or Surface	--	All flaws are surface breaking				Judgment. Only flaws in cladding that would influence brittle fracture of the vessel are brittle. Material properties assigned to clad flaws are that of the underlying material, be it base or weld.
	Orientation	--	*All circumferential.*				Observation: All flaws observed in PVRUF & Shoreham were lack of inter-run fusion defects, and cladding is always deposited circumferentially
	Density basis	--	No surface flaws observed. Density is 1/1000 that of the observed buried flaws in cladding of vessels examined by PNNL. If there is more than one clad layer then there are no clad flaws.				Judgment
	Aspect ratio basis	--	Observations on buried flaws				Judgment
	Depth basis	--	Depth of all surface flaws is the actual clad thickness rounded up to the nearest 1/100th of the total vessel wall thickness.				Judgment.

Table B-2 Summary of Reactor Vessel-Specific Inputs for Flaw Distribution (cont.)

Variable			Oconee	Beaver Valley	Palisades	Calvert Cliffs	Notes
Plate	Truncation Limit	[in]	0.433				Judgment. Twice the depth of the largest flaw observed in all PNNL plate inspections.
	Buried or Surface	--	All flaws are buried				Observation
	Orientation	--	Half of the simulated flaws are circumferential, half are axial.				Observation & Physics: No observed orientation preference, and no reason to suspect one (other than laminations which are benign.
	Density basis	--	1/10 of small weld flaw density, 1/40 of large weld flaw density of the PVRUF data				Judgment. Supported by limited data.
	Aspect ratio basis	--	Same as for PVRUF welds				Judgment
	Depth basis	--	Same as for PVRUF welds				Judgment. Supported by limited data.

9. TWCF calculated at 60 EFPY using correlation from Reference 27: 4.67E-09 Events per year

APPENDIX C
BEAVER VALLEY UNIT 1 PROBSBFD OUTPUT

C-1: 10 Year ISI Only

```
                 STRUCTURAL RELIABILITY AND RISK ASSESSMENT (SRRA)
 WESTINGHOUSE        MONTE-CARLO SIMULATION PROGRAM PROBSBFD        VERSION
1.0

===========================================================================
==
     INPUT VARIABLES FOR CASE  3: BV1 HUCD 10 YR ISI ONLY

     NCYCLE =     80            NFAILS =  1001           NTRIAL =  1000
     NOVARS =     19            NUMSET =     2           NUMISI =     5
     NUMSSC =      4            NUMTRC =     4           NUMFMD =     4

   VARIABLE        DISTRIBUTION      MEDIAN       DEVIATION    SHIFT    USAGE
  NO.   NAME       TYPE      LOG      VALUE       OR FACTOR    MV/SD   NO. SUB

   1   FIFDepth    - CONSTANT -    2.0000D-02                           1  SET
   2   IFlawDen    - CONSTANT -    3.6589D-03                           2  SET
   3   ICy-ISI     - CONSTANT -    1.0000D+01                           1  ISI
   4   DCy-ISI     - CONSTANT -    8.0000D+01                           2  ISI
   5   MV-Depth    - CONSTANT -    1.5000D-02                           3  ISI
   6   SD-Depth    - CONSTANT -    1.8500D-01                           4  ISI
   7   CEff-ISI    - CONSTANT -    1.0000D+00                           5  ISI
   8   Aspect1     - CONSTANT -    2.0000D+00                           1  SSC
   9   Aspect2     - CONSTANT -    6.0000D+00                           2  SSC
  10   Aspect3     - CONSTANT -    1.0000D+01                           3  SSC
  11   Aspect4     - CONSTANT -    9.9000D+01                           4  SSC
  12   NoTr/Cy     - CONSTANT -    7.0000D+00                           1  TRC
  13   FCGThld     - CONSTANT -    1.5000D+00                           2  TRC
  14   FCGR-UC     NORMAL    NO    0.0000D+00   1.0000D+00     .00      3  TRC
  15   DKINFile    - CONSTANT -    1.0000D+00                           4  TRC
  16   Percent1    - CONSTANT -    5.6175D+01                           1  FMD
  17   Percent2    - CONSTANT -    3.0283D+01                           2  FMD
  18   Percent3    - CONSTANT -    3.9086D+00                           3  FMD
  19   Percent4    - CONSTANT -    9.6333D+00                           4  FMD

  INFORMATION GENERATED FROM FAVLOADS.DAT FILE
  AND SAVED IN DKINSAVE.DAT FILE:

  WALL THICKNESS =  8.0360 INCH

  FLAW DEPTH  MINIMUM K AND MAXIMUM K FOR

    TYPE 1 WITH AN ASPECT RATIO OF   2.

    8.03600D-02    2.41927D+00    1.03655D+01
    1.47862D-01    3.22858D+00    1.40170D+01
    4.01800D-01    1.29279D+01    1.75751D+01
    6.02700D-01    1.41327D+01    2.09080D+01
    8.03600D-01    1.49423D+01    2.33544D+01
    1.60720D+00    1.45812D+01    2.72710D+01
    2.41080D+00    1.02448D+01    2.63600D+01
    4.01800D+00    2.35823D+00    2.78623D+01
```

C-1: 10 Year ISI Only (cont.)

TYPE 2 WITH AN ASPECT RATIO OF 6.

8.03600D-02	3.63673D+00	1.56338D+01
1.47862D-01	4.95557D+00	2.15454D+01
4.01800D-01	1.90999D+01	2.63794D+01
6.02700D-01	2.31650D+01	3.16223D+01
8.03600D-01	2.48064D+01	3.60464D+01
1.60720D+00	2.65025D+01	4.51155D+01
2.41080D+00	2.31198D+01	4.76172D+01
4.01800D+00	1.54934D+01	5.27667D+01

TYPE 3 WITH AN ASPECT RATIO OF 10.

8.03600D-02	3.98451D+00	1.71374D+01
1.47862D-01	5.29827D+00	2.30393D+01
4.01800D-01	2.02922D+01	2.81955D+01
6.02700D-01	2.51750D+01	3.36684D+01
8.03600D-01	2.69393D+01	3.84779D+01
1.60720D+00	2.92755D+01	4.91684D+01
2.41080D+00	2.74642D+01	5.45509D+01
4.01800D+00	2.02195D+01	6.28814D+01

TYPE 4 WITH AN ASPECT RATIO OF 99.

8.03600D-02	6.51796D+00	1.75511D+01
1.60720D-01	1.01756D+01	2.28059D+01
2.41080D-01	1.54398D+01	2.23553D+01
4.01800D-01	2.18696D+01	2.94323D+01
6.02700D-01	2.69582D+01	3.66108D+01
8.03600D-01	2.88204D+01	4.17713D+01
1.60720D+00	3.37365D+01	5.67413D+01
2.41080D+00	3.35927D+01	6.64759D+01

AVERAGE CALCULATED VALUES FOR: Surface Flaw Density with FCG and ISI

NUMBER FAILED = 0 NUMBER OF TRIALS = 1000

DEPTH (WALL/400) AND FLAW DENSITY FOR ASPECT RATIOS OF 2, 6, 10 AND 99

8	4.4254D-04	1.4320D-04	1.4728D-05	4.7035D-05
9	0.0000D+00	8.8686D-05	1.4703D-05	2.7347D-05
10	0.0000D+00	4.4175D-06	9.2631D-07	7.2598D-07
11	0.0000D+00	2.2821D-07	5.9150D-08	7.0131D-08
12	0.0000D+00	2.2665D-07	2.9099D-08	0.0000D+00
13	0.0000D+00	0.0000D+00	2.8861D-08	0.0000D+00

C-2: ISI Every 10 Years

```
                    STRUCTURAL RELIABILITY AND RISK ASSESSMENT (SRRA)
 WESTINGHOUSE           MONTE-CARLO SIMULATION PROGRAM PROBSBFD       VERSION
1.0

============================================================================
==
      INPUT VARIABLES FOR CASE  2: BV1 HUCD 10 YR ISI INT

      NCYCLE =     80            NFAILS =  1001            NTRIAL =  1000
      NOVARS =     19            NUMSET =     2            NUMISI =     5
      NUMSSC =      4            NUMTRC =     4            NUMFMD =     4

    VARIABLE        DISTRIBUTION      MEDIAN      DEVIATION    SHIFT    USAGE
   NO.   NAME       TYPE     LOG       VALUE      OR FACTOR    MV/SD   NO. SUB

    1  FIFDepth    - CONSTANT -    2.0000D-02                            1  SET
    2  IFlawDen    - CONSTANT -    3.6589D-03                            2  SET
    3  ICy-ISI     - CONSTANT -    1.0000D+01                            1  ISI
    4  DCy-ISI     - CONSTANT -    1.0000D+01                            2  ISI
    5  MV-Depth    - CONSTANT -    1.5000D-02                            3  ISI
    6  SD-Depth    - CONSTANT -    1.8500D-01                            4  ISI
    7  CEff-ISI    - CONSTANT -    1.0000D+00                            5  ISI
    8  Aspect1     - CONSTANT -    2.0000D+00                            1  SSC
    9  Aspect2     - CONSTANT -    6.0000D+00                            2  SSC
   10  Aspect3     - CONSTANT -    1.0000D+01                            3  SSC
   11  Aspect4     - CONSTANT -    9.9000D+01                            4  SSC
   12  NoTr/Cy     - CONSTANT -    7.0000D+00                            1  TRC
   13  FCGThld     - CONSTANT -    1.5000D+00                            2  TRC
   14  FCGR-UC     NORMAL    NO    0.0000D+00   1.0000D+00      .00      3  TRC
   15  DKINFile    - CONSTANT -    1.0000D+00                            4  TRC
   16  Percent1    - CONSTANT -    5.6175D+01                            1  FMD
   17  Percent2    - CONSTANT -    3.0283D+01                            2  FMD
   18  Percent3    - CONSTANT -    3.9086D+00                            3  FMD
   19  Percent4    - CONSTANT -    9.6333D+00                            4  FMD

  INFORMATION GENERATED FROM FAVLOADS.DAT FILE
  AND SAVED IN DKINSAVE.DAT FILE:

  WALL THICKNESS =  8.0360 INCH

  FLAW DEPTH  MINIMUM K AND MAXIMUM K FOR

    TYPE 1 WITH AN ASPECT RATIO OF   2.

    8.03600D-02    2.41927D+00    1.03655D+01
    1.47862D-01    3.22858D+00    1.40170D+01
    4.01800D-01    1.29279D+01    1.75751D+01
    6.02700D-01    1.41327D+01    2.09080D+01
    8.03600D-01    1.49423D+01    2.33544D+01
    1.60720D+00    1.45812D+01    2.72710D+01
    2.41080D+00    1.02448D+01    2.63600D+01
    4.01800D+00    2.35823D+00    2.78623D+01
```

C-2: ISI Every 10 Years (cont.)

TYPE 2 WITH AN ASPECT RATIO OF 6.

8.03600D-02	3.63673D+00	1.56338D+01
1.47862D-01	4.95557D+00	2.15454D+01
4.01800D-01	1.90999D+01	2.63794D+01
6.02700D-01	2.31650D+01	3.16223D+01
8.03600D-01	2.48064D+01	3.60464D+01
1.60720D+00	2.65025D+01	4.51155D+01
2.41080D+00	2.31198D+01	4.76172D+01
4.01800D+00	1.54934D+01	5.27667D+01

TYPE 3 WITH AN ASPECT RATIO OF 10.

8.03600D-02	3.98451D+00	1.71374D+01
1.47862D-01	5.29827D+00	2.30393D+01
4.01800D-01	2.02922D+01	2.81955D+01
6.02700D-01	2.51750D+01	3.36684D+01
8.03600D-01	2.69393D+01	3.84779D+01
1.60720D+00	2.92755D+01	4.91684D+01
2.41080D+00	2.74642D+01	5.45509D+01
4.01800D+00	2.02195D+01	6.28814D+01

TYPE 4 WITH AN ASPECT RATIO OF 99.

8.03600D-02	6.51796D+00	1.75511D+01
1.60720D-01	1.01756D+01	2.28059D+01
2.41080D-01	1.54398D+01	2.23553D+01
4.01800D-01	2.18696D+01	2.94323D+01
6.02700D-01	2.69582D+01	3.66108D+01
8.03600D-01	2.88204D+01	4.17713D+01
1.60720D+00	3.37365D+01	5.67413D+01
2.41080D+00	3.35927D+01	6.64759D+01

AVERAGE CALCULATED VALUES FOR: Surface Flaw Density with FCG and ISI

NUMBER FAILED = 0 NUMBER OF TRIALS = 1000

DEPTH (WALL/400) AND FLAW DENSITY FOR ASPECT RATIOS OF 2, 6, 10 AND 99

8	4.3486D-08	1.2355D-08	1.2447D-09	4.0471D-09
9	0.0000D+00	6.1902D-09	9.9626D-10	1.8380D-09
10	0.0000D+00	1.8825D-10	3.7663D-11	2.6218D-11
11	0.0000D+00	4.7355D-12	1.6752D-12	1.3302D-12
12	0.0000D+00	3.5199D-12	4.3837D-13	0.0000D+00
13	0.0000D+00	0.0000D+00	3.0423D-13	0.0000D+00

APPENDIX D
BEAVER VALLEY UNIT 1 PTS TRANSIENTS

Table D-1 PTS Transient Descriptions for BV1

Count	TH Case #	System Failure	Operator Action	HZP	Dominant*
1	002	3.59 cm [1.414 in] surge line break	None.	No	No
2	003	5.08 cm [2 in] surge line break	None.	No	No
3	007	2.54 cm [8 in] surge line break	None.	No	Yes at 32, 60, 100, 200 EFPY
4	009	2.54 cm [16 in] hot leg break	None.	No	Yes at 32, 60, 100, 200 EFPY
5	014	Reactor/turbine trip w/one stuck open pressurizer SRV	None.	No	No
6	031	Reactor/turbine trip w/feed and bleed (Operator open all pressurizer PORVs and use all charging/HHSI pumps)	None.	No	No
7	034	Reactor/turbine trip w/two stuck open pressurizer SRV's	None.	No	No
8	056	10.16 cm [4.0 in] surge line break	None.	Yes	Yes at 32, 60, 100, 200 EFPY
9	059	Reactor/turbine trip w/one stuck open pressurizer SRV which recloses at 3,000 s.	None.	No	No
10	060	Reactor/turbine trip w/one stuck open pressurizer SRV which recloses at 6,000 s.	None.	No	Yes at 32, 60, 100 EFPY
11	061	Reactor/turbine trip w/two stuck open pressurizer SRV which recloses at 3,000 s.	None.	No	No
12	062	Reactor/turbine trip w/two stuck open pressurizer SRV which recloses at 6,000 s.	None.	No	No
13	064	Reactor/turbine trip w/two stuck open pressurizer SRV's	None.	Yes	No
14	065	Reactor/turbine trip w/two stuck open pressurizer SRV's and HHSI failure	Operator opens all ASDVs 5 minutes after HHSI would have come on.	No	No

Table D-1 PTS Transient Descriptions for BV1					
Count	**TH Case #**	**System Failure**	**Operator Action**	**HZP**	**Dominant***
15	066	Reactor/turbine trip w/two stuck open pressurizer SRV's. One valve recloses at 3000 seconds while the other valve remains open.	None.	No	No
16	067	Reactor/turbine trip w/two stuck open pressurizer SRV's. One valve recloses at 6000 seconds while the other valve remains open.	None.	No	No
17	068	Reactor/turbine trip w/two stuck open pressurizer SRV's that reclose at 6000 s with HHSI failure.	Operator opens all ASDVs 5 minutes after HHSI would have come on.	No	No
18	069	Reactor/turbine trip w/two stuck open pressurizer SRVs which reclose at 3,000 s.	None.	Yes	No
19	070	Reactor/turbine trip w/two stuck open pressurizer SRVs which reclose at 6,000 s.	None.	Yes	No
20	071	Reactor/turbine trip w/one stuck open pressurizer SRV which recloses at 6,000 s.	None.	Yes	Yes at 32 EFPY
21	072	Reactor/turbine trip w/one stuck open pressurizer SRV with HHSI failure.	Operator opens all ASDVs 5 minutes after HHSI would have come on.	No	No
22	073	Reactor/turbine trip w/one stuck open pressurizer SRV with HHSI failure	Operator open all ASDVs 5 minutes after HHSI would have come on.	Yes	No
23	074	Main steam line break with AFW continuing to feed affected generator	None.	No	No
24	076	Reactor/turbine trip w/full MFW to all 3 SGs (MFW maintains SG level near top).	Operator trips reactor coolant pumps.	Yes	No
25	078	Reactor/turbine trip with failure of MFW and AFW.	Operator opens all ASDVs to let condensate fill SGs.	No	No
26	081	Main Steam Line Break with AFW continuing to feed affected generator and with HHSI failure initially.	Operator opens ADVs (on intact generators). HHSI is restored after CFTs discharge 50%.	No	No
27	082	Reactor/turbine trip w/one stuck open pressurizer SRV (recloses at 6000 s) and with HHSI failure.	Operator opens all ASDVs 5 minutes after HHSI would have started.	No	No

Table D-1 PTS Transient Descriptions for BV1					
Count	**TH Case #**	**System Failure**	**Operator Action**	**HZP**	**Dominant***
28	083	2.54 cm [1.0 in] surge line break with HHSI failure and motor driven AFW failure. MFW is tripped. Level control failure causes all steam generators to be overfed with turbine AFW, with the level maintained at top of SGs.	Operator trips RCPs. Operator opens all ASDVs 5 minutes after HHSI would have come on.	No	No
29	092	Reactor/turbine trip w/two stuck open pressurizer SRV's, one recloses at 3000 s.	None.	Yes	No
30	093	Reactor/turbine trip w/two stuck open pressurizer SRV's. One valve recloses at 6000 seconds while the other valve remains open.	None.	Yes	No
31	094	Reactor/turbine trip w/one stuck open pressurizer SRV.	None.	Yes	No
32	097	Reactor/turbine trip w/one stuck open pressurizer SRV which recloses at 3,000 s.	None.	Yes	Yes at 32, 60 EFPY
33	102	Main steam line break with AFW continuing to feed affected generator for 30 minutes.	Operator controls HHSI 30 minutes after allowed. Break is assumed to occur inside containment so that the operator trips the RCPs due to adverse containment conditions.	No	Yes at 100, 200 EFPY
34	103	Main steam line break with AFW continuing to feed affected generator for 30 minutes.	Operator controls HHSI 30 minutes after allowed. Break is assumed to occur inside containment so that the operator trips the RCPs due to adverse containment conditions.	Yes	Yes at 60, 100, 200 EFPY
35	104	Main steam line break with AFW continuing to feed affected generator for 30 minutes.	Operator controls HHSI 60 minutes after allowed. Break is assumed to occur inside containment so that the operator trips the RCPs due to adverse containment conditions.	No	Yes at 100, 200 EFPY
36	105	Main steam line break with AFW continuing to feed affected generator for 30 minutes.	Operator controls HHSI 60 minutes after allowed. Break is assumed to occur inside containment so that the operator trips the RCPs due to adverse containment conditions.	Yes	No
37	106	Main steam line break with AFW continuing to feed affected generator.	Operator controls HHSI 30 minutes after allowed. Break is assumed to occur inside containment so that the operator trips the RCPs due to adverse containment conditions.	No	No

Table D-1 PTS Transient Descriptions for BV1					
Count	**TH Case #**	**System Failure**	**Operator Action**	**HZP**	**Dominant***
38	107	Main steam line break with AFW continuing to feed affected generator.	Operator controls HHSI 30 minutes after allowed. Break is assumed to occur inside containment so that the operator trips the RCPs due to adverse containment conditions.	Yes	No
39	108	Small steam line break (simulated by sticking open all SG-A SRVs) with AFW continuing to feed affected generator for 30 minutes.	Operator controls HHSI 30 minutes after allowed.	Yes	No
40	109	Small steam line break (simulated by sticking open all SG-A SRVs) with AFW continuing to feed affected generator for 30 minutes.	Operator controls HHSI 30 minutes after allowed. Break is assumed to occur inside containment so that the operator trips the RCPs due to adverse containment conditions.	Yes	No
41	110	Small steam line break (simulated by sticking open all SG-A SRVs) with AFW continuing to feed affected generator for 30 minutes	Operator controls HHSI 60 minutes after allowed.	No	Yes at 200 EFPY
42	111	Small steam line break (simulated by sticking open all SG-A SRVs) with AFW continuing to feed affected generator for 30 minutes.	Operator controls HHSI 60 minutes after allowed. Break is assumed to occur inside containment so that the operator trips the RCPs due to adverse containment conditions.	Yes	No
43	112	Small steam line break (simulated by sticking open all SG-A SRVs) with AFW continuing to feed affected generator.	Operator controls HHSI 30 minutes after allowed. Break is assumed to occur inside containment so that the operator trips the RCPs due to adverse containment conditions.	No	No
44	113	Small steam line break (simulated by sticking open all SG-A SRVs) with AFW continuing to feed affected generator.	Operator controls HHSI 30 minutes after allowed. Break is assumed to occur inside containment so that the operator trips the RCPs due to adverse containment conditions.	Yes	No
45	114	7.18 cm [2.828 in] surge line break, summer conditions (HHSI, LHSI temp = 55°F, Accumulator Temp = 105°F), heat transfer coefficient increased 30% (modeled by increasing heat transfer surface area by 30% in passive heat structures).	None.	No	No
46	115	7.18 cm [2.828 in] cold leg break	None.	No	No
47	116	14.366 cm [5.657 in] cold leg break with break area increased 30%	None.	No	No

Table D-1 PTS Transient Descriptions for BV1

Count	TH Case #	System Failure	Operator Action	HZP	Dominant*
48	117	14.366 cm [5.657 in] cold leg break, summer conditions (HHSI, LHSI temp = 55°F, Accumulator Temp = 105°F)	None.	No	No
49	118	Small steam line break (simulated by sticking open all SG-A SRVs) with AFW continuing to feed affected generator	None.	No	No
50	119	Reactor/turbine trip w/two stuck open pressurizer SRV which recloses at 6,000 s	Operator controls HHSI (1 minute delay). Updated control logic.	No	No
51	120	Reactor/turbine trip w/two stuck open pressurizer SRV which recloses at 6,000 s	Operator controls HHSI (10 minute delay). Updated control logic.	No	No
52	121	Reactor/turbine trip w/two stuck open pressurizer SRV which recloses at 3,000 s	Operator controls HHSI (1 minute delay). Updated control logic.	Yes	No
53	122	Reactor/turbine trip w/two stuck open pressurizer SRVs which reclose at 6,000 s	Operator controls HHSI (1 minute delay). Updated control logic.	Yes	No
54	123	Reactor/turbine trip w/two stuck open pressurizer SRVs which reclose at 3,000 s	Operator controls HHSI (10 minute delay). Updated control logic.	Yes	Yes at 32 EFPY
55	124	Reactor/turbine trip w/two stuck open pressurizer SRVs which reclose at 6,000 s	Operator controls HHSI (10 minute delay). Updated control logic.	Yes	No
56	125	Reactor/turbine trip w/one stuck open pressurizer SRV which recloses at 6,000 s	Operator controls HHSI (1 minute delay). Updated control logic.	No	No
57	126	Reactor/turbine trip w/one stuck open pressurizer SRV which recloses at 6,000 s	Operator controls HHSI (10 minute delay). Updated control logic.	No	Yes at 32, 60, 100 EFPY
58	127	Reactor/turbine trip w/one stuck open pressurizer SRV which recloses at 6,000 s	Operator controls HHSI (1 minute delay). Updated control logic.	Yes	No

Table D-1 PTS Transient Descriptions for BV1					
Count	**TH Case #**	**System Failure**	**Operator Action**	**HZP**	**Dominant***
59	128	Reactor/turbine trip w/one stuck open pressurizer SRV which recloses at 3,000 s	Operator controls HHSI (1 minute delay). Updated control logic.	Yes	No
60	129	Reactor/turbine trip w/one stuck open pressurizer SRV which recloses at 6,000 s	Operator controls HHSI (10 minute delay). Updated control logic.	Yes	Yes at 32, 60 EFPY
61	130	Reactor/turbine trip w/one stuck open pressurizer SRV which recloses at 3,000 s	Operator controls HHSI (10 minute delay). Updated control logic.	Yes	Yes at 32, 60, 100 EFPY

Notes:
1. TH – Thermal hydraulics
2. LOCA – Loss-of-coolant accident
3. SBLOCA – Small-break loss-of-coolant accident
4. MBLOCA – Medium-break loss-of-coolant accident
5. LBLOCA – Large-break loss-of-coolant accident
6. HZP – Hot-zero power
7. SRV – Safety and relief valve
8. MSLB – Main steam line break
9. AFW – Auxiliary feedwater
10. HPI – High-pressure injection
11. RCPs – Reactor coolant pumps

* The arbitrary definition of a dominant transient is a transient that contributes 1% or more of the total Through-Wall Cracking Failure (TWCF).

APPENDIX E
BEAVER VALLEY UNIT 1 FAVPOST OUTPUT

E-1: 10 Year ISI Only

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************************************************
*                                              *
*              WELCOME TO FAVOR                *
*                                              *
*   FRACTURE ANALYSIS OF VESSELS: OAK RIDGE    *
*                VERSION 05.1                  *
*                                              *
*     FAVPOST MODULE: POSTPROCESSOR MODULE     *
*  COMBINES TRANSIENT INITIAITING FREQUENCIES  *
*        WITH RESULTS OF PFM ANALYSIS          *
*                                              *
*     PROBLEMS OR QUESTIONS REGARDING FAVOR    *
*            SHOULD BE DIRECTED TO             *
*                                              *
*               TERRY DICKSON                  *
*        OAK RIDGE NATIONAL LABORATORY         *
*                                              *
*          e-mail: dicksontl@ornl.gov          *
*                                              *
************************************************

***********************************************************
* This computer program was prepared as an account of     *
* work sponsored by the United States Government          *
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* Department of Energy, nor the United States Nuclear     *
* Regulatory Commission, nor any of their employees,      *
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* assumes any legal liability or responsibility for the   *
* accuracy, completeness, or usefulness of any            *
* information, apparatus, product, or process disclosed,  *
* or represents that its use would not infringe           *
* privately-owned rights.                                 *
***********************************************************


DATE: 04-Nov-2005  TIME: 10:28:56


FAVPOST INPUT  FILE NAME                     = postbv.in
FAVPFM  OUTPUT FILE CONTAINING PFMI ARRAY = INITIATE.DAT
FAVPFM  OUTPUT FILE CONTAINING PFMF ARRAY = FAILURE.DAT
FAVPOST OUTPUT FILE NAME                     = 70000.out


   *********************************
   * NUMBER OF SIMULATIONS =  70000 *
   *********************************
```

E-1: 10 Year ISI Only (cont.)

TRANSIENT NUMBER	CONDITIONAL PROBABILITY OF INITIATION CPI=P(I|E) MEAN CPI	95th % CPI	99th % CPI	CONDITIONAL PROBABILITY OF FAILURE CPF=P(F|E) MEAN CPF	95th % CPF	99th % CPF	RATIO CPFmn/CPImn
2	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000
3	7.6374E-07	0.0000E+00	0.0000E+00	5.5035E-10	0.0000E+00	0.0000E+00	0.0007
7	2.7571E-03	5.6821E-03	3.3310E-02	2.2371E-05	1.9975E-04	2.1081E-04	0.0081
9	3.2069E-03	6.4951E-03	4.2593E-02	1.5386E-05	1.3201E-04	2.5868E-04	0.0048
14	3.8125E-08	0.0000E+00	0.0000E+00	1.8254E-12	0.0000E+00	0.0000E+00	0.0000
31	3.7578E-06	0.0000E+00	5.3739E-06	9.7546E-09	0.0000E+00	0.0000E+00	0.0026
34	3.2068E-06	0.0000E+00	3.1596E-06	5.4857E-09	0.0000E+00	0.0000E+00	0.0017
56	3.0133E-03	5.9261E-03	3.5996E-02	1.1795E-05	1.8407E-04	1.9278E-04	0.0039
59	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000
60	1.6254E-05	3.3482E-04	4.7535E-05	1.6188E-05	3.3482E-04	4.6936E-05	0.9960
61	3.2188E-05	1.0222E-03	4.2646E-04	9.7772E-06	1.6071E-04	1.2352E-04	0.3038
62	6.7799E-06	0.0000E+00	1.1850E-05	3.7461E-06	0.0000E+00	2.5715E-07	0.5525
64	5.9938E-05	6.6389E-04	8.3149E-04	1.4250E-07	0.0000E+00	0.0000E+00	0.0024
65	1.6724E-07	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000
66	3.0302E-05	1.0222E-03	4.0011E-04	2.8466E-07	0.0000E+00	7.6395E-08	0.0094
67	2.5244E-06	0.0000E+00	1.2032E-06	1.8558E-09	0.0000E+00	0.0000E+00	0.0007
68	7.3375E-07	0.0000E+00	0.0000E+00	3.8818E-07	0.0000E+00	0.0000E+00	0.5290
69	4.6965E-04	2.6882E-03	5.9400E-03	3.4299E-04	2.3854E-03	3.4546E-03	0.7303
70	1.2588E-04	1.4959E-03	2.5843E-03	2.4716E-05	1.4958E-03	1.6818E-04	0.1963
71	1.1163E-05	0.0000E+00	4.2542E-05	1.0078E-05	0.0000E+00	3.4650E-05	0.9028
72	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000
73	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000
74	9.4097E-07	0.0000E+00	0.0000E+00	1.4446E-07	0.0000E+00	0.0000E+00	0.1535
76	1.5000E-12	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000
78	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000
81	3.0845E-10	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000
82	2.0239E-09	0.0000E+00	0.0000E+00	1.9682E-09	0.0000E+00	0.0000E+00	0.9725
83	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000
92	2.5221E-04	2.6882E-03	4.3698E-03	2.4069E-06	1.3492E-04	1.7547E-05	0.0095
93	2.5221E-04	2.6882E-03	4.3698E-03	2.4069E-06	1.3492E-04	1.7547E-05	0.0095
94	9.3226E-10	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000
97	4.2917E-05	1.3064E-03	3.8919E-04	4.1429E-05	1.3064E-03	3.5267E-04	0.9653
102	7.6743E-06	0.0000E+00	1.8991E-05	3.8212E-07	0.0000E+00	5.7475E-08	0.0498

E-1: 10 Year ISI Only (cont.)

103	7.4101E-05	1.7535E-03	1.2240E-03	1.0560E-05	3.3704E-04	1.0577E-04	0.1425
104	7.6743E-06	0.0000E+00	1.8991E-05	3.8212E-07	0.0000E+00	5.7475E-08	0.0498
105	9.7900E-07	0.0000E+00	0.0000E+00	2.3627E-07	0.0000E+00	0.0000E+00	0.2413
106	7.5240E-06	0.0000E+00	1.9911E-05	3.0346E-07	0.0000E+00	6.1572E-08	0.0403
107	6.2004E-05	1.6508E-03	8.8517E-04	1.2311E-05	3.3756E-04	1.3610E-04	0.1986
108	4.6067E-08	0.0000E+00	0.0000E+00	5.4710E-11	0.0000E+00	0.0000E+00	0.0012
109	3.6621E-07	0.0000E+00	0.0000E+00	1.2338E-08	0.0000E+00	0.0000E+00	0.0337
110	4.6067E-08	0.0000E+00	0.0000E+00	5.4710E-11	0.0000E+00	0.0000E+00	0.0012
111	3.6621E-07	0.0000E+00	0.0000E+00	1.2338E-08	0.0000E+00	0.0000E+00	0.0337
112	4.1576E-08	0.0000E+00	0.0000E+00	5.2737E-11	0.0000E+00	0.0000E+00	0.0013
113	2.5716E-07	0.0000E+00	0.0000E+00	1.8363E-08	0.0000E+00	0.0000E+00	0.0714
114	2.1511E-05	8.1510E-04	2.8659E-04	5.8566E-08	0.0000E+00	0.0000E+00	0.0027
115	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000
116	1.0473E-05	5.9788E-04	7.0627E-05	1.4761E-08	0.0000E+00	0.0000E+00	0.0014
117	5.1424E-05	5.0747E-04	8.3700E-04	2.1688E-08	0.0000E+00	0.0000E+00	0.0004
118	9.3071E-08	0.0000E+00	0.0000E+00	4.5498E-10	0.0000E+00	0.0000E+00	0.0049
119	5.4721E-06	0.0000E+00	1.2451E-05	1.7745E-08	0.0000E+00	0.0000E+00	0.0032
120	1.6463E-05	7.3530E-04	9.9671E-05	1.0717E-05	7.3525E-04	3.0519E-05	0.6510
121	6.0575E-05	2.4970E-03	6.5834E-04	9.1184E-08	0.0000E+00	3.4630E-09	0.0015
122	6.0575E-05	2.4970E-03	6.5834E-04	9.1115E-09	0.0000E+00	0.0000E+00	0.0002
123	2.1671E-04	2.4970E-03	2.9800E-03	1.8248E-04	2.1693E-03	2.4295E-03	0.8420
124	7.2595E-05	2.4970E-03	8.1638E-04	1.3063E-05	1.1486E-03	3.5391E-05	0.1799
125	1.0084E-07	0.0000E+00	0.0000E+00	2.5450E-11	0.0000E+00	0.0000E+00	0.0003
126	4.0844E-06	0.0000E+00	1.3341E-06	3.9031E-06	0.0000E+00	1.1204E-06	0.9556
127	2.3342E-08	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000
128	2.3342E-08	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000
129	2.8388E-06	0.0000E+00	1.4336E-07	1.7696E-06	0.0000E+00	9.2881E-09	0.6234
130	1.2370E-05	0.0000E+00	1.8447E-05	1.1955E-05	0.0000E+00	1.6861E-05	0.9665

NOTES: CPI IS CONDITIONAL PROBABILITY OF CRACK INITIATION, P(I|E)
CPF IS CONDITIONAL PROBABILITY OF TWC FAILURE, P(F|E)

E-1: 10 Year ISI Only (cont.)

```
*****************************************************
*     PROBABILITY DISTRIBUTION FUNCTION (HISTOGRAM)     *
*        FOR THE FREQUENCY OF CRACK INITIATION          *
*****************************************************
```

FREQUENCY OF CRACK INITIATION (PER REACTOR-OPERATING YEAR)	RELATIVE DENSITY (%)	CUMULATIVE DISTRIBUTION (%)
0.0000E+00	0.2186	0.2186
2.3948E-06	97.9957	98.2143
7.1845E-06	1.0371	99.2514
1.1974E-05	0.3443	99.5957
1.6764E-05	0.1600	99.7557
2.1553E-05	0.0643	99.8200
2.6343E-05	0.0514	99.8714
3.1133E-05	0.0343	99.9057
3.5922E-05	0.0214	99.9271
4.0712E-05	0.0086	99.9357
4.5502E-05	0.0086	99.9443
5.0291E-05	0.0057	99.9500
5.5081E-05	0.0043	99.9543
5.9871E-05	0.0043	99.9586
6.4660E-05	0.0043	99.9629
6.9450E-05	0.0100	99.9729
7.4240E-05	0.0057	99.9786
7.9029E-05	0.0014	99.9800
8.3819E-05	0.0014	99.9814
1.0298E-04	0.0014	99.9829
1.0777E-04	0.0029	99.9857
1.4129E-04	0.0014	99.9871
1.4608E-04	0.0029	99.9900
1.5566E-04	0.0014	99.9914
1.6524E-04	0.0014	99.9929
1.7003E-04	0.0014	99.9943
1.7482E-04	0.0014	99.9957
1.9877E-04	0.0014	99.9971
2.6104E-04	0.0014	99.9986
4.7657E-04	0.0014	100.0000

```
=============================================
==        Summary Descriptive Statistics     ==
=============================================

Minimum                          = 0.0000E+00
Maximum                          = 4.7418E-04
Range                            = 4.7418E-04

Number of Simulations            =   70000

5th Percentile                   = 2.2105E-10
Median                           = 5.8229E-08
95.0th Percentile                = 2.3948E-06
99.0th Percentile                = 6.0234E-06
```

E-1: 10 Year ISI Only (cont.)

99.9th Percentile	= 3.0334E-05
Mean	= 5.4171E-07
Standard Deviation	= 3.5373E-06
Standard Error	= 1.3370E-08
Variance (unbiased)	= 1.2512E-11
Variance (biased)	= 1.2512E-11
Moment Coeff. of Skewness	= 5.6550E+01
Pearson's 2nd Coeff. of Skewness	= 4.5943E-01
Kurtosis	= 5.6636E+03

```
*************************************************************
*      PROBABILITY DISTRIBUTION FUNCTION (HISTOGRAM)        *
*      FOR THROUGH-WALL CRACKING FREQUENCY (FAILURE)        *
*************************************************************
```

FREQUENCY OF TWC FAILURES (PER REACTOR-OPERATING YEAR)	RELATIVE DENSITY (%)	CUMULATIVE DISTRIBUTION (%)
0.0000E+00	9.3643	9.3643
9.0896E-08	90.2371	99.6014
2.7269E-07	0.2129	99.8143
4.5448E-07	0.0529	99.8671
6.3627E-07	0.0257	99.8929
8.1806E-07	0.0200	99.9129
9.9985E-07	0.0157	99.9286
1.1816E-06	0.0043	99.9329
1.3634E-06	0.0071	99.9400
1.5452E-06	0.0100	99.9500
1.7270E-06	0.0143	99.9643
1.9088E-06	0.0057	99.9700
2.0906E-06	0.0057	99.9757
2.2724E-06	0.0043	99.9800
2.8178E-06	0.0014	99.9814
2.9996E-06	0.0014	99.9829
3.3631E-06	0.0014	99.9843
3.5449E-06	0.0014	99.9857
3.9085E-06	0.0029	99.9886
4.0903E-06	0.0014	99.9900
4.2721E-06	0.0014	99.9914
6.6354E-06	0.0014	99.9929
7.7261E-06	0.0014	99.9943
1.0453E-05	0.0014	99.9957
1.2089E-05	0.0014	99.9971
1.3907E-05	0.0014	99.9986
1.8088E-05	0.0014	100.0000

```
=============================================
==      Summary Descriptive Statistics     ==
=============================================
```

E-1: 10 Year ISI Only (cont.)

```
Minimum                                  = 0.0000E+00
Maximum                                  = 1.7997E-05
Range                                    = 1.7997E-05

Number of Simulations                    =  70000

5th Percentile                           = 0.0000E+00
Median                                   = 2.2644E-12
95.0th Percentile                        = 9.0896E-08
99.0th Percentile                        = 5.9436E-08
99.9th Percentile                        = 7.0119E-07

Mean                                     = 5.0405E-09
Standard Deviation                       = 1.2772E-07
Standard Error                           = 4.8272E-10
Variance (unbiased)                      = 1.6311E-14
Variance (biased)                        = 1.6311E-14
Moment Coeff. of Skewness                = 8.8158E+01
Pearson's 2nd Coeff. of Skewness         =-1.2494E+00
Kurtosis                                 = 9.8765E+03
```

```
***********************************************************
*  FRACTIONALIZATION OF FREQUENCY OF CRACK INITIATION     *
*      AND THROUGH-WALL CRACKING FREQUENCY (FAILURE) -    *
*       WEIGHTED BY TRANSIENT INITIATING FREQUENCIES      *
***********************************************************
```

	% of total frequency of crack initiation	% of total frequency of of TWC failure
2	0.00	0.00
3	0.02	0.00
7	15.92	14.79
9	5.97	2.61
14	0.00	0.00
31	0.00	0.00
34	0.00	0.00
56	76.28	31.09
59	0.00	0.00
60	0.09	10.11
61	0.01	0.37
62	0.00	0.01
64	0.00	0.00
65	0.00	0.00
66	0.01	0.01
67	0.00	0.00
68	0.00	0.00
69	0.00	0.18
70	0.00	0.01
71	0.01	1.13
72	0.00	0.00
73	0.00	0.00
74	0.00	0.00
76	0.00	0.00

E-1: 10 Year ISI Only (cont.)

78	0.00	0.00
81	0.00	0.00
82	0.00	0.00
83	0.00	0.00
92	0.01	0.01
93	0.01	0.01
94	0.00	0.00
97	0.04	3.88
102	0.18	2.74
103	0.16	2.23
104	0.17	0.93
105	0.00	0.07
106	0.01	0.01
107	0.01	0.10
108	0.01	0.00
109	0.01	0.01
110	0.01	0.00
111	0.00	0.01
112	0.00	0.00
113	0.00	0.00
114	0.47	0.14
115	0.00	0.00
116	0.05	0.00
117	0.25	0.01
118	0.00	0.00
119	0.00	0.00
120	0.00	0.29
121	0.00	0.00
122	0.00	0.00
123	0.01	0.69
124	0.00	0.04
125	0.00	0.00
126	0.19	19.41
127	0.00	0.00
128	0.00	0.00
129	0.01	0.86
130	0.08	8.23
TOTALS	100.00	100.00

```
**********************************************************
*   FRACTIONALIZATION OF FREQUENCY OF CRACK INITIATION   *
*     AND THROUGH-WALL CRACKING FREQUENCY (FAILURE) -    *
*                          BY                            *
*               RPV BELTLINE MAJOR REGION                *
*                  BY PARENT SUBREGION                   *
*                                                        *
*      WEIGHTED BY % CONTRIBUTION OF EACH TRANSIENT      *
*         TO FREQUENCY OF CRACK INITIATION AND           *
*        THROUGH-WALL CRACKING FREQUENCY (FAILURE)       *
**********************************************************
```

E-1: 10 Year ISI Only (cont.)

MAJOR REGION	RTndt (MAX)	% of total flaws	% of total frequency of crack initiation	% of total through-wall crack frequency cleavage	ductile	total

DATE: 04-Nov-2005 TIME: 10:30:14

E-2: ISI Every 10 Years

```
**************************************************
*                                                *
*                WELCOME TO FAVOR                *
*                                                *
*   FRACTURE ANALYSIS OF VESSELS: OAK RIDGE     *
*                  VERSION 05.1                  *
*                                                *
*     FAVPOST MODULE: POSTPROCESSOR MODULE      *
*  COMBINES TRANSIENT INITIAITING FREQUENCIES   *
*        WITH RESULTS OF PFM ANALYSIS           *
*                                                *
*     PROBLEMS OR QUESTIONS REGARDING FAVOR     *
*             SHOULD BE DIRECTED TO             *
*                                                *
*                 TERRY DICKSON                  *
*          OAK RIDGE NATIONAL LABORATORY         *
*                                                *
*          e-mail: dicksontl@ornl.gov            *
*                                                *
**************************************************

************************************************************
* This computer program was prepared as an account of     *
* work sponsored by the United States Government          *
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* Regulatory Commission, nor any of their employees,      *
* nor any of their contractors, subcontractors, or their  *
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* assumes any legal liability or responsibility for the   *
* accuracy, completeness, or usefulness of any            *
* information, apparatus, product, or process disclosed,  *
* or represents that its use would not infringe           *
* privately-owned rights.                                 *
************************************************************


DATE: 04-Nov-2005  TIME: 10:56:03


FAVPOST INPUT  FILE NAME                    = postbv.in
FAVPFM  OUTPUT FILE CONTAINING PFMI ARRAY = INITIATE.DAT
FAVPFM  OUTPUT FILE CONTAINING PFMF ARRAY = FAILURE.DAT
FAVPOST OUTPUT FILE NAME                    = 70000.out


   **********************************
   * NUMBER OF SIMULATIONS =  70000 *
   **********************************
```

E-2: ISI Every 10 Years (cont.)

TRANSIENT NUMBER	CONDITIONAL PROBABILITY OF INITIATION CPI=P(I|E) MEAN CPI	95th % CPI	99th % CPI	CONDITIONAL PROBABILITY OF FAILURE CPF=P(F|E) MEAN CPF	95th % CPF	99th % CPF	RATIO CPFmn/CPImn
2	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000
3	7.3099E-07	0.0000E+00	0.0000E+00	4.5869E-10	0.0000E+00	0.0000E+00	0.0006
7	2.6095E-03	5.3460E-03	3.1509E-02	2.3240E-05	1.6898E-04	2.2607E-04	0.0089
9	3.0658E-03	6.2771E-03	4.0860E-02	1.7123E-05	1.8425E-04	2.5602E-04	0.0056
14	2.2735E-08	0.0000E+00	0.0000E+00	1.0890E-13	0.0000E+00	0.0000E+00	0.0000
31	3.6059E-06	0.0000E+00	3.3987E-06	4.8059E-09	0.0000E+00	0.0000E+00	0.0013
34	2.8045E-06	0.0000E+00	6.7039E-07	2.5594E-09	0.0000E+00	0.0000E+00	0.0009
56	2.8825E-03	5.6301E-03	3.4746E-02	1.2533E-05	1.5293E-04	1.6910E-04	0.0043
59	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000
60	1.1138E-05	0.0000E+00	4.0644E-05	1.1125E-05	0.0000E+00	4.0497E-05	0.9988
61	2.9916E-05	6.2439E-04	3.4900E-04	9.3981E-06	1.6377E-04	9.9742E-05	0.3141
62	4.7339E-06	0.0000E+00	6.2706E-06	1.9666E-06	0.0000E+00	2.1643E-08	0.4154
64	5.4989E-05	1.9871E-03	5.5502E-04	6.0010E-08	0.0000E+00	0.0000E+00	0.0011
65	1.5967E-07	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000
66	2.8258E-05	6.2439E-04	3.1813E-04	3.0198E-07	0.0000E+00	2.5370E-08	0.0107
67	2.3731E-06	0.0000E+00	6.8662E-07	8.1027E-10	0.0000E+00	0.0000E+00	0.0003
68	4.8167E-07	0.0000E+00	0.0000E+00	1.5477E-07	0.0000E+00	0.0000E+00	0.3213
69	4.4360E-04	1.8769E-03	5.9474E-03	3.2319E-04	1.4112E-03	3.9874E-03	0.7286
70	6.7323E-05	1.9871E-03	7.8804E-04	1.3026E-05	2.8848E-04	4.8209E-05	0.1935
71	2.9876E-06	0.0000E+00	2.9198E-07	2.9787E-06	0.0000E+00	2.5586E-07	0.9970
72	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000
73	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000
74	8.1491E-07	0.0000E+00	0.0000E+00	4.4065E-08	0.0000E+00	0.0000E+00	0.0541
76	8.3064E-11	0.0000E+00	0.0000E+00	2.4171E-13	0.0000E+00	0.0000E+00	0.0029
78	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000
81	1.9977E-10	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000
82	2.0484E-12	0.0000E+00	0.0000E+00	1.3286E-12	0.0000E+00	0.0000E+00	0.6486
83	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000
92	2.3879E-04	1.8768E-03	2.1861E-03	2.3813E-06	4.7982E-05	1.3929E-05	0.0100
93	2.3879E-04	1.8768E-03	2.1861E-03	2.3813E-06	4.7982E-05	1.3929E-05	0.0100
94	1.1629E-09	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000
97	1.9455E-05	7.4525E-04	5.5437E-05	1.9451E-05	7.4525E-04	5.5437E-05	0.9998

E-2: ISI Every 10 Years (cont.)

102	7.6059E-06	0.0000E+00	1.7988E-05	3.4031E-07	0.0000E+00	3.2659E-08	0.0447
103	7.2086E-05	1.2014E-03	1.0838E-03	9.3180E-06	2.5639E-04	9.0875E-05	0.1293
104	7.6059E-06	0.0000E+00	1.7988E-05	3.4031E-07	0.0000E+00	3.2659E-08	0.0447
105	7.8222E-07	0.0000E+00	0.0000E+00	3.9488E-08	0.0000E+00	0.0000E+00	0.0505
106	7.7355E-06	0.0000E+00	1.9001E-05	3.1620E-07	0.0000E+00	3.6654E-08	0.0409
107	6.0651E-05	1.1241E-03	7.4550E-04	1.1002E-05	2.8868E-04	1.1158E-04	0.1814
108	3.4049E-08	0.0000E+00	0.0000E+00	1.5682E-10	0.0000E+00	0.0000E+00	0.0046
109	3.6270E-07	0.0000E+00	0.0000E+00	3.1459E-09	0.0000E+00	0.0000E+00	0.0087
110	3.4049E-08	0.0000E+00	0.0000E+00	1.5682E-10	0.0000E+00	0.0000E+00	0.0046
111	3.6270E-07	0.0000E+00	0.0000E+00	3.1459E-09	0.0000E+00	0.0000E+00	0.0087
112	2.9426E-08	0.0000E+00	0.0000E+00	1.5351E-10	0.0000E+00	0.0000E+00	0.0052
113	2.9348E-07	0.0000E+00	0.0000E+00	8.7048E-09	0.0000E+00	0.0000E+00	0.0297
114	1.5853E-05	4.8071E-04	1.1595E-04	1.6236E-08	0.0000E+00	0.0000E+00	0.0010
115	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000
116	9.9888E-06	3.4660E-04	6.2966E-05	8.9747E-09	0.0000E+00	0.0000E+00	0.0009
117	4.1320E-05	5.8616E-04	6.6857E-04	1.9783E-08	0.0000E+00	0.0000E+00	0.0005
118	9.1276E-08	0.0000E+00	0.0000E+00	4.3537E-10	0.0000E+00	0.0000E+00	0.0048
119	5.1182E-06	0.0000E+00	9.4538E-06	8.1800E-09	0.0000E+00	0.0000E+00	0.0016
120	1.1915E-05	3.3334E-04	5.4145E-05	6.9078E-06	0.0000E+00	6.5630E-06	0.5797
121	5.0631E-05	1.7120E-03	4.5095E-04	2.8141E-08	0.0000E+00	1.0207E-11	0.0006
122	5.0631E-05	1.7120E-03	4.5095E-04	6.3711E-09	0.0000E+00	0.0000E+00	0.0001
123	1.8872E-04	1.7120E-03	2.8001E-03	1.5743E-04	1.2600E-03	2.2751E-03	0.8342
124	5.8284E-05	1.7120E-03	6.4131E-04	8.0551E-06	1.9826E-04	2.3633E-05	0.1382
125	7.4039E-08	0.0000E+00	0.0000E+00	5.9480E-11	0.0000E+00	0.0000E+00	0.0008
126	2.3632E-06	0.0000E+00	1.2557E-07	2.2871E-06	0.0000E+00	9.7627E-08	0.9678
127	1.5164E-08	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000
128	1.5164E-08	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000
129	8.7384E-07	0.0000E+00	0.0000E+00	8.5868E-07	0.0000E+00	0.0000E+00	0.9826
130	8.3755E-06	0.0000E+00	7.5345E-06	8.3596E-06	0.0000E+00	7.1473E-06	0.9981

NOTES: CPI IS CONDITIONAL PROBABILITY OF CRACK INITIATION, P(I|E)
CPF IS CONDITIONAL PROBABILITY OF TWC FAILURE, P(F|E)

E-2: ISI Every 10 Years (cont.)

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**********************************************************
*      PROBABILITY DISTRIBUTION FUNCTION (HISTOGRAM)     *
*         FOR THE FREQUENCY OF CRACK INITIATION          *
**********************************************************
```

FREQUENCY OF CRACK INITIATION (PER REACTOR-OPERATING YEAR)	RELATIVE DENSITY (%)	CUMULATIVE DISTRIBUTION (%)
0.0000E+00	0.1957	0.1957
1.5531E-06	96.8100	97.0057
4.6594E-06	1.7143	98.7200
7.7657E-06	0.5614	99.2814
1.0872E-05	0.2714	99.5529
1.3978E-05	0.1286	99.6814
1.7084E-05	0.0814	99.7629
2.0191E-05	0.0500	99.8129
2.3297E-05	0.0314	99.8443
2.6403E-05	0.0257	99.8700
2.9510E-05	0.0271	99.8971
3.2616E-05	0.0157	99.9129
3.5722E-05	0.0114	99.9243
3.8828E-05	0.0100	99.9343
4.1935E-05	0.0057	99.9400
4.5041E-05	0.0043	99.9443
4.8147E-05	0.0029	99.9471
5.1253E-05	0.0100	99.9571
5.4360E-05	0.0029	99.9600
5.7466E-05	0.0043	99.9643
6.0572E-05	0.0071	99.9714
6.3678E-05	0.0014	99.9729
6.6785E-05	0.0014	99.9743
6.9891E-05	0.0029	99.9771
7.2997E-05	0.0014	99.9786
8.2316E-05	0.0014	99.9800
8.5422E-05	0.0029	99.9829
8.8529E-05	0.0043	99.9871
9.1635E-05	0.0014	99.9886
9.4741E-05	0.0014	99.9900
1.0095E-04	0.0014	99.9914
1.1338E-04	0.0014	99.9929
1.4755E-04	0.0014	99.9943
1.5997E-04	0.0014	99.9957
2.3142E-04	0.0014	99.9971
2.5316E-04	0.0014	99.9986
3.0597E-04	0.0014	100.0000

```
============================================
==      Summary Descriptive Statistics    ==
============================================
```

E-2: ISI Every 10 Years (cont.)

Minimum	= 0.0000E+00
Maximum	= 3.0752E-04
Range	= 3.0752E-04
Number of Simulations	= 70000
5th Percentile	= 1.9684E-10
Median	= 5.3514E-08
95.0th Percentile	= 1.5531E-06
99.0th Percentile	= 6.2086E-06
99.9th Percentile	= 3.0074E-05
Mean	= 5.0694E-07
Standard Deviation	= 2.9934E-06
Standard Error	= 1.1314E-08
Variance (unbiased)	= 8.9603E-12
Variance (biased)	= 8.9601E-12
Moment Coeff. of Skewness	= 4.3241E+01
Pearson's 2nd Coeff. of Skewness	= 5.0806E-01
Kurtosis	= 3.2246E+03

```
************************************************************
*      PROBABILITY DISTRIBUTION FUNCTION (HISTOGRAM)       *
*      FOR THROUGH-WALL CRACKING FREQUENCY (FAILURE)       *
************************************************************
```

FREQUENCY OF TWC FAILURES (PER REACTOR-OPERATING YEAR)	RELATIVE DENSITY (%)	CUMULATIVE DISTRIBUTION (%)
0.0000E+00	9.6271	9.6271
3.7659E-08	89.6043	99.2314
1.1298E-07	0.3486	99.5800
1.8829E-07	0.1214	99.7014
2.6361E-07	0.0714	99.7729
3.3893E-07	0.0486	99.8214
4.1424E-07	0.0329	99.8543
4.8956E-07	0.0200	99.8743
5.6488E-07	0.0171	99.8914
6.4020E-07	0.0100	99.9014
7.1551E-07	0.0114	99.9129
7.9083E-07	0.0057	99.9186
8.6615E-07	0.0057	99.9243
9.4147E-07	0.0086	99.9329
1.0168E-06	0.0071	99.9400
1.0921E-06	0.0043	99.9443
1.1674E-06	0.0043	99.9486
1.2427E-06	0.0029	99.9514
1.3181E-06	0.0057	99.9571
1.3934E-06	0.0014	99.9586

E-2: ISI Every 10 Years (cont.)

1.4687E-06	0.0029	99.9614
1.5440E-06	0.0014	99.9629
1.6193E-06	0.0029	99.9657
1.6946E-06	0.0014	99.9671
1.8453E-06	0.0029	99.9700
1.9959E-06	0.0029	99.9729
2.0712E-06	0.0014	99.9743
2.2972E-06	0.0014	99.9757
2.3725E-06	0.0014	99.9771
2.4478E-06	0.0029	99.9800
2.6738E-06	0.0014	99.9814
2.9750E-06	0.0014	99.9829
3.2010E-06	0.0014	99.9843
3.2763E-06	0.0014	99.9857
3.5023E-06	0.0014	99.9871
3.6529E-06	0.0014	99.9886
3.8035E-06	0.0014	99.9900
4.1048E-06	0.0014	99.9914
4.3307E-06	0.0014	99.9929
4.4814E-06	0.0014	99.9943
4.7073E-06	0.0014	99.9957
5.2346E-06	0.0014	99.9971
6.8915E-06	0.0014	99.9986
7.4941E-06	0.0014	100.0000

```
============================================
==       Summary Descriptive Statistics    ==
============================================

Minimum                               = 0.0000E+00
Maximum                               = 7.4564E-06
Range                                 = 7.4564E-06

Number of Simulations                 =   70000

5th Percentile                        = 0.0000E+00
Median                                = 1.8724E-12
95.0th Percentile                     = 3.7659E-08
99.0th Percentile                     = 5.2893E-08
99.9th Percentile                     = 6.2944E-07

Mean                                  = 4.0995E-09
Standard Deviation                    = 7.6551E-08
Standard Error                        = 2.8934E-10
Variance (unbiased)                   = 5.8600E-15
Variance (biased)                     = 5.8600E-15
Moment Coeff. of Skewness             = 5.2718E+01
Pearson's 2nd Coeff. of Skewness      =-1.9366E+00
Kurtosis                              = 3.6577E+03
```

E-2: ISI Every 10 Years (cont.)

```
**********************************************************
*  FRACTIONALIZATION OF FREQUENCY OF CRACK INITIATION    *
*      AND THROUGH-WALL CRACKING FREQUENCY (FAILURE) -   *
*       WEIGHTED BY TRANSIENT INITIATING FREQUENCIES     *
**********************************************************
```

	% of total frequency of crack initiation	% of total frequency of of TWC failure
2	0.00	0.00
3	0.02	0.00
7	14.70	17.42
9	5.52	4.14
14	0.00	0.00
31	0.00	0.00
34	0.00	0.00
56	78.22	41.56
59	0.00	0.00
60	0.07	9.01
61	0.01	0.49
62	0.00	0.01
64	0.00	0.00
65	0.00	0.00
66	0.01	0.02
67	0.00	0.00
68	0.00	0.00
69	0.00	0.25
70	0.00	0.01
71	0.00	0.31
72	0.00	0.00
73	0.00	0.00
74	0.00	0.00
76	0.00	0.00
78	0.00	0.00
81	0.00	0.00
82	0.00	0.00
83	0.00	0.00
92	0.01	0.01
93	0.01	0.02
94	0.00	0.00
97	0.02	1.92
102	0.19	0.78
103	0.16	2.43
104	0.21	0.71
105	0.00	0.01
106	0.00	0.01
107	0.01	0.14
108	0.00	0.00
109	0.00	0.00
110	0.00	0.00
111	0.00	0.00
112	0.00	0.00
113	0.00	0.00
114	0.36	0.04

E-2: ISI Every 10 Years (cont.)

115	0.00	0.00
116	0.06	0.00
117	0.20	0.01
118	0.00	0.00
119	0.00	0.00
120	0.00	0.18
121	0.00	0.00
122	0.00	0.00
123	0.01	0.72
124	0.00	0.03
125	0.00	0.00
126	0.10	11.83
127	0.00	0.00
128	0.00	0.00
129	0.01	0.63
130	0.06	7.28
TOTALS	100.00	100.00

DATE: 04-Nov-2005 TIME: 10:57:18

APPENDIX F
INPUTS FOR THE PALISADES PILOT PLANT EVALUATION

A summary of the NDE inspection history based on Regulatory Guide 1.150 and pertinent input data for Palisades is as follows:

1. Number of ISIs performed (relative to initial pre-service and 10-year interval inspections) for full penetration Category B-A, B-D, and B-J reactor vessel welds assuming all of the candidate welds were inspected: 2 (covering all welds of the specified categories).

2. The inspections performed covered: 100% for 13 Category B-A welds, >90% but <100% for 6 Category B-A welds, <90% for 8 Category B-A welds, and 100% of all Category B-D and B-J welds.

3. Number of indications found during most recent inservice inspection: 11
   This number includes consideration of the following additional information:

   a. Indications found that were reportable: 0
   b. Indications found that were within acceptable limits: 11
   c. Indications/anomalies currently being monitored: 0

4. Full penetration relief requests for the RV submitted and accepted by the NRC: 2 relief requests for limited converage for 12 welds

5. Fluence distribution at inside surface of RV beltline until end of life (EOL): see Figure F-1 taken from the NRC PTS Risk Study [7], Figure 4.3.

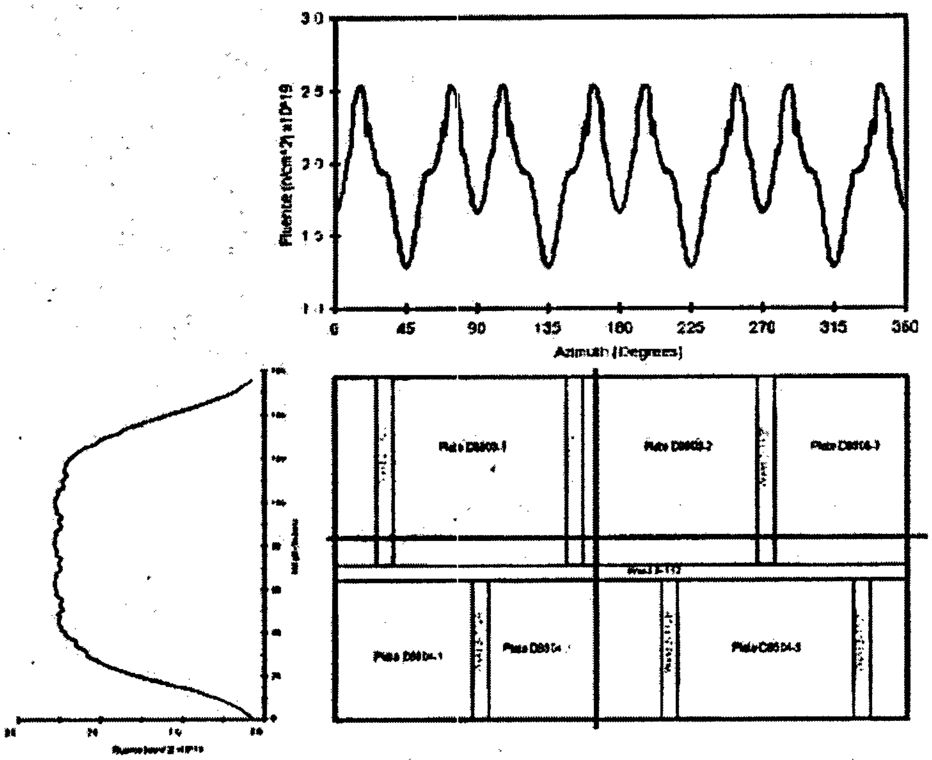

Figure F-1 Rollout Diagram of Beltline Materials and Representative Fluence Maps for Palisades

6. Reactor vessel cladding details:

   a. Thickness: 0.25 inches

   b. Material properties (assumed to be independent of temperature):

      1) Thermal conductivity (Btu/hr-ft-°F), K=10.0

      2) Specific heat (Btu/LBM-°F),C=0.120

      3) Density (LBM/ft^3).RHO=489.00

      4) Young's Modulus of Elasticity (KSI), E=22800

      5) Thermal expansion coefficient (°F^{-1}), ALPHA=0.00000945

      6) Poisson's Ratio, V=0.3

   c. Material including copper and nickel content: Material properties assigned to clad flaws are that of the underlying material be it base metal or weld. These properties are identified in Table F-1. This is consistent with the NRC PTS Risk Study [7].

   d. Material property uncertainties:

      1) Bead width: 1 inch – bead widths vary for all plants. Based on the NRC PTS Risk Study [7], a nominal dimension of 1 inch is selected for all analyses because this parameter is not expected to influence significantly the predicted vessel failure probabilities.

2) Truncation limit: Cladding thickness rounded to the next 1/100th of the total reactor vessel thickness to be consistent with the NRC PTS Risk Study [7].

3) Surface flaw depth: 0.263 inch

4) All flaws are surface-breaking. Only flaws in cladding that would influence brittle fracture of the reactor vessel are brittle. This is consistent with the NRC PTS Risk Study [7].

e. Additional cladding properties are identified in Table F-2.

7. Base metal:

a. Wall thickness: 8.5 inches

b. Material properties (assumed to be independent of temperature):

1) Thermal conductivity (Btu/hr-ft-°F), K=24.0

2) Specific heat (Btu/LBM-°F),C=0.120

3) Density (LBM/ft^3).RHO=489.00

4) Young's Modulus of Elasticity (KSI), E=28000

5) Thermal expansion coefficient (°F^{-1}), ALPHA=0.00000777

6) Poisson's Ratio, V=0.3

7) Other material properties are identified in Table F-1

Table F-1 Palisades-Specific Material Values Drawn from the RVID (see Ref. 7 Table 4.1)

Major Material Region Description				Cu [wt%]	Ni [wt%]	P [wt%]	Un-Irradiated RT_{NDT}		RT_{PTS} @60 EFPY
#	Type	Heat	Location				[°F]	Method	
1	Axial Weld	3-112A	lower	0.213	1.010	0.019	- 56	Generic	276.4
2	Axial Weld	3-112B	lower	0.213	1.010	0.019	- 56	Generic	285.3
3	Axial Weld	3-112C	lower	0.213	1.010	0.019	- 56	Generic	285.3
4	Axial Weld	2-112A	upper	0.213	1.010	0.019	- 56	Generic	285.8
5	Axial Weld	2-112B	upper	0.213	1.010	0.019	- 56	Generic	276.7
6	Axial Weld	2-112C	upper	0.213	1.010	0.019	- 56	Generic	285.8
7	Circ Weld	9-112	intermediate	0.203	1.018	0.013	- 56	Generic	270.3
8	Plate	D3804-1	lower	0.190	0.480	0.016	0	ASME NB-2331	261.9
9	Plate	D3804-2	lower	0.190	0.500	0.015	-30	MTEB 5-2	230.5
10	Plate	D3804-3	lower	0.120	0.550	0.010	-25	MTEB 5-2	170.0
11	Plate	D3803-1	upper	0.240	0.510	0.009	-5	ASME NB-2331	261.5
12	Plate	D3803-2	upper	0.240	0.520	0.010	-30	MTEB 5-2	242.4
13	Plate	D3803-3	upper	0.240	0.500	0.011	-5	ASME NB-2331	268.1

8. Weld metal details: Details of information used in addressing weld-specific information are taken directly from the NRC PTS Risk Study [7], Table 4.2. Summaries are reproduced as Table F-2.

Table F-2 Summary of Reactor Vessel-Specific Inputs for Flaw Distribution

Variable		Oconee	Beaver Valley	Palisades	Calvert Cliffs	Notes
Inner Radius (to cladding)	[in]	85.5	78.5	86	86	Vessel specific info
Base Metal Thickness	[in]	8.438	7.875	8.5	8.675	Vessel specific info
Total Wall Thickness	[in]	8.626	8.031	8.75	8.988	Vessel specific info

Variable			Oconee	Beaver Valley	Palisades	Calvert Cliffs	Notes
SAW Weld	Volume fraction	[%]	97%				100% - SMAW% - REPAIR%
	Thru-Wall Bead Thickness	[in]	0.1875	0.1875	0.1875	0.1875	All plants report plant specific dimensions of 3/16-in.
	Truncation Limit	[in]	1				Judgment. Approx. 2X the size of the largest non-repair flaw observed in PVRUF & Shoreham.
	Buried or Surface	--	All flaws are buried				Observation
	Orientation	--	Circ flaws in circ welds, axial flaws in axial welds.				Observation: Virtually all of the weld flaws in PVRUF & Shoreham were aligned with the welding direction because they were lack of sidewall fusion defects.
	Density basis	--	Shoreham density				Highest of observations
	Aspect ratio basis	--	Shoreham & PVRUF observations				Statistically similar distributions from Shoreham and PVRUF were combined to provide more robust estimates, when based on judgment the amount data were limited and/or insufficient to identify different trends for aspect ratios for flaws in the two vessels.
	Depth basis	--	Shoreham & PVRUF observations				Statistically similar distributions combined to provide more robust estimates

Table F-2 Summary of Reactor Vessel-Specific Inputs for Flaw Distribution (cont.)

Variable			Oconee	Beaver Valley	Palisades	Calvert Cliffs	Notes
SMAW Weld	*Volume fraction*	[%]	1%				Upper bound to all plant specific info provided by Steve Byrne (Westinghouse – Windsor).
	Thru-Wall Bead Thickness	[in]	0.21	0.20	0.22	0.25	Oconee is generic value based on average of all plants specific values (including Shoreham & PVRUF data). Other values are plant specific as reported by Steve Byrne.
	Truncation Limit	[in]	1				Judgment. Approx. 2X the size of the largest non-repair flaw observed in PVRUF & Shoreham.
	Buried or Surface	--	All flaws are buried				Observation
	Orientation	--	Circ flaws in circ welds, axial flaws in axial welds.				Observation: Virtually all of the weld flaws in PVRUF & Shoreham were aligned with the welding direction because they were lack of sidewall fusion defects.
	Density basis	--	Shoreham density				Highest of observations

Table F-2 Summary of Reactor Vessel-Specific Inputs for Flaw Distribution (cont.)

	Variable		Oconee	Beaver Valley	Palisades	Calvert Cliffs	Notes
Repair Weld	Volume fraction	[%]	2%				Judgment. A rounded integral percentage that exceeds the repaired volume observed for Shoreham and for PVRUF, which was 1.5%.
	Thru-Wall Bead Thickness	[in]	0.14				Generic value: As observed in PVRUF and Shoreham by PNNL.
	Truncation Limit	[in]	2				Judgment. Approx. 2X the largest repair flaw found in PVRUF & Shoreham. Also based on maximum expected width of repair cavity.
	Buried or Surface	--	All flaws are buried				Observation
	Orientation	--	Circ flaws in circ welds, axial flaws in axial welds.				The repair flaws had complex shapes and orientations that were not aligned with either the axial or circumferential welds; for consistency with the available treatments of flaws by the FAVOR code, a common treatment of orientations was adopted for flaws in SAW/SMAW and repair welds.
	Density basis	--	Shoreham density				Highest of observations
	Aspect ratio basis	--	Shoreham & PVRUF observations				Statistically similar distributions from Shoreham and PVRUF were combined to provide more robust estimates, when based on judgment the amount data were limited and/or insufficient to identify different trends for aspect ratios for flaws in the two vessels.
	Depth basis	--	Shoreham & PVRUF observations				Statistically similar distributions combined to provide more robust estimates

Table F-2 Summary of Reactor Vessel-Specific Inputs for Flaw Distribution (cont.)

Variable			Oconee	Beaver Valley	Palisades	Calvert Cliffs	Notes
Cladding	Actual Thickness	[in]	0.188	0.156	0.25	0.313	Vessel specific info
	# of Layers	[#]	1	2	2	2	Vessel specific info
	Bead Width	[in]	1				Bead widths of 1 to 5-in. characteristic of machine deposited cladding. Bead widths down to ½-in. can occur over welds. Nominal dimension of 1-in. selected for all analyses because this parameter is not expected to influence significantly the predicted vessel failure probabilities. May need to refine this estimate later, particularly for Oconee who reported a 5-in bead width.
	Truncation Limit	[in]	Actual clad thickness rounded to the nearest 1/100th of the total vessel wall thickness				Judgment & computational convenience
	Surface flaw depth in FAVOR	[in]	0.259	0.161	0.263	0.360	
	Buried or Surface	--	All flaws are surface breaking				Judgment. Only flaws in cladding that would influence brittle fracture of the vessel are brittle. Material properties assigned to clad flaws are that of the underlying material, be it base or weld.
	Orientation	--	All circumferential.				Observation: All flaws observed in PVRUF & Shoreham were lack of inter-run fusion defects, and cladding is always deposited circumferentially
	Density basis	--	No surface flaws observed. Density is 1/1000 that of the observed buried flaws in cladding of vessels examined by PNNL. If there is more than one clad layer then there are no clad flaws.				Judgment
	Aspect ratio basis	--	Observations on buried flaws				Judgment
	Depth basis	--	Depth of all surface flaws is the actual clad thickness rounded up to the nearest 1/100th of the total vessel wall thickness.				Judgment.

Table F-2 Summary of Reactor Vessel-Specific Inputs for Flaw Distribution (cont.)

Variable			Oconee	Beaver Valley	Palisades	Calvert Cliffs	Notes
Plate	Truncation Limit	[in]	0.433				Judgment. Twice the depth of the largest flaw observed in all PNNL plate inspections.
	Buried or Surface	--	All flaws are buried				Observation
	Orientation	--	Half of the simulated flaws are circumferential, half are axial.				Observation & Physics: No observed orientation preference, and no reason to suspect one (other than laminations which are benign.
	Density basis	--	1/10 of small weld flaw density, 1/40 of large weld flaw density of the PVRUF data				Judgment. Supported by limited data.
	Aspect ratio basis	--	Same as for PVRUF welds				Judgment
	Depth basis	--	Same as for PVRUF welds				Judgment. Supported by limited data.

9. TWCF calculated at 60 EFPY using correlation from Reference 27: 6.42E-09 Events per year

APPENDIX G
PALISADES PROBSBFD OUTPUT

G-1: 10 Year ISI Only

```
                    STRUCTURAL RELIABILITY AND RISK ASSESSMENT (SRRA)
WESTINGHOUSE            MONTE-CARLO SIMULATION PROGRAM PROBSBFD    VERSION 1.0

==============================================================================
      INPUT VARIABLES FOR CASE  2: PAL 10 YEAR ISI ONLY

      NCYCLE =     80          NFAILS =  1001          NTRIAL =  1000
      NOVARS =     19          NUMSET =     2          NUMISI =     5
      NUMSSC =      4          NUMTRC =     4          NUMFMD =     4

    VARIABLE       DISTRIBUTION        MEDIAN      DEVIATION   SHIFT    USAGE
  NO.   NAME       TYPE      LOG       VALUE      OR FACTOR   MV/SD   NO. SUB

    1  FIFDepth    - CONSTANT -     3.0000D-02                          1  SET
    2  IFlawDen    - CONSTANT -     3.6589D-03                          2  SET
    3  ICy-ISI     - CONSTANT -     1.0000D+01                          1  ISI
    4  DCy-ISI     - CONSTANT -     8.0000D+01                          2  ISI
    5  MV-Depth    - CONSTANT -     1.5000D-02                          3  ISI
    6  SD-Depth    - CONSTANT -     1.8500D-01                          4  ISI
    7  CEff-ISI    - CONSTANT -     1.0000D+00                          5  ISI
    8  Aspect1     - CONSTANT -     2.0000D+00                          1  SSC
    9  Aspect2     - CONSTANT -     6.0000D+00                          2  SSC
   10  Aspect3     - CONSTANT -     1.0000D+01                          3  SSC
   11  Aspect4     - CONSTANT -     9.9000D+01                          4  SSC
   12  NoTr/Cy     - CONSTANT -     1.3000D+01                          1  TRC
   13  FCGThld     - CONSTANT -     1.5000D+00                          2  TRC
   14  FCGR-UC     NORMAL    NO     0.0000D+00  1.0000D+00     .00      3  TRC
   15  DKINFile    - CONSTANT -     1.0000D+00                          4  TRC
   16  Percent1    - CONSTANT -     7.8870D+01                          1  FMD
   17  Percent2    - CONSTANT -     1.0720D+01                          2  FMD
   18  Percent3    - CONSTANT -     4.3807D+00                          3  FMD
   19  Percent4    - CONSTANT -     6.0298D+00                          4  FMD

  INFORMATION GENERATED FROM FAVLOADS.DAT FILE
  AND SAVED IN DKINSAVE.DAT FILE:

  WALL THICKNESS =  8.7500 INCH

  FLAW DEPTH  MINIMUM K AND MAXIMUM K FOR

    TYPE 1 WITH AN ASPECT RATIO OF   2.

    8.75000D-02    2.69285D+00    1.08492D+01
    1.61000D-01    3.60064D+00    1.46562D+01
    4.37500D-01    1.26609D+01    2.00367D+01
    6.56250D-01    1.49279D+01    2.39231D+01
    8.75000D-01 .  1.53491D+01    2.67406D+01
    1.75000D+00    1.37876D+01    3.14212D+01
    2.62500D+00    8.13906D+00    3.01520D+01
    4.37500D+00   -2.32655D+00    2.91175D+01

    TYPE 2 WITH AN ASPECT RATIO OF   6.

    8.75000D-02    4.04516D+00    1.64003D+01
```

G-1: 10 Year ISI Only (cont.)

1.61000D-01	5.52109D+00	2.25832D+01
4.37500D-01	1.80126D+01	3.03772D+01
6.56250D-01	2.31235D+01	3.61026D+01
8.75000D-01	2.65795D+01	4.11957D+01
1.75000D+00	2.62424D+01	5.18633D+01
2.62500D+00	2.10650D+01	5.45640D+01
4.37500D+00	9.61580D+00	5.85179D+01

TYPE 3 WITH AN ASPECT RATIO OF 10.

8.75000D-02	4.43154D+00	1.79837D+01
1.61000D-01	5.90218D+00	2.41564D+01
4.37500D-01	1.90406D+01	3.24750D+01
6.56250D-01	2.45354D+01	3.85918D+01
8.75000D-01	2.87821D+01	4.40958D+01
1.75000D+00	2.91774D+01	5.64674D+01
2.62500D+00	2.54877D+01	6.25646D+01
4.37500D+00	1.38132D+01	7.03917D+01

TYPE 4 WITH AN ASPECT RATIO OF 99.

8.75000D-02	7.10780D+00	1.85180D+01
1.75000D-01	1.00487D+01	2.59141D+01
2.62500D-01	1.38195D+01	2.86661D+01
4.37500D-01	2.16458D+01	3.45538D+01
6.56250D-01	2.85157D+01	4.23747D+01
8.75000D-01	3.03911D+01	4.83133D+01
1.75000D+00	3.36289D+01	6.57043D+01
2.62500D+00	3.16032D+01	7.68320D+01

AVERAGE CALCULATED VALUES FOR: Surface Flaw Density with FCG and ISI

NUMBER FAILED = 0 NUMBER OF TRIALS = 1000

DEPTH (WALL/400) AND FLAW DENSITY FOR ASPECT RATIOS OF 2, 6, 10 AND 99

12	2.5402D-04	4.5317D-06	1.3489D-06	2.1739D-06
13	5.8986D-06	1.9521D-05	7.3792D-06	9.7312D-06
14	0.0000D+00	7.0234D-06	3.2977D-06	4.3086D-06
15	0.0000D+00	1.9775D-06	1.1450D-06	1.7029D-06
16	0.0000D+00	5.8037D-07	4.0809D-07	6.4975D-07
17	0.0000D+00	3.4736D-07	1.5441D-07	2.2919D-07
18	0.0000D+00	1.5414D-07	8.8627D-08	1.7208D-07
19	0.0000D+00	9.1024D-08	6.1738D-08	5.0696D-08
20	0.0000D+00	0.0000D+00	3.6375D-08	8.2449D-08
21	0.0000D+00	0.0000D+00	0.0000D+00	3.2256D-08
23	0.0000D+00	2.7971D-08	0.0000D+00	0.0000D+00
25	0.0000D+00	0.0000D+00	1.1041D-08	0.0000D+00
26	0.0000D+00	2.6821D-08	0.0000D+00	0.0000D+00
27	0.0000D+00	0.0000D+00	0.0000D+00	1.4338D-08
29	0.0000D+00	0.0000D+00	1.0518D-08	0.0000D+00
31	0.0000D+00	0.0000D+00	0.0000D+00	1.3440D-08

G-2: ISI Every 10 Years

```
                 STRUCTURAL RELIABILITY AND RISK ASSESSMENT (SRRA)
WESTINGHOUSE         MONTE-CARLO SIMULATION PROGRAM PROBSBFD    VERSION 1.0

=============================================================================
     INPUT VARIABLES FOR CASE  2: PAL 10 YEAR INT

     NCYCLE =    80             NFAILS =  1001            NTRIAL =  1000
     NOVARS =    19             NUMSET =     2            NUMISI =     5
     NUMSSC =     4             NUMTRC =     4            NUMFMD =     4

   VARIABLE       DISTRIBUTION      MEDIAN       DEVIATION    SHIFT   USAGE
  NO.   NAME      TYPE      LOG      VALUE       OR FACTOR    MV/SD  NO. SUB

   1  FIFDepth    - CONSTANT -    3.0000D-02                          1  SET
   2  IFlawDen    - CONSTANT -    3.6589D-03                          2  SET
   3  ICy-ISI     - CONSTANT -    1.0000D+01                          1  ISI
   4  DCy-ISI     - CONSTANT -    1.0000D+01                          2  ISI
   5  MV-Depth    - CONSTANT -    1.5000D-02                          3  ISI
   6  SD-Depth    - CONSTANT -    1.8500D-01                          4  ISI
   7  CEff-ISI    - CONSTANT -    1.0000D+00                          5  ISI
   8  Aspect1     - CONSTANT -    2.0000D+00                          1  SSC
   9  Aspect2     - CONSTANT -    6.0000D+00                          2  SSC
  10  Aspect3     - CONSTANT -    1.0000D+01                          3  SSC
  11  Aspect4     - CONSTANT -    9.9000D+01                          4  SSC
  12  NoTr/Cy     - CONSTANT -    1.3000D+01                          1  TRC
  13  FCGThld     - CONSTANT -    1.5000D+00                          2  TRC
  14  FCGR-UC     NORMAL    NO    0.0000D+00   1.0000D+00      .00    3  TRC
  15  DKINFile    - CONSTANT -    1.0000D+00                          4  TRC
  16  Percent1    - CONSTANT -    7.8870D+01                          1  FMD
  17  Percent2    - CONSTANT -    1.0720D+01                          2  FMD
  18  Percent3    - CONSTANT -    4.3807D+00                          3  FMD
  19  Percent4    - CONSTANT -    6.0298D+00                          4  FMD

 INFORMATION GENERATED FROM FAVLOADS.DAT FILE
 AND SAVED IN DKINSAVE.DAT FILE:

 WALL THICKNESS =  8.7500 INCH

 FLAW DEPTH  MINIMUM K AND MAXIMUM K FOR

   TYPE 1 WITH AN ASPECT RATIO OF   2.

   8.75000D-02    2.69285D+00    1.08492D+01
   1.61000D-01    3.60064D+00    1.46562D+01
   4.37500D-01    1.26609D+01    2.00367D+01
   6.56250D-01    1.49279D+01    2.39231D+01
   8.75000D-01    1.53491D+01    2.67406D+01
   1.75000D+00    1.37876D+01    3.14212D+01
   2.62500D+00    8.13906D+00    3.01520D+01
   4.37500D+00   -2.32655D+00    2.91175D+01

   TYPE 2 WITH AN ASPECT RATIO OF   6.

   8.75000D-02    4.04516D+00    1.64003D+01
```

G-2: ISI Every 10 Years (cont.)

1.61000D-01	5.52109D+00	2.25832D+01
4.37500D-01	1.80126D+01	3.03772D+01
6.56250D-01	2.31235D+01	3.61026D+01
8.75000D-01	2.65795D+01	4.11957D+01
1.75000D+00	2.62424D+01	5.18633D+01
2.62500D+00	2.10650D+01	5.45640D+01
4.37500D+00	9.61580D+00	5.85179D+01

TYPE 3 WITH AN ASPECT RATIO OF 10.

8.75000D-02	4.43154D+00	1.79837D+01
1.61000D-01	5.90218D+00	2.41564D+01
4.37500D-01	1.90406D+01	3.24750D+01
6.56250D-01	2.45354D+01	3.85918D+01
8.75000D-01	2.87821D+01	4.40958D+01
1.75000D+00	2.91774D+01	5.64674D+01
2.62500D+00	2.54877D+01	6.25646D+01
4.37500D+00	1.38132D+01	7.03917D+01

TYPE 4 WITH AN ASPECT RATIO OF 99.

8.75000D-02	7.10780D+00	1.85180D+01
1.75000D-01	1.00487D+01	2.59141D+01
2.62500D-01	1.38195D+01	2.86661D+01
4.37500D-01	2.16458D+01	3.45538D+01
6.56250D-01	2.85157D+01	4.23747D+01
8.75000D-01	3.03911D+01	4.83133D+01
1.75000D+00	3.36289D+01	6.57043D+01
2.62500D+00	3.16032D+01	7.68320D+01

AVERAGE CALCULATED VALUES FOR: Surface Flaw Density with FCG and ISI

NUMBER FAILED = 0 NUMBER OF TRIALS = 1000

DEPTH (WALL/400) AND FLAW DENSITY FOR ASPECT RATIOS OF 2, 6, 10 AND 99

12	1.2465D-10	1.8940D-12	5.5678D-13	9.1111D-13
13	1.9983D-12	5.5048D-12	2.0459D-12	2.7226D-12
14	0.0000D+00	9.6570D-13	4.5289D-13	5.8811D-13
15	0.0000D+00	1.2835D-13	7.5032D-14	1.0930D-13
16	0.0000D+00	1.8170D-14	1.2594D-14	1.8759D-14
17	0.0000D+00	5.2179D-15	2.1701D-15	2.9926D-15
18	0.0000D+00	9.4118D-16	6.6938D-16	9.6145D-16
19	0.0000D+00	2.9809D-16	1.7580D-16	1.4879D-16
20	0.0000D+00	0.0000D+00	4.8987D-17	9.2976D-17
21	0.0000D+00	0.0000D+00	0.0000D+00	1.4658D-17
23	0.0000D+00	2.2110D-18	0.0000D+00	0.0000D+00
25	0.0000D+00	0.0000D+00	1.5152D-19	0.0000D+00
26	0.0000D+00	2.1470D-19	0.0000D+00	0.0000D+00
27	0.0000D+00	0.0000D+00	0.0000D+00	2.4461D-20
29	0.0000D+00	0.0000D+00	7.9308D-21	0.0000D+00
31	0.0000D+00	0.0000D+00	0.0000D+00	5.2922D-22

APPENDIX H
PALISADES PTS TRANSIENTS

Table H-1 PTS Transient Descriptions for Palisades

Count	TH Case #	System Failure	Operator Action	HZP	HiK	Dominant*
1	2	3.59 cm (1.414 in) surge line break. Containment sump recirculation included in the analysis.	None	No	Yes	No
2	16	Turbine/reactor trip with 2 stuck-open ADVs on SG-A combined with controller failure resulting in the flow from two AFW pumps into affected steam generator.	Operator starts second AFW pump. Operator isolates AFW to affected SG at 30 minutes after initiation. Operator assumed to throttle HPI if auxiliary feedwater is running with SG wide range level > -84% and RCS subcooling > 25 F. HPI is throttled to maintain pressurizer level between 40 and 60 %.	No	No	No
3	18	Turbine/reactor trip with 1 stuck-open ADV on SG-A. Failure of both MSIVs (SG-A and SG-B) to close.	Operator does not isolate AFW on affected SG. Normal AFW flow assumed (200 gpm). Operator assumed to throttle HPI if auxiliary feedwater is running with SG wide range level > -84% and RCS subcooling > 25 F. HPI is throttled to maintain pressurizer level between 40 and 60 %.	No	No	No
4	19	Reactor trip with 1 stuck-open ADV on SG-A.	None. Operator does not throttle HPI.	Yes	No	Yes at 60, 200, 500 EFPY
5	22	Turbine/reactor trip with loss of MFW and AFW.	Operator depressurizes through ADVs and feeds SG's using condensate booster pumps. Operators maintain a cooldown rate within technical specification limits and throttle condensate flow at 84 % level in the steam generator.	No	No	No
6	24	Main steam line break with the break assumed to be inside containment causing containment spray actuation.	None	No	No	No
7	26	Main steam line break with the break assumed to be inside containment causing containment spray actuation.	Operator isolates AFW to affected SG at 30 minutes after initiation.	No	No	No

Table H-1	PTS Transient Descriptions for Palisades					
Count	**TH Case #**	**System Failure**	**Operator Action**	**HZP**	**HiK**	**Dominant***
8	27	Main steam line break with controller failure resulting in the flow from two AFW pumps into affected steam generator. Break assumed to be inside containment causing containment spray actuation.	Operator starts second AFW pump.	No	No	No
9	29	Main steam line break with break assumed to be inside containment causing containment spray actuation.	None. Operator does not throttle HPI.	Yes	No	No
10	31	Turbine/reactor trip with failure of MFW and AFW. Containment spray actuation assumed due to PORV discharge.	Operator maintains core cooling by "feed and bleed" using HPI to feed and two PORVs to bleed.	No	No	No
11	32	Turbine/reactor trip with failure of MFW and AFW. Containment spray actuation assumed due to PORV discharge.	Operator maintains core cooling by "feed and bleed" using HPI to feed and two PORV to bleed. AFW is recovered 15 minutes after initiation of "feed and bleed" cooling. Operator closes PORVs when SG level reaches 60 percent.	No	No	No
12	34	Main steam line break concurrent with a single tube failure in SG-A due to MSLB vibration.	Operator isolates AFW to affected SG at 15 minutes after initiation. Operator trips RCPs assuming that they do not trip as a result of the event. Operator assumed to throttle HPI if auxiliary feedwater is running with SG wide range level > -84% and RCS subcooling > 25 F. HPI is throttled to maintain pressurizer level between 40 and 60 %.	No	No	No
13	40	40.64 cm (16 in) hot leg break. Containment sump recirculation included in the analysis.	None. Operator does not throttle HPI.	No	Yes	Yes at 32, 60, 200, 500 EFPY
14	42	Turbine/reactor trip with two stuck open pressurizer SRVs. Containment spray is assumed not to actuate.	Operator assumed to throttle HPI if auxiliary feedwater is running with SG wide range level > -84% and RCS subcooling > 25 F. HPI is throttled to maintain pressurizer level between 40 and 60 %.	No	No	No

Table H-1	PTS Transient Descriptions for Palisades					
Count	**TH Case #**	**System Failure**	**Operator Action**	**HZP**	**HiK**	**Dominant***
15	48	Two stuck-open pressurizer SRVs that reclose at 6000 sec after initiation. Containment spray is assumed not to actuate.	None. Operator does not throttle HPI.	Yes	No	Yes at 32 EFPY
16	49	Main steam line break with the break assumed to be inside containment causing containment spray actuation.	Operator isolates AFW to affected SG at 30 minutes after initiation. Operator does not throttle HPI.	Yes	No	No
17	50	Main steam line break with controller failure resulting in the flow from two AFW pumps into affected steam generator. Break assumed to be inside containment causing containment spray actuation.	Operator starts second AFW pump. Operator does not throttle HPI.	Yes	No	No
18	51	Main steam line break with failure of both MSIVs to close. Break assumed to be inside containment causing containment spray actuation.	Operator does not isolate AFW on affected SG. Operator does not throttle HPI.	Yes	No	No
19	52	Reactor trip with 1 stuck-open ADV on SG-A. Failure of both MSIVs (SG-A and SG-B) to close.	Operator does not isolate AFW on affected SG. Normal AFW flow assumed (200 gpm). Operator does not throttle HPI.	Yes	No	Yes at 500 EFPY
20	53	Turbine/reactor trip with two stuck-open pressurizer SRVs that reclose at 6000 sec after initiation. Containment spray is assumed not to actuate.	None. Operator does not throttle HPI.	No	No	Yes at 500 EFPY
21	54	Main steam line break with failure of both MSIVs to close. Break assumed to be inside containment causing containment spray actuation.	Operator does not isolate AFW on affected SG. Operator does not throttle HPI.	No	No	Yes at 32, 60, 200, 500 EFPY
22	55	Turbine/reactor trip with 2 stuck-open ADVs on SG-A combined with controller failure resulting in the flow from two AFW pumps into affected steam generator.	Operator starts second AFW pump.	No	No	Yes at 32, 60, 200, 500 EFPY
23	58	10.16 cm (4 in) cold leg break. Winter conditions assumed (HPI and LPI injection temp = 40 F, Accumulator temp = 60 F)	None. Operator does not throttle HPI.	No	Yes	Yes at 32, 60, 200, 500 EFPY

Table H-1 PTS Transient Descriptions for Palisades

Count	TH Case #	System Failure	Operator Action	HZP	HiK	Dominant*
24	59	10.16 cm (4 in) cold leg break. Summer conditions assumed (HPI and LPI injection temp = 100 F, Accumulator temp = 90 F)	None. Operator does not throttle HPI.	No	Yes	Yes at 500 EFPY
25	60	5.08 cm (2 in) surge line break. Winter conditions assumed (HPI and LPI injection temp = 40 F, Accumulator temp = 60 F)	None. Operator does not throttle HPI.	No	Yes	Yes at 60, 200, 500 EFPY
26	61	7.18 cm (2.8 in) cold leg break. Summer conditions assumed (HPI and LPI injection temp = 100 F, Accumulator temp = 90 F)	None. Operator does not throttle HPI.	No	Yes	No
27	62	20.32 cm (8 in) cold leg break. Winter conditions assumed (HPI and LPI injection temp = 40 F, Accumulator temp = 60 F)	None. Operator does not throttle HPI.	No	Yes	Yes at 32, 60, 200, 500 EFPY
28	63	14.37 cm (5.656 in) cold leg break. Winter conditions assumed (HPI and LPI injection temp = 40 F, Accumulator temp = 60 F)	None. Operator does not throttle HPI.	No	Yes	Yes at 60, 200, 500 EFPY
29	64	10.16 cm (4 in) surge line break. Summer conditions assumed (HPI and LPI injection temp = 100 F, Accumulator temp = 90 F)	None. Operator does not throttle HPI.	No	Yes	Yes at 32, 60, 200, 500 EFPY
30	65	One stuck-open pressurizer SRV that recloses at 6000 sec after initiation. Containment spray is assumed not to actuate.	None. Operator does not throttle HPI.	Yes	No	Yes at 32, 60, 200, 500 EFPY

Notes:
1. TH ### – Thermal hydraulics run number ###
2. LOCA – Loss-of-coolant accident
3. SBLOCA – Small-break loss-of-coolant accident
4. MBLOCA – Medium-break loss-of-coolant accident
5. LBLOCA – Large-break loss-of-coolant accident
6. HZP – Hot-zero power
7. ADV – Atmospheric dump valve
8. SRV – Safety and relief valve
9. MSLB – Main steam line break
10. AFW – Auxiliary feedwater
11. HPI – High-pressure injection
12. RCP – Reactor coolant pump
13. SG – Steam generator

* The arbitrary definition of a dominant transient is a transient that contributes 1% or more of the total Through-Wall Cracking Failure (TWCF).

APPENDIX I
PALISADES FAVPOST OUTPUT

I-1: 10 Year ISI only

```
***********************************************
*                                             *
*               WELCOME TO FAVOR              *
*                                             *
*   FRACTURE ANALYSIS OF VESSELS: OAK RIDGE   *
*                 VERSION 05.1                *
*                                             *
*     FAVPOST MODULE: POSTPROCESSOR MODULE    *
*  COMBINES TRANSIENT INITIAITING FREQUENCIES *
*        WITH RESULTS OF PFM ANALYSIS         *
*                                             *
*     PROBLEMS OR QUESTIONS REGARDING FAVOR   *
*             SHOULD BE DIRECTED TO           *
*                                             *
*                TERRY DICKSON                *
*         OAK RIDGE NATIONAL LABORATORY       *
*                                             *
*          e-mail: dicksontl@ornl.gov         *
*                                             *
***********************************************

***********************************************************
* This computer program was prepared as an account of     *
* work sponsored by the United States Government          *
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* employees, makes any warranty, expressed or implied, or *
* assumes any legal liability or responsibility for the   *
* accuracy, completeness, or usefulness of any            *
* information, apparatus, product, or process disclosed,  *
* or represents that its use would not infringe           *
* privately-owned rights.                                 *
***********************************************************

DATE: 17-Aug-2005  TIME: 13:36:46

FAVPOST INPUT  FILE NAME                       = postpl.in
FAVPFM  OUTPUT FILE CONTAINING PFMI ARRAY = INITIATE.DAT
FAVPFM  OUTPUT FILE CONTAINING PFMF ARRAY = FAILURE.DAT
FAVPOST OUTPUT FILE NAME                       = 80000.out


   **********************************
   * NUMBER OF SIMULATIONS =  80000 *
   **********************************
```

I-1: 10 Year ISI only (cont.)

TRANSIENT NUMBER	CONDITIONAL PROBABILITY OF INITIATION CPI=P(I|E) MEAN CPI	95th % CPI	99th % CPI	CONDITIONAL PROBABILITY OF FAILURE CPF=P(F|E) MEAN CPF	95th % CPF	99th % CPF	RATIO CPFmn/CPImn
2	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000
16	8.5514E-11	0.0000E+00	0.0000E+00	6.4577E-12	0.0000E+00	0.0000E+00	0.0755
18	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000
19	2.0472E-07	0.0000E+00	0.0000E+00	1.4761E-07	0.0000E+00	0.0000E+00	0.7210
22	1.9203E-10	0.0000E+00	0.0000E+00	3.5754E-11	0.0000E+00	0.0000E+00	0.1862
24	2.1733E-07	0.0000E+00	0.0000E+00	2.0921E-08	0.0000E+00	0.0000E+00	0.0963
26	2.1733E-07	0.0000E+00	0.0000E+00	2.1644E-08	0.0000E+00	0.0000E+00	0.0996
27	6.1144E-06	0.0000E+00	7.3401E-06	1.7443E-06	0.0000E+00	2.0580E-06	0.2853
29	1.2217E-07	0.0000E+00	0.0000E+00	6.6567E-08	0.0000E+00	0.0000E+00	0.5449
31	6.0427E-06	0.0000E+00	7.2749E-06	1.3184E-06	0.0000E+00	1.0400E-06	0.2182
32	1.7253E-07	0.0000E+00	0.0000E+00	1.2111E-07	0.0000E+00	0.0000E+00	0.7020
34	1.0526E-06	0.0000E+00	0.0000E+00	1.6479E-07	0.0000E+00	0.0000E+00	0.1566
40	1.7124E-03	4.0206E-03	2.4764E-02	9.2478E-05	3.1386E-04	1.3383E-03	0.0540
42	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000
48	1.4380E-04	2.7147E-03	2.1555E-03	1.4279E-04	2.7081E-03	2.1525E-03	0.9930
49	6.4972E-08	0.0000E+00	0.0000E+00	7.8356E-09	0.0000E+00	0.0000E+00	0.1206
50	1.3736E-05	3.1047E-04	7.6335E-05	4.6895E-06	0.0000E+00	2.4951E-05	0.3414
51	6.8134E-05	8.5399E-04	9.3788E-04	3.4744E-05	4.4063E-04	5.0756E-04	0.5099
52	2.4938E-07	0.0000E+00	0.0000E+00	1.7929E-07	0.0000E+00	0.0000E+00	0.7189
53	1.0285E-07	0.0000E+00	0.0000E+00	7.8964E-08	0.0000E+00	0.0000E+00	0.7678
54	1.2929E-04	1.2053E-03	2.0543E-03	6.4735E-05	6.6773E-04	1.0681E-03	0.5007
55	5.2116E-07	0.0000E+00	0.0000E+00	4.1702E-07	0.0000E+00	0.0000E+00	0.8002
58	5.3040E-05	1.9353E-03	6.0871E-04	5.6862E-06	2.3038E-04	4.9345E-05	0.1072
59	4.6920E-06	0.0000E+00	3.9194E-06	2.4849E-07	0.0000E+00	1.1407E-07	0.0530
60	1.2442E-05	8.3152E-04	5.6793E-05	1.4351E-06	4.7793E-05	5.6811E-06	0.1153
61	3.3870E-07	0.0000E+00	0.0000E+00	8.0845E-09	0.0000E+00	0.0000E+00	0.0239
62	8.9226E-04	4.2246E-03	1.1498E-02	7.0150E-05	5.1237E-04	7.1432E-04	0.0786
63	3.4598E-04	3.6819E-03	5.8713E-03	3.2526E-05	3.3214E-04	5.0588E-04	0.0940
64	5.1177E-04	2.7766E-03	6.7856E-03	6.4962E-05	4.3520E-04	8.0933E-04	0.1269
65	4.8006E-05	1.6992E-03	4.8467E-04	4.7426E-05	1.6732E-03	4.7726E-04	0.9879

I-1: 10 Year ISI only (cont.)

```
NOTES: CPI IS CONDITIONAL PROBABILITY OF CRACK INITIATION, P(I|E)
       CPF IS CONDITIONAL PROBABILITY OF TWC FAILURE, P(F|E)

    ***********************************************************
    *      PROBABILITY DISTRIBUTION FUNCTION (HISTOGRAM)      *
    *         FOR THE FREQUENCY OF CRACK INITIATION           *
    ***********************************************************
```

FREQUENCY OF CRACK INITIATION (PER REACTOR-OPERATING YEAR)	RELATIVE DENSITY (%)	CUMULATIVE DISTRIBUTION (%)
0.0000E+00	9.5813	9.5813
9.4162E-07	89.5063	99.0875
2.8249E-06	0.5238	99.6113
4.7081E-06	0.1488	99.7600
6.5913E-06	0.0675	99.8275
8.4746E-06	0.0363	99.8638
1.0358E-05	0.0313	99.8950
1.2241E-05	0.0225	99.9175
1.4124E-05	0.0125	99.9300
1.6008E-05	0.0075	99.9375
1.7891E-05	0.0075	99.9450
1.9774E-05	0.0038	99.9487
2.1657E-05	0.0063	99.9550
2.3540E-05	0.0038	99.9587
2.5424E-05	0.0038	99.9625
2.9190E-05	0.0025	99.9650
3.1073E-05	0.0038	99.9687
3.2957E-05	0.0025	99.9712
3.4840E-05	0.0013	99.9725
3.6723E-05	0.0038	99.9762
3.8606E-05	0.0025	99.9787
4.0490E-05	0.0025	99.9812
4.6139E-05	0.0013	99.9825
4.8023E-05	0.0025	99.9850
5.1789E-05	0.0013	99.9862
5.3672E-05	0.0013	99.9875
5.5556E-05	0.0013	99.9887
7.0621E-05	0.0013	99.9900
8.3804E-05	0.0013	99.9912
8.5687E-05	0.0013	99.9925
8.7571E-05	0.0013	99.9937
1.0075E-04	0.0013	99.9950
1.1582E-04	0.0025	99.9975
1.3653E-04	0.0013	99.9987
1.8550E-04	0.0013	100.0000

I-1: 10 Year ISI only (cont.)

```
============================================
==       Summary Descriptive Statistics   ==
============================================

Minimum                              = 0.0000E+00
Maximum                              = 1.8644E-04
Range                                = 1.8644E-04

Number of Simulations                =  80000

5th Percentile                       = 0.0000E+00
Median                               = 1.4422E-09
95.0th Percentile                    = 9.4162E-07
99.0th Percentile                    = 1.7191E-06
99.9th Percentile                    = 1.0776E-05

Mean                                 = 1.1076E-07
Standard Deviation                   = 1.5009E-06
Standard Error                       = 5.3064E-09
Variance (unbiased)                  = 2.2526E-12
Variance (biased)                    = 2.2526E-12
Moment Coeff. of Skewness            = 6.3441E+01
Pearson's 2nd Coeff. of Skewness     = 2.2139E-01
Kurtosis                             = 5.5931E+03

**********************************************************
*      PROBABILITY DISTRIBUTION FUNCTION (HISTOGRAM)     *
*      FOR THROUGH-WALL CRACKING FREQUENCY (FAILURE)     *
**********************************************************
```

FREQUENCY OF TWC FAILURES (PER REACTOR-OPERATING YEAR)	RELATIVE DENSITY (%)	CUMULATIVE DISTRIBUTION (%)
0.0000E+00	17.3700	17.3700
4.9226E-07	82.4238	99.7938
1.4768E-06	0.1012	99.8950
2.4613E-06	0.0375	99.9325
3.4458E-06	0.0187	99.9513
4.4303E-06	0.0113	99.9625
5.4148E-06	0.0100	99.9725
6.3993E-06	0.0050	99.9775
7.3839E-06	0.0025	99.9800
8.3684E-06	0.0025	99.9825
9.3529E-06	0.0025	99.9850
1.0337E-05	0.0013	99.9862
1.1322E-05	0.0013	99.9875
1.2306E-05	0.0025	99.9900
1.4275E-05	0.0025	99.9925
1.7229E-05	0.0013	99.9937
1.8214E-05	0.0013	99.9950
2.4121E-05	0.0013	99.9962
3.3966E-05	0.0013	99.9975

I-1: 10 Year ISI only (cont.)

4.8733E-05	0.0013	99.9987
9.7959E-05	0.0013	100.0000

```
===========================================
==      Summary Descriptive Statistics    ==
===========================================

Minimum                               = 0.0000E+00
Maximum                               = 9.7467E-05
Range                                 = 9.7467E-05

Number of Simulations                 =  80000

5th Percentile                        = 0.0000E+00
Median                                = 2.6070E-11
95.0th Percentile                     = 4.9226E-07
99.0th Percentile                     = 1.9186E-07
99.9th Percentile                     = 1.6080E-06

Mean                                  = 1.5364E-08
Standard Deviation                    = 4.5949E-07
Standard Error                        = 1.6246E-09
Variance (unbiased)                   = 2.1113E-13
Variance (biased)                     = 2.1113E-13
Moment Coeff. of Skewness             = 1.4562E+02
Pearson's 2nd Coeff. of Skewness      = 9.0898E-02
Kurtosis                              = 2.7526E+04
```

```
**********************************************************
*  FRACTIONALIZATION OF FREQUENCY OF CRACK INITIATION    *
*      AND THROUGH-WALL CRACKING FREQUENCY (FAILURE) -   *
*       WEIGHTED BY TRANSIENT INITIATING FREQUENCIES     *
**********************************************************
```

	% of total frequency of crack initiation	% of total frequency of of TWC failure
2	0.00	0.00
16	0.00	0.00
18	0.00	0.00
19	0.16	0.93
22	0.00	0.00
24	0.00	0.00
26	0.11	0.08
27	0.21	0.45
29	0.00	0.00
31	0.09	0.16
32	0.00	0.00
34	0.02	0.02
40	61.35	23.84
42	0.00	0.00
48	0.10	0.74
49	0.00	0.00

I-1: 10 Year ISI only (cont.)

50	0.01	0.02
51	0.01	0.03
52	0.06	0.32
53	0.09	0.48
54	0.81	2.82
55	0.96	5.53
58	13.03	10.90
59	1.10	0.42
60	2.48	2.04
61	0.06	0.01
62	7.31	4.49
63	2.36	1.74
64	3.86	3.40
65	5.84	41.57
TOTALS	100.00	100.00

DATE: 17-Aug-2005 TIME: 13:37:51

I-2: ISI Every 10 Years

```
**************************************************
*                                                *
*                WELCOME TO FAVOR                *
*                                                *
*    FRACTURE ANALYSIS OF VESSELS: OAK RIDGE     *
*                 VERSION 05.1                   *
*                                                *
*      FAVPOST MODULE: POSTPROCESSOR MODULE      *
*  COMBINES TRANSIENT INITIAITING FREQUENCIES    *
*         WITH RESULTS OF PFM ANALYSIS           *
*                                                *
*      PROBLEMS OR QUESTIONS REGARDING FAVOR     *
*              SHOULD BE DIRECTED TO             *
*                                                *
*                 TERRY DICKSON                  *
*         OAK RIDGE NATIONAL LABORATORY          *
*                                                *
*           e-mail: dicksontl@ornl.gov           *
*                                                *
**************************************************

************************************************************
* This computer program was prepared as an account of     *
* work sponsored by the United States Government          *
* Neither the United States, nor the United States        *
* Department of Energy, nor the United States Nuclear     *
* Regulatory Commission, nor any of their employees,      *
* nor any of their contractors, subcontractors, or their  *
* employees, makes any warranty, expressed or implied, or *
* assumes any legal liability or responsibility for the   *
* accuracy, completeness, or usefulness of any            *
* information, apparatus, product, or process disclosed,  *
* or represents that its use would not infringe           *
* privately-owned rights.                                 *
************************************************************


DATE: 17-Aug-2005  TIME: 13:48:45

FAVPOST INPUT  FILE NAME                     = postpl.in
FAVPFM  OUTPUT FILE CONTAINING PFMI ARRAY = INITIATE.DAT
FAVPFM  OUTPUT FILE CONTAINING PFMF ARRAY = FAILURE.DAT
FAVPOST OUTPUT FILE NAME                     = 80000.out


   **********************************
   * NUMBER OF SIMULATIONS =   80000 *
   **********************************
```

I-2: ISI Every 10 Years (cont.)

TRANSIENT NUMBER	CONDITIONAL PROBABILITY OF INITIATION CPI=P(I|E) MEAN CPI	95th % CPI	99th % CPI	CONDITIONAL PROBABILITY OF FAILURE CPF=P(F|E) MEAN CPF	95th % CPF	99th % CPF	RATIO CPFmn/CPImn
2	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000
16	7.7445E-13	0.0000E+00	0.0000E+00	4.0453E-14	0.0000E+00	0.0000E+00	0.0522
18	1.5330E-12	0.0000E+00	0.0000E+00	1.3949E-12	0.0000E+00	0.0000E+00	0.9099
19	9.5976E-08	0.0000E+00	0.0000E+00	7.0366E-08	0.0000E+00	0.0000E+00	0.7332
22	1.6973E-12	0.0000E+00	0.0000E+00	3.0494E-13	0.0000E+00	0.0000E+00	0.1797
24	1.8918E-07	0.0000E+00	0.0000E+00	1.9286E-08	0.0000E+00	0.0000E+00	0.1019
26	1.8918E-07	0.0000E+00	0.0000E+00	2.1409E-08	0.0000E+00	0.0000E+00	0.1132
27	6.0143E-06	0.0000E+00	1.2200E-05	1.9166E-06	0.0000E+00	3.8500E-06	0.3187
29	1.4942E-07	0.0000E+00	0.0000E+00	1.1331E-07	0.0000E+00	0.0000E+00	0.7584
31	5.6325E-06	0.0000E+00	9.9944E-06	1.5013E-06	0.0000E+00	1.8825E-06	0.2665
32	6.3302E-08	0.0000E+00	0.0000E+00	4.5718E-08	0.0000E+00	0.0000E+00	0.7222
34	6.9279E-07	0.0000E+00	0.0000E+00	1.1530E-07	0.0000E+00	0.0000E+00	0.1664
40	1.7563E-03	3.5046E-03	2.6632E-02	1.0662E-04	5.3379E-04	1.4595E-03	0.0607
42	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000
48	1.6777E-04	1.9595E-03	2.4975E-03	1.6654E-04	1.9556E-03	2.4899E-03	0.9927
49	4.2963E-08	0.0000E+00	0.0000E+00	5.9375E-09	0.0000E+00	0.0000E+00	0.1382
50	1.4198E-05	2.3876E-04	9.8468E-05	5.3103E-06	1.0187E-04	3.5054E-05	0.3740
51	7.3994E-05	6.9359E-04	1.0986E-03	3.9840E-05	4.5202E-04	6.2532E-04	0.5384
52	1.2878E-07	0.0000E+00	0.0000E+00	9.6447E-08	0.0000E+00	0.0000E+00	0.7489
53	3.0031E-08	0.0000E+00	0.0000E+00	2.1896E-08	0.0000E+00	0.0000E+00	0.7291
54	1.4119E-04	9.8601E-04	1.6965E-03	7.2979E-05	6.2722E-04	6.3054E-04	0.5169
55	2.7149E-07	0.0000E+00	0.0000E+00	2.2231E-07	0.0000E+00	0.0000E+00	0.8188
58	4.6551E-05	4.7271E-04	6.0087E-04	6.1176E-06	1.6444E-04	5.9645E-05	0.1314
59	4.0810E-06	0.0000E+00	6.0089E-06	3.3603E-07	0.0000E+00	1.9652E-07	0.0823
60	1.0945E-05	1.8544E-04	6.6125E-05	1.6980E-06	6.5520E-05	7.6249E-06	0.1551
61	2.2645E-07	0.0000E+00	0.0000E+00	1.0152E-08	0.0000E+00	0.0000E+00	0.0448
62	9.1091E-04	2.8994E-03	1.3298E-02	8.3097E-05	7.1500E-04	7.1829E-04	0.0912
63	3.3398E-04	1.5885E-03	4.5267E-03	3.5907E-05	5.1263E-04	5.9140E-04	0.1075
64	5.3039E-04	1.6958E-03	7.6957E-03	7.4704E-05	5.1773E-04	9.3671E-04	0.1408
65	5.4910E-05	1.0905E-03	5.8651E-04	5.4180E-05	1.0833E-03	5.8057E-04	0.9867

I-2: ISI Every 10 Years (cont.)

NOTES: CPI IS CONDITIONAL PROBABILITY OF CRACK INITIATION, P(I|E)
CPF IS CONDITIONAL PROBABILITY OF TWC FAILURE, P(F|E)

PROBABILITY DISTRIBUTION FUNCTION (HISTOGRAM)
FOR THE FREQUENCY OF CRACK INITIATION

FREQUENCY OF CRACK INITIATION (PER REACTOR-OPERATING YEAR)	RELATIVE DENSITY (%)	CUMULATIVE DISTRIBUTION (%)
0.0000E+00	9.4225	9.4225
1.0787E-06	89.7512	99.1737
3.2360E-06	0.4288	99.6025
5.3934E-06	0.1513	99.7537
7.5507E-06	0.0838	99.8375
9.7081E-06	0.0425	99.8800
1.1865E-05	0.0187	99.8987
1.4023E-05	0.0225	99.9212
1.6180E-05	0.0150	99.9362
1.8338E-05	0.0075	99.9437
2.0495E-05	0.0125	99.9562
2.2652E-05	0.0087	99.9650
2.4810E-05	0.0038	99.9687
2.6967E-05	0.0038	99.9725
2.9124E-05	0.0013	99.9737
3.1282E-05	0.0025	99.9762
3.3439E-05	0.0013	99.9775
3.5596E-05	0.0025	99.9800
4.2068E-05	0.0013	99.9812
4.4226E-05	0.0025	99.9837
4.6383E-05	0.0025	99.9862
4.8541E-05	0.0013	99.9875
5.0698E-05	0.0013	99.9887
5.2855E-05	0.0013	99.9900
5.7170E-05	0.0025	99.9925
6.3642E-05	0.0013	99.9937
6.5799E-05	0.0013	99.9950
8.3058E-05	0.0013	99.9962
1.0247E-04	0.0013	99.9975
1.4346E-04	0.0013	99.9987
2.1466E-04	0.0013	100.0000

Summary Descriptive Statistics

Minimum	= 0.0000E+00
Maximum	= 2.1358E-04
Range	= 2.1358E-04

I-2: ISI Every 10 Years (cont.)

Number of Simulations	= 80000
5th Percentile	= 0.0000E+00
Median	= 1.4375E-09
95.0th Percentile	= 1.0787E-06
99.0th Percentile	= 1.8108E-06
99.9th Percentile	= 1.1985E-05
Mean	= 1.1279E-07
Standard Deviation	= 1.4163E-06
Standard Error	= 5.0072E-09
Variance (unbiased)	= 2.0058E-12
Variance (biased)	= 2.0057E-12
Moment Coeff. of Skewness	= 7.3721E+01
Pearson's 2nd Coeff. of Skewness	= 2.3892E-01
Kurtosis	= 8.5766E+03

```
**********************************************************
*      PROBABILITY DISTRIBUTION FUNCTION (HISTOGRAM)      *
*      FOR THROUGH-WALL CRACKING FREQUENCY (FAILURE)      *
**********************************************************
```

FREQUENCY OF TWC FAILURES (PER REACTOR-OPERATING YEAR)	RELATIVE DENSITY (%)	CUMULATIVE DISTRIBUTION (%)
0.0000E+00	14.5263	14.5263
3.0683E-07	85.0700	99.5963
9.2049E-07	0.2087	99.8050
1.5342E-06	0.0675	99.8725
2.1478E-06	0.0263	99.8988
2.7615E-06	0.0250	99.9238
3.3751E-06	0.0138	99.9375
3.9888E-06	0.0087	99.9463
4.6025E-06	0.0087	99.9550
5.2161E-06	0.0075	99.9625
5.8298E-06	0.0075	99.9700
6.4434E-06	0.0038	99.9738
7.0571E-06	0.0038	99.9775
7.6708E-06	0.0013	99.9788
8.8981E-06	0.0025	99.9813
9.5117E-06	0.0025	99.9838
1.0739E-05	0.0025	99.9862
1.3194E-05	0.0013	99.9875
1.3807E-05	0.0013	99.9887
1.5648E-05	0.0013	99.9900
1.7489E-05	0.0038	99.9937
1.9330E-05	0.0013	99.9950
2.1785E-05	0.0013	99.9962
2.3626E-05	0.0013	99.9975
2.9763E-05	0.0013	99.9987

I-2: ISI Every 10 Years (cont.)

```
6.0446E-05          0.0013      100.0000


==============================================
==      Summary Descriptive Statistics      ==
==============================================

Minimum                              = 0.0000E+00
Maximum                              = 6.0752E-05
Range                                = 6.0752E-05

Number of Simulations                =  80000

5th Percentile                       = 0.0000E+00
Median                               = 3.9848E-11
95.0th Percentile                    = 3.0683E-07
99.0th Percentile                    = 2.1221E-07
99.9th Percentile                    = 2.1785E-06

Mean                                 = 1.6655E-08
Standard Deviation                   = 3.4892E-07
Standard Error                       = 1.2336E-09
Variance (unbiased)                  = 1.2175E-13
Variance (biased)                    = 1.2174E-13
Moment Coeff. of Skewness            = 9.3708E+01
Pearson's 2nd Coeff. of Skewness     = 1.3084E-01
Kurtosis                             = 1.3132E+04


************************************************************
*  FRACTIONALIZATION OF FREQUENCY OF CRACK INITIATION      *
*      AND THROUGH-WALL CRACKING FREQUENCY (FAILURE) -     *
*       WEIGHTED BY TRANSIENT INITIATING FREQUENCIES       *
************************************************************
```

	% of total frequency of crack initiation	% of total frequency of of TWC failure
2	0.00	0.00
16	0.00	0.00
18	0.00	0.00
19	0.17	0.90
22	0.00	0.00
24	0.00	0.00
26	0.11	0.08
27	0.24	0.50
29	0.00	0.00
31	0.08	0.13
32	0.00	0.00
34	0.01	0.01
40	63.92	27.54
42	0.00	0.00
48	0.12	0.77

I-2: ISI Every 10 Years (cont.)

49	0.00	0.00
50	0.01	0.02
51	0.01	0.03
52	0.16	0.91
53	0.04	0.18
54	0.68	2.28
55	0.82	4.53
58	11.29	9.77
59	0.75	0.41
60	2.05	2.01
61	0.04	0.01
62	7.00	3.98
63	2.29	1.63
64	4.15	3.76
65	6.07	40.54
TOTALS	100.00	100.00

DATE: 17-Aug-2005 TIME: 13:49:41

APPENDIX J
INPUTS FOR THE OCONEE UNIT 1 PILOT PLANT EVALUATION

A summary of the NDE inspection history based on Regulatory Guide 1.150 and pertinent input data for OC1 is as follows:

1. Number of inservice inspections performed (relative to initial pre-service and 10 year interval inspections) for full penetration category B-A, B-D, and B-J vessel welds assuming all of the candidate welds were inspected: 3 (covering all welds of the specified categories).

2. The inspections performed covered: 62 total examinations. 23 items with 100% coverage, 22 items with < 90% coverage and 17 items with coverage >90% but less than 100%.

3. Number of indications found during most recent inservice inspection: 44
This number includes consideration of the following additional information.

   a. Indications found that were reportable: 0
   b. Indications found that were within acceptable limits: 44
   c. Indications/anomalies currently being monitored: 0

4. Full Penetration Relief requests for the reactor vessel submitted and accepted by the NRC: 2 relief requests for limited coverage for 22 items, as noted in item 2

5. Fluence distribution at inside surface of RV Beltline until end of life is shown in: see Figure J-1 taken from the NRC PTS Risk Study [7], Figure 4.1.

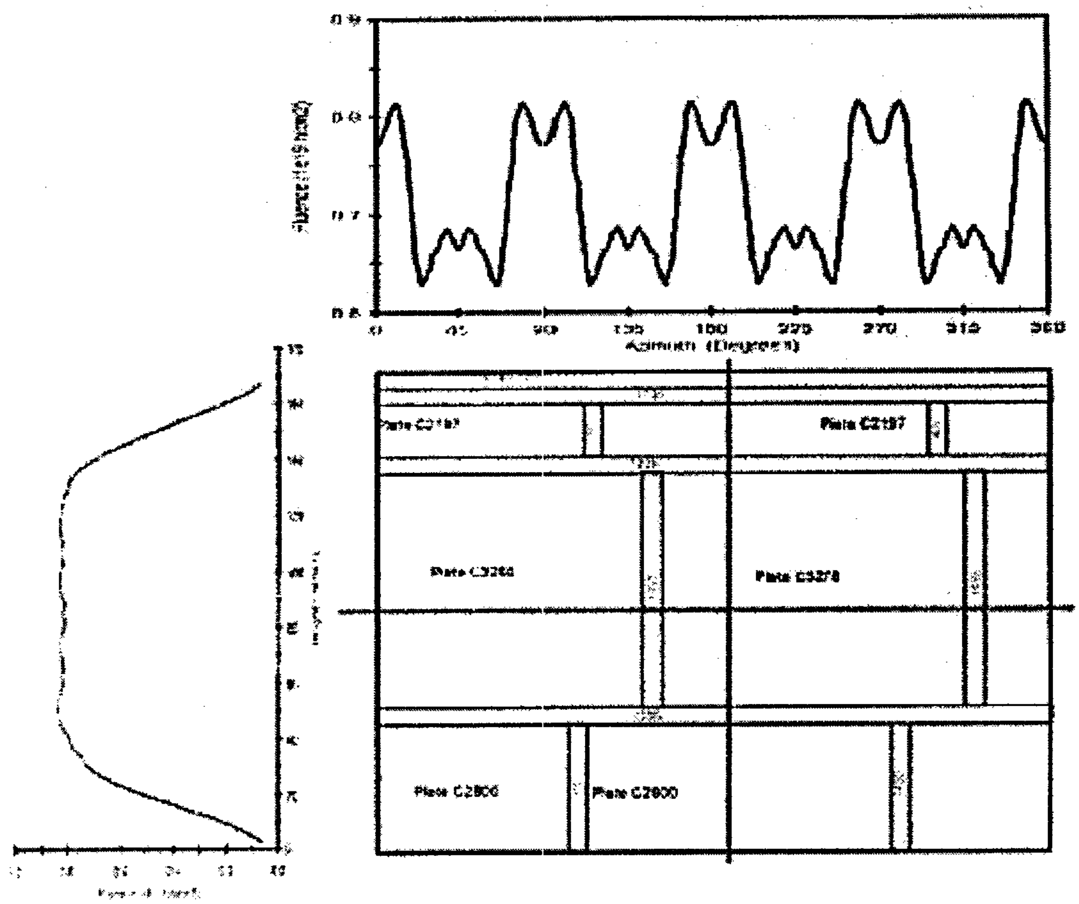

Figure J-1 Rollout Diagram of Beltline Materials and Representative Fluence Maps for OC1

6. Vessel cladding details (Table 4.2 of Reference 7):
   a. Number of layers: 1
   b. Thickness: 0.188
   c. Material properties (assumed to be independent of temperature):
      i. Thermal conductivity (BTU/HR-FT-F), K = 10.0
      ii. Specific Heat (BTU/LBM-F), C = 0.120
      iii. Density (LBM/FT^3), ρ = 489.00
      iv. Young's Modulus of Elasticity (KSI), E = 22800
      v. Thermal Expansion Coefficient (F^{-1}), α = 0.00000945
      vi. Poisson's Ratio, ν = 0.3
   d. Material including copper and nickel content: Material properties assigned to clad flaws are that of the underlying material be it base or weld. This is consistent with the PTS evaluation (Reference 7).
   e. Material property uncertainties:
      i. Bead width: 1 inch – bead widths vary for all plants. Based on Reference 7 a nominal dimension of 1 inch is selected for all analyses because this parameter is not expected to significantly influence the predicted vessel failure probabilities.

ii. Truncation Limit: Cladding thickness rounded up to the next 1/100th of the total vessel thickness to be consistent with PTS evaluation.
iii. Surface flaw depth: 0.03 x 8.626 = 0.259 in

7. Base metal (Reference 7):
   a. Wall thickness: 8.438 inches
   b. Material properties (assumed to be independent of temperature):
      i. Thermal conductivity (BTU/HR-FT-F), K = 24.0
      ii. Specific Heat (BTU/LBM-F), C = 0.120
      iii. Density (LBM/FT^3), ρ = 489.00
      iv. Young's Modulus of Elasticity (KSI), E = 28000
      v. Thermal Expansion Coefficient (F^{-1}), α = 0.00000777
      vi. Poisson's Ratio, v = 0.3
      i. Other material properties are identified in Table J-1.

Table J-1 OC1-Specific Material Values Drawn from the RVID (see Ref. 7 Table 4.1)

Major Material Region Description				Cu [wt%]	Ni [wt%]	P [wt%]	Un-Irradiated RT_{NDT}		RT_{PTS} @60 EFPY
#	Type	Heat	Location				[°F]	Method	
1	Axial Weld	SA-1430	Lower	0.190	0.570	0.017	-5	B&W Generic	
2	Axial Weld	SA-1493	Intermediate	0.190	0.570	0.017	-5	B&W Generic	
3	Axial Weld	SA-1073	Upper	0.210	0.640	0.025	-5	B&W Generic	
4	Circ Weld	SA-1585	Lower	0.220	0.540	0.016	-5	B&W Generic	
5	Circ Weld	SA-1229	Intermediate	0.230	0.590	0.021	10	ASME NB-2331	
6	Circ Weld	SA-1135	Upper	0.230	0.520	0.011	- 5	B&W Generic	
7	Plate	C-2800	Lower	0.110	0.630	0.012	1	B&W Generic	
8	Plate	C3265-1	Intermediate	0.100	0.500	0.015	1	B&W Generic	
9	Plate	C3278-1	Intermediate	0.120	0.600	0.010	1	B&W Generic	
10	Plate	C2197-2	Upper	0.150	0.500	0.008	1	B&W Generic	
11	Forging	ZV2861	Upper	0.160	0.650	0.006	3	B&W Generic	

8. Weld metal details: Details of information used in addressing weld-specific information are taken directly from the NRC PTS Risk Study [7], Table 4.2. Summaries are reproduced as Table J-2.

Table J-2 Summary of Reactor Vessel-Specific Inputs for Flaw Distribution

Variable		Oconee	Beaver Valley	Palisades	Calvert Cliffs	Notes
Inner Radius (to cladding)	[in]	85.5	78.5	86	86	Vessel specific info
Base Metal Thickness	[in]	8.438	7.875	8.5	8.675	Vessel specific info
Total Wall Thickness	[in]	8.626	8.031	8.75	8.988	Vessel specific info

Variable			Oconee	Beaver Valley	Palisades	Calvert Cliffs	Notes
SAW Weld	Volume fraction	[%]	97%				100% - SMAW% - REPAIR%
	Thru-Wall Bead Thickness	[in]	0.1875	0.1875	0.1875	0.1875	All plants report plant specific dimensions of 3/16-in.
	Truncation Limit	[in]	1				Judgment. Approx. 2X the size of the largest non-repair flaw observed in PVRUF & Shoreham.
	Buried or Surface	--	All flaws are buried				Observation
	Orientation	--	Circ flaws in circ welds, axial flaws in axial welds.				Observation: Virtually all of the weld flaws in PVRUF & Shoreham were aligned with the welding direction because they were lack of sidewall fusion defects.
	Density basis	--	Shoreham density				Highest of observations
	Aspect ratio basis	--	Shoreham & PVRUF observations				Statistically similar distributions from Shoreham and PVRUF were combined to provide more robust estimates, when based on judgment the amount data were limited and/or insufficient to identify different trends for aspect ratios for flaws in the two vessels.
	Depth basis	--	Shoreham & PVRUF observations				Statistically similar distributions combined to provide more robust estimates

Table J-2 Summary of Reactor Vessel-Specific Inputs for Flaw Distribution (cont.)

	Variable		Oconee	Beaver Valley	Palisades	Calvert Cliffs	Notes
SMAW Weld	Volume fraction	[%]	1%				Upper bound to all plant specific info provided by Steve Byrne (Westinghouse – Windsor).
	Thru-Wall Bead Thickness	[in]	0.21	0.20	0.22	0.25	Oconee is generic value based on average of all plants specific values (including Shoreham & PVRUF data). Other values are plant specific as reported by Steve Byrne.
	Truncation Limit	[in]	1				Judgment. Approx. 2X the size of the largest non-repair flaw observed in PVRUF & Shoreham.
	Buried or Surface	--	All flaws are buried				Observation
	Orientation	--	Circ flaws in circ welds, axial flaws in axial welds.				Observation: Virtually all of the weld flaws in PVRUF & Shoreham were aligned with the welding direction because they were lack of sidewall fusion defects.
	Density basis	--	Shoreham density				Highest of observations
	Aspect ratio basis	--	Shoreham & PVRUF observations				Statistically similar distributions from Shoreham and PVRUF were combined to provide more robust estimates, when based on judgment the amount data were limited and/or insufficient to identify different trends for aspect ratios for flaws in the two vessels.
	Depth basis	--	Shoreham & PVRUF observations				Statistically similar distributions combined to provide more robust estimates

Table J-2 Summary of Reactor Vessel-Specific Inputs for Flaw Distribution (cont.)

Variable			Oconee	Beaver Valley	Palisades	Calvert Cliffs	Notes
Repair Weld	Volume fraction	[%]	2%				Judgment. A rounded integral percentage that exceeds the repaired volume observed for Shoreham and for PVRUF, which was 1.5%.
	Thru-Wall Bead Thickness	[in]	0.14				Generic value: As observed in PVRUF and Shoreham by PNNL.
	Truncation Limit	[in]	2				Judgment. Approx. 2X the largest repair flaw found in PVRUF & Shoreham. Also based on maximum expected width of repair cavity.
	Buried or Surface	--	All flaws are buried				Observation
	Orientation	--	Circ flaws in circ welds, axial flaws in axial welds.				The repair flaws had complex shapes and orientations that were not aligned with either the axial or circumferential welds; for consistency with the available treatments of flaws by the FAVOR code, a common treatment of orientations was adopted for flaws in SAW/SMAW and repair welds.
	Density basis	--	Shoreham density				Highest of observations
	Aspect ratio basis	--	Shoreham & PVRUF observations				Statistically similar distributions from Shoreham and PVRUF were combined to provide more robust estimates, when based on judgment the amount data were limited and/or insufficient to identify different trends for aspect ratios for flaws in the two vessels.
	Depth basis	--	Shoreham & PVRUF observations				Statistically similar distributions combined to provide more robust estimates

Table J-2 Summary of Reactor Vessel-Specific Inputs for Flaw Distribution (cont.)

Variable			Oconee	Beaver Valley	Palisades	Calvert Cliffs	Notes
Cladding	Actual Thickness	[in]	0.188	0.156	0.25	0.313	Vessel specific info
	# of Layers	[#]	1	2	2	2	Vessel specific info
	Bead Width	[in]	1				Bead widths of 1 to 5-in. characteristic of machine deposited cladding. Bead widths down to ½-in. can occur over welds. Nominal dimension of 1-in. selected for all analyses because this parameter is not expected to influence significantly the predicted vessel failure probabilities. May need to refine this estimate later, particularly for Oconee who reported a 5-in bead width.
	Truncation Limit	[in]	Actual clad thickness rounded to the nearest 1/100th of the total vessel wall thickness				Judgment & computational convenience
	Surface flaw depth in FAVOR	[in]	0.259	0.161	0.263	0.360	
	Buried or Surface	--	All flaws are surface breaking				Judgment. Only flaws in cladding that would influence brittle fracture of the vessel are brittle. Material properties assigned to clad flaws are that of the underlying material, be it base or weld.
	Orientation	--	All circumferential.				Observation: All flaws observed in PVRUF & Shoreham were lack of inter-run fusion defects, and cladding is always deposited circumferentially
	Density basis	--	No surface flaws observed. Density is 1/1000 that of the observed buried flaws in cladding of vessels examined by PNNL. If there is more than one clad layer then there are no clad flaws.				Judgment
	Aspect ratio basis	--	Observations on buried flaws				Judgment
	Depth basis	--	Depth of all surface flaws is the actual clad thickness rounded up to the nearest 1/100th of the total vessel wall thickness.				Judgment.

Table J-2 Summary of Reactor Vessel-Specific Inputs for Flaw Distribution (cont.)

Variable			Oconee	Beaver Valley	Palisades	Calvert Cliffs	Notes
Plate	Truncation Limit	[in]	0.433				Judgment. Twice the depth of the largest flaw observed in all PNNL plate inspections.
	Buried or Surface	--	All flaws are buried				Observation
	Orientation	--	Half of the simulated flaws are circumferential, half are axial.				Observation & Physics: No observed orientation preference, and no reason to suspect one (other than laminations which are benign.
	Density basis	--	1/10 of small weld flaw density, 1/40 of large weld flaw density of the PVRUF data				Judgment. Supported by limited data.
	Aspect ratio basis	--	Same as for PVRUF welds				Judgment
	Depth basis	--	Same as for PVRUF welds				Judgment. Supported by limited data.

9. TWCF calculated at 500 EFPY using correlation from Reference 27: 7.18E-09 Events per year

APPENDIX K
OCONEE UNIT 1 PROBSBFD OUTPUT

K-1: 10 Year ISI only

```
                    STRUCTURAL RELIABILITY AND RISK ASSESSMENT (SRRA)
WESTINGHOUSE           MONTE-CARLO SIMULATION PROGRAM PROBSBFD    VERSION 1.0

=============================================================================
     INPUT VARIABLES FOR CASE  3: OC1 10 YEAR ISI ONLY

     NCYCLE =    80            NFAILS =  1001           NTRIAL =  1000
     NOVARS =    19            NUMSET =     2           NUMISI =     5
     NUMSSC =     4            NUMTRC =     4           NUMFMD =     4

   VARIABLE       DISTRIBUTION       MEDIAN       DEVIATION     SHIFT    USAGE
  NO.   NAME      TYPE       LOG      VALUE       OR FACTOR     MV/SD   NO. SUB

   1  FIFDepth    - CONSTANT -     3.0000D-02                            1  SET
   2  IFlawDen    - CONSTANT -     3.6589D-03                            2  SET
   3  ICy-ISI     - CONSTANT -     1.0000D+01                            1  ISI
   4  DCy-ISI     - CONSTANT -     8.0000D+01                            2  ISI
   5  MV-Depth    - CONSTANT -     1.5000D-02                            3  ISI
   6  SD-Depth    - CONSTANT -     1.8500D-01                            4  ISI
   7  CEff-ISI    - CONSTANT -     1.0000D+00                            5  ISI
   8  Aspect1     - CONSTANT -     2.0000D+00                            1  SSC
   9  Aspect2     - CONSTANT -     6.0000D+00                            2  SSC
  10  Aspect3     - CONSTANT -     1.0000D+01                            3  SSC
  11  Aspect4     - CONSTANT -     9.9000D+01                            4  SSC
  12  NoTr/Cy     - CONSTANT -     1.2000D+01                            1  TRC
  13  FCGThld     - CONSTANT -     1.5000D+00                            2  TRC
  14  FCGR-UC     NORMAL    NO     0.0000D+00   1.0000D+00     .00       3  TRC
  15  DKINFile    - CONSTANT -     1.0000D+00                            4  TRC
  16  Percent1    - CONSTANT -     6.7450D+01                            1  FMD
  17  Percent2    - CONSTANT -     2.0769D+01                            2  FMD
  18  Percent3    - CONSTANT -     3.9642D+00                            3  FMD
  19  Percent4    - CONSTANT -     7.8166D+00                            4  FMD

 INFORMATION GENERATED FROM FAVLOADS.DAT FILE
 AND SAVED IN DKINSAVE.DAT FILE:

 WALL THICKNESS =  8.6260 INCH

 FLAW DEPTH  MINIMUM K AND MAXIMUM K FOR

   TYPE 1 WITH AN ASPECT RATIO OF   2.

   8.62600D-02    2.26895D+00    1.06757D+01
   1.58718D-01    3.02106D+00    1.44232D+01
   4.31300D-01    1.30893D+01    2.08943D+01
   6.46950D-01    1.39096D+01    2.49826D+01
   8.62600D-01    1.44263D+01    2.80058D+01
   1.72520D+00    1.30110D+01    3.31903D+01
   2.58780D+00    7.51977D+00    3.23837D+01
   4.31300D+00   -2.67288D+00    3.20852D+01

   TYPE 2 WITH AN ASPECT RATIO OF   6.
```

K-1: 10 Year ISI only (cont.)

8.62600D-02	3.40901D+00	1.61172D+01
1.58718D-01	4.63620D+00	2.21942D+01
4.31300D-01	1.99455D+01	3.13897D+01
6.46950D-01	2.33230D+01	3.76625D+01
8.62600D-01	2.45197D+01	4.30412D+01
1.72520D+00	2.46021D+01	5.46183D+01
2.58780D+00	1.95704D+01	5.81373D+01
4.31300D+00	8.31986D+00	6.38027D+01

TYPE 3 WITH AN ASPECT RATIO OF 10.

8.62600D-02	3.73472D+00	1.76698D+01
1.58718D-01	4.95671D+00	2.37364D+01
4.31300D-01	2.11257D+01	3.35265D+01
6.46950D-01	2.53490D+01	4.01563D+01
8.62600D-01	2.66367D+01	4.59818D+01
1.72520D+00	2.73025D+01	5.94651D+01
2.58780D+00	2.36720D+01	6.65485D+01
4.31300D+00	1.21426D+01	7.64376D+01

TYPE 4 WITH AN ASPECT RATIO OF 99.

8.62600D-02	6.74437D+00	1.82354D+01
1.72520D-01	9.55233D+00	2.55450D+01
2.58780D-01	1.62039D+01	2.74271D+01
4.31300D-01	2.37153D+01	3.58624D+01
6.46950D-01	2.70360D+01	4.44287D+01
8.62600D-01	2.84566D+01	5.07281D+01
1.72520D+00	3.19293D+01	6.96665D+01
2.58780D+00	2.97815D+01	8.22041D+01

AVERAGE CALCULATED VALUES FOR: Surface Flaw Density with FCG and ISI

NUMBER FAILED = 0 NUMBER OF TRIALS = 1000

DEPTH (WALL/400) AND FLAW DENSITY FOR ASPECT RATIOS OF 2, 6, 10 AND 99

12	2.2380D-04	1.0377D-05	1.4547D-06	1.1205D-05
13	6.5980D-06	4.0083D-05	7.1947D-06	1.1813D-05
14	0.0000D+00	1.2906D-05	2.8652D-06	2.3081D-06
15	0.0000D+00	3.4523D-06	1.0131D-06	4.5211D-07
16	0.0000D+00	1.1683D-06	2.9704D-07	2.7150D-07
17	0.0000D+00	5.0981D-07	1.5720D-07	1.2084D-07
18	0.0000D+00	3.1177D-07	3.5675D-08	7.1479D-08
19	0.0000D+00	1.2295D-07	5.8386D-08	0.0000D+00
20	0.0000D+00	0.0000D+00	2.2976D-08	0.0000D+00
22	0.0000D+00	5.7099D-08	0.0000D+00	0.0000D+00
24	0.0000D+00	0.0000D+00	0.0000D+00	2.2058D-08
25	0.0000D+00	5.4884D-08	1.0551D-08	0.0000D+00
28	0.0000D+00	0.0000D+00	1.0078D-08	2.1150D-08

K-2: ISI Every 10 Years

```
                STRUCTURAL RELIABILITY AND RISK ASSESSMENT (SRRA)
WESTINGHOUSE        MONTE-CARLO SIMULATION PROGRAM PROBSBFD    VERSION 1.0

==============================================================================
      INPUT VARIABLES FOR CASE  2: OC1 10 YEAR INTERVAL

      NCYCLE =    80           NFAILS =  1001          NTRIAL =  1000
      NOVARS =    19           NUMSET =     2          NUMISI =     5
      NUMSSC =     4           NUMTRC =     4          NUMFMD =     4

    VARIABLE        DISTRIBUTION      MEDIAN      DEVIATION    SHIFT    USAGE
   NO.   NAME       TYPE      LOG      VALUE      OR FACTOR    MV/SD   NO. SUB

    1  FIFDepth    - CONSTANT -    3.0000D-02                           1  SET
    2  IFlawDen    - CONSTANT -    3.6589D-03                           2  SET
    3  ICy-ISI     - CONSTANT -    1.0000D+01                           1  ISI
    4  DCy-ISI     - CONSTANT -    1.0000D+01                           2  ISI
    5  MV-Depth    - CONSTANT -    1.5000D-02                           3  ISI
    6  SD-Depth    - CONSTANT -    1.8500D-01                           4  ISI
    7  CEff-ISI    - CONSTANT -    1.0000D+00                           5  ISI
    8  Aspect1     - CONSTANT -    2.0000D+00                           1  SSC
    9  Aspect2     - CONSTANT -    6.0000D+00                           2  SSC
   10  Aspect3     - CONSTANT -    1.0000D+01                           3  SSC
   11  Aspect4     - CONSTANT -    9.9000D+01                           4  SSC
   12  NoTr/Cy     - CONSTANT -    1.2000D+01                           1  TRC
   13  FCGThld     - CONSTANT -    1.5000D+00                           2  TRC
   14  FCGR-UC     NORMAL    NO    0.0000D+00    1.0000D+00      .00    3  TRC
   15  DKINFile    - CONSTANT -    1.0000D+00                           4  TRC
   16  Percent1    - CONSTANT -    6.7450D+01                           1  FMD
   17  Percent2    - CONSTANT -    2.0769D+01                           2  FMD
   18  Percent3    - CONSTANT -    3.9642D+00                           3  FMD
   19  Percent4    - CONSTANT -    7.8166D+00                           4  FMD

  INFORMATION GENERATED FROM FAVLOADS.DAT FILE
  AND SAVED IN DKINSAVE.DAT FILE:

  WALL THICKNESS =  8.6260 INCH

  FLAW DEPTH  MINIMUM K AND MAXIMUM K FOR

    TYPE 1 WITH AN ASPECT RATIO OF   2.

    8.62600D-02    2.26895D+00    1.06757D+01
    1.58718D-01    3.02106D+00    1.44232D+01
    4.31300D-01    1.30893D+01    2.08943D+01
    6.46950D-01    1.39096D+01    2.49826D+01
    8.62600D-01    1.44263D+01    2.80058D+01
    1.72520D+00    1.30110D+01    3.31903D+01
    2.58780D+00    7.51977D+00    3.23837D+01
    4.31300D+00   -2.67288D+00    3.20852D+01

    TYPE 2 WITH AN ASPECT RATIO OF   6.
```

K-2: ISI Every 10 Years (cont.)

8.62600D-02	3.40901D+00	1.61172D+01
1.58718D-01	4.63620D+00	2.21942D+01
4.31300D-01	1.99455D+01	3.13897D+01
6.46950D-01	2.33230D+01	3.76625D+01
8.62600D-01	2.45197D+01	4.30412D+01
1.72520D+00	2.46021D+01	5.46183D+01
2.58780D+00	1.95704D+01	5.81373D+01
4.31300D+00	8.31986D+00	6.38027D+01

TYPE 3 WITH AN ASPECT RATIO OF 10.

8.62600D-02	3.73472D+00	1.76698D+01
1.58718D-01	4.95671D+00	2.37364D+01
4.31300D-01	2.11257D+01	3.35265D+01
6.46950D-01	2.53490D+01	4.01563D+01
8.62600D-01	2.66367D+01	4.59818D+01
1.72520D+00	2.73025D+01	5.94651D+01
2.58780D+00	2.36720D+01	6.65485D+01
4.31300D+00	1.21426D+01	7.64376D+01

TYPE 4 WITH AN ASPECT RATIO OF 99.

8.62600D-02	6.74437D+00	1.82354D+01
1.72520D-01	9.55233D+00	2.55450D+01
2.58780D-01	1.62039D+01	2.74271D+01
4.31300D-01	2.37153D+01	3.58624D+01
6.46950D-01	2.70360D+01	4.44287D+01
8.62600D-01	2.84566D+01	5.07281D+01
1.72520D+00	3.19293D+01	6.96665D+01
2.58780D+00	2.97815D+01	8.22041D+01

AVERAGE CALCULATED VALUES FOR: Surface Flaw Density with FCG and ISI

NUMBER FAILED = 0 NUMBER OF TRIALS = 1000

DEPTH (WALL/400) AND FLAW DENSITY FOR ASPECT RATIOS OF 2, 6, 10 AND 99

12	1.3580D-10	5.4482D-12	7.5613D-13	6.1767D-12
13	2.8117D-12	1.4377D-11	2.5387D-12	4.4630D-12
14	0.0000D+00	2.2869D-12	5.0820D-13	4.3208D-13
15	0.0000D+00	2.9908D-13	8.6948D-14	4.2493D-14
16	0.0000D+00	4.7816D-14	1.1866D-14	1.3716D-14
17	0.0000D+00	1.0793D-14	2.7598D-15	2.7273D-15
18	0.0000D+00	2.8658D-15	3.3064D-16	8.9749D-16
19	0.0000D+00	6.3484D-16	2.5927D-16	0.0000D+00
20	0.0000D+00	0.0000D+00	5.0956D-17	0.0000D+00
22	0.0000D+00	1.1431D-17	0.0000D+00	0.0000D+00
24	0.0000D+00	0.0000D+00	0.0000D+00	5.0464D-18
25	0.0000D+00	1.4911D-18	3.6983D-19	0.0000D+00
28	0.0000D+00	0.0000D+00	2.2911D-20	2.7483D-19

APPENDIX L
OCONEE UNIT 1 PTS TRANSIENTS

Table L-1 PTS Transient Descriptions for OC1

Count	TH Case #	System Failure	Operator Action	HZP	Hi K	Dominant*
1	8	2.54 cm [1 in] surge line break with 1 stuck open safety valve in SG-A.	None	No	No	No
2	12	2.54 cm [1 in] surge line break with 1 stuck open safety valve in SG-A.	HPI throttled to maintain 27.8 K [50° F] subcooling margin	No	No	No
3	15	2.54 cm [1 in] surge line break with HPI Failure	At 15 minutes after transient initiation, operator opens all TBVs to lower primary system pressure and allow CFT and LPI injection.	No	No	No
4	27	MSLB without trip of turbine driven emergency feedwater.	Operator throttles HPI to maintain 27.8 K [50° F] subcooling margin.	No	No	No
5	28	Reactor/turbine trip with 1 stuck open safety valve in SG-A	None	No	No	No
6	29	Reactor/turbine trip with 1 stuck open safety valve in SG-A and a second stuck open safety valve in SG-B	None	No	No	No
7	30	Reactor/turtine trip with 1 stuck open safety valve in SG-A	None	Yes	No	No
8	31	Reactor/turbine trip with 1 stuck open safety valve in SG-A and a second stuck open safety valve in SG-B	None	Yes	No	No
9	36	Reactor/turbine trip with 1 stuck open safety valve in SG-A and a second stuck open safety valve in SG-B	Operator throttles HPI to maintain 27.8 K [50° F] subcooling and 304.8 cm [120 in] pressurizer level.	No	No	No
10	37	Reactor/turbine trip with 1 stuck open safety valve in SG-A	Operator throttles HPI to maintain 27.8 K [50° F] subcooling and 304.8 cm [120 in] pressurizer level.	Yes	No	No
11	38	Reactor/turbine trip with 1 stuck open safety valve in SG-A and a second stuck open safety valve in SG-B	Operator throttles HPI to maintain 27.8 K [50° F] subcooling and 304.8 cm [120 in] pressurizer level.	Yes	No	No

Table L-1 PTS Transient Descriptions for OC1						
Count	**TH Case #**	**System Failure**	**Operator Action**	**HZP**	**Hi K**	**Dominant***
12	44	2.54 cm [1 in] surge line break with HPI Failure	At 15 minutes after initiation, operators open all TBVs to depressurize the system to the CFT setpoint. When the CFTs are 50 percent discharged, HPI is assumed to be recovered. The TBVs are assumed remain open for the duration of the transient.	No	No	No
13	89	Reactor/turbine trip with Loss of MFW and EFW.	Operator opens all TBVs to depressurize the secondary side to below the condensate booster pump shutoff head so that these pumps feed the steam generators. Booster pumps are assumed to be initially uncontrolled so that the steam generators are overfilled (609 cm [240 in] startup level). Operator controls booster pump flow to maintain SG level at 76 cm [30 in] due to continued RCP operation. Operator also throttles HPI to maintain 55 K [100EF] subcooling and a pressurizer level of 254 cm [100 in]. The TBVs are kept fully opened due to operator error.	No	No	No
14	90	Reactor/turbine trip with 2 stuck open safety valves in SG-A	Operator throttles HPI 20 minutes after 2.7 K [5°F] subcooling and 254 cm [100"] pressurizer level is reached [throttling criteria is 27.8 K [50°F] subcooling].	No	No	No

Table L-1 PTS Transient Descriptions for OC1						
Count	**TH Case #**	**System Failure**	**Operator Action**	**HZP**	**Hi K**	**Dominant***
15	98	Reactor/turbine trip with loss of MFW and EFW	Operator opens all TBVs to depressurize the secondary side to below the condensate booster pump shutoff head so that these pumps feed the steam generators. Booster pumps are assumed to be initially uncontrolled so that the steam generators are overfilled (610 cm [240 in] startup level). Operator controls booster pump flow to maintain SG level at 76 cm [30 in] due to continued RCP operation. Operator also throttles HPI to maintain 55 K [100EF] subcooling and a pressurizer level of 254 cm [100 in]. The TBVs are kept fully opened due to operator error.	Yes	No	No
16	99	MSLB with trip of turbine driven EFW by MSLB Circuitry	HPI is throttled 20 minutes after 2.7 K [5°F] subcooling and 254 cm [100"] pressurizer level is reached (throttling criteria is 27.8 K [50°F] subcooling).	No	No	No
17	100	MSLB with trip of turbine driven EFW by MSLB Circuitry	Operator throttles HPI 20 minutes after 2.7 K [5°F] subcooling and 254 cm [100"] pressurizer level is reached (throttling criteria is 27.8 K [50°F] subcooling).	Yes	No	No
18	101	MSLB without trip of turbine driven EFW by MSLB Circuitry	Operator throttles HPI to maintain 27.8 K [50° F] subcooling margin (throttling criteria is 27.8 K [50°F] subcooling).	Yes	No	No
19	102	Reactor/turbine trip with 2 stuck open safety valves in SG-A	Operator throttles HPI 20 minutes after 2.77 K [5°F] subcooling and 254 cm [100 in] pressurizer level is reached (throttling criteria is 27 K [50°F] subcooling).	Yes	No	No

Table L-1 PTS Transient Descriptions for OC1						
Count	**TH Case #**	**System Failure**	**Operator Action**	**HZP**	**Hi K**	**Dominant***
20	109	Stuck open pressurizer safety valve. Valve recloses at 6000 secs [RCS low pressure point].	None	No	Yes	No
21	110	5.08 cm [2 inch] surge line break with HPI failure	At 15 minutes after transient initiation, operator opens both TBV to lower primary system pressure and allow CFT and LPI injection.	No	Yes	Yes at 1000 EFPY
22	111	2.54 cm [1 in] surge line break with HPI failure	At 15 minutes after initiation, operator opens all TBVs to lower primary pressure and allow CFT and LPI injection. When the CFTs are 50% discharged, HPI is recovered. At 3000 seconds after initiation, operator starts throttling HPI to 55 K [100°F] subcooling and 254 cm [100"] pressurizer level.	No	Yes	No
23	112	Stuck open pressurizer safety valve. Valve recloses at 6000 secs.	After valve recloses, operator throttles HPI 1 minute after 2.7 K [5°F] subcooling and 254 cm [100"] pressurizer level is reached (throttling criteria is 27 K [50°F] subcooling)	No	Yes	No
24	113	Stuck open pressurizer safety valve. Valve recloses at 6000 secs.	After valve recloses, operator throttles HPI 10 minutes after 2.7 K [5°F] subcooling and 254 cm [100"] pressurizer level is reached (throttling criteria is 27.8 K [50°F] subcooling)	No	Yes	No
25	114	Stuck open pressurizer safety valve. Valve recloses at 3000 secs.	After valve recloses, operator throttles HPI 1 minute after 2.7 K [5°F] subcooling and 254 cm [100"] pressurizer level is reached (throttling criteria is 50°F subcooling)	No	Yes	No
26	115	Stuck open pressurizer Safety Valve. Valve recloses at 3000 secs.	After valve recloses, operator throttles HPI 10 minutes after 2.7 K [5°F] subcooling and 254 cm [100"] pressurizer level is reached (throttling criteria is 50°F subcooling)	No	Yes	No

Table L-1 PTS Transient Descriptions for OC1

Count	TH Case #	System Failure	Operator Action	HZP	Hi K	Dominant*
27	116	Stuck open pressurizer safety valve and HPI failure	At 15 minutes after initiation, operator opens all TBVs to lower primary pressure and allow CFT and LPI injection. When the CFTs are 50% discharged, HPI is recovered. The HPI is throttled 20 minutes after 2.7 K [5°F] subcooling and 254 cm [100"] pressurizer level is reached (throttling criteria is 50°F subcooling).	No	Yes	No
28	117	Stuck open pressurizer safety valve and HPI failure	At 15 minutes after initiation, operator opens all TBV to lower primary pressure and allow CFT and LPI injection. When the CFTs are 50% discharged, HPI is recovered. The SRV is closed 5 minutes after HPI recovered. HPI is throttled at 1 minute after 2.7 K [5°F] subcooling and 254 cm [100"] pressurizer level is reached (throttling criteria is 27.8 K [50°F] subcooling).	No	Yes	No
29	119	2.54 cm [1 in] surge line break with HPI Failure	At 15 minutes after transient initiation, the operator opens all turbine bypass valves to lower primary system pressure and allow core flood tank and LPI injection.	Yes	Yes	No
30	120	2.54 cm [1 in] surge line break with HPI Failure	At 15 minutes after sequence initiation, operators open all TBVs to depressurize the system to the CFT setpoint. When the CFTs are 50 percent discharged, HPI is assumed to be recovered. The TBVs are assumed remain opened for the duration of the transient.	Yes	Yes	No
31	121	Stuck open pressurizer safety valve. Valve recloses at 6000 secs .	Operator throttles HPI at 1 minute after 2.7 K [5°F] subcooling and 254 cm [100"] pressurizer level is reached [throttling criteria is 27.8 K [50°F] subcooling].	Yes	Yes	No

Table L-1 PTS Transient Descriptions for OC1						
Count	**TH Case #**	**System Failure**	**Operator Action**	**HZP**	**Hi K**	**Dominant***
32	122	Stuck open pressurizer safety valve. Valve recloses at 6000 secs.	Operator throttles HPI at 10 minutes after 2.7 K [5°F] subcooling and 254 cm [100"] pressurizer level is reached (throttling criteria is 27.8 K [50°F] subcooling).	Yes	Yes	Yes at 32, 60, 500, 1000 EFPY
33	123	Stuck open pressurizer safety valve. Valve recloses at 3000 secs.	Operator throttles HPI at 1 minute after 2.7 K [5°F] subcooling and 254 cm [100"] pressurizer level is reached (throttling criteria is 27.8 K [50°F] subcooling).	Yes	Yes	No
34	124	Stuck open pressurizer safety valve. Valve recloses at 3000 secs.	Operator throttles HPI at 10 minutes after 2.7 K [5°F] subcooling and 254 cm [100"] pressurizer level is reached (throttling criteria is 27.8 K [50°F] subcooling).	Yes	Yes	Yes at 60, 500, 1000 EFPY
35	125	Stuck open pressurizer safety valve and HPI Failure	At 15 minutes after initiation, operator opens all TBVs to lower primary pressure and allow CFT and LPI injection. When the CFTs are 50% discharged, HPI is recovered. HPI is throttled 20 minutes after 2.7 K [5°F] subcooling and 254 cm [100"] pressurizer level is reached (throttling criteria is 27.8 K [50°F] subcooling).	Yes	Yes	No
36	126	Stuck open pressurizer safety valve and HPI Failure	At 15 minutes after initiation, operator opens all TBVs to lower primary pressure and allow CFT and LPI injection. When the CFTs are 50% discharged, HPI is recovered. SRV is closed at 5 minutes after HPI is recovered. HPI is throttled at 1 minute after 2.7 K [5°F] subcooling and 254 cm [100"] pressurizer level is reached (throttling criteria is 27.8 K [50°F] subcooling).	Yes	Yes	No

Table L-1 PTS Transient Descriptions for OC1						
Count	**TH Case #**	**System Failure**	**Operator Action**	**HZP**	**Hi K**	**Dominant***
37	127	SGTR with a stuck open SRV in SG-B. A reactor trip is assumed to occur at the time of the tube rupture. Stuck safety relief valve is assumed to reclose 10 minutes after initiation.	Operator trips RCP's 1 minute after initiation. Operator also throttles HPI 10 minutes after 2.77 K [5° F] subcooling and 254 cm [100 in] pressurizer level is reached (assumed throttling criteria is 27 K [50°F] subcooling).	Yes	Yes	No
38	141	8.19 cm [3.22 in] surge line break [Break flow area increased by 30% from 7.18 cm [2.828 in] break].	None	No	Yes	Yes at 500, 1000 EFPY
39	142	6.01 cm [2.37 in] surge line break [Break flow area decreased by 30% from 7.18 cm [2.828 in] break].	None	No	Yes	No
40	145	4.34 cm [1.71 in] surge line break [Break flow area increased by 30% from 3.81 cm [1.5 in] break]. Winter conditions assumed [HPI, LPI temp = 277 K [40° F] and CFT temp = 294 K [70° F]].	None	No	Yes	No
41	146	TT/RT with stuck open pzr SRV [valve flow area reduced by 30 percent]. Summer conditions assumed [HPI, LPI temp = 302 K [85° F] and CFT temp = 310 K [100° F]]. Vent valves do not function.	None	No	Yes	No
42	147	TT/RT with stuck open pzr SRV. Summer conditions assumed [HPI, LPI temp = 302 K [85° F] and CFT temp = 310 K [100° F]].	None	No	Yes	No
43	148	TT/RT with partially stuck open pzr SRV [flow area equivalent to 1.5 in diameter opening]. HTC coefficients increased by 1.3.	None	No	Yes	No
44	149	TT/RT with stuck open pzr SRV. SRV assumed to reclose at 3000 secs. Operator does not throttle HPI.	None	No	Yes	No

Table L-1 PTS Transient Descriptions for OC1

Count	TH Case #	System Failure	Operator Action	HZP	Hi K	Dominant*
45	154	8.53 cm [3.36 in] surge line break [Break flow area reduced by 30% from 10.16 cm [4 in] break]. Vent valves do not function. ECC suction switch to the containment sump included in the analysis.	None	No	Yes	No
46	156	40.64 cm [16 in] hot leg break. ECC suction switch to the containment sump included in the analysis.	None	No	Yes	Yes at 500, 1000 EFPY
47	160	14.37 cm [5.656 in] surge line break. ECC suction switch to the containment sump included in the analysis.	None	No	Yes	Yes at 500, 1000 EFPY
48	164	20.32 cm [8 inch] surge line break. ECC suction switch to the containment sump included in the analysis.	None	No	Yes	Yes at 60, 500, 1000 EFPY
49	165	Stuck open pressurizer safety valve. Valve recloses at 6000 secs [RCS low pressure point].	None	Yes	Yes	Yes at 32, 60, 500, 1000 EFPY
50	168	TT/RT with stuck open pzr SRV. SRV assumed to reclose at 3000 secs. Operator does not throttle HPI.	None	Yes	Yes	Yes at 500, 1000 EFPY
51	169	TT/RT with stuck open pzr SRV [valve flow area reduced by 30 percent]. Summer conditions assumed [HPI, LPI temp = 302 K [85° F] and CFT temp = 310 K [100° F]]. Vent valves do not function.	None	Yes	Yes	No
52	170	TT/RT with stuck open pzr SRV. Summer conditions assumed [HPI, LPI temp = 302 K [85° F] and CFT temp = 310 K [100° F]].	None	Yes	Yes	No
53	171	TT/RT with partially stuck open pzr SRV [flow area equivalent to 1.5 in diameter opening]. HTC coefficients increased by 1.3.	None	Yes	Yes	No

Table L-1 PTS Transient Descriptions for OC1						
Count	**TH Case #**	**System Failure**	**Operator Action**	**HZP**	**Hi K**	**Dominant***
54	172	10.16 cm [4 in] cold leg break. ECC suction switch to the containment sump included in the analysis.	None	No	Yes	Yes at 1000 EFPY
55	178	8.53 cm [3.36 in] surge line break [Break flow area reduced by 30% from 10.16 cm [4 in] break]. Vent valves do not function. ECC suction switch to the containment sump included in the analysis.	None	No	Yes	No

Notes:
1. TH – Thermal hydraulics
2. LOCA – Loss-of-coolant accident
3. SBLOCA – Small-break loss-of-coolant accident
4. MBLOCA – Medium-break loss-of-coolant accident
5. LBLOCA – Large-break loss-of-coolant accident
6. HZP – Hot-zero power
7. SRV – Safety and relief valve
8. MSLB – Main steam line break
9. AFW – Auxiliary feedwater
10. HPI – High-pressure injection
11. RCPs – Reactor coolant pumps

* The arbitrary definition of a dominant transient is a transient that contributes 1% or more of the total Through-Wall Cracking Failure (TWCF).

APPENDIX M
OCONEE UNIT 1 FAVPOST OUTPUT

M-1: 10 Year ISI only

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               ************************************************
               *                                              *
               *               WELCOME TO FAVOR               *
               *                                              *
               *    FRACTURE ANALYSIS OF VESSELS: OAK RIDGE   *
               *                 VERSION 05.1                 *
               *                                              *
               *      FAVPOST MODULE: POSTPROCESSOR MODULE    *
               *   COMBINES TRANSIENT INITIAITING FREQUENCIES *
               *        WITH RESULTS OF PFM ANALYSIS          *
               *                                              *
               *     PROBLEMS OR QUESTIONS REGARDING FAVOR    *
               *             SHOULD BE DIRECTED TO            *
               *                                              *
               *                TERRY DICKSON                 *
               *        OAK RIDGE NATIONAL LABORATORY         *
               *                                              *
               *          e-mail: dicksontl@ornl.gov          *
               *                                              *
               ************************************************

          ***********************************************************
          * This computer program was prepared as an account of     *
          * work sponsored by the United States Government          *
          * Neither the United States, nor the United States        *
          * Department of Energy, nor the United States Nuclear     *
          * Regulatory Commission, nor any of their employees,      *
          * nor any of their contractors, subcontractors, or their  *
          * employees, makes any warranty, expressed or implied, or *
          * assumes any legal liability or responsibility for the   *
          * accuracy, completeness, or usefulness of any            *
          * information, apparatus, product, or process disclosed,  *
          * or represents that its use would not infringe           *
          * privately-owned rights.                                 *
          ***********************************************************

               DATE: 08-Sep-2005  TIME: 17:17:09

               FAVPOST INPUT  FILE NAME                    = postoc.in
               FAVPFM  OUTPUT FILE CONTAINING PFMI ARRAY = INITIATE.DAT
               FAVPFM  OUTPUT FILE CONTAINING PFMF ARRAY = FAILURE.DAT
               FAVPOST OUTPUT FILE NAME                    = 90000.out


                  **********************************
                  * NUMBER OF SIMULATIONS =  90000 *
                  **********************************
```

M-1: 10 Year ISI only (cont.)

	CONDITIONAL PROBABILITY OF INITIATION CPI=P(I|E)			CONDITIONAL PROBABILITY OF FAILURE CPF=P(F|E)			
TRANSIENT NUMBER	MEAN CPI	95th % CPI	99th % CPI	MEAN CPF	95th % CPF	99th % CPF	RATIO CPFmn/CPImn
8	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000
12	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000
15	7.2898E-09	0.0000E+00	0.0000E+00	2.3438E-19	0.0000E+00	0.0000E+00	0.0000
27	2.9384E-07	0.0000E+00	0.0000E+00	5.2252E-10	0.0000E+00	0.0000E+00	0.0018
28	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000
29	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000
30	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000
31	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000
36	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000
37	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000
38	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000
44	7.3156E-07	0.0000E+00	0.0000E+00	6.3417E-07	0.0000E+00	0.0000E+00	0.8669
89	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000
90	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000
98	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000
99	5.8411E-08	0.0000E+00	0.0000E+00	1.0665E-09	0.0000E+00	0.0000E+00	0.0183
100	1.3730E-07	0.0000E+00	0.0000E+00	5.9118E-08	0.0000E+00	0.0000E+00	0.4306
101	4.3136E-07	0.0000E+00	0.0000E+00	1.0206E-09	0.0000E+00	0.0000E+00	0.0024
102	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000
109	3.3999E-08	0.0000E+00	0.0000E+00	3.3386E-08	0.0000E+00	0.0000E+00	0.9820
110	2.6322E-04	1.6630E-03	3.3311E-03	2.5634E-06	4.7298E-05	1.3155E-05	0.0097
111	1.5338E-09	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000
112	9.0580E-11	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000
113	2.7127E-08	0.0000E+00	0.0000E+00	2.6545E-08	0.0000E+00	0.0000E+00	0.9786
114	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000
115	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000
116	1.5678E-07	0.0000E+00	0.0000E+00	1.8680E-11	0.0000E+00	0.0000E+00	0.0001
117	4.8853E-06	0.0000E+00	9.8372E-06	7.2052E-09	0.0000E+00	0.0000E+00	0.0015
119	8.1150E-06	2.5484E-04	4.9562E-05	7.1294E-08	0.0000E+00	0.0000E+00	0.0088
120	3.8499E-06	0.0000E+00	1.8546E-06	3.3791E-06	0.0000E+00	1.3591E-06	0.8777
121	1.1616E-09	0.0000E+00	0.0000E+00	2.4623E-15	0.0000E+00	0.0000E+00	0.0000
122	6.7435E-05	2.8618E-03	6.6407E-04	6.7433E-05	2.8618E-03	6.6407E-04	1.0000

M-1: 10 Year ISI only (cont.)

123	1.1616E-09	0.0000E+00	0.0000E+00	2.4623E-15	0.0000E+00	0.0000E+00	0.0000
124	2.0309E-05	1.3429E-03	1.3420E-04	2.0269E-05	1.3429E-03	1.3407E-04	0.9980
125	6.2610E-06	2.5158E-04	2.5355E-05	1.9360E-08	0.0000E+00	0.0000E+00	0.0031
126	8.7233E-08	0.0000E+00	0.0000E+00	8.2837E-12	0.0000E+00	0.0000E+00	0.0001
127	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000
141	1.0863E-05	2.5369E-04	7.5651E-05	2.2179E-07	0.0000E+00	0.0000E+00	0.0204
142	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000
145	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000
146	1.1757E-07	0.0000E+00	0.0000E+00	1.2582E-08	0.0000E+00	0.0000E+00	0.1070
147	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000
148	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000
149	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000
154	6.8954E-07	0.0000E+00	0.0000E+00	3.7947E-09	0.0000E+00	0.0000E+00	0.0055
156	2.8339E-03	6.1751E-03	3.1558E-02	1.2301E-05	9.7288E-05	1.7165E-04	0.0043
160	1.2999E-03	2.9990E-03	1.5041E-02	1.7600E-05	1.9275E-04	2.6694E-04	0.0135
164	1.2683E-03	3.4267E-03	1.5518E-02	8.9460E-06	1.1956E-04	1.1128E-04	0.0071
165	5.2553E-05	7.1010E-04	5.4086E-04	5.2549E-05	7.1010E-04	5.4086E-04	0.9999
168	2.4637E-05	1.4671E-03	1.9041E-04	2.4591E-05	1.4671E-03	1.8976E-04	0.9981
169	1.3342E-05	3.7174E-04	9.1620E-05	6.1163E-07	0.0000E+00	9.3081E-12	0.0458
170	4.6746E-10	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000
171	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000
172	7.1525E-06	2.6205E-04	3.9921E-05	4.7288E-08	0.0000E+00	0.0000E+00	0.0066
178	6.8954E-07	0.0000E+00	0.0000E+00	3.7947E-09	0.0000E+00	0.0000E+00	0.0055

NOTES: CPI IS CONDITIONAL PROBABILITY OF CRACK INITIATION, P(I|E)
CPF IS CONDITIONAL PROBABILITY OF TWC FAILURE, P(F|E)

M-1: 10 Year ISI only (cont.)

```
*************************************************************
*      PROBABILITY DISTRIBUTION FUNCTION (HISTOGRAM)       *
*         FOR THE FREQUENCY OF CRACK INITIATION            *
*************************************************************
```

FREQUENCY OF CRACK INITIATION (PER REACTOR-OPERATING YEAR)	RELATIVE DENSITY (%)	CUMULATIVE DISTRIBUTION (%)
0.0000E+00	0.8300	0.8300
2.9933E-07	96.4467	97.2767
8.9800E-07	1.4756	98.7522
1.4967E-06	0.4778	99.2300
2.0953E-06	0.2444	99.4744
2.6940E-06	0.1344	99.6089
3.2927E-06	0.0867	99.6956
3.8913E-06	0.0533	99.7489
4.4900E-06	0.0378	99.7867
5.0887E-06	0.0311	99.8178
5.6874E-06	0.0256	99.8433
6.2860E-06	0.0167	99.8600
6.8847E-06	0.0222	99.8822
7.4834E-06	0.0122	99.8944
8.0820E-06	0.0133	99.9078
8.6807E-06	0.0111	99.9189
9.2794E-06	0.0044	99.9233
9.8780E-06	0.0078	99.9311
1.0477E-05	0.0033	99.9344
1.1075E-05	0.0056	99.9400
1.1674E-05	0.0056	99.9456
1.2273E-05	0.0033	99.9489
1.2871E-05	0.0011	99.9500
1.3470E-05	0.0033	99.9533
1.4069E-05	0.0033	99.9567
1.5865E-05	0.0022	99.9589
1.6463E-05	0.0022	99.9611
1.7062E-05	0.0022	99.9633
1.7661E-05	0.0011	99.9644
1.8259E-05	0.0011	99.9656
1.8858E-05	0.0022	99.9678
2.0055E-05	0.0022	99.9700
2.0654E-05	0.0011	99.9711
2.1253E-05	0.0022	99.9733
2.1851E-05	0.0033	99.9767
2.2450E-05	0.0011	99.9778
2.4246E-05	0.0011	99.9789
2.4845E-05	0.0022	99.9811
2.6042E-05	0.0011	99.9822
2.7239E-05	0.0011	99.9833
2.7838E-05	0.0011	99.9844
2.8437E-05	0.0011	99.9856
2.9634E-05	0.0011	99.9867
3.1430E-05	0.0011	99.9878

M-1: 10 Year ISI only (cont.)

3.2029E-05	0.0011	99.9889
3.3226E-05	0.0022	99.9911
3.4423E-05	0.0011	99.9922
3.7417E-05	0.0011	99.9933
3.8015E-05	0.0011	99.9944
3.8614E-05	0.0011	99.9956
4.0410E-05	0.0011	99.9967
4.2206E-05	0.0011	99.9978
4.7594E-05	0.0011	99.9989
5.8969E-05	0.0011	100.0000

```
=============================================
==      Summary Descriptive Statistics     ==
=============================================

Minimum                              = 0.0000E+00
Maximum                              = 5.9268E-05
Range                                = 5.9268E-05

Number of Simulations                =  90000

5th Percentile                       = 1.0021E-11
Median                               = 6.3992E-09
95.0th Percentile                    = 2.9933E-07
99.0th Percentile                    = 1.2085E-06
99.9th Percentile                    = 7.7328E-06

Mean                                 = 9.6315E-08
Standard Deviation                   = 7.2357E-07
Standard Error                       = 2.4119E-09
Variance (unbiased)                  = 5.2356E-13
Variance (biased)                    = 5.2355E-13
Moment Coeff. of Skewness            = 3.4633E+01
Pearson's 2nd Coeff. of Skewness     = 3.9933E-01
Kurtosis                             = 1.7340E+03
```

```
***********************************************************
*      PROBABILITY DISTRIBUTION FUNCTION (HISTOGRAM)      *
*      FOR THROUGH-WALL CRACKING FREQUENCY (FAILURE)      *
***********************************************************
```

FREQUENCY OF TWC FAILURES (PER REACTOR-OPERATING YEAR)	RELATIVE DENSITY (%)	CUMULATIVE DISTRIBUTION (%)
0.0000E+00	58.0878	58.0878
1.0195E-07	41.7589	99.8467
3.0584E-07	0.0911	99.9378
5.0973E-07	0.0200	99.9578
7.1362E-07	0.0100	99.9678

M-1: 10 Year ISI only (cont.)

9.1751E-07	0.0078	99.9756
1.1214E-06	0.0078	99.9833
1.5292E-06	0.0022	99.9856
1.7331E-06	0.0022	99.9878
1.9370E-06	0.0011	99.9889
2.1408E-06	0.0011	99.9900
2.3447E-06	0.0011	99.9911
2.5486E-06	0.0011	99.9922
3.3642E-06	0.0022	99.9944
3.7720E-06	0.0022	99.9967
3.9759E-06	0.0011	99.9978
7.4420E-06	0.0011	99.9989
2.0083E-05	0.0011	100.0000

```
=============================================
==      Summary Descriptive Statistics      ==
=============================================

Minimum                                = 0.0000E+00
Maximum                                = 2.0185E-05
Range                                  = 2.0185E-05

Number of Simulations                  =  90000

5th Percentile                         = 0.0000E+00
Median                                 = 0.0000E+00
95.0th Percentile                      = 1.0195E-07
99.0th Percentile                      = 2.2874E-08
99.9th Percentile                      = 2.2130E-07

Mean                                   = 2.0641E-09
Standard Deviation                     = 8.1215E-08
Standard Error                         = 2.7072E-10
Variance (unbiased)                    = 6.5958E-15
Variance (biased)                      = 6.5958E-15
Moment Coeff. of Skewness              = 1.8585E+02
Pearson's 2nd Coeff. of Skewness       =-1.6013E-01
Kurtosis                               = 4.3419E+04


**********************************************************
*  FRACTIONALIZATION OF FREQUENCY OF CRACK INITIATION    *
*     AND THROUGH-WALL CRACKING FREQUENCY (FAILURE) -    *
*       WEIGHTED BY TRANSIENT INITIATING FREQUENCIES     *
**********************************************************
```

	% of total frequency of crack initiation	% of total frequency of of TWC failure
8	0.00	0.00
12	0.00	0.00
15	0.00	0.00
27	0.00	0.00

M-1: 10 Year ISI only (cont.)

28	0.00	0.00
29	0.00	0.00
30	0.00	0.00
31	0.00	0.00
36	0.00	0.00
37	0.00	0.00
38	0.00	0.00
44	0.00	0.00
89	0.00	0.00
90	0.00	0.00
98	0.00	0.00
99	0.00	0.00
100	0.00	0.00
101	0.00	0.00
102	0.00	0.00
109	0.00	0.01
110	1.16	0.49
111	0.00	0.00
112	0.00	0.00
113	0.00	0.02
114	0.00	0.00
115	0.00	0.00
116	0.00	0.00
117	0.00	0.00
119	0.00	0.00
120	0.00	0.01
121	0.00	0.00
122	0.61	28.54
123	0.00	0.00
124	0.21	9.75
125	0.00	0.00
126	0.00	0.00
127	0.00	0.00
141	1.25	1.27
142	0.00	0.00
145	0.00	0.00
146	0.01	0.02
147	0.00	0.00
148	0.00	0.00
149	0.00	0.00
154	0.09	0.03
156	27.80	6.96
160	30.94	29.68
164	36.76	13.00
165	0.14	6.31
168	0.07	3.44
169	0.12	0.23
170	0.00	0.00
171	0.00	0.00
172	0.83	0.25
178	0.01	0.00
TOTALS	100.00	100.00

M-2: ISI Every 10 Years

```
**************************************************
*                                                *
*               WELCOME TO FAVOR                 *
*                                                *
*    FRACTURE ANALYSIS OF VESSELS: OAK RIDGE     *
*                 VERSION 05.1                   *
*                                                *
*      FAVPOST MODULE: POSTPROCESSOR MODULE      *
*   COMBINES TRANSIENT INITIAITING FREQUENCIES   *
*         WITH RESULTS OF PFM ANALYSIS           *
*                                                *
*      PROBLEMS OR QUESTIONS REGARDING FAVOR     *
*             SHOULD BE DIRECTED TO              *
*                                                *
*                TERRY DICKSON                   *
*         OAK RIDGE NATIONAL LABORATORY          *
*                                                *
*          e-mail: dicksontl@ornl.gov            *
*                                                *
**************************************************

************************************************************
* This computer program was prepared as an account of     *
* work sponsored by the United States Government          *
* Neither the United States, nor the United States        *
* Department of Energy, nor the United States Nuclear     *
* Regulatory Commission, ncr any of their employees,      *
* nor any of their contractors, subcontractors, or their  *
* employees, makes any warranty, expressed or implied, or *
* assumes any legal liability or responsibility for the   *
* accuracy, completeness, or usefulness of any            *
* information, apparatus, product, or process disclosed,  *
* or represents that its use would not infringe           *
* privately-owned rights.                                 *
************************************************************


     DATE: 08-Sep-2005  TIME: 16:13:04

     FAVPOST INPUT  FILE NAME                     = postoc.in
     FAVPFM  OUTPUT FILE CONTAINING PFMI ARRAY = INITIATE.DAT
     FAVPFM  OUTPUT FILE CONTAINING PFMF ARRAY = FAILURE.DAT
     FAVPOST OUTPUT FILE NAME                     = 90000.out


        **********************************
        * NUMBER OF SIMULATIONS =  90000 *
        **********************************
```

M-2: ISI Every 10 Years (cont.)

TRANSIENT NUMBER	CONDITIONAL PROBABILITY OF INITIATION CPI=P(I|E) MEAN CPI	95th % CPI	99th % CPI	CONDITIONAL PROBABILITY OF FAILURE CPF=P(F|E) MEAN CPF	95th % CPF	99th % CPF	RATIO CPFmn/CPImn
8	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000
12	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000
15	1.7815E-09	0.0000E+00	0.0000E+00	9.3921E-13	0.0000E+00	0.0000E+00	0.0005
27	1.4039E-07	0.0000E+00	0.0000E+00	6.6856E-09	0.0000E+00	0.0000E+00	0.0476
28	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000
29	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000
30	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000
31	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000
36	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000
37	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000
38	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000
44	5.3390E-07	0.0000E+00	0.0000E+00	4.9845E-07	0.0000E+00	0.0000E+00	0.9336
89	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000
90	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000
98	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000
99	2.5350E-08	0.0000E+00	0.0000E+00	1.1771E-08	0.0000E+00	0.0000E+00	0.4643
100	1.7558E-07	0.0000E+00	0.0000E+00	1.4948E-07	0.0000E+00	0.0000E+00	0.8514
101	1.7367E-07	0.0000E+00	0.0000E+00	2.3133E-09	0.0000E+00	0.0000E+00	0.0133
102	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000
109	7.8421E-09	0.0000E+00	0.0000E+00	7.7661E-09	0.0000E+00	0.0000E+00	0.9903
110	2.4559E-04	9.8081E-04	3.3825E-03	3.4652E-06	1.6053E-04	1.4147E-05	0.0141
111	1.9195E-10	0.0000E+00	0.0000E+00	2.1778E-14	0.0000E+00	0.0000E+00	0.0001
112	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000
113	5.7623E-09	0.0000E+00	0.0000E+00	5.7071E-09	0.0000E+00	0.0000E+00	0.9904
114	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000
115	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000
116	4.2378E-08	0.0000E+00	0.0000E+00	1.7003E-10	0.0000E+00	0.0000E+00	0.0040
117	3.4748E-06	0.0000E+00	7.9204E-06	1.9228E-08	0.0000E+00	0.0000E+00	0.0055
119	6.3411E-06	8.7838E-05	4.6421E-05	2.0663E-07	0.0000E+00	0.0000E+00	0.0326
120	3.7181E-06	0.0000E+00	2.6462E-06	3.3372E-06	0.0000E+00	2.1160E-06	0.8976
121	9.5832E-11	0.0000E+00	0.0000E+00	1.1044E-11	0.0000E+00	0.0000E+00	0.1152
122	8.2527E-05	1.1594E-03	8.5743E-04	8.2526E-05	1.1594E-03	8.5743E-04	1.0000

M-2: ISI Every 10 Years (cont.)

123	9.5832E-11	0.0000E+00	0.0000E+00	1.1044E-11	0.0000E+00	0.0000E+00	0.1152
124	2.2553E-05	5.8212E-04	1.8636E-04	2.2501E-05	5.8212E-04	1.8553E-04	0.9977
125	4.8152E-06	8.5566E-05	2.6193E-05	8.0652E-08	0.0000E+00	0.0000E+00	0.0167
126	2.9183E-08	0.0000E+00	0.0000E+00	2.8699E-10	0.0000E+00	0.0000E+00	0.0098
127	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000
141	7.9520E-06	1.1655E-04	6.6654E-05	3.9604E-07	0.0000E+00	0.0000E+00	0.0498
142	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000
145	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000
146	7.9326E-08	0.0000E+00	0.0000E+00	3.8870E-08	0.0000E+00	0.0000E+00	0.4900
147	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000
148	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000
149	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000
154	3.1638E-07	0.0000E+00	0.0000E+00	2.4759E-08	0.0000E+00	0.0000E+00	0.0783
156	2.7343E-03	6.0920E-03	2.9727E-02	1.3731E-05	3.8866E-04	1.8007E-04	0.0050
160	1.2415E-03	2.2956E-03	1.5514E-02	2.2394E-05	6.6379E-04	3.0192E-04	0.0180
164	1.2169E-03	2.3202E-03	1.5756E-02	1.1117E-05	3.4747E-04	1.1747E-04	0.0091
165	7.5936E-05	1.1789E-03	7.1367E-04	7.5936E-05	1.1789E-03	7.1367E-04	1.0000
168	2.7524E-05	6.4298E-04	2.5502E-04	2.7467E-05	6.4298E-04	2.5487E-04	0.9980
169	1.0686E-05	1.9109E-04	8.9740E-05	9.4985E-07	0.0000E+00	3.4677E-10	0.0889
170	5.5133E-11	0.0000E+00	0.0000E+00	1.5098E-12	0.0000E+00	0.0000E+00	0.0274
171	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000
172	5.4843E-06	7.9106E-05	3.7104E-05	1.1567E-07	0.0000E+00	0.0000E+00	0.0211
178	3.1638E-07	0.0000E+00	0.0000E+00	2.4759E-08	0.0000E+00	0.0000E+00	0.0783

NOTES: CPI IS CONDITIONAL PROBABILITY OF CRACK INITIATION, P(I|E)
CPF IS CONDITIONAL PROBABILITY OF TWC FAILURE, P(F|E)

M-2: ISI Every 10 Years (cont.)

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*********************************************************
*      PROBABILITY DISTRIBUTION FUNCTION (HISTOGRAM)    *
*         FOR THE FREQUENCY OF CRACK INITIATION         *
*********************************************************
```

FREQUENCY OF CRACK INITIATION (PER REACTOR-OPERATING YEAR)	RELATIVE DENSITY (%)	CUMULATIVE DISTRIBUTION (%)
0.0000E+00	0.8700	0.8700
1.0899E-06	98.5233	99.3933
3.2698E-06	0.3711	99.7644
5.4496E-06	0.0944	99.8589
7.6295E-06	0.0533	99.9122
9.8093E-06	0.0267	99.9389
1.1989E-05	0.0122	99.9511
1.4169E-05	0.0044	99.9556
1.6349E-05	0.0089	99.9644
1.8529E-05	0.0111	99.9756
2.0709E-05	0.0044	99.9800
2.2888E-05	0.0044	99.9844
2.5068E-05	0.0033	99.9878
2.7248E-05	0.0022	99.9900
3.3788E-05	0.0033	99.9933
3.5967E-05	0.0011	99.9944
3.8147E-05	0.0011	99.9956
4.6867E-05	0.0022	99.9978
5.9946E-05	0.0011	99.9989
2.1471E-04	0.0011	100.0000

```
============================================
==      Summary Descriptive Statistics    ==
============================================
```

Minimum	= 0.0000E+00
Maximum	= 2.1580E-04
Range	= 2.1580E-04
Number of Simulations	= 90000
5th Percentile	= 1.1016E-11
Median	= 6.2400E-09
95.0th Percentile	= 1.0899E-06
99.0th Percentile	= 1.4158E-06
99.9th Percentile	= 7.1299E-06
Mean	= 9.6929E-08
Standard Deviation	= 9.9040E-07
Standard Error	= 3.3013E-09
Variance (unbiased)	= 9.8090E-13
Variance (biased)	= 9.8088E-13
Moment Coeff. of Skewness	= 1.2603E+02
Pearson's 2nd Coeff. of Skewness	= 2.9361E-01

M-2: ISI Every 10 Years (cont.)

Kurtosis = 2.5418E+04

```
*************************************************************
*      PROBABILITY DISTRIBUTION FUNCTION (HISTOGRAM)        *
*      FOR THROUGH-WALL CRACKING FREQUENCY (FAILURE)        *
*************************************************************
```

FREQUENCY OF TWC FAILURES (PER REACTOR-OPERATING YEAR)	RELATIVE DENSITY (%)	CUMULATIVE DISTRIBUTION (%)
0.0000E+00	55.2600	55.2600
4.6338E-08	44.3778	99.6378
1.3901E-07	0.1667	99.8044
2.3169E-07	0.0700	99.8744
3.2437E-07	0.0256	99.9000
4.1704E-07	0.0289	99.9289
5.0972E-07	0.0133	99.9422
6.0239E-07	0.0122	99.9544
6.9507E-07	0.0044	99.9589
7.8774E-07	0.0078	99.9667
8.8042E-07	0.0022	99.9689
9.7310E-07	0.0056	99.9744
1.0658E-06	0.0022	99.9767
1.1584E-06	0.0011	99.9778
1.2511E-06	0.0044	99.9822
1.3438E-06	0.0033	99.9856
1.4365E-06	0.0011	99.9867
1.5292E-06	0.0022	99.9889
1.7145E-06	0.0011	99.9900
1.8999E-06	0.0011	99.9911
2.1779E-06	0.0011	99.9922
3.3827E-06	0.0011	99.9933
3.7534E-06	0.0011	99.9944
4.0314E-06	0.0011	99.9956
4.7728E-06	0.0011	99.9967
5.6996E-06	0.0011	99.9978
5.9776E-06	0.0011	99.9989
9.1286E-06	0.0011	100.0000

```
=============================================
==      Summary Descriptive Statistics     ==
=============================================
```

Minimum	= 0.0000E+00
Maximum	= 9.1749E-06
Range	= 9.1749E-06
Number of Simulations	= 90000
5th Percentile	= 0.0000E+00
Median	= 0.0000E+00
95.0th Percentile	= 4.6338E-08

M-2: ISI Every 10 Years (cont.)

```
99.0th Percentile                    = 2.6796E-08
99.9th Percentile                    = 3.2437E-07

Mean                                 = 2.1755E-09
Standard Deviation                   = 5.6060E-08
Standard Error                       = 1.8687E-10
Variance (unbiased)                  = 3.1427E-15
Variance (biased)                    = 3.1427E-15
Moment Coeff. of Skewness            = 9.4559E+01
Pearson's 2nd Coeff. of Skewness     =-2.1751E-01
Kurtosis                             = 1.1887E+04
```

```
************************************************************
*  FRACTIONALIZATION OF FREQUENCY OF CRACK INITIATION      *
*      AND THROUGH-WALL CRACKING FREQUENCY (FAILURE) -     *
*       WEIGHTED BY TRANSIENT INITIATING FREQUENCIES       *
************************************************************
```

	% of total frequency of crack initiation	% of total frequency of of TWC failure
8	0.00	0.00
12	0.00	0.00
15	0.00	0.00
27	0.00	0.00
28	0.00	0.00
29	0.00	0.00
30	0.00	0.00
31	0.00	0.00
36	0.00	0.00
37	0.00	0.00
38	0.00	0.00
44	0.00	0.01
89	0.00	0.00
90	0.00	0.00
98	0.00	0.00
99	0.00	0.00
100	0.00	0.00
101	0.00	0.00
102	0.00	0.00
109	0.00	0.00
110	1.22	0.63
111	0.00	0.00
112	0.00	0.00
113	0.00	0.01
114	0.00	0.00
115	0.00	0.00
116	0.00	0.00
117	0.00	0.00
119	0.00	0.01
120	0.00	0.01
121	0.00	0.00
122	0.70	31.40

M-2: ISI Every 10 Years (cont.)

123	0.00	0.00
124	0.18	8.09
125	0.00	0.00
126	0.00	0.00
127	0.00	0.00
141	0.91	1.69
142	0.00	0.00
145	0.00	0.00
146	0.00	0.05
147	0.00	0.00
148	0.00	0.00
149	0.00	0.00
154	0.07	0.39
156	26.68	7.15
160	30.45	24.43
164	38.82	14.53
165	0.17	7.73
168	0.06	2.68
169	0.09	0.56
170	0.00	0.00
171	0.00	0.00
172	0.64	0.63
178	0.01	0.01
TOTALS	100.00	100.00

DATE: 08-Sep-2005   TIME: 16:14:07