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NOT IN ADAMS

Docket Number 50-346

License Number NPF-3

Serial Number 2735

October 17, 2001

U.S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, DC. 20555-0001

Subject: Supplemental Information in Response to NRC Bulletin 2001-01, "Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles"

Ladies and Gentlemen:

The attached provides supplemental information concerning the Davis-Besse Nuclear Power Station, Unit 1 (DBNPS) response (Serial Number 2731, dated September 4, 2001) to Nuclear Regulatory Commission (NRC) Bulletin 2001-01, "Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles." Portions of this information were discussed with members of the NRC staff on October 3 and 11, 2001. In addition, the DBNPS and NRC staffs are scheduled to meet and discuss this information and additional NRC crack growth modeling information on October 24, 2001.

This submittal provides updated and additional information in support of the basis for the continued safe operation of the Davis-Besse Nuclear Power Station (DBNPS) until its next scheduled refueling outage commencing in March 2002, at which time the Control Rod Drive Mechanism (CRDM) nozzles and Reactor Pressure Vessel (RPV) head penetrations will undergo qualified visual inspections or appropriate supplemental inspections.

In May 1996, during a refueling outage, the RPV head was inspected. No leakage was identified, and these results have been recently verified by a re-review of the video tapes obtained from that inspection. The RPV head was mechanically cleaned at the end of the outage. Subsequent inspections of the RPV head in the next two refueling outages (1998 and 2000), also did not identify any leakage in the CRDM nozzle-to-head areas that could be inspected. Video tapes taken during these inspections have also been re-reviewed.

A088

Accordingly, using the end of the outage in 1996 as the postulated worst-case time for an axial crack to reach a through-wall condition, the projected time for the crack to reach its critical through-wall circumferential size was determined based on the results from an Framatome ANP assessment. This RV Head Nozzle and Weld Safety Assessment demonstrates the postulated crack will take approximately 7.5 years to manifest into an ASME Code allowable crack size. Applying this 7.5 years to the May 1996 inspection projects the worst-case allowable crack size being reached in November 2003. It is important to note the allowable crack size will still maintain an ASME Code safety factor of three.

A Finite Element Gap Analysis was performed by Structural Integrity Associates to verify the gaps between the CRDM nozzles and the RVP head during normal operation would permit through-wall leakage from any nozzle or through-weld cracks in the J-groove weld to be observed via boric acid crystal deposits. This analysis concluded that all but four nozzle/penetration interfaces would show visible leakage. These four nozzles are in the least stressed area of the RPV head, and where no leakage attributed to circumferential cracks has been observed at any other plants.

The DBNPS staff is continuing to be involved in and monitoring industry developments regarding CRDM nozzle/penetration cracking, and modifying its inspection plans as appropriate.

Based on the previous inspections conducted, re-reviewed inspection videos, analyses that have been performed concerning crack growth rates, the ability to identify cracking, and industry evaluations and findings, it is concluded there is reasonable assurance that the DBNPS will continue to operate safely to the next refueling outage scheduled for March 2002.

If you have any question or comments, please contact Mr. David H. Lockwood, Manager, Regulatory Affairs, at (419) 321-8450.

Very truly yours,

B Worley

/s

Enclosure and Attachments

cc: J. E. Dyer, Regional Administrator, NRC Region III
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Utility Radiological Safety Board

SUPPLEMENTAL INFORMATION

IN RESPONSE TO

NRC BULLETIN 2001-01

FOR

DAVIS-BESSE NUCLEAR POWER STATION

UNIT NUMBER 1

This letter is submitted pursuant to 10 CFR 50.54(f) and contains supplemental information concerning the response (Serial 2371, dated September 4, 2001) to NRC Bulletin 2001-01, "Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles," for the Davis-Besse Nuclear Power Station, Unit Number 1.

I, Guy G. Campbell, state that (1) I am Vice President - Nuclear of the FirstEnergy Nuclear Operating Company, (2) I am duly authorized to execute and file this certification on behalf of the Toledo Edison Company and The Cleveland Electric Illuminating Company, and (3) the statements set forth herein are true and correct to the best of my knowledge, information and belief.

For: G. G. Campbell, Vice President - Nuclear

By: L W Worley
L. W. Worley, Director - Support Services

Affirmed and subscribed before me this 17th day of October, 2001

Laura A. Jennison
Notary Public, State of Ohio
Laura A. Jennison
My Commission Expires on August 16, 2006.

SUPPLEMENTAL INFORMATION IN RESPONSE TO NRC BULLETIN 2001-01

The Davis-Besse Nuclear Power Station, Unit 1 (DBNPS) submitted its response to NRC Bulletin 2001-01, "Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles" in FirstEnergy Nuclear Operating Company (FENOC) letter Serial Number 2371, dated September 4, 2001. Portions of this information have been discussed with members of the NRC staff on October 3 and 11, 2001.

SUMMARY

This submittal provides updated and additional information in support of the basis for the continued safe operation of the Davis-Besse Nuclear Power Station (DBNPS) until its next scheduled refueling outage commencing in March 2002, at which time the Control Rod Drive Mechanism (CRDM) nozzles and Reactor Pressure Vessel (RPV) head penetrations will undergo qualified visual inspections.

In May 1996, during a refueling outage, the RPV head was inspected. No leakage was identified, and these results have been recently verified by a re-review of the video tapes obtained from that inspection. The RPV head was mechanically cleaned at the end of the outage. Subsequent inspections of the RPV head in the next two refueling outages (1998 and 2000), also did not identify any leakage in the CRDM nozzle-to-head areas that could be inspected. Video tapes taken during these inspections have also been re-reviewed.

Accordingly, using the end of the outage in 1996 as the postulated worst-case time for an axial crack to reach a through-wall condition, the projected time for the crack to reach its critical through-wall circumferential size was determined based on the results from an Framatome ANP assessment. This RV Head Nozzle and Weld Safety Assessment demonstrates the postulated crack will take approximately 7.5 years to manifest into an ASME Code allowable crack size. Applying this 7.5 years to the May 1996 inspection projects the worst-case allowable crack size being reached in November 2003. It is important to note the allowable crack size will still maintain an ASME Code safety factor of three.

A Finite Element Gap Analysis was performed by Structural Integrity Associates to verify the gaps between the CRDM nozzles and the RVP head during normal operation would permit through-wall leakage from any nozzle or through-weld cracks in the J-groove weld to be observed via boric acid crystal deposits. This analysis concluded that all but four nozzle/penetration interfaces would show visible leakage. These four nozzles are in the least stressed area of the RPV head, and where no leakage attributed to circumferential cracks has been observed at any other plants.

The DBNPS staff is continuing to be involved in and monitoring industry developments regarding CRDM nozzle/penetration cracking, and modifying its inspection plans as appropriate.

Based on the previous inspections conducted, re-reviewed inspection videos, analyses that have been performed concerning crack growth rates, the ability to identify cracking, and industry evaluations and findings, it is concluded there is reasonable assurance that the DBNPS will continue to operate safely to the next refueling outage scheduled for March 2002.

PLANT DESIGN

The DBNPS has a Babcock & Wilcox (B&W) nuclear steam supply system. The design is similar to other B&W 177-fuel assembly plants, except that DBNPS is of the raised-loop design. The DBNPS has 69 Control Rod Drive Mechanism (CRDM) nozzles of which 61 are used for CRDMs, 7 are spare, and one is used for the Reactor Pressure Vessel (RPV) continuous head vent. Each CRDM nozzle is constructed of Inconel Alloy 600 and is attached to the RPV head by an Inconel Alloy 182 J-groove weld. The DBNPS is unique in the B&W fleet in that it is the only unit that has a RPV head continuous vent that allows for the movement of coolant around the interior of the head, thereby minimizing the stagnation of hot coolant in the top of the head and trapping of air or oxygen.

PREVIOUS INSPECTION RESULTS

In FENOC letter Serial Number 2731, the past inspections of the DBNPS Reactor Pressure Vessel (RPV) head were discussed. As a result of NRC staff questions, supplemental information to and amplification of that discussion is provided in the following.

The inspections performed during the 10th, 11th, and 12th Refueling Outage (10RFO, conducted April 8 to June 2, 1996; 11RFO, conducted April 10, to May 23, 1998; and, 12RFO, conducted April 1 to May 18, 2000) consisted of a whole head visual inspection of the RPV head in accordance with the DBNPS Boric Acid Control Program pursuant to Generic Letter 88-05, "Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants." The visual inspections were conducted by remote camera and included below insulation inspections of the RPV bare head such that the Control Rod Drive Mechanism (CRDM) nozzle penetrations were viewed. During 10RFO, 65 of 69 nozzles were viewed, during 11RFO, 50 of 69 nozzles were viewed, and during 12RFO, 45 of 69 nozzles were viewed. It should be noted that 19 of the obscured nozzles in 12RFO were also those obscured in 11RFO. Following 11RFO, the RPV head was mechanically cleaned in localized areas as limited by the service structure design. Following 12RFO, the RPV head was cleaned with demineralized water to the extent possible to provide a clean head for evaluating future inspection results.

The affected areas of accumulated boric acid crystal deposits were video taped, and have subsequently been reviewed with specific focus on boric acid crystal deposits with reference to the CRDM nozzle penetration leakage as previously observed at the Oconee Nuclear Station, Unit 3 (ONS-3) and at Arkansas Nuclear One, Unit 1 (ANO-1). During the 12RFO inspection, 24 of the 69 nozzles were obscured by boric acid crystal deposits that were clearly attributable to leaking motor tube flanges from the center CRDMs. A further subsequent review of the video tapes has been conducted and corroborates the previous statements and conclusions stated in letter Serial Number 2731 that the results of this review did not identify any boric acid crystal deposits that would have been attributed to leakage from the CRDM nozzle penetrations, but were indicative of CRDM flange leakage. Included as Attachments 2 and 3 are the inspection results for 10RFO, 11RFO and 12RFO, and a figure representing these nozzle locations, respectively. Att 2 3

In summary, results from previous inspections of the CRDM nozzle penetrations provide reasonable assurance for the continued safe operation of the DBNPS until the next refueling outage in March 2002.

ANALYTICAL WORK PERFORMED:

RV Head Nozzle and Weld Safety Assessment

Att 4

Attachment 4, Framatome ANP's non-proprietary document FRA-ANP 51-5012567-01, "RV Head Nozzle and Weld Safety Assessment," provides an assessment that demonstrates safe operation of the Babcock & Wilcox (B&W)-designed nuclear steam supply systems with the potential for primary water stress corrosion cracking (PWSCC) of RPV head penetration nozzles. The document addresses the assumed presence of PWSCC in either the nozzle base material or the partial penetration welds used in the attachment to the RPV head and the risk assessment with regard to nozzle integrity over a period of time.

Using the Framatome ANP assessment, the DBNPS feels assured in operating until the next scheduled refueling outage. This is based on the worst case scenario that a visible nozzle axial crack leak developed immediately after start-up from 10RFO in May 1996, and was from one of the 19 drives that could not be inspected in 1998 (11RFO) or the 24 drives that could not be inspected in 2000 (12RFO). The Framatome ANP assessment concluded that such a crack would take 3.5 to 10 years to grow circumferentially through wall. The DBNPS has assumed the 3.5 year value since 3.5 years is based upon multiple crack sites merging together consistent with that which was observed at Oconee 3. This results in the development of a worst case through wall circumferential crack development by November 1999 (May 1996 plus 3.5 years). The Framatome ANP assessment further concluded that this crack would be expected to take an additional 4 years to grow to maximum ASME Code allowable crack size of 270 degrees. Continuing to apply this to the DBNPS's worst case scenario results in the potential to reach a maximum allowable crack size on one of the obscured CRDM nozzles (from 1998 and 2000 inspections) by November 2003. Because this date is beyond the date for the planned March 2002 refueling outage, the DBNPS has concluded that there is reasonable assurance that DBNPS will continue to operate safely to the start of 13RFO, scheduled for March 2002.

Finite Element Gap Analysis

As discussed with the NRC staff during a telephone conference call on October 3, 2001, the DBNPS contracted with Structural Integrity Associates (SIA) to perform a finite element analysis of the RPV head penetrations and nozzles. This analysis was performed to verify that gaps would exist between the CRDM nozzles and the RPV head during normal operation. These gaps would permit through-wall leakage from any nozzle or through-weld cracks in the J-groove weld to be observed via boric acid crystal deposits. This plant-specific stress analysis used the DBNPS as-built nozzle and RPV head dimensions. The analysis does not include the effects of primary system pressure in the nozzle gap area that would tend to further open the gaps. The SIA analysis is included herein as Attachment 5 and provides assurance that leakage will be visible on all but four (4) of the sixty-nine (69) nozzle/penetration interfaces. However, the four nozzle/penetration interfaces where it could not be assured that leakage would be visible are nozzle numbers 1, 2, 3, and 4, which are in the center of the RPV head. As documented in the industry history of circumferential cracks observed to date, no leakage attributable to circumferential cracks has been observed in this area from any of the inspections conducted by other licensees. Therefore, based on the verification of inspection results conducted at DBNPS, industry historical results of CDRM nozzle leakage and the finite element analysis performed, it is concluded that no leakage from the CRDM nozzle/head interface has previously occurred at the DBNPS, and through-weld cracking was not present. AH5

INDUSTRY EXPERIENCE & FINDINGS

Since discovery of Alloy 600 cracking at VC Summer and ONS, the DBNPS has been following activities and planning site-specific activities to assure that the Reactor Coolant System pressure boundary integrity is maintained. These activities have included participation in industry groups that are extensively analyzing and characterizing the phenomenological attributes of the cracking issue, and developing sophisticated means of detecting and, as necessary, repairing identified cracks. The findings at other plants are being communicated among the industry in a timely manner which allows aggressive evaluation of the nature, extensiveness and implications of the cracking to ensure the issue is understood as completely as possible, and ensures the development of conservative decision-making. It is through these continuing efforts as well as ongoing plant-specific efforts that the DBNPS can also conclude that there is reasonable assurance that the DBNPS will operate safely to its next refueling outage, scheduled to commence in March 2002.

ALARA ISSUES

In NRC Bulletin 2001-01, page 8, the NRC identified that nozzle penetration activities have the potential for large personnel exposure. Plants have experienced 15 to 40 rem during recent CRDM nozzle activities. The bulletin states that all activities related to the inspection of nozzles should be planned and implemented to keep personnel exposures as low as reasonably achievable (ALARA). As discussed in its initial response to the bulletin, the DBNPS will perform qualified visual inspections or appropriate supplemental inspection of the CRDM nozzle penetrations during its refueling outage scheduled to commence in March 2002. Inspection of these penetrations between now and March 2002, and then again during the refueling outage would significantly increase the personnel exposures. Since the continued safe operation of the DBNPS can be reasonably assured to the beginning of the next refueling outage, completing additional inspections before then would not be consistent with ALARA principles.

CONCLUSION

Based on the previous inspections conducted, analyses that have been performed concerning crack growth rates, the ability to identify cracking, and industry evaluations and findings, it is concluded that there is reasonable assurance that the DBNPS will continue to operate safely to the start of 13RFO, scheduled in March 2002.

Nozzle No.	Core Locat.	Quadrant	1996 Inspection results	1998 Inspection results	2000 Inspection results
			See Note 1.0		
1	H8	1		*Flange Leak Evident*	*Flange Leak Evident*
2	G7	4		*Flange Leak Evident*	*Flange Leak Evident*
3	**G9**	**1**		***Flange Leak Evident***	***Flange Leak Evident***
4	K9	2		*Flange Leak Evident*	*Flange Leak Evident*
5	K7	3		Flange Leak Evident	Flange Leak Evident
6	**F8**	**1**		**Flange Leak Evident**	**Flange Leak Evident**
7	H10	2		Flange Leak Evident	Flange Leak Evident
8	L8	3		No Leak Observed	No Leak Observed
9	H6	4		No Leak Observed	No Leak Observed
10	F6	4		No Leak Observed	No Leak Observed
11	**F10**	**1**		**Flange Leak Evident**	**Flange Leak Evident**
12	L10	2		No Leak Observed	No Leak Observed
13	L6	3		No Leak Recorded	No Leak Observed
14	E7	4		Flange Leak Evident	Flange Leak Evident
15	E9	1		Flange Leak Evident	Flange Leak Evident
16	G11	1		Flange Leak Evident	Flange Leak Evident
17	K11	2		No Leak Observed	No Leak Observed
18	M9	2		No Leak Recorded	No Leak Observed
19	M7	3		No Leak Observed	No Leak Recorded
20	K5	3		No Leak Observed	No Leak Observed
21	G5	4		No Leak Observed	No Leak Observed
22	D8	1		Flange Leak Evident	Flange Leak Evident
23	H12	2		No Leak Observed	No Leak Observed
24	N8	3		No Leak Recorded	No Leak Recorded
25	H4	4		No Leak Recorded	No Leak Observed
26	E5	4		No Leak Recorded	No Leak Observed
27	E11	1		Flange Leak Evident	Flange Leak Evident
28	M11	2		No Leak Recorded	No Leak Observed
29	M5	3		No Leak Recorded	No Leak Observed
30	D6	4		No Leak Observed	No Leak Observed
31	**D10**	**1**		**Flange Leak Evident**	**Flange Leak Evident**
32	F12	1		Flange Leak Evident	Flange Leak Evident
33	L12	2		No Leak Recorded	No Leak Observed
34	N10	2		No Leak Recorded	No Leak Observed
35	N6	3		No Leak Recorded	No Leak Recorded
36	L4	3		No Leak Recorded	No Leak Observed
37	F4	4		No Leak Recorded	No Leak Observed
38	C7	4		No Leak Recorded	Flange Leak Evident
39	C9	1		Flange Leak Evident	Flange Leak Evident
40	G13	1		Flange Leak Evident	Flange Leak Evident
41	K13	2		No Leak Recorded	No Leak Observed
42	O9	2		No Leak Recorded	No Leak Recorded
43	O7	3		No Leak Recorded	No Leak Recorded
44	K3	3		No Leak Recorded	No Leak Observed
45	G3	4		No Leak Recorded	No Leak Observed
46	D4	4		No Leak Recorded	No Leak Observed
47	D12	1		Flange Leak Evident	Flange Leak Evident

Nozzle No.	Core Locat.	Quadrant	1996 Inspection results	1998 Inspection results	2000 Inspection results
48	N12	2		No Leak Recorded	No Leak Observed
49	N4	3		No Leak Recorded	No Leak Observed
50	C5	4		No Leak Recorded	No Leak Observed
51	**C11**	**1**		**Flange Leak Evident**	**Flange Leak Evident**
52	E13	1		No Leak Recorded	Flange Leak Evident
53	M13	2		No Leak Recorded	No Leak Observed
54	O11	2		No Leak Recorded	No Leak Observed
55	O5	3		No Leak Recorded	No Leak Recorded
56	M3	3		No Leak Recorded	No Leak Observed
57	E3	4		No Leak Recorded	No Leak Observed
58	B8	1		No Leak Recorded	Flange Leak Evident
59	H14	2		No Leak Recorded	No Leak Observed
60	P8	3		No Leak Recorded	No Leak Recorded
61	H2	4		No Leak Recorded	No Leak Observed
62	B6	4		No Leak Recorded	No Leak Observed
63	B10	1		No Leak Recorded	Flange Leak Evident
64	F14	1		No Leak Recorded	Flange Leak Evident
65	L14	2		No Leak Recorded	No Leak Observed
66	P10	2		No Leak Recorded	No Leak Recorded
67	P6	3		No Leak Recorded	No Leak Recorded
68	L2	3		No Leak Recorded	No Leak Observed
69	F2	4		No Leak Recorded	No Leak Observed

Filed as h/RCS leakage issues/nozzle review Table

Notes:

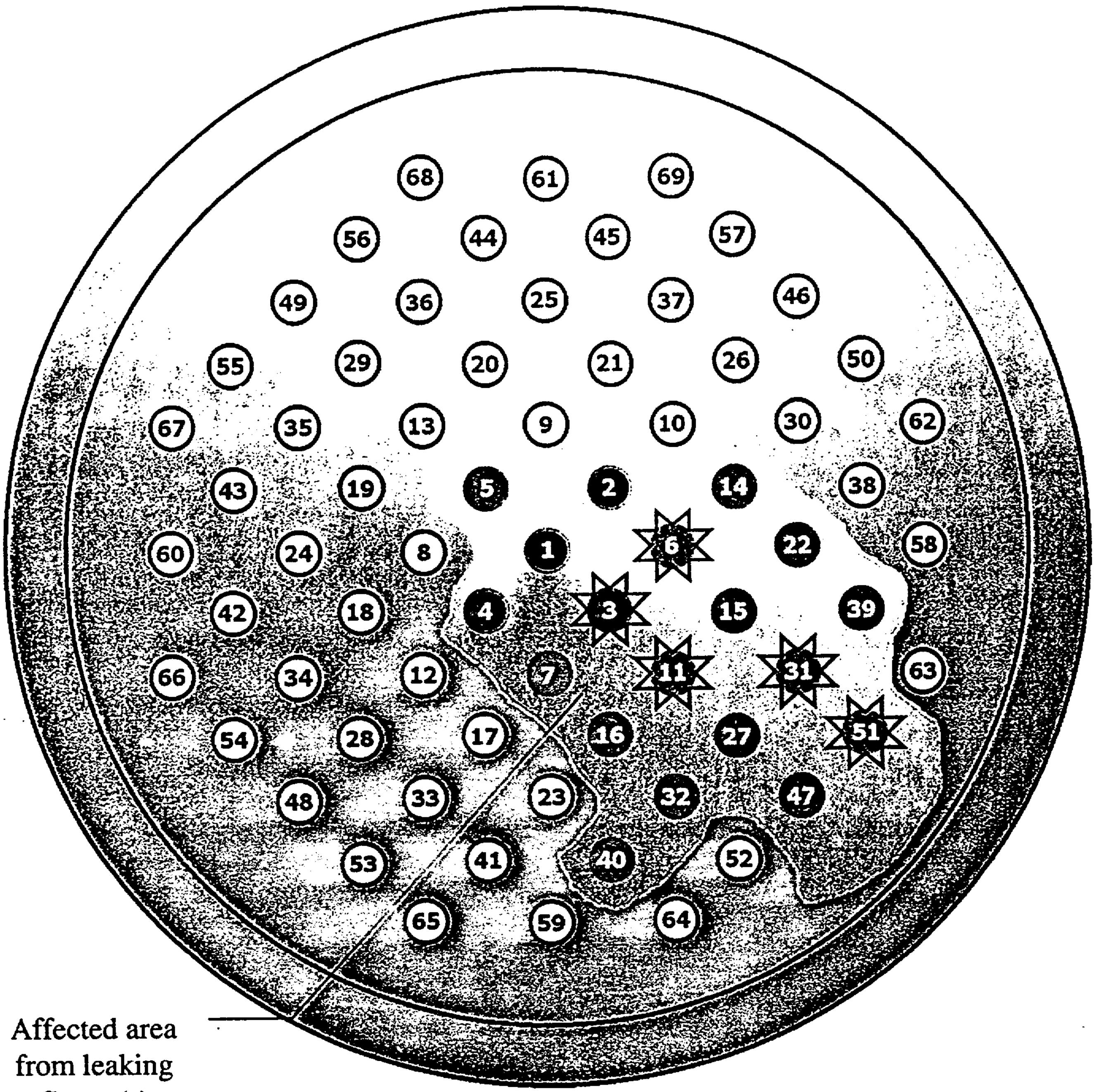
1 In 1996 during 10 RFO, the entire RPV head was inspected.
Since the video was void of head orientation narration, each specific nozzle view could not be correlated.

Bold letters indicate leaking CRDM bolting flanges discovered and repaired during 12 RFO (April 2000).
No Leak Observed = Visual Inspection Satisfactory, No Video Record Required.
No Leak Recorded = Nozzle inspection recorded on videotape
Italicized text indicates nozzles that are not expected to show leakage due to insufficient gap.

RPV Head Inspection Results

3 pages follow

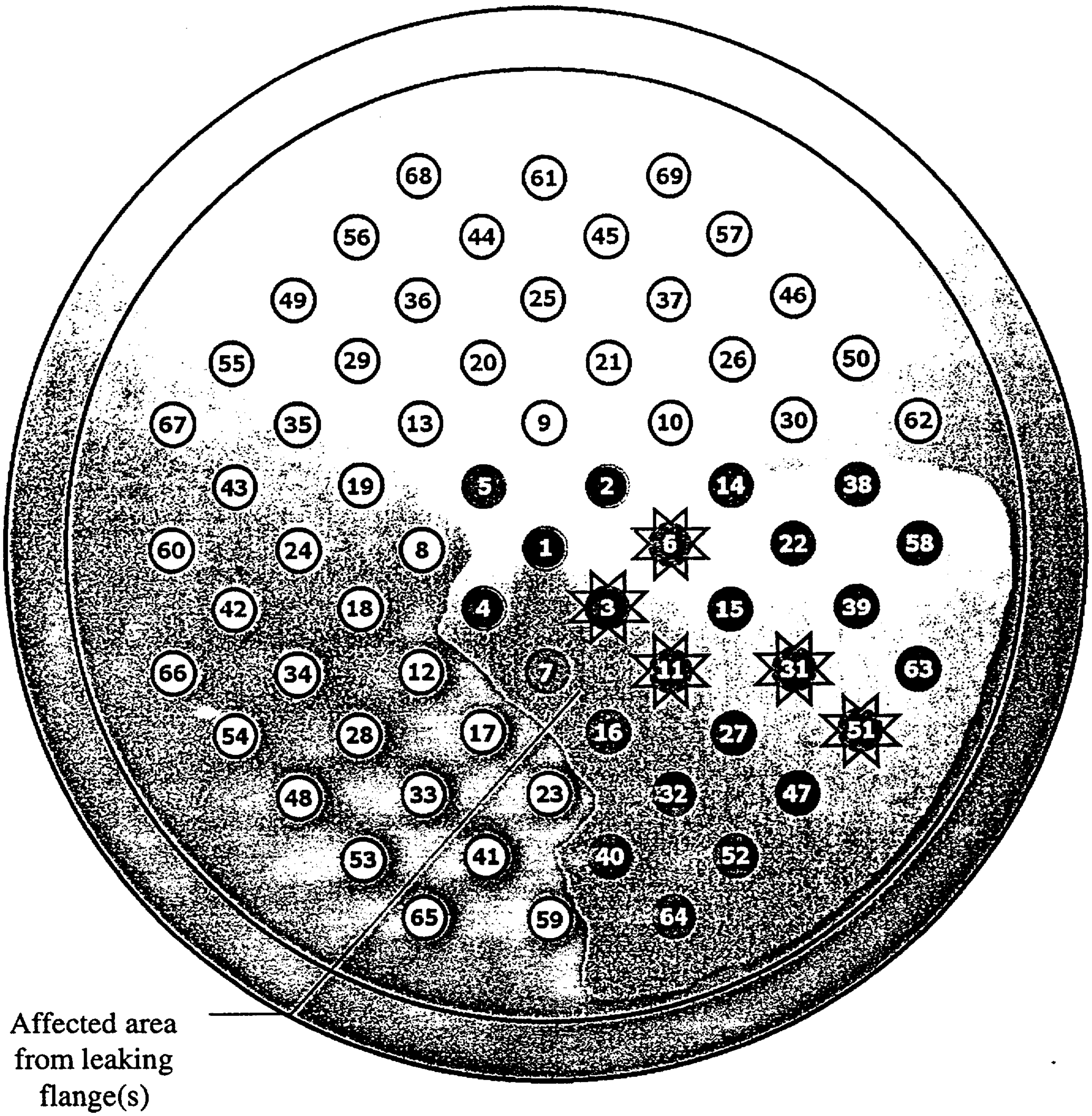
RPV Head 11 RFO Inspection Results



Affected area from leaking flange(s)

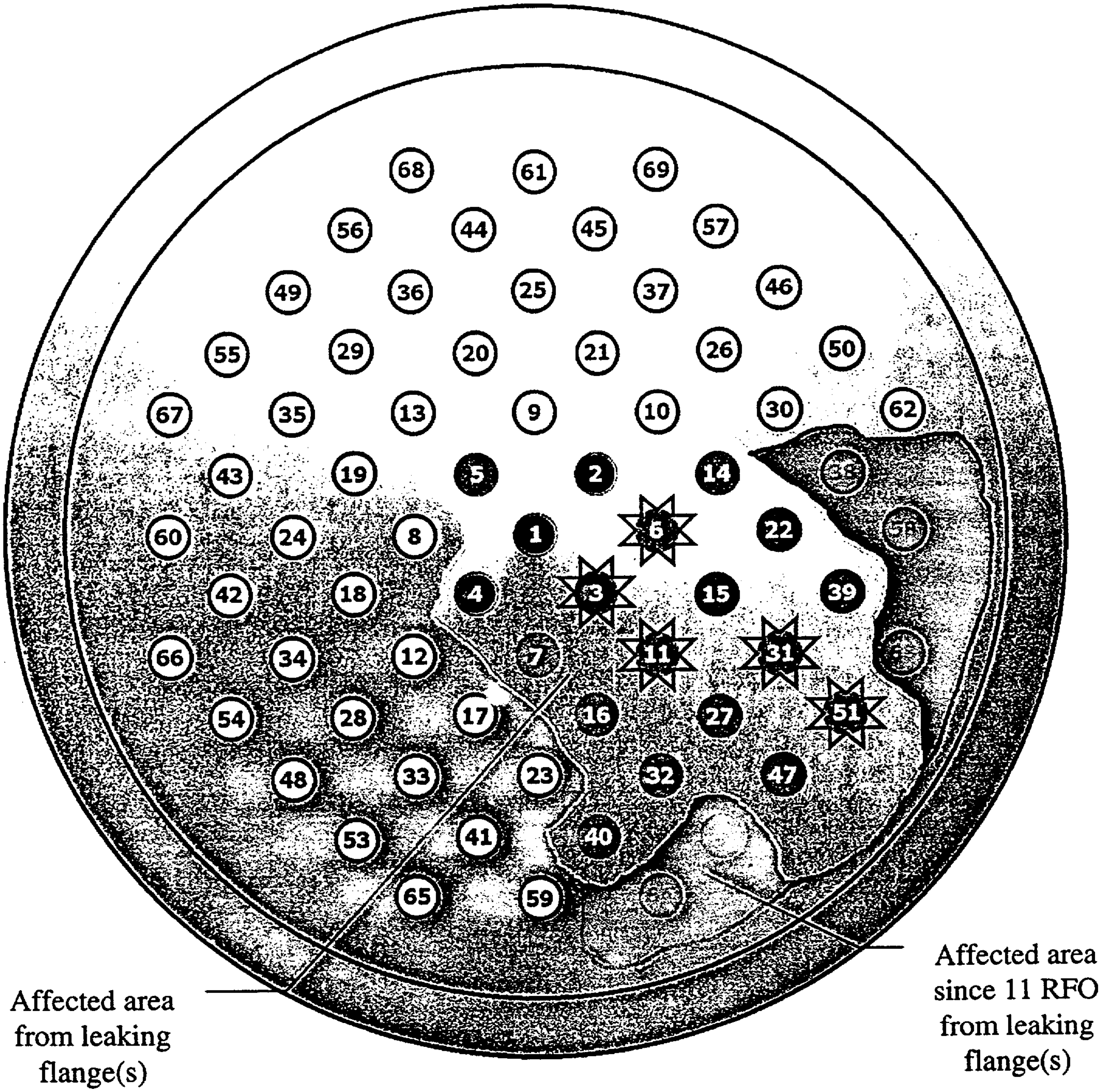
- No leakage identified
- Evaluated not to have sufficient gap to exhibit leakage
- Insufficient gap with leaking flange
- Nozzle obscured by boron
- Nozzle obscured by boron with leaking flange

RPV Head 12 RFO Inspection Results



- No leakage identified
- Evaluated not to have sufficient gap to exhibit leakage
- Insufficient gap with leaking flange
- Nozzle obscured by boron
- Nozzle obscured by boron with leaking flange

RPV Head 11 & 12 RFO Inspection Results



- No leakage identified
- Evaluated not to have sufficient gap to exhibit leakage
- Insufficient gap with leaking flange
- Nozzle obscured by boron
- Nozzle obscured by boron with leaking flange
- Newly affected, since 11 RFO, by leaking flange(s)

FRA-ANP 51-502567-01, "RV Head Nozzle and Weld Safety Assessment"
Summary Description

The attached document FRA-ANP 51-5012567-01 is a non-proprietary updated version of a previously proprietary FTI document (51-5011603-01). This document is the primary basis document for the DBNPS's assertion that it is acceptable for the plant to continue to operate until its next scheduled refueling outage scheduled to start in March 2002.

The most important portions of this document are Sections 3 and 4.

Fifty-six (56) pages follow

ATT 4
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The attached document FRA-ANP 51-5012567-01 is a non-proprietary updated version of a previously proprietary FTI document (51-5011603-01). This document is the primary basis document for Davis-Besse's assertion that it is acceptable for the plant top continue to operate until it's next scheduled refueling outage scheduled to start in March 2002.

The most important portion of this document is section 4.

It should be noted that Davis-Besse has contracted with SIA and submitted to the NRC SIA's plant specific stress analysis which closely follows the stress analysis performed by FTI for all B&W plants which is covered in section 3.

20440-8 (1/2001)

FRAMATOME ANP

ENGINEERING INFORMATION RECORD

Document Identifier 51 - 5012567 - 01

Title RV HEAD NOZZLE AND WELD SAFETY ASSESSMENT

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This document provides a safety assessment of reactor vessel head nozzles and welds that could potentially be susceptible to PWSCC in B&W-design reactors. Revision 01 incorporates information observed at ONS-2, a risk assessment, and editorial changes.

Table of Contents

1.0 Purpose

The purpose of this report is to provide an assessment that demonstrates safe operation of B&W-design nuclear steam supply systems with the potential for primary water stress corrosion cracking (PWSCC) of reactor vessel (RV) head penetration nozzles. This document addresses the assumed presence of PWSCC in either the nozzle base material or the partial penetration (or "J-groove") welds used in their attachment to the RV head. This safety assessment applies to the RV heads for the following nuclear stations:

Plant[a]	Owner
Davis-Besse (D-B)	First Energy Nuclear Operating Company
Oconee Nuclear Station Units 1, 2, and 3 (ONS-1, -2, and -3)	Duke Energy Corporation
Arkansas Nuclear One Unit 1 (ANO-1)	Entergy Operations, Incorporated
Crystal River Unit 3 (CR-3)	Florida Power Corporation
Three Mile Island Unit 1 (TMI-1)	Exelon Corporation
[a] Note: This group will subsequently be identified as the "B&WOG plants."	

Drawing on the applicable results presented in several B&WOG documents and the results of additional stress, structural, flaw tolerance and fracture mechanics analysis, the objective of this document is met. In addition, the results of a review of the existing safety analyses (Section 8) shows that defense in depth is assured.

2.0 Introduction

Cracking was first observed in a CRDM nozzle at the French pressurized water reactor (PWR) Bugey Unit 3 in 1991. Since that time, the U.S. nuclear industry has developed safety assessments (References 1-3) and several utilities have proactively inspected control rod drive mechanism (CRDM) nozzles considered to be susceptible to PWSCC.

On April 1, 1997, the Nuclear Regulatory Commission (NRC) issued Generic Letter 97-01 (Reference 4). The B&W Owners Group (B&WOG) submitted BAW-2301 (Reference 5) in response to Generic Letter 97-01, which provided details of an integrated inspection plan to address the potential degradation of RV head penetration nozzles at B&WOG plants. [It is noted that the B&WOG plants have two types of RV head penetration nozzles, which consist of CRDM nozzles at all the plants and thermocouple nozzles at ONS-1[a] and TMI-1 only.]

All B&W-design reactors were designed, fabricated, erected, constructed, tested, and continue to be inspected in compliance with 10CFR50.55a (Reference 6). In particular, the RV head penetration nozzles were designed, fabricated, and manufactured to have a low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture in accordance with General Design Criterion 14 of Appendix A to 10CFR50. The Alloy 600 material utilized for these RV head penetration nozzles is an austenitic material that is very ductile and meets the requirements set forth in General Design Criterion 31 of Appendix A to 10CFR50. Finally, accessibility to the RV head is available to assess the structural and leak tight integrity of the RV head penetration nozzles in compliance with General Design Criterion 32 of Appendix A to 10CFR50.

The discovery of the J-groove Alloy 182 weld cracking at ONS-1 and the circumferentially-oriented flaw indications revealed at ONS-3, ONS-2, and ANO-1 have introduced new concerns that must be addressed. This document provides a bounding safety assessment to address the potential severity of these concerns at ONS, ANO-1, and the other B&WOG plants.

2.1 Background

The 1993 B&WOG safety evaluation (Reference 3) presented a stress analysis, crack growth analysis, leakage assessment, and wastage assessment for potential inside surface PWSCC of the B&W-design CRDM nozzles. Based on the results of the stress analysis performed, it was concluded that the peak hoop stresses are greater than axial stresses on the inside surface of the nozzle. Also, the maximum hoop stress is similar for both the center and peripheral nozzles. Thus, if an inside surface crack were to develop in a CRDM nozzle due to PWSCC, the cracks would mainly be axially oriented. It was conservatively concluded that safe operation of the B&W-design plants will not be affected for at least six years (operating with adequate leakage to corrode the RV head), and that within this time, the leak will be detected during a walk-down inspection of

[a] The thermocouple nozzles were removed from ONS-1 at EOC-19.

the RV head area. Thus, the potential for cracking of CRDM nozzles does not present a near-term safety concern.

The same nozzle containing a through-wall crack at Bugey-3 also exhibited an indication of circumferential cracking on its outside surface. In this case, the initiation and propagation of the axial crack preceded exposure of the outer surface of the nozzle above the weld in the annulus to leaking reactor coolant. An addendum to the B&WOG safety evaluation was prepared to address this concern in December 1993 (Reference 7). It was concluded in this evaluation that ample leakage through the penetration would occur to allow detection. In addition, the occurrence of nozzle detachment is highly unlikely during the design life of the B&WOG plants since actions would be taken to repair the nozzle prior to a nuclear safety concern existing.

During a CRDM nozzle inspection at Ringhals Unit 2 in 1992, an indication was detected in the nozzle-to-vessel (J-groove) weld at one penetration. The indication was not indicative of PWSCC; rather, the indication was attributed to a weld defect that occurred during fabrication of the CRDM nozzle to the RV weld. The B&WOG took action to address this concern by acquiring additional data from several sources. First, the data from Ringhals Units 2 and 4 and data from a cancelled Westinghouse reactor, Shearon Harris, were acquired from the Westinghouse Owners Group (WOG). Second, the B&WOG performed an inspection of the RV head from Midland Unit 1, which was a cancelled nuclear station fabricated by B&W.

Another addendum to the B&WOG safety evaluation was prepared to analyze these data (Reference 8). This evaluation included a statistical review and analysis of the J-groove weld inspection data and a stress analysis of the CRDM J-groove weld to determine the minimum weld area that is required to meet the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code primary shear stress limits. It was shown in this report that the maximum areas of weld lack of fusion detected for the Midland Unit 1, Shearon Harris, and Ringhals Unit 2 RV closure heads are well below the ASME Code allowable limits for weld structural integrity. It was concluded that a large margin exists between the statistical bound of the total lack of weld fusion areas in the Midland Unit 1 head and the ASME Code allowable limits. Therefore, the observed lack of fusion areas do not give rise to a safety concern.

In addition, Generic Letter 97-01 requested a description of resin intrusions that may have occurred at the B&WOG plants. The B&WOG response (Reference 5) included a review of plant historical records regarding sulfate excursions. Also, the results of primary water chemistry analysis at each of the B&WOG plants were reviewed for excursions from out-of-specification conditions. Based on these data, it was concluded that the potential for intergranular attack (IGA) or stress corrosion cracking (SCC) of CRDM and thermocouple nozzles was very low.

2.2 B&WOG Plant Inspections

All B&WOG Plants

As described in References 3 and 5, leakage of B&W-design flanges has previously been experienced at each of the B&W-design plants, and visual inspections of the RV head area have been implemented so that flange leaks can be identified and repaired as soon as possible. Primary water that exits from a leaking flange quickly flashes to steam, leaving behind a "snow" of boric acid crystals. Exposure of the RV head to dry boric acid crystals from this type of leakage has not resulted in wastage of the RV head.

The B&WOG utilities have included plans to visually inspect the CRDM nozzle area to determine if leakage is observed on top of the RV head, which would indicate through-wall cracking has occurred, during their outages. In addition, walk-down inspections have been implemented in response to NRC Generic Letter 88-05 (Reference 9) at each of the B&WOG plants. The walk-down inspections include an enhanced visual inspection of the gasket area and RV head during every refueling outage (12-24 months). The B&W closure head and service structure design provides access for a visual or boroscopic examination of the CRDM nozzle area, since the insulation is not resting on the RV head (see Figure 1). If any leaks or boric acid crystal deposits are noted during inspection of the RV head area, an evaluation of the source of the leak and the extent of any wastage is performed. This program has shown to be effective, as evidenced at ONS and ANO-1. These visual examinations provide an acceptable level of quality and safety and are in accordance with 10CFR50.55a and General Design Criteria 30 of Appendix A to 10 CFR50.

BAW-2301 (Reference 5) also describes the ASME B&PV Code Section XI, Article IWB 2500 inspections performed by all the B&WOG plants. In addition, a plant-specific inspection of a CRDM nozzle and a thermocouple nozzle was performed by TMI-1 in 1982 as a result of intergranular attack on the steam generator tubes.

Most recently, NRC Bulletin 2001-01, "Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles" (Reference 10) was issued, requesting plant-specific information regarding the structural integrity of the RV head nozzles and extent of leakage and cracking that has been found to date. Information was also requested regarding inspections and repairs that have been completed and those planned in the future to satisfy regulatory requirements, and the basis for concluding that those plans will ensure compliance with the applicable regulatory requirements. Each of the B&WOG member utilities prepared a response that provides this information (References 11-15). A summary of the plant-specific inspections is described below.

Oconee Unit 2

Duke Energy volunteered to perform an inspection[b] of all 69 CRDM nozzles (from the nozzle ID) at ONS-2, which was ranked as one of the B&WOG plants potentially susceptible to PWSCC, in 1994. All indications identified at ONS-2 in 1994 were confined to nozzle number 23 and consisted of 20 indications predominantly axial in orientation. These indications were detected with an eddy current technique and confirmed with dye penetrant testing. An ultrasonic technique could not size the indications on nozzle number 23 because they were too shallow; therefore, the depth was conservatively assumed to be 2 mm (0.079 inch). These indications were subsequently identified as "craze-type" flaw indications. The 1994 eddy current results of a group of eleven other nozzles (numbers 16, 45, 46, 50, 52, 56, 57, 60, 63, and 65) indicated high noise areas, with nozzle number 63 exhibiting the most severe noise of the group. Both ultrasonic and dye penetrant examinations were completed on nozzle number 63 without identifying any indications. Based on this additional information, this group of nozzles with high noise was collectively dispositioned as non-reportable indications.

Both rotating eddy current and dye penetrant examinations were completed on nozzle numbers 23 and 63 during the re-inspection work at ONS-2 in 1996.[c] Multiple indications were observed (i.e., small craze-type flaw indications) that were predominantly axial in nature. Separate eddy current data acquisitions were completed on both nozzles before and after use of a honing cleaning technique to evaluate the effect of the cleaning on the eddy current results. They were confirmed to be in the same location as the high noise areas detected with eddy current in nozzle number 63 in 1994, thus explaining the cause of the noise that was previously unknown.

In 1999, rotating eddy current inspection results for nozzle numbers 23 and 63 at ONS-2 again showed no significant change when the data were compared to the 1994 and 1996 results. Thus it was concluded that the indications had not grown or changed since the 1994 inspection. Rotating eddy current results in 1999 on nozzle numbers 16, 21, 46, 50, 62, and 68 were also obtained and evaluated against the 1994 results. As with all the previous data comparisons, no significant change in the data was observed when compared to the 1994 data.

[b] Non-destructive examination techniques were developed and initially qualified for ID inspection of CRDM nozzles.
[c] Significant development work was completed to improve both the eddy current and liquid penetrant methodologies.

Thus, it was concluded that the indications in these nozzles had not grown or changed since the 1994 inspection.

ONS-2 most recently performed a routine visual inspection of the RV head during a refueling outage in April 2001. Boric acid crystals were observed at four CRDM nozzles (numbers 4, 6, 18, and 30). Liquid penetrant examination identified OD crack-like axial indications below the weld on all four nozzles. Ultrasonic examinations showed that these indications were OD-initiated and that none of the indications were through-wall. An OD-initiated circumferential indication, 0.07 inch in depth and 1.25 inch in length (approximately 36 degrees circumferential extent), was noted above the weld on nozzle number 18 (Reference 11). Eddy current examinations of the ID of the nozzles revealed shallow craze-type flaw clusters in all four nozzles that were distributed around the entire ID circumference (i.e., 360°, above the weld). Based on these results, the leak path was through the interface between the nozzle and the J-groove weld.

The repair at ONS-2 consisted of an automated repair. The four CRDM nozzles were roll-expanded in the upper portion of the RV head in the area of the repair. The bottom portions of the CRDM nozzles were machined out to an elevation above the original structural weld. The machined RV head bores in the area for the new weld and the weld preparations in the CRDM nozzles were liquid penetrant examined to verify that there were no rejectable indications in these areas. The CRDM nozzles were welded to the RV head in accordance with the ASME Code using the temper bead technique using Alloy 52 weld material. The welds were then liquid penetrant and ultrasonic examined. The final operation was to perform an abrasive water jet remediation of the rolled and welded regions.

<u>*Oconee Unit 1*</u>

As part of a routine visual inspection of the RV head during the ONS-1 refueling outage (November 2000), boric acid crystals were observed at one CRDM nozzle location (number 21) and at five of the eight thermocouple nozzle locations.

Eddy current examination of the inside surfaces of the thermocouple nozzles showed that all eight nozzles contained crack-like indications and that these were predominantly axial in orientation. Ultrasonic examinations from the inside surface of the thermocouple nozzles allowed the weld size to be determined and the axial crack-like indications to be located. Liquid penetrant examination of the J-groove welds (after boring out the nozzles), showed that some cracks had penetrated through the nozzle walls, and that the orientation of these cracks was predominantly axial at the plane where the cracks penetrated into the welds. All eight (8) of these nozzles have been removed by sealing the RV head penetration with a more corrosion resistant material (Reference 11).

Eddy current examination of the inside surfaces of CRDM nozzle 21 and seven other locations (42, 49, 55, 56, 61, 67, and 68) was performed. All eight of the CRDM nozzles contained craze-type indications located in clusters in the uphill region, both above and below the weld. Ultrasonic examinations were performed on the inside surface of 18 nozzles (numbers 17, 21, 22, 28, 34, 42, 47, 48, 49, 52, 54, 55, 56, 61, 62, 66, 67, and 68). No crack-like indications were detected. A liquid penetrant examination was performed on the partial penetration weld of nozzle 21. Two Code acceptable small rounded indications were found. After lightly grinding and performing another penetrant examination, a 0.75 inch radial indication running at a slightly skewed angle across the fillet weld was identified. This crack was ground out of the weld and nozzle material. It extended into the nozzle material approximately 0.4 inch and ran radially out from the nozzle penetrating through the weld and through the butter layer in one location. This crack was identified as the leak source since the annulus was exposed prior to the crack being fully removed.

In summary, cracking was identified in the CRDM nozzle J-groove weld and continued from the weld into the OD of the nozzle. Cracking was also identified in the weld and nozzle ID of the eight thermocouple nozzles. The cracking mechanism was attributed to PWSCC. All indications at these nine (9) locations were removed and weld repairs were performed.

Oconee Unit 3

As part of a routine visual inspection of the RV head during an ONS-3 outage (February 2001), boric acid crystals were observed at nine CRDM nozzle locations (numbers 3, 7, 11, 23, 28, 34, 50, 56, and 63). In addition to observations of through-wall axial flaws above the weld, outside surface circumferential indications (relatively deep and located below the weld) were present on four nozzles (numbers 11, 23, 50, and 56). Also, outside surface circumferential indications above the weld (one through-wall and one nearly through-wall) were present on two nozzles (numbers 50 and 56). Another outside surface circumferential indication above the weld, which was relatively shallow (0.22 inch deep), was present on nozzle number 23. It appeared that these particular cracks initiated from the nozzle OD following exposure to leaking primary water from a through-wall axial flaw (similar to the Bugey-3 cracking). This would require an axial crack, and ultimately a leak path, to either propagate through the weld or the nozzle surface adjacent to the weld. The fact that nozzles 50 and 23 had a circumferential crack at the OD that had not propagated through-wall supports the assertion that an axial flaw is needed prior to a circumferential flaw being formed. The existence of seven (7) other nozzles with at least one axial indication connected to the OD surface of the nozzle at ONS-3 also supports this position.

Liquid penetrant examination of the weld excavation area in nozzle number 34 also revealed a circumferentially oriented flaw indication. Several PT examinations performed during the excavation process revealed that this particular indication spanned about 2-inches in length and was located in the J-groove weld. This indication did not appear to extend to the root of the weld. Shallow axially oriented inside surface indications were also observed in areas of craze-type cracking above and below the weld (similar to those observed at ONS-1 and 2) in virtually all the nozzles examined.

The circumferential cracks discovered at ONS-3 ranged from about 2-mm (0.079 inch) in depth to through-wall. In nozzle number 11, a circumferentially oriented OD crack (below the weld) crossed the path of three axial cracks, the circumferential crack was 0.380 inch deep (61% through-wall), with a 31% circumferential extent (Reference 11). In nozzle number 56, a circumferentially oriented OD crack (above the weld) was through-wall and extended approximately 165° around the nozzle. The circumferential crack in nozzle number 50 was nearly through-wall (i.e., pinhole indications were observed on the ID during liquid penetrant testing) and extended approximately 59° around the nozzle. The circumferential extent and through-thickness depth (from the OD) of nozzle number 23 were 66° around the nozzle and 0.22 inch (Reference 11).

All through-wall cracking observed below the weld appears to have initiated near the toe of the fillet weld. It is most likely associated with the residual weld stresses introduced in this area during manufacturing. Some shallow cracking was also observed on the outer surface at the end of several nozzles (e.g., number 28 and 56).

Ultrasonic examinations were subsequently performed on an additional set of nine nozzles (numbers 4, 8, 10, 14, 19, 22, 47, 64, and 65). From these nine nozzles, eight of them showed no indications. Nozzle number 4, however, showed four shallow axially oriented flaws, all on the inside surface and above the weld. Also, eddy current examinations were performed for these additional nine nozzles. For nozzle number 4, the eddy current results confirmed the findings from the ultrasonic examination. In addition, the eddy current examination revealed shallow craze-type flaw clusters that were found in four nozzles (numbers 8, 10, 14, and 22) and distributed around the entire ID circumference (360°, above the weld).

Arkansas Nuclear One Unit 1

Following shutdown for a scheduled refueling outage (March 2001), ANO-1 performed a routine visual inspection of the RV head area. This inspection revealed boric acid crystals in the area of one CRDM nozzle (number 56). Based on the visual inspection results, liquid penetrant (PT), ultrasonic (UT), and eddy current (ET) examinations of the CRDM nozzle and PT of the J-groove weld were also performed. The leak path was determined to be an axial flaw in the nozzle outside diameter that extended beyond each side of the weld. The PT examination identified a circumferential crack, approximately 0.70 inch long in the outside diameter of the CRDM nozzle below the J-groove weld (Reference 12). This crack branched twice and each of the three resulting tributaries extended off-axial (nearly axial) up to the weld fusion line. There is no firm evidence that any cracking occurred in the weld. The UT examination indicated that the subsurface dimensions of the crack extended in a circumferential and then off-axial direction below the weld and in an axial direction through the nozzle past the weld to a termination point 1.3 inches above the weld on the nozzle OD (in the annulus region). The flaw depth dimension was estimated to be a maximum of 0.20 inch into the nozzle wall and thus never penetrating to the nozzle ID surface. It would appear from the NDE evidence that the cracking was confined to the nozzle material, which became the leak path. The ET and PT examination confirmed that the crack had not propagated to the inside diameter of the CRDM nozzle. This flaw is consistent with the PWSCC experience that has occurred at ONS-1. This event also reaffirmed the effectiveness of examining the RV closure head for leaks as means of assuring the avoidance of a safety concern.

3.0 Stress Analysis Efforts

3.1 Summary of Stress Analyses Performed

Nonlinear elastic-plastic finite element analysis was performed in 1993 to characterize stresses in the Alloy 600 CRDM nozzle (SB-167 tube material), the low alloy steel head, the stainless steel cladding in the head, and the Alloy 182 weld material used for the partial penetration weld and butter between the nozzle and head. The purpose of this analysis was to determine the preferential direction for cracking based on the relative magnitude of longitudinal (axial) and circumferential (hoop) stresses. Results were also used to predict crack growth by PWSCC and to support leakage assessments for postulated through-wall cracks in the nozzle wall (Reference 3).

Two bounding nozzle configurations were addressed in the 1993 stress analysis, the center nozzle and one of the outermost peripheral nozzles (hillside nozzle). Taking advantage of full symmetry of the top of the reactor vessel head, the center nozzle was analyzed using a two-dimensional model. Since the outer hillside nozzle penetrates the head at an angle of 38.5 degrees, a 180 degree three-dimensional model was utilized at this location to address the more complicated stress fields associated with an oblique penetration, due in part to ovalization of the nozzle under pressure and thermal loads. The following loading conditions were considered in the 1993 stress analysis to determine long-term sustained stress in the nozzle and weld materials (and to a lesser extent in the head):

1) Shrink fit of the nozzle within the head during installation (0.0010 inch nominal diametric interference).
2) Simulated welding of the nozzle to the head (heatup of the weld material to 2470 °F and cooldown to develop residual stresses).
3) Cold hydrostatic testing of the completed head assembly at a pressure of 3125 psig.
4) Steady state operation at a temperature of 600 °F and a pressure of 2250 psig.

Residual stresses from the welding process are strongly dependent on plastic deformation in the nozzle. Yield strengths for B&W-design plants range from 31 ksi to 64 ksi. For the higher yield strength nozzles, more residual stress is locked in as the weld puddle cools from its molten state. The 1993 stress analysis used the 64 ksi nozzle yield strength as a bounding value.

3.2 Nozzle and Weld Stresses

Since at most locations the inside surface hoop stress is higher than the axial stress, the preferential direction for cracking is axial (in a radial plane relative to the nozzle). Exceptions occur at the lower end of the nozzle and above the weld. Some circumferential cracking may occur on the outside surface of the nozzle, just below the weld, where hoop and axial stresses are similar in magnitude on the uphill side. Axial stresses would also promote the propagation of OD initiated circumferential cracks above the weld.

3.3 Flaw Growth Evaluations

Evaluations of flaw growth from PWSCC have been performed for the J-groove weld and CRDM nozzle as discussed below. Axial ID nozzle flaws were addressed in the original safety evaluation for cracking of B&W-design CRDM nozzles (Reference 3).

3.3.1 Axial J-Groove Weld and OD CRDM Nozzle Flaws

As discussed above, the dominant hoop stress in the J-groove weld would promote axial cracking of this Alloy 182 material. Due to the relatively high crack growth rates observed in autoclave tests with this weld metal in a PWR environment (Reference 16), and considering the increasing stress gradient away from the inside surface of the weld, crack growth through the J-groove weld would be expected. Although the flaw would arrest at the low alloy steel RV head (see Section 4.0), the flaw would continue to grow into the Alloy 600 CRDM nozzle, as seen at ONS-1 (Reference 17).

Calculations were performed to predict the time it would take to grow an axial outside surface flaw (OD flaw) through the nozzle to the inside surface. Assuming a length-to-depth ratio of six, using the Peter Scott crack growth model for Alloy 600 in a PWR environment, and considering the highest stressed location, it would take almost four years for an axial OD flaw that is initially 0.5 mm (0.02 inch) deep to grow through-wall. It has already been reported (Reference 3) that it would take at least four more years for a through-wall flaw to extend two inches above the weld, thereby creating a leak path into the annular region between the nozzle and head.

3.3.2 Circumferential OD CRDM Flaws

Since the OD surface hoop stresses in the weld are about two times the surface axial stresses; flaws originating at this location should be oriented in an axial plane. Development of a leak path through the weld to the annulus between the nozzle and RV head would however, expose the outside surface of the nozzle to the primary water environment. Since there is a band of high axial stress on the outside of the nozzle just above the weld, initiation of a circumferential crack at this location is a concern. Based on experience at ONS-3, the development of an axial leak path through the weld and/or nozzle would precede initiation of a circumferential OD flaw on the outside surface of the nozzle above the weld. Furthermore, as observed at ANO-1, deposits of boric acid crystals on the top of the head would provide evidence of a leak path prior to the initiation of a circumferential OD flaw. For the purpose of performing crack growth calculations, it is conservatively assumed that a small flaw, 0.5 mm (0.02 inch) in depth, initiates immediately after the plant is returned to service. Using 0.5 mm (0.02 inch) as the initial depth of an isolated OD initiated circumferential flaw above the weld, it would take more than 10 years for a short (l/a = 6) semi-elliptical surface flaw to grow through-wall. At ONS-3, following the growth of an axial flaw to the annulus between the nozzle and head, there were apparently several initiation sites that linked to form a long circumferential OD outside surface crack above the weld, extending nearly half way around the circumference. Such a flaw could grow through-wall in 3.5 years. Even then, it would take another 4 years for the through-wall flaw to grow another 25% around the circumference. The remaining ligament, which would then be 25% of the original circumference, would still be sufficient to preclude gross net-section failure (nozzle ejection). This ligament satisfies primary stress limits using a safety factor of 3 (Reference 18).

Lack of fusion weld defects between the nozzle and weld, of the type detected at Ringhals Unit 2 and at the cancelled Shearon Harris and Midland plants, should also be considered in light of the potential for CRDM nozzle J-groove weld cracking. This flaw is described as a "wrap-around" circumferential flaw along the cylindrical surface at the nozzle-to-weld interface. As discussed in Reference 8, there may be up to 67% lack of fusion between the nozzle and weld before the ASME Code primary shear stress limits are violated. It has been calculated that it would take two years for a 0.25-inch wrap-around flaw to grow to the 67% limit. This is based on a conservative value of 45 ksi for the average radial stress between the nozzle and weld, and utilizes the high crack growth rates observed in laboratory testing for Alloy 182 weld metal (Reference 16). Based on observations at ONS-1, ONS-2, ONS-3, and ANO-1, where there was no evidence of wrap-around cracking between the nozzle and weld, this is an extremely conservative crack growth prediction.

A 2-inch long circumferentially oriented flaw indication was observed in nozzle 34 at ONS-3. It was located in the weld material and spiraled from a distance of $1^1/_8$ inch from the OD of the nozzle on the uphill side to 0.75 inch from the nozzle as it went about 45° around the weld. Being located in the weld, this laminar-type anomaly is not considered to be a safety concern, since it did not provide a leak path to the environment and it could not lead to ejection of the nozzle.

4.0 Flaw Growth Into the RV Head

A crack, propagating through the J-groove weld by PWSCC, will eventually grow to the RV head (low alloy steel) and the CRDM nozzle (Alloy 600). It is expected that the resultant crack will continue to propagate through the CRDM nozzle material as observed at ONS-1 (Reference 17) and ONS-3, in a direction determined by the residual stress distribution. However, continued flaw growth into the low alloy steel is not expected to occur.

Stress corrosion cracking (SCC) of carbon and low alloy steels is not expected to be a problem under BWR or PWR conditions (Reference 19). SCC of steels containing up to 5% chromium is most frequently observed in caustic and nitrate solutions and in media containing hydrogen sulfide (References 20 and 21). A recent review of literature results was performed by Framatome ANP, which also concluded that SCC of low alloy steel materials is non-credible in PWR environments (Reference 22). Based on this information, SCC is not expected to be a concern for low alloy steel exposed to primary water.

Instead, an interdendritic crack propagating from the J-groove weld area is expected to blunt and cease propagation. This has been shown to be the case for interdendritic SCC of stainless steel cladding cracks in charging pumps (References 23 and 24) and by recent events with PWSCC of Alloy 600 weld materials at ONS-1 and VC Summer (References 25 and 26). Although a PWSCC-initiated flaw may continue to propagate by fatigue crack growth into the low alloy steel head, this is considered to be insignificant over several operating cycles based on anticipated cyclic loads. Since borated water will now be in contact with the low alloy steel, corrosion wastage of the material is expected to occur. This is addressed in Section 6.0 below.

5.0 Leakage Assessment

The B&WOG has performed leakage assessments for various potential leak scenarios expected prior to the recent leak events at ONS-1, ONS-2, ONS-3, and ANO-1. The results from these assessments are documented in detail in Appendix A. The recent experience, however, indicates that the leak rates are apparently very low based on the amount of boric acid crystals observed on leaking nozzles. It was estimated that approximately 0.5 in^3 was present around CRDM nozzle number 21 at ONS-1. In the case of the ONS-1 thermocouple nozzles, five (5) were suspected to have leaks while the other three (3) did not exhibit evidence of boric acid crystals. The examinations subsequently performed on all eight (8) nozzles revealed cracking that would strongly suggest a leak path. It is reasoned that a small leak and narrow annulus can lead to "leak plugging" by the formation of less dense metal oxides in the annulus. Thermal cycling is anticipated to lead to starting or re-initiating a weeping type leak. Therefore, leakage is anticipated to be minimal until a long axial flaw (i.e., approximately the length of the RV head penetration) develops above the weld.

6.0 Wastage Assessment

The purpose of this section is to assess the potential damage that can occur to the RV head as a result of a leaking CRDM nozzle or J-groove weld. Two areas of concern are considered in this discussion:

1) General corrosion damage to the reactor vessel head as a result of exiting boric acid crystals and borated steam condensing on the head insulation from a through-wall crack in a CRDM nozzle or J-groove weld.

2) Corrosion damage both within and in the vicinity of the reactor vessel head penetration due to boric acid corrosion resulting from a through-wall crack in the CRDM nozzle or J-groove weld.

A leaking CRDM nozzle or J-groove weld is of concern because the leaking primary coolant, containing boron in the form of boric acid, can be very corrosive to carbon and low alloy steel materials when subjected to certain environmental conditions. Several studies have been performed to determine these conditions.

A description of the testing performed and their respective results is given in References 3 and 27.

Reference 3 includes a corrosion damage assessment for a variety of conditions and leakage rates assumed to occur with CRDM nozzles. As noted above in Section 5, similar assumptions can be made for the case of leakage that is associated with PWSCC of RV head J-groove welds.

It was determined in Reference 3 that this type of leakage would lead to corrosion of the RV head penetration, at a maximum volumetric metal loss rate of 1.07 in^3/yr. Three defect profiles were postulated to model this level of corrosion for a time period of six years. It was concluded through an ASME B&PV Code evaluation for membrane stresses in the RV head, that safe operation of the plant would not be affected as a result of this level of corrosion of the RV head penetration.

Finally, it was concluded that safe operation of the B&W-design plants will not be affected for at least six years, and that within this time, the leak will be detected during a walk-down inspection of the RV head area. It should be noted that this minimum six-year period represents corrosion of the RV head at the maximum rate of 1.07 in^3/yr, which would only occur when a sufficient leakage rate has been realized. Thus, the potential for cracking of CRDM nozzles and RV head J-groove welds does not present a near-term safety concern. The validity of these assumptions and conclusions was recently verified by the detection of boric acid crystal deposits around CRDM and thermocouple nozzles and the subsequent identification of RV head J-groove weld leakage at ONS-1, ONS-2, ONS-3, and ANO-1. In all cases, only minimal corrosion (wastage) was observed.

7.0 Loose Parts Assessment

As noted earlier, circumferential cracking has been observed on the outside surface of leaking CRDM nozzles at ONS-3. This cracking occurred at the toe of the fillet weld that forms part of the structural attachment to the RV head. In some of these nozzles through-wall axial cracking has also been observed in the nozzle base metal below the weld. Thus, there is a concern that a through-wall circumferential crack could link up with two or more through-wall axial cracks and form a loose part. An assessment of the potential consequences associated with CRDM nozzle fragmentation has been performed (Reference 28). The potential transport of fragments originating at the reactor vessel head penetration were identified and evaluated.

If a piece of the CRDM nozzle were to break away, it could potentially end up in one of three places. The first location is the stainless steel plate around the column weldments (plenum cover) where it would not have an impact on any safety or operational issue in the plant (see Figure 2). The second location is through the gaps around the periphery of the plenum cover and would likely end up in the steam generator, potentially damaging the tubes or tube welds. A fragment lodged within a single tube could, as a result of motion induced by the flow through the tube, cause wear of the tube at the point of contact with the inside surface. Although unlikely, this could eventually result in a small through-wall flaw in the tube, causing a primary-to-secondary leak, which can be detected by monitoring procedures already in place at the plant. Once detected, the plant operators would follow the technical specification action statements to shut down if the leak became significant. This does not introduce any new or unanalyzed event. While this location may cause equipment damage, it is not a safety concern. The third possibility, which could be a safety concern, is that the pieces could enter any one of the 69 column weldments through which the control rod spiders descend (see Figures 5 and 6). It has been calculated that there is a 25% chance or greater for a loose piece to enter one of the column weldments. This is simply based on an area ratio of the column weldments in the upper head and the fact that low cross flow velocities in this region would tend to allow debris to fall vertically. In addition, the leadscrews could tend to guide the debris such that the probability of entering the column weldment may be much higher than 25%. If fragments enter the column weldments, they will likely be stopped on one of the control rod guide tube brazements where relatively small fragments (< ¾ inch) would be capable of precluding complete control rod insertion.

Based on experience at ONS-3, circumferential and axial cracking below the weld is accompanied by through-wall axial cracking at and above the weld. The ONS experience coupled with the extensive examinations performed in Europe, and the stress analysis results described in Section 3.0 indicate that the predominant cracking orientation is axial.

In addition, there have been a total of 27 non-leaking nozzles at both ONS-1 and ONS-3 subjected to both eddy current and ultrasonic examinations. Very shallow craze-type cracks were revealed above and below the welds. No OD cracks were detected at the nozzle-to-weld intersection (below the weld) for these 27 nozzles. In each case, these nozzles were found to be free of cracking. These observations and results support the assertion that there is a high probability that detectable leakage would precede the development of a loose part.

8.0 Safety Analysis Review

In this section, the plant safety analyses will be reviewed to determine if a safety issue exists and to provide justification that the consequences of a failure of a single CRDM nozzle are bounded by the existing plant safety analyses and will support plant restart and continued operation.

Loss of coolant accidents (LOCA) and non-LOCA safety analyses are performed to justify that the nuclear power plants can be safely shut down following postulated accidents. Although these analyses do not specifically consider failure (i.e., complete severence) of a CRDM nozzle, they consider events that have more limiting consequences. LOCA analyses typically postulate breaks in RCS pipes from those within the plant makeup capacity up to and including a double-ended guillotine break of the hot leg to demonstrate acceptable core cooling in the short term as well as the long term. Non-LOCA safety analyses specifically postulate a control rod ejection accident, although the CRDM nozzle remains intact. The rod ejection event postulates that the CRDM flange bolts fail and the control rod is ejected out of the CRDM housing. These plant safety analyses are reviewed in the following paragraphs to determine if a more substantial safety issue exists based on the leaks that have been observed at ONS-1, ONS-2, ONS-3, and ANO-1. Where applicable, additional margin is identified to further support plant restart and continued safe operation.

As described in the previous sections, once a crack initiates, it is estimated that it may take up to six years for it to migrate through the CRDM components and begin to leak at undetectable rates. Detection of such minor leaks that grow at slow rates is by visual inspections of the CRDM nozzles as noted with the ONS-1, ONS-2, ONS-3, and ANO-1 outages. These routine inspections of the potentially affected areas will identify if any leak has initiated well before the weld or component could fail catastrophically. The detected cracks have grown predominantly in the axial direction, although some circumferential cracks have been observed near the weld. These as-found circumferential and axial cracks have been evaluated, and it was concluded that the structural integrity of the component retains sufficient margin to ensure continued safe operation of the plant. In addition, the maximum projected growth rate from the boric acid corrosion of the RV head penetration from a minor leak would not propagate into adjacent CRDM nozzle failures. Therefore, simultaneous catastrophic failure of multiple nozzles will not be postulated.

Since failure of multiple CRDM nozzles is not considered credible, the primary concern is the failure of a single nozzle. This unlikely, yet postulated failure leads to RCS inventory loss and less core shutdown margin for the plant safety analyses. These aspects are addressed in the following paragraphs relative to the consequences already included in the existing LOCA and non-LOCA plant safety analyses.

Loss-of-Coolant Accident

Plant LOCA analyses do not specifically analyze the potential failure of the reactor vessel or any of the attached nozzles, but they do postulate break sizes from 0.01 ft^2 to 14.2 ft^2 in area in any RCS pipe. A break in a CRDM from a crack that formed, propagated without detection, and failed catastrophically would be bounded by the RCS inventory losses considered in the existing plant LOCA analyses. Also, this break location is favorable from a core cooling standpoint, in that it is on the hot side of the core, such that no emergency core cooling system (ECCS) fluid is bypassed directly out of the break. That means that all the ECCS liquid is available for core cooling. The core shutdown for this event is assured by the insertion of the remaining control rods, augmented by the soluble boron reactivity control via the boron in the ECCS injection fluid.

Despite the fact that the existing LOCA analyses bound the CRDM nozzle failure with respect to inventory loss, there remains additional margin based on the credited rod worth and the RCS leakage detection systems. In the small break LOCA analyses, minimum control rod worths are credited. The control rod of highest worth is assumed to be stuck out of the core, and only a fraction of the remaining worth is used in demonstrating that at least a 1 percent shutdown margin exists at hot zero power conditions.

The RCS leakage detection systems are required by the plant technical specifications to detect unidentified leak rates of 1 gpm or greater. If the leak rate is higher, the plant will be shut down, and a controlled cooldown will be initiated. The makeup system will provide sufficient inventory and boron control. Insertion of the control rods will not be inhibited, and the core reactivity will be controlled. Following reactor shutdown, the consequences of a CRDM nozzle failure are decreased, thereby providing additional assurance that a safe shutdown is not compromised by the leakage that has been found or postulated to propagate during a single operating cycle with a leak in a CRDM nozzle.

Non-LOCA Safety Analyses

The plant non-LOCA safety analyses, for which consequences can be more severe if the core is not completely shut down, assume that the highest worth control rod is stuck out of the core, and at least a 1 percent shutdown margin exists at hot zero power conditions. Also, the consequences of a control rod ejection accident (CREA) are explicitly analyzed and included in the individual plant Final Safety Analysis Report (FSAR). Limitations are also imposed on each core design to limit the worth of any ejected control rod worth at hot full power to a value much less than the value assumed in the accident analyses.

The standard NRC-approved methodology (for Framatome ANP) consists of (1) calculating the maximum single ejected rod worth throughout cycle life, (2) verifying that the limits bound these maximum worths after augmenting by a 15 percent uncertainty, and (3) verifying that the core operating (rod index) limits preserve the calculational basis of the maximum worth. Because the typical analysis methodology uses the core average power response, the results of the calculation are sensitive to the total amount of reactivity inserted, not the number of control rods ejected. Consequently, the existing analysis will remain bounding for any number of ejected control rods, provided the total reactivity inserted into the core remains less than the values analyzed and reported in the FSAR. This provides additional margin, such that the consequences for the unlikely failure of a single CRDM nozzle will not be more severe than that already considered by each new fuel cycle for a limiting control rod ejection accident scenario.

9.0 Risk Assessment for CRDM Nozzle Cracks

The purpose of this section is to provide a risk analysis to supplement and support the deterministic safety assessment. The other sections of this safety assessment report describe the traditional engineering assessment of the CRDM nozzle cracks, including deterministic issues such as the impact upon safety margins and defense-in-depth. This deterministic analysis provides the source material upon which the risk assessment is based. This risk analysis estimates the core damage frequency (CDF) associated with operation with potentially undetected CRDM nozzle cracks, such as those found recently at ONS.

9.1 Potential Risks from CRDM Nozzle Cracking

Potential risks associated with the possibility of undiscovered CRDM nozzle cracks include:

- LOCA due to CRDM nozzle rupture or detachment
- Anticipated Transient Without Scram (ATWS) due to rod ejection accident (one or multiple)
- ATWS due to loose parts blocking control rods
- Damage due to CRDM and nozzle missile during accident

Of these, LOCA is considered to be the most important from a core damage frequency or risk perspective.

The random nature of crack initiation and growth makes it highly unlikely that multiple circumferential cracks will reach critical size in different CRDM nozzles at the same time. This assertion is made because the mean-time-to-failure in any given CRDM nozzle population is randomly (and widely) distributed. Even if there were several CRDM nozzles with unrevealed degradation, the loads administered during a plant transient would not impact them in identical ways. Because of nonhomogeneous crack initiation and growth, one CRDM nozzle failure time would precede the other(s). The recent B&WOG plant experience (see Sections 2.2 and 9.2) supports this assertion. The evidence of crack extent for the observed OD circumferential cracks above the J-groove weld indicates a random distribution of crack lengths. Therefore, simultaneous crack-initiated failures of redundant CRDM nozzles are very unlikely, and the risk from multiple CRDM nozzle failures due to cracking is very small.

In addition, the plant shutdown margins (if evaluated realistically) are such that several CRDM nozzle failures could be tolerated before the risk would increase over that of a single CRDM nozzle failure. Even with conservative success criteria for reactor trip, two or three CRDM nozzle failures could easily be tolerated from a reactivity standpoint. Therefore, it is concluded that the reactivity accidents (rod ejection or control rod blockage by loose parts) are not credible risk contributors, because of the number of simultaneous CRDM nozzle failures that would be required.

Missiles generated by CRDM nozzle failures are also not credible risk contributors. Even in the unlikely event of a CRDM nozzle detachment, the missile shields will prevent consequential damage to the reactor building or other safety systems.

Therefore, the risk impact that will be addressed and quantified is the risk from a LOCA. This analysis will estimate the incremental CDF due to a LOCA caused by a CRDM nozzle that may fail during operation due to undiscovered cracks.

9.2 Identification of CRDM Nozzle Cracks that are a Risk Concern

Of particular concern are circumferential cracks above the weld. A circumferential crack of sufficient extent may cause a large leak or a LOCA due to gross structural failure (net-section collapse) of the CRDM nozzle pressure boundary. The OD of the CRDM nozzle just above the J-groove weld (which is normally dry) is the only region on the CRDM nozzle pressure boundary where there is high axial stress relative to hoop stress. This region is susceptible to circumferential PWSCC cracks only if there is a source of primary water to the nozzle penetration annulus, such as might occur if there is a through-wall (TW) axial crack initiated from the ID of the CRDM nozzle or a crack in the J-groove weld.

Although axial CRDM nozzle cracks and J-groove weld cracks have occurred, they are not likely to result in a significant LOCA directly. The primary risk concern with the axial cracks and weld cracks is that the primary water leakage through the crack can provide moisture to the CRDM nozzle exterior and promote OD PWSCC.

Recent B&WOG experience (see Section 2.2) has included several axial cracks propagating through the J-groove weld or the area of the CRDM nozzle near the weld. At ONS- 3, there were nine CRDM nozzles with TW axial cracks above the J-groove welds or in the welds themselves. These cracks provided a path for primary water to the OD of the CRDM nozzle (in the annulus area just above the weld) where subsequent inspection indicated that three of these had indications of OD circumferential cracks above the weld. At ANO-1, there was an axial OD crack in the nozzle below the weld (i.e., in the area that is normally wetted) that extended to above the weld on the nozzle OD, thus wetting the OD area above the weld. At ONS-1, there was a crack through the J-groove weld of a CRDM nozzle that wetted the OD of the nozzle in the annular region above the weld. And at ONS-2, four CRDM nozzles were found with axial cracks that caused leakage to the annular region. Of these, one had indications of an OD circumferential crack above the weld. This experience suggests that there is a risk of a LOCA-sized CRDM nozzle failure from OD circumferential cracks that may be initiated due to primary water leaking into the annulus area from undetected ID-initiated TW axial, other OD-initiated axial, or J-groove weld cracks.

The four above-the-weld OD circumferential cracks at ONS-2 and ONS-3 were repaired along with the other crack indications. Excavations to clear these indications extended up to 180 degrees in the circumferential direction, and complete characterization of the indications was not recorded due to the aggressive nature of the excavations. However, subsequent examinations of ultrasonic test (UT) data taken before the excavations have indicated the circumferential extents of the cracks to be approximately 36 degrees (ONS-2 nozzle 18), 66 degrees (ONS-3 nozzle 23), 59 degrees (ONS-3 nozzle 50), and 165 degrees (ONS-3 nozzle 56) (see Section 2.2).

It is also possible that an ID-initiated crack could grow circumferentially to fail the CRDM nozzle pressure boundary directly. These cracks have been considered in this risk assessment. However, the operating history and probabilistic fracture mechanics analysis of ID-initiated circumferential cracks indicates that the likelihood of this failure mode is very small due to the nature of the stresses on the ID of the CRDM nozzle. To support this assertion, a probabilistic fracture mechanics prediction was made (Reference 29) of the ID-initiated TW crack frequency using the CHECWORKS computer code (Reference 30), the EPRI tool for predicting time to Alloy 600 PWSCC. The CHECWORKS analysis shows that the expected frequency of ID-initiated circumferential cracks is much less than the expected frequency of ID-initiated axial cracks. For the worst-case B&WOG plant, CHECWORKS predicts a median cumulative probability of 0.07 over the (60 year) plant life of getting an ID-initiated above-the-weld TW circumferential crack. This is a frequency of approximately 0.002 per reactor-year if averaged over the remaining plant life. That frequency is insignificant relative to the probability of axial cracks that may contribute to OD PWSCC (which is discussed further in Section 9.3.1). Therefore, the focus of the risk assessment is on the scenarios for circumferential OD CRDM nozzle cracking, and the risk estimated for OD-initiated circumferential cracks is considered representative of the overall risk.

With respect to impact upon risk, the only CRDM nozzle cracks that are risk significant are those where detectable symptoms of the degradation are not identified (and acted upon) prior to total failure of the nozzle. The risk analysis discussed below estimates the probability that CRDM nozzle failure will occur before a successful visual inspection detects telltale boron crystals on the exterior of the reactor vessel head.

9.3 OD Circumferential Crack Risk

The risk analysis focuses on scenarios in which an OD circumferential crack can grow to failure, causing a LOCA. The OD circumferential crack grows on the CRDM nozzle pressure boundary as a result of PWSCC caused by CRDM nozzle of J-groove weld leakage that wets the exterior of the CRDM nozzle in the annulus around the head penetration. The incremental core damage frequency for a LOCA induced by OD circumferential CRDM nozzle cracking is the product of the following factors:

- Frequency of weld or nozzle leak that wets OD of CRDM nozzle in the susceptible location
- Probability that CRDM nozzle leakage is undetected
- Time-dependent probability that total failure of CRDM nozzle will occur due to undetected crack initiation and growth on nozzle OD
- Probability of core damage from resulting LOCA

These events are shown as headers on the event tree (Figure 5) in which sequences that result in core damage are shown. Estimates of the event tree probabilities and initiating event frequency are provided in the following sections.

9.3.1 Probability of Weld or Nozzle Leak

In this section, the frequency of CRDM nozzle leaks that may wet the OD above the weld is estimated. It is assumed that some CRDM nozzles may be in service with near-TW nozzle cracks or weld cracks that may surface in the next fuel cycle. The CRDM nozzle cracks of interest are those above weld axial cracks and weld cracks that may leak primary water to the exterior of the CRDM nozzle in the annulus region of the RV head penetration.

The CRDM nozzle leak rate has been estimated from the recent inspection experience at ONS-1, ONS-2, ONS-3, and ANO-1. At these four plants, boron crystal deposits indicated leakage at 15 CRDM nozzles, including one at ONS-1, four at ONS-2, nine at ONS-3, and one at ANO-1 (see Section 2.2). It is uncertain how long these CRDM nozzles have been leaking. For the purpose of estimating a leak frequency, it will be assumed that half of these 15 leaks appeared during the most recent fuel cycle and half in the previous fuel cycle. It is likely that some of these leaks were actually present in earlier refueling outages, but were not identified as nozzle leaks at that time. Therefore, 15 leaking CRDM nozzles in approximately twelve plant-years (four plants times two cycles times 1.5 years per fuel cycle), gives an average frequency of approximately 1.25 leaking CRDM nozzles per reactor-year.

A prediction was also made (Reference 29) of the ID-initiated TW crack frequency using CHECWORKS (Reference 30). CHECWORKS is an empirical code, and the recent inspection results were taken into account by adjusting the crack initiation reference times. The results of the CHECWORKS analysis of ID-initiated cracking are dominated by the contribution from axial cracking, which was over two orders of magnitude more likely than circumferential ID cracking to cause a TW crack above the weld. The results of this analysis predict a median frequency of ID-initiated TW cracks above the J-groove weld of 0.52 per reactor-year averaged over the remaining plant life (assuming 60 year life). However, the CHECWORKS analysis is for ID-initiated nozzle cracking only. CHECWORKS (as it is now configured) is not designed for OD-initiated cracks above the weld or for J-groove weld cracking. Hence, as a prediction for CRDM nozzle leak frequency, it may underestimate. Therefore, to be conservative the mean value of 1.25 CRDM nozzle leaks per reactor-year, which was estimated from the plant experience, will be used in the risk assessment.

9.3.2 Probability that CRDM Nozzle Leakage is Undetected

A human reliability analysis (Reference 31) has been performed to estimate the human error probability (HEP) for the utility's inspection personnel failing to detect boron crystal deposits on the RV head that are indicative of a CRDM nozzle leak. CRDM nozzle leakage will be detectable through the accumulation of boron crystals on the top of the RV head around the base of the affected CRDM nozzles. It is assumed that any CRDM nozzle crack is undetectable until a through-wall crack (or weld crack) deposits boron crystals on the exterior of the RV head.

For OD-initiated above-the-weld cracks, the fracture mechanics model that is most relevant to the human reliability analysis is how long it takes, once wetted, for an OD crack to initiate and grow to the critical size for CRDM nozzle failure. This time (estimated in Section 9.3.4) will indicate how many opportunities (refueling outages) there may be to detect the boron crystals before total failure of the CRDM nozzle. Another factor important to the risk assessment is when the boron crystal deposition will be visible relative to the growth of the circumferential cracking.

The OD PWSCC failure mechanism requires a moist environment from the presence of primary water in either the liquid or steam state. Primary water from a leaking CRDM nozzle or weld will be deposited into the nozzle penetration annulus. During steady state operation there is a small radial clearance in the annulus above the weld to the surface of the RV head (see Appendix A). The primary leakage into the annulus may initially be very small, as might be the case for a pinhole leak, or somewhat larger, but there can only be PWSCC when there is a sufficient rate of leakage to keep the annulus area moist. Very small leaks are not likely to provide the appropriate environment in the annulus initially, considering the temperatures and pressures on top of the reactor vessel. The rate of boron crystal deposition will also be dependent upon the size of the leak. Moderate-sized leaks will generate boron crystals rapidly. For smaller leaks, there may be some time before significant boron crystals accumulate. However, as the boron crystals build up in and around the annulus, their presence will tend to trap moisture below. It is also possible that for a small leak there may be intermittent "leak plugging" and a weeping type leak (see Section 5) as the buildup of boron crystals intermittently "vents." Hence, it is reasonable to conclude that the environment required for the initiation of PWSCC on the OD of the CRDM nozzle (i.e., above the weld), whether it be from a small or moderate leak, will coincide roughly with the presence of visible boron crystal deposits.

9.3.2.1 Reactor Vessel Head Inspections

As a result of Generic Letter 97-01 (Reference 4), the B&WOG licensees have made a commitment to perform timely inspections of CRDM nozzles (and other vessel closure head penetrations). This commitment is maintained by permanent addition of a task item/work order into the refueling outage schedule program. Discovery of (new) boron on the head will result in the finding being placed in the licensee's Corrective Action Program (CAP).

CRDM nozzle flange leaks have occurred on several past occasions at B&WOG plants. Boric acid crystal buildup from these leaks may have masked indications of CRDM nozzle leakage in the past, and may have contributed to the exterior circumferential OD cracks at ONS not being detected by an inspection sooner.

CRDM nozzle flange leakage in the past was not considered to be unusual; however, once discovered, the CRDM nozzle flange was repaired to stop the leakage. Part of the repair process was to replace the gasket. The B&WOG licensees have been gradually repairing flanges and replacing gaskets since about May 1989. To date, nearly all of the B&WOG plant CRDM nozzle flange gaskets have been replaced with a stainless steel/graphite gasket, which, according to operating experience, are less prone to leakage. The number of CRDM nozzle flanges that still have the old gaskets is currently quite small (total of about a dozen over all of the B&WOG plants). Any flange leakage from one of these few remaining old-style gaskets would be quite evident, and would be promptly addressed.

Over the last five to seven years, the RV head inspections have become increasingly more meaningful because of utility efforts to clean the head of boron deposits resulting from past CRDM nozzle flange leakage and other sources. A clean RV head will make new boron crystals at the nozzle penetrations more evident, and reduce the likelihood that the leakage will be missed or masked by other sources of boron on the RV head.

The method of RV head inspection for indications of boron varies among the B&WOG plants. The methods vary from a simple visual inspection to the use of a mobile RV head robot with an attached video camera. However, none of the B&WOG plants have insulation directly on the reactor vessel head that may impact visual inspections. With all of the methods, the RV head inspection process is simple and straightforward, such that a written procedure is not necessary for a successful inspection. For the visual method, the RV head is observed through eight or nine access panels in the service structure with a high intensity portable light. The farthest an inspector would be from a CRDM nozzle is five feet. To ensure completeness, the inspection is carried out with a paper map of CRDM nozzle locations. The visual inspection method requires approximately two hours to complete. Other methods, such as use of a boroscope (i.e., camera on a stick) or RV head robot, result in a permanent record of the inspection on videotape. These methods also rely on the use of a paper map of CRDM nozzle locations to ensure completeness of the inspection.

9.3.2.2 Estimate of Human Error Probability for Visual Inspections

HEPs have been estimated for failure of the visual inspections using a combination of the Human Cognitive Reliability Model (Reference 32) and Swain and Guttman's Handbook (Reference 33). Since, visual inspections will occur with each refueling outage, a time-dependent failure probability is estimated considering the inspections to occur at two-year intervals. This is conservative since refueling cycles range between 18 and 24 months.

The human reliability analysis considered three ways in which the inspection process can fail to detect the boron crystals that are indicative of a CRDM nozzle leak. These include failure to conduct the inspection, failure to observe the boron crystals on the RV head when present, or failure to identify boron crystals resulting from a CRDM nozzle leak due to masking by other sources of boron (i.e., from CRDM nozzle flange leakage).

The human reliability analysis estimates that the HEP for failure of the visual inspection to detect signs of CRDM nozzle leakage at the first opportunity is $6.0x10^{-2}$. The human reliability analysis also estimates the failure probability for a second and third inspection (spaced at refueling outage intervals) of finding the same leaking CRDM nozzle, given failure of the previous inspection(s). The probability of repeatedly failing to detect the boron deposits at consecutive inspections (assumed two-year intervals) is a dependent relationship. The impact of this dependency (aside from the crack growing larger) is that the boron deposit will be more prominent and more difficult to miss at the next inspection. However, there is also the possibility that errors made in previous inspections will be repeated, the most important error being failure to perform the inspection. The human reliability analysis balances these competing dependencies, and appropriately adjusts the HEP with each subsequent outage. After failure of the first visual inspection, the dependency for repeatedly failing to perform the inspection is conservatively assumed to be stronger than the dependency of the boron deposit being more evident with time, thus causing the HEP to increase with additional opportunities. The human reliability analysis estimates that the HEP for failure to detect the CRDM nozzle leakage on the second opportunity is $6.5x10^{-2}$ and that it is 0.11 for the third and each subsequent outage. These HEPs are conservative considering the increased future emphasis on effective visual inspections of the reactor vessel head penetrations. Conservative HEPs have been used to encompass the uncertainty that is generally present in HEP estimates.

9.3.3 Probability of OD Crack Initiation

The time-to-OD-crack-initiation, once the exterior of the nozzle is wetted with primary water, is unknown. Computer codes used to predict time-to-PWSCC initiation are unreliable for OD PWSCC because the environment on the exterior of the CRDM nozzle (i.e., exterior to the pressure boundary) may be different than on the ID of the nozzle, especially in terms of boron concentration and length of time wetted. Therefore, to be conservative, the risk analysis assumes that the time-to-OD-crack-initiation is zero for all CRDM nozzles with exterior primary water wetting.

This approach is conservative with respect to the observations of the 15 leaking CRDM nozzles that were recently found at ONS and ANO-1. Only four of these

CRDM nozzles had indications of OD circumferential cracking above the weld. It is unknown specifically how long each of these CRDM nozzles has been leaking or whether OD cracks would have initiated on the others if leakage had continued undetected. Therefore, since a valid time-dependent model for OD PWSCC crack initiation is unavailable, it is conservative to assume that OD crack initiation will occur in 100% of the CRDM nozzles that have leakage into the annular region above the weld.

This approach (100% crack initiation with zero time-to-initiation) bounds the uncertainty associated with the lack of probabilistic fracture mechanics data for OD PWSCC crack initiation.

9.3.4 Time to Total Failure of CRDM Nozzle

A probabilistic fracture mechanics analysis (Reference 34) was performed to determine the probability of net-section failure of CRDM nozzles after initiation of above-the-weld OD circumferential cracking. The probabilistic fracture mechanics model was built around the deterministic crack growth model (Reference 18) described in Section 3.3.2. The crack growth model uses the Peter Scott model with worst case stresses. The probability of gross net-section failure is determined by performing a Monte Carlo simulation on a typical B&W-designed CRDM nozzle by varying the defining parameters of crack growth and size used in the deterministic fracture mechanics analysis. Available industry data were used to define distributions for key variables and conservatism was used where the data were sparse.

For the initial flaw distribution, the calculation was performed using parameters representative of the nozzle cracks found at ONS. For example, UT exams of the four above-the-weld OD circumferential crack indications at ONS-2 (nozzle 18) and ONS-3 (nozzles 23, 50, 56) indicate circumferential extents of approximately 36 degrees, 66 degrees, 59 degrees, and 165 degrees, respectively (see Section 2.2). It is unknown whether each of these cracks grew from a single OD initiation site, or from several initiation sites that linked together to form a long circumferential OD surface crack. Therefore, the initial flaw size used in the Monte Carlo simulation is a shallow semi-elliptical flaw with a circumferential extent uniformly distributed between zero and 180 degrees. Postulating a single flaw with an initially long circumferential extent is an approximation of the possibility of multiple linked initiation sites. The ONS plant experience is consistent with a uniform distribution of initial flaw extent and this approach is reasonable in light of the sparse industry data available for OD flaw distributions. A practical upper limit for this **initial** flaw distribution is a circumferential extent of 180 degrees, which is related to the nature of the stresses on the surface of the CRDM nozzle above the weld. On the nozzle OD above the weld, crack initiation in the circumferential direction may be driven by the axial bending stresses that are related to the proximity of the weld and shrink fit zones, and these are different on the uphill and downhill side of the nozzle. This approach of postulating an initial flaw as long as 180 degrees in circumferential extent bounds the uncertainty from scarcity of probabilistic fracture mechanics data for OD flaw distributions and multiple initiation sites.

Another source of uncertainty is the crack growth rate for OD-initiated PWSCC. The flaw growth rate distribution used in the Monte Carlo simulation is based upon industry data for PWSCC. Parameters affecting growth rate, such as stress intensity and temperature, were distributed in the Monte Carlo model to address uncertainty. Figure 6 illustrates the resulting crack growth rate distribution that was assumed in the Monte Carlo simulation. However, it is unknown whether the difference in environment between the nozzle exterior and interior may affect the growth rate for OD PWSCC. The approach used in this risk assessment to ensure that the uncertainty associated with crack growth rate is bounded, is to benchmark the Monte Carlo simulation results for time-to-TW crack against the plant observations. If the crack growth data are reasonable, the Monte-Carlo simulation should predict TW crack times consistent with the plant observations.

The results of the Monte Carlo simulation for TW cracking are illustrated by the histogram shown in Figure 7. The Monte Carlo simulation results are consistent with the plant experience. Of the 15 leaking CRDM nozzles found at ONS and ANO-1, two had above-the-weld OD circumferential cracks that were TW or almost TW (ONS-3 nozzles 50 and 56). To reach the equivalent percentage of TW cracks in the Monte Carlo simulation (i.e., 13.3% of the samples) required 4.2 years. Anecdotal evidence suggests that some of the nozzles at ONS may have been leaking for as much as 5 to 10 years. Therefore, the results of the Monte Carlo simulation appear to be reasonable or conservative with respect to experience.

The Monte Carlo simulation was used to grow the initial flaws to failure using the crack growth model described in Section 3.3.2 and stress distributions that are characteristic of the most exterior nozzles (i.e., the highest angle of penetration). For this analysis, failure is defined as insufficient ligament to meet ASME Code primary stress limits, which corresponds to a circumferential crack extent of approximately 292 degrees or 81% (Reference 18). The failure definition is conservative since the threshold ligament is based on satisfying primary stress limits using a safety factor of 3 (and 1.5 for emergency and faulted conditions). The failure definition also does not take credit for the Technical Specification required 1 gpm leak detection capability, which as described in Appendix A may occur at a somewhat smaller crack extent depending upon the radial clearance in the penetration annulus. A conservative failure definition is appropriate for this risk assessment considering the current weakness in industry understanding of OD PWSCC and because it may bound uncertainties inherent in the probabilistic fracture mechanics data.

The results of the Monte Carlo simulation are illustrated by the histograms shown in Figures 7 and 8. Based on the Monte Carlo simulation, an OD-initiated crack above the J-groove weld would take a mean time of 8.9 years to grow to a through-wall state, and a mean time of 28 years to result in nozzle failure (or LOCA) due to net-section stress. For comparison, 97.5% of the time-to-failure distribution (non-parametric) is greater than the point estimate reported in Section 3.3.2 (7.5 years for a crack to reach 75% circumferential extent). This reflects the conservatism that is inherent in the deterministic approach.

The time-to-failure histogram (Figure 8) has been partitioned into two-year probability increments to correspond to the worst-case visual inspection intervals for B&WOG plants (i.e., plants with a two-year fuel cycle). The table (see Figure 8 inset) shows the probability that the OD crack will grow to failure within the time indicated assuming there is no detection by visual inspection (boron crystals). The opportunities for detection will be added at two-year intervals in the event tree quantification (see Figure 5).

9.3.5 Probability of Core Damage

The most likely consequence of CRDM nozzle failure (critical size crack) is leakage that is within the capacity of the makeup system. If a complete severance of the CRDM nozzle occurs, the break size will be within the range of what most B&WOG PRAs identify as a medium break LOCA. However, a smaller break size could result if there is a partial failure of the nozzle.

The conditional probability of core damage given a small- or medium-sized LOCA can be readily determined from the plant-specific B&WOG PRAs. In a B&WOG PRA, the conditional core damage probability (CCDP) for a medium break LOCA is on average worse than for a small break LOCA. Therefore, as a representative value, the risk assessment uses the average CCDP for a medium-break LOCA from a survey of the B&WOG PRAs, which is approximately $4x10^{-3}$ (Reference 35). Use of this CCDP is conservative because plant mitigation response will be better for a break at the top of the vessel than for the LOCAs typically considered in the PRAs (see Section 8.0).

9.3.6 Risk Results for OD PWSCC

The estimated frequency and probabilities in the preceding sections are used to quantify the event tree shown in Figure 5. The event tree shows the progression of sequences starting with the initiating event "CRDM leaks." Each sequence can result in success (e.g., no core damage) or failure/core damage, as noted by the "S" and the "CD" in the "Success or Core Damage" column. One sequence is identified as "CD Residual," recognizing that inspection and crack growth may continue beyond the eight years explicitly modeled in the event tree. The contribution from these residual sequences is not significant. In the event tree, at each decision point (success or failure), the conditional failure probability (as estimated in the previous sections) is shown. Multiplying the appropriate branch failure probabilities results in the frequency of core damage for each sequence. Only sequences that result in core damage are quantified. When summed, the sequence frequencies provide an estimate of the CDF due to OD PWSCC of the CRDM nozzles, which has a mean value of 3.4×10^{-7} per reactor-year.

Uncertainty in these results has been addressed via the use of conservative assumptions and the use of bounding data inputs for the probabilistic fracture mechanics. In particular, bounding assumptions were made for crack initiation time, initial flaw distribution, and multiple crack initiation sites. The crack growth rates used appear to produce results consistent with the plant observations of TW cracks. Other conservatisms include the human error probability for visual inspections, nozzle failure definition, and LOCA mitigation failure probability. Therefore, it is concluded that the CDF results produced by this risk assessment are reasonable in light of the limited industry knowledge base for this failure mechanism.

The estimated core damage frequency (3.4×10^{-7} per reactor-year) compares favorably to the risk acceptance guidelines contained in Regulatory Guide 1.174 (Reference 36) for core damage frequency. Per these guidelines, the risk of operation with potentially undiscovered CRDM nozzle cracks is categorized as "very small." Regulatory Guide 1.174 also has acceptance guidelines for large early release frequency (LERF). The effect of the nozzle cracks upon LERF is insignificant because the containment safeguards systems are not affected by CRDM nozzle cracking. The reactor vessel missile shields preclude consequential damage to the containment building in the unlikely event of CRDM nozzle detachment. No other collateral damage has been identified that may affect containment safeguards systems. Therefore it is concluded that the risk associated with CRDM nozzle cracking at B&WOG plants is small and consistent with the Commission's Safety Goal Policy.

The public health risk associated with the CRDM nozzle cracking is correspondingly very small. For example, the conditional population dose for a medium break LOCA core damage accident at a typical B&WOG plant (ONS) is 1.1E4 person-rem (Reference 37). For the estimated core damage frequency of 3.4×10^{-7} per reactor-year, this corresponds to a public health risk of only 3.7×10^{-4} person-rem/reactor-year, which is insignificant.

According to Regulatory Guide 1.174, risk insights should be considered in an integrated fashion with traditional deterministic evaluations (such as those discussed in Sections 1 through 8). The deterministic and risk evaluations taken together indicate that safety margins and defense-in-depth are not significantly affected by the CRDM nozzle cracking. With effective visual inspections, the nozzle cracking does not significantly increase the LOCA frequency that is assumed in B&WOG PRAs, nor does it increase the frequency above the level that is assumed for design basis accidents. The consequences of a CRDM nozzle failure are less severe than the LOCAs assumed in the FSAR analyses. Also, the CRDM nozzle cracking has no effect on core damage mitigation, containment safeguards, or emergency planning effectiveness. Therefore, this risk analysis concludes that the risk to the public due to CRDM nozzle cracking is acceptable. The risk analysis also supports the findings of the deterministic analyses, which is that visual inspections of the RV head will discover signs of CRDM nozzle leakage before there is a significant likelihood of total failure of a CRDM nozzle due to PWSCC.

10.0 Summary and Conclusions

A safety assessment has been performed to address the potential for PWSCC cracking of RV head penetration nozzles and welds at the B&WOG plants. It addresses both axially and circumferentially oriented flaws that have been observed in the Alloy 600 CRDM nozzles as well as axial/radial flaws observed in the Alloy 182 J-groove partial penetration welds used to attach Alloy 600 CRDM nozzles to low alloy steel RV heads. This safety assessment utilizes and builds upon the existing analyses performed for CRDM nozzle PWSCC (References 3, 7, and 8).

The results of detailed stress analysis of the nozzle and weld regions of the RV head demonstrate that the circumferential, or hoop, stress is generally higher than the axial stress at the same location. On the downhill side of the nozzle, the ratio of hoop stress to axial stress is about 2/1, and on the uphill side it is about 3/2. In the weld region, hoop stresses are about two times the axial stress at the same location. It can therefore be concluded that if PWSCC cracking were to occur, flaws would predominantly be oriented in a longitudinal, or axial, plane, and as such would not promote catastrophic failure of the nozzle by ejection.

Based on laboratory test data for Alloy 182 weld metal in a PWR environment, crack growth through the J-groove weld could occur rapidly (i.e., within one or two years). Although continued crack growth into the low alloy steel head would not be expected due to the low susceptibility of this material to SCC, flaws in the weld metal could continue to grow into the Alloy 600 CRDM nozzle, as seen at ONS-1 and ONS-3. It has been predicted that it would take almost four years for an axial OD nozzle flaw to grow through-wall to the inside surface. At this point, a leak path into the annular region between the nozzle and head could be present, depending on the location of the original flaw in the nozzle.

Any circumferential flaw above the weld on the outside surface of the nozzle should not be considered a safety concern. A short, isolated flaw would take more than 10 years to grow through-wall, while a long circumferential (where multiple flaws have joined) could grow from the outside surface to the inside surface in about 3.5 years. In neither case would the structural integrity of the nozzle be compromised to the point that the nozzle would fail by ejection.

Circumferential cracking has also been observed on the outside surface of CRDM nozzles at ONS-3, at the toe of the fillet weld that forms part of the structural attachment to the reactor vessel head. Since these cracks are located at or below the weld, and not in the reactor coolant pressure boundary, they are not considered to be a safety concern from the standpoint of gross structural failure or release of radioactive water. Due to the proximity of associated through-wall cracking below the weld, however, there is a concern that a through-wall circumferential crack could link up with two or more through-wall axial cracks and form a loose part.

Based on experience at ONS-3, circumferential and axial cracking below the weld is accompanied by through-wall axial cracking at and above the weld, as evidenced by deposits of boric acid crystals on the top of the head. It is concluded from these results and observations that detectable leakage would precede the development of a loose part.

Concerns relating to a lack of fusion type weld defect between the nozzle and weld have been addressed by considering the growth of a postulated "wrap-around" circumferential flaw along the cylindrical surface at the nozzle-to-weld interface. Utilizing radial stresses between the nozzle and weld and PWSCC crack growth rates for Alloy 182 weld metal, it has been calculated that it would take two years for a 17.5% wrap-around flaw to grow to an allowable 67% flaw size.

It has also been demonstrated by a detailed stress analysis that annular gaps develop between the CRDM nozzle and the RV head in the RV head penetration of the B&WOG plants. In the event of a through-wall crack in the J-groove weld or the portion of the CRDM nozzle in the annulus, these gaps form the natural leakage path for the RCS coolant to the OD of the RV head. Assuming a designed 0.0010 inch nominal diametric interference, the minimum calculated radial gap is 0.001 inch for both the center nozzle and the outermost nozzle designs. The average or representative radial gaps for the center nozzle and the outermost nozzle are 0.0016 inch, and 0.002 inch, respectively.

Axial flaws are anticipated to be predominant at both the ID and OD of the CRDM nozzle based on the magnitude of the hoop stresses, although circumferential flaws can be envisioned on the OD and have been observed (both above and below the weld). Axial flaws within the J-groove weld are also the most plausible flaws due to high hoop stresses. These types of cracks are envisioned to break the surface as pinhole type cracks or as tight PWSCC cracks. These tight cracks would result in very low leakage rates as evidenced by the low volume of boric acid crystals found in the vicinity of CRDM nozzles at ONS-1, ONS-2, ONS-3, and ANO-1. It was estimated that approximately 0.5 in^3 was present around CRDM nozzle number 21 at ONS-1. However, observable leakage is expected to occur well before crack propagation would reach ASME Code limits.

It has been shown that, assuming a large portion of the nozzle cross-section contains a through-wall circumferential crack, there is ample room for leakage to occur before approaching the net section limit ligament. This will allow a detectable leakage of steam through this large crack, thereby providing ample warning to prevent the failure of the nozzle. In addition, evidence indicates that the nozzles are in an oval shape due to interaction with the closure head deformation. Therefore, there are gaps between the nozzle and the head that will provide sufficient leak paths for a fairly large volume of steam to escape thereby providing leak detection.

The allowable lack of fusion size was previously determined to be 67%, or 8.4 inches of circumferential extent. Also, the critical lack of fusion size was determined to be 85% or 10.6 inches of the circumference. The leakage rates were predicted using the same methodology as used for the evaluation of the axial crack. It was determined that a crack length of 7.5 inches is required for the center nozzle design to achieve a 1 gpm leak rate. Similarly, it was established that a crack length of 5.0 inches is required for the outermost nozzle design to achieve a leak rate of 1 gpm. Since these cracks are less than the allowable lack of fusion crack length of 8.4 inches, it is concluded that these types of cracks will be detected by the plant's leak detection capability.

Boric acid corrosion concerns were addressed for a variety of conditions and leakage rates potentially assumed to occur. It was determined that corrosion of the RV head penetration, at a maximum volumetric metal loss rate of 1.07 in^3/yr would be possible. Various defect profiles were postulated to model this level of corrosion for a time period of six years. It was concluded that safe operation of the plant would not be affected as a result of this level of corrosion and that within this time, the leak will be detected during a walk-down inspection of the RV head area.

All of the observed through-wall CRDM cracks in the B&WOG plants have been traced to origination in the vicinity of the weld and not at the end of CRDM nozzle. Failures in the end of the nozzle have the potential to generate loose parts that could relocate within the RCS and compromise equipment operation or fuel-clad barrier integrity. Given the current knowledge of the residual stresses in the CRDM nozzles, FRA-ANP has concluded that the through-wall axial cracks present below the weld initiate at the toe of the weld. These cracks are not expected to propagate to the point that a loose part will be generated before some leakage is visible. Therefore, all aspects of the CRDM cracks have been considered from a safety analysis perspective. This review has concluded that simultaneous multiple CRDM nozzles will not fail and that the failure of a single CRDM nozzle is bounded by both the LOCA and non-LOCA plant analyses already completed to support current plant operation.

A loose part evaluation was performed to evaluate the potential for loose parts from a failed CRDM nozzle to potentially enter a control rod guide tube and prevent the control rod assembly from being fully inserted. It was concluded that there was at least a 25 percent chance of a loose part entering the guide tube and potentially impairing successful operation of that assembly. The LOCA and non-LOCA analyses assume that the control rod of highest worth is stuck out of the core. In addition, only a fraction of the remaining worth is used in demonstrating that at least a 1 percent shutdown margin exists at hot zero power conditions.

It has been demonstrated through risk analysis that the risk from potentially undetected CRDM nozzle cracks is "very small" per the guidelines of Regulatory Guide 1.174. The estimated core damage frequency due to OD PWSCC of the CRDM nozzles is 3.4×10^{-7} per reactor-year. Conservative assumptions are made in the risk assessment to address uncertainty in the estimates of human reliability, probabilistic fracture mechanics, and plant mitigation response. Taken together with the results of the deterministic analyses, the risk analysis demonstrates that visual inspections of the reactor vessel head will be sufficient to minimize public risk. The visual inspections will discover signs of CRDM nozzle or penetration weld leakage before there is a significant likelihood that the leakage will cause CRDM nozzle structural failure or detachment due to outside diameter PWSCC.

Finally, all evidence to date suggests that it will require several years for the material to degrade to the point that total failure of the component could occur. During that time, if a crack should form, leakage of primary coolant on to the RV head can be identified through routine visual inspections. The component can then be repaired and returned to service without jeopardizing the health and safety of the public.

As a result of the previously described activities and evaluations performed by the B&WOG, the following conclusions have been reached regarding degradation of CRDM nozzles, thermocouple nozzles, and RV head attachment welds at B&WOG plants:

1) The B&WOG plant safety evaluation (Reference 3) remains valid.

2) The B&WOG utilities comply with 10CFR50.55a and continue to meet the intent of General Design Criteria 14, 30, 31, and 32 of Appendix A of 10CFR50.

3) The potential for the B&WOG plants to have sulfur-induced IGA or SCC of CRDM and thermocouple nozzles is very low (Reference 5).

4) The risk to the public due to CRDM nozzle cracking is "very small" and acceptable per the guidelines of Regulatory Guide 1.174.

5) Visual inspections of the reactor vessel head will discover signs of CRDM nozzle leakage before there is a significant likelihood of total failure of a CRDM nozzle due to PWSCC.

6) Inspections, other than visual examinations in accordance with GL 88-05, are not necessary from a safety perspective.

7) One of the most susceptible B&WOG plants, ONS-2, has inspected all 69 CRDM nozzles in 1994 and two follow-up inspections on the nozzles identified with flaw-like indications. Recent observations at ONS-1, ONS-2, ONS-3, and ANO-1 have also added credence to the safety assessments that have been performed.

8) All B&WOG utilities continue to perform visual inspections of the RV head in accordance with their respective Generic Letter 88-05 and Bulletin 2001-01 responses.

9) The B&WOG will continue to share B&WOG plant inspection data and participate in agreed upon joint Owners Group (e.g., MRP) activities with the U.S. nuclear industry on this issue.

10) The B&WOG will continue to monitor this issue.

11.0 References

1) "Safety Evaluation of the Potential for and Consequences of Reactor Vessel Head Penetration Alloy 600 ID Initiated Penetration Cracking," CEN-607, May 1993.

2) "Alloy 600 Reactor Vessel Head Adapter Tube Cracking Safety Evaluation," WCAP-13565, Rev. 1, February 1993.

3) "Safety Evaluation for B&W-Designed Reactor Vessel Head Control Rod Drive Mechanism Nozzle Cracking," BAW-10190P (B&W Owners Group Proprietary), May 1993.

4) Generic Letter 97-01: "Degradation of Control Rod Drive Mechanism Nozzle And Other Vessel Closure Head Penetrations,' U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, Washington, DC, April 1, 1997.

5) "B&WOG Integrated Response to Generic Letter 97-01: 'Degradation of Control Rod Drive Mechanism Nozzle And Other Vessel Closure Head Penetrations'," BAW-2301, July 1997.

6) Title 10 of the Code of Federal Regulations, Part 50, U.S. Nuclear Regulatory Commission, Washington, DC.

7) "External Circumferential Crack Growth Analysis for B&W-Design Reactor Vessel Head Control Rod Drive Mechanism Nozzles," BAW-10190P, Addendum 1 (B&W Owners Group Proprietary), December 1993.

8) "Safety Evaluation for Control Rod Drive Mechanism Nozzle J-Groove Weld," BAW-10190P, Addendum 2 (B&W Owners Group Proprietary), December 1997.

9) Generic Letter No. 88-05, "Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants," U.S. Nuclear Regulatory Commission, March 17, 1988.

10) Bulletin 2001-01: "Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles," U.S. Nuclear Regulatory Commission, August 3, 2001.

11) "Oconee Nuclear Station Units 1, 2, & 3, Docket Nos. 50-269, 270, and 287, Response to NRC Bulletin 2001-01: Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles," Duke Energy letter from W.R. McCollum to U.S. Nuclear Regulatory Commission, August 28, 2001.

12) "Arkansas Nuclear One – Unit 1, Docket No. 50-313, License No. DPR-51, 30 Day Response to NRC Bulletin 2001-01 for ANO-1; Circumferential Cracking of VHP Nozzles," Entergy letter from C. Anderson to U.S. Nuclear Regulatory Commission, September 4, 2001 (1CAN090102).

13) "Crystal River Unit 3 – Response to NRC Bulletin 2001-01, "Circumferential Cracking of Reactor Pressure Vessel Head Penetration

Nozzles," Progress Energy letter from D.E. Young to U.S. Nuclear Regulatory Commission, August 30, 2001 (3F0801-06).

14) Exelon/AmerGen Response to NRC Bulletin 2001-01, "Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles," letter from J.A. Benjamin to U. S. Nuclear Regulatory Commission, August 31, 2001 (RS-01-182).

15) "Response to NRC Bulletin 2001-01, "Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles," letter from G.G. Campbell to U.S. Regulatory Commission, September 4, 2001.

16) "Crack Growth of Alloy 182 Weld Metal in PWR Environments (PWRMRP-21)," TR-1000037, Electric Power Research Institute, June 2000.

17) "Reactor Coolant System Pressure Boundary Leakage Due to Cracks Found in Several Small Bore Reactor Vessel Head Penetrations," Licensee Event Report 2000-006-00, Oconee Nuclear Station 1, Docket Number 05000-269, January 2, 2001.

18) Killian, D.E., "OC-3 CRDM Nozzle Circumferential Flaw Evaluations," 32-5012403-00, April 2001 (FRA-ANP Proprietary).

19) Vreeland, D.C., "Corrosion of Carbon Steel and Low Alloy Steels in Primary Systems of Water-Cooled Nuclear Reactors," Presented at Netherlands-Norwegian Reactor School, Kjeller, Norway, August 1963.

20) Uhlig, H.H., ed., Corrosion Handbook, Wiley, New York, 1948, p 125.

21) Shvartz, G.C., and Kristal, H.M., Corrosion of Chemical Apparatus, Chapman and Hall, London, 1959, pp 53-70.

22) Moore, K.E., "Stress Corrosion Cracking of Low Alloy Steel," 51-5012047-00, March 2001 (FRA-ANP Proprietary).

23) "Cracking in Charging Pump Casing Cladding," IE Information Notice No. 80-38, Nuclear Regulatory Commission, October 31, 1980.

24) "Boric Acid Corrosion of Charging Pump Casing Caused by Cladding Cracks," IE Information Notice 94-63, Nuclear Regulatory Commission, August 30, 1994.

25) Snow, F., "PT Inspection Report Resolution," 51-5011639-00, February 2001.

26) V.C. Summer Nuclear Station Alpha Hot Leg Evaluation and Repair, Presentation to the Nuclear Regulatory Commission Region II, December 20, 2000.

27) "Boric Acid Corrosion Guidebook – Recommended Guidance for Addressing Boric Acid Corrosion and Leakage Reduction Issues," TR-102748, Electric Power Research Institute, April 1995.

28) Wiemer, J.A., "Loose CRDM Nozzle Components in RCS," 51-5012057-00, April 2001.

29) Xu, Hongqing, "CHECWORKS RHNM PWSCC Risk Assessment," 51-5013250-00, June 2001 (FRA-ANP Proprietary).

30) 30) CHECWORKS PWR Vessel and Internals Application: RPV Head Nozzle Module, Version 1.0, User Guide," TR-103198-P8, Final Report, December 1998, Electric Power Research Institute, Palo Alto, CA.

31) Levinson, S.H., "Probability that CRDM Leakage is Detected," 32-5013324-00, June 2001.

32) W. Hannaman, "Human Cognitive Reliability Model for PRA Analysis," RP-2847-1, Electric Power Research Institute, December 1984.

33) D. Swain and H. E. Guttman, "Handbook of Human-Reliability Analysis with Emphasis on Nuclear Power Plant Applications/Final Report," Sandia National Laboratories, prepared for the U. S. Nuclear Regulatory Commission, SAND80-0200, NUREG/CR-1278, August 1983.

34) Mazurkiewicz, S.M., "Monte Carlo Evaluation of Circumferential Flaws in B&WOG CRDM Nozzles," 32-5013346-01, August 2001 (FRA-ANP Proprietary).

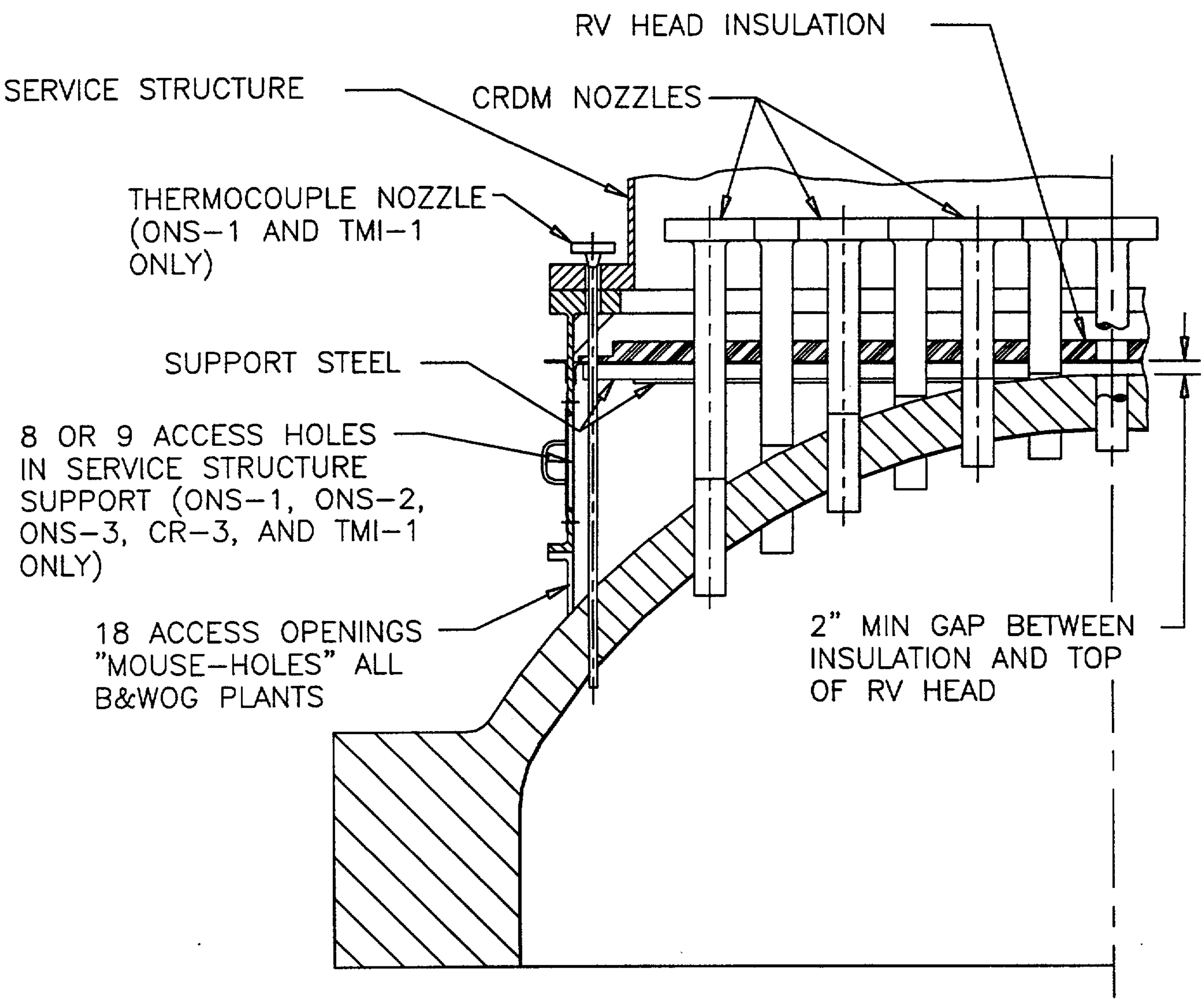
35) Enzinna, R.S. and Levinson, S.H., "B&WOG PSA Comparison," 47-5006733-00, January 2000.

36) Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," U.S. Nuclear Regulatory Commission, July 1998.

37) Oconee Nuclear Station PRA, Revision 2, Duke Energy Corporation, December 1996.

38) "Leakage Assessment Through CRDM Nozzle and Closure Head," BAW-2213 (B&W Owners Group Proprietary), June 1994.

Figure 1. Side View Schematic of B&W-Design Reactor Vessel Head, CRDM Nozzles, Thermocouple Nozzles, and Insulation.



Note: The thermocouple nozzles were removed from ONS-1 at EOC-19.

Figure 2. Plenum Cover Assembly.

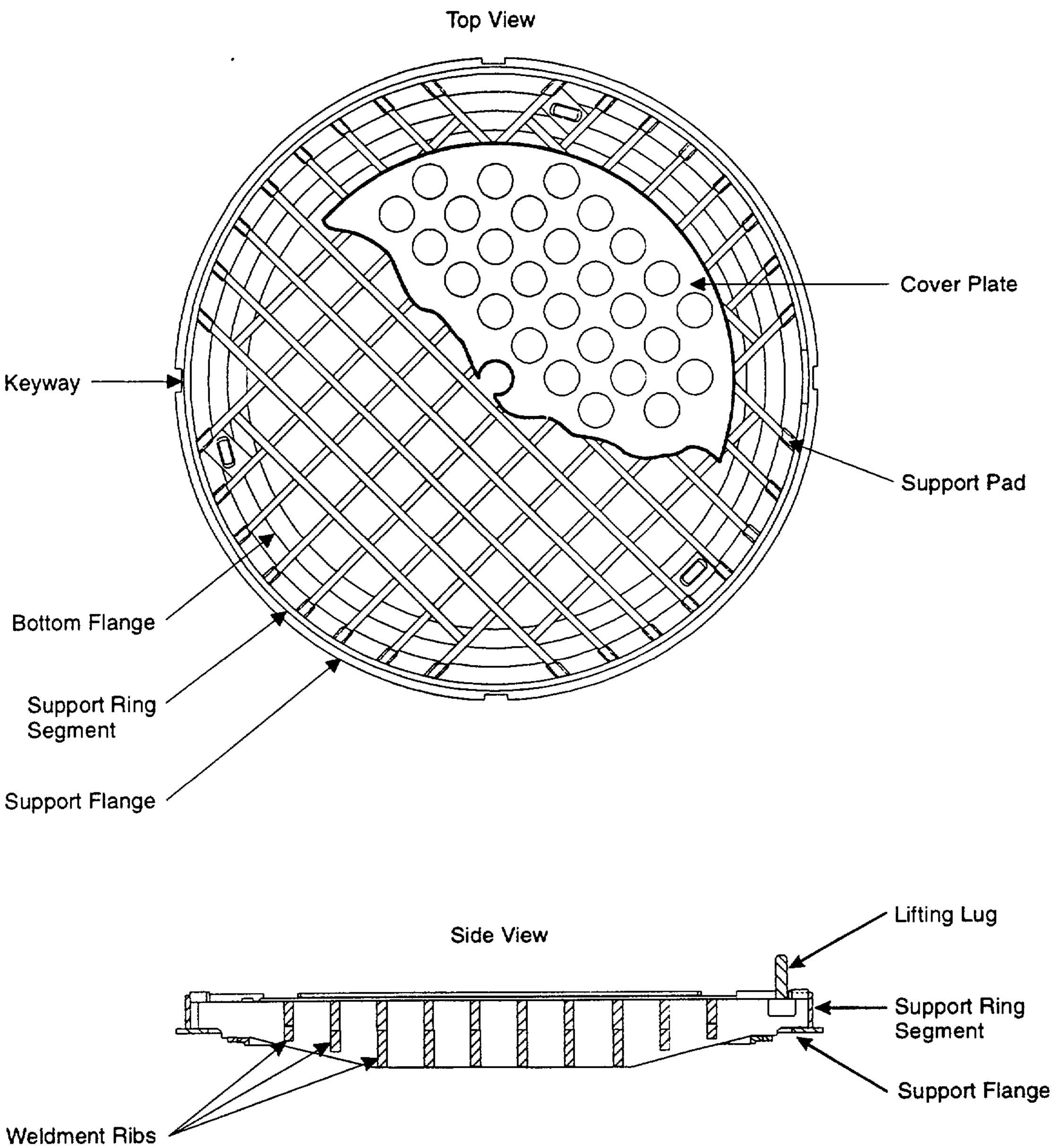


Figure 3. Control Rod Spider Assembly.

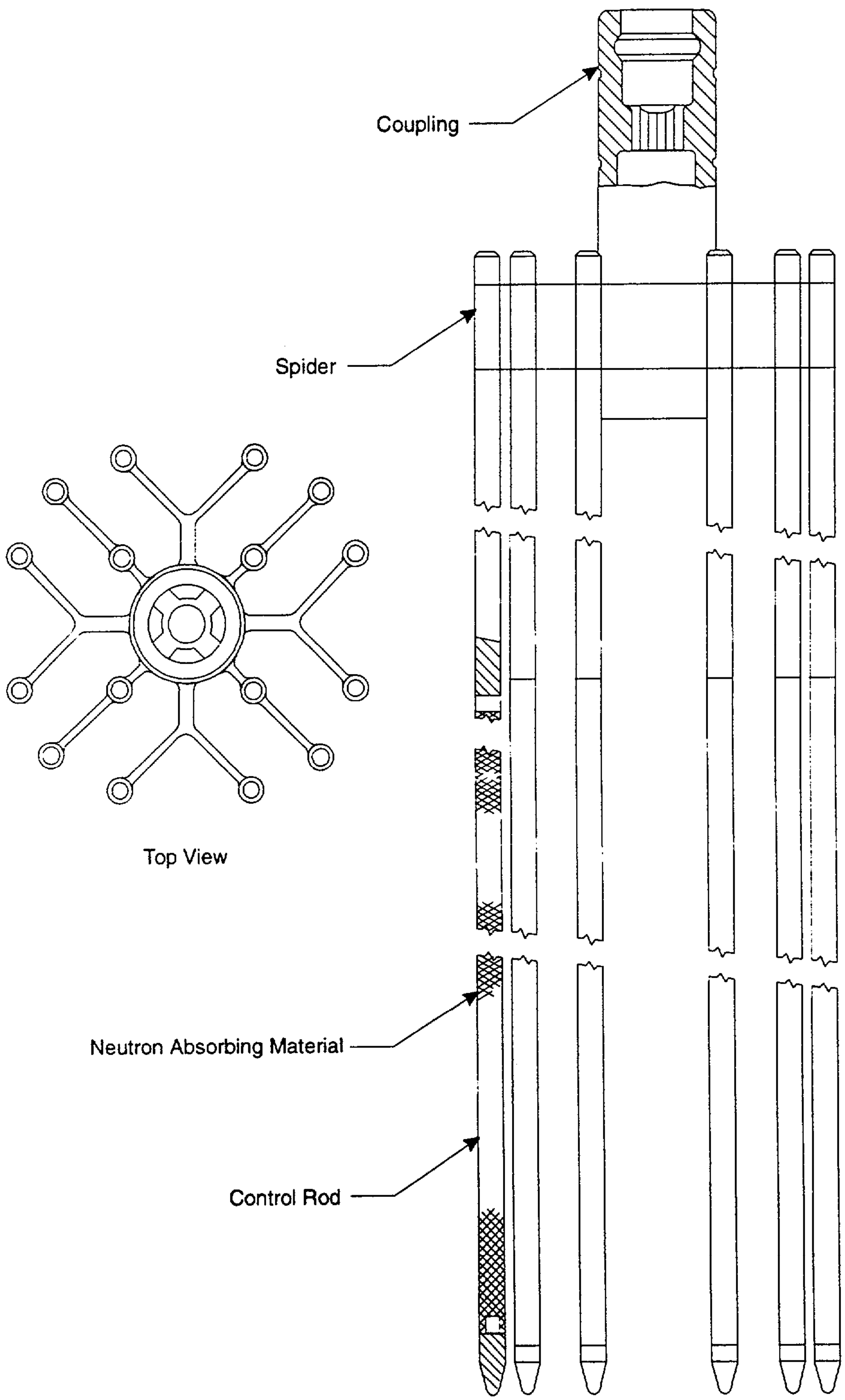


Figure 4. Control Rod Guide Brazement Assembly

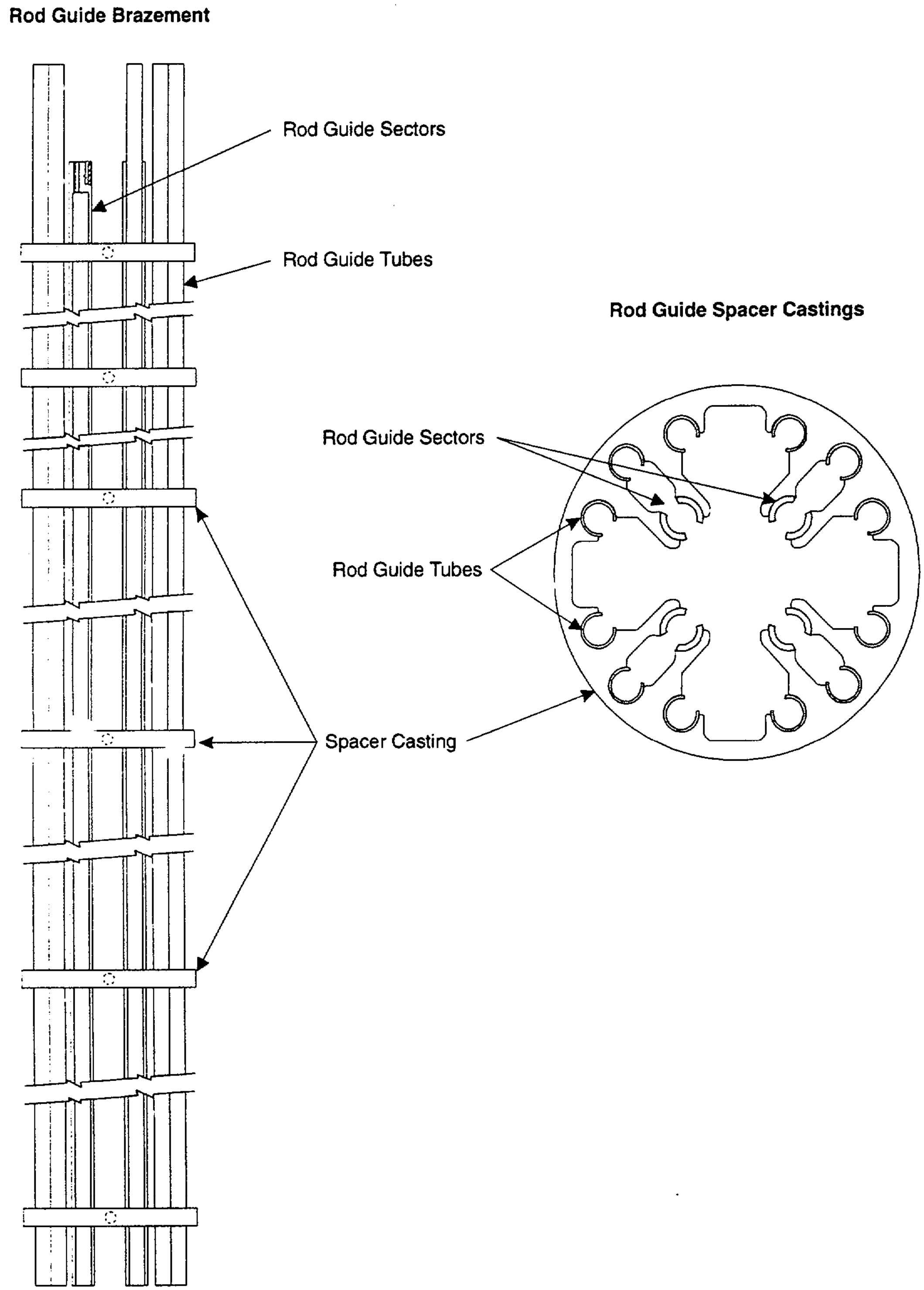


Figure 5. Event Tree for Frequency of Core Damage from Outside Diameter PWSCC in B&WOG CRDM Nozzle

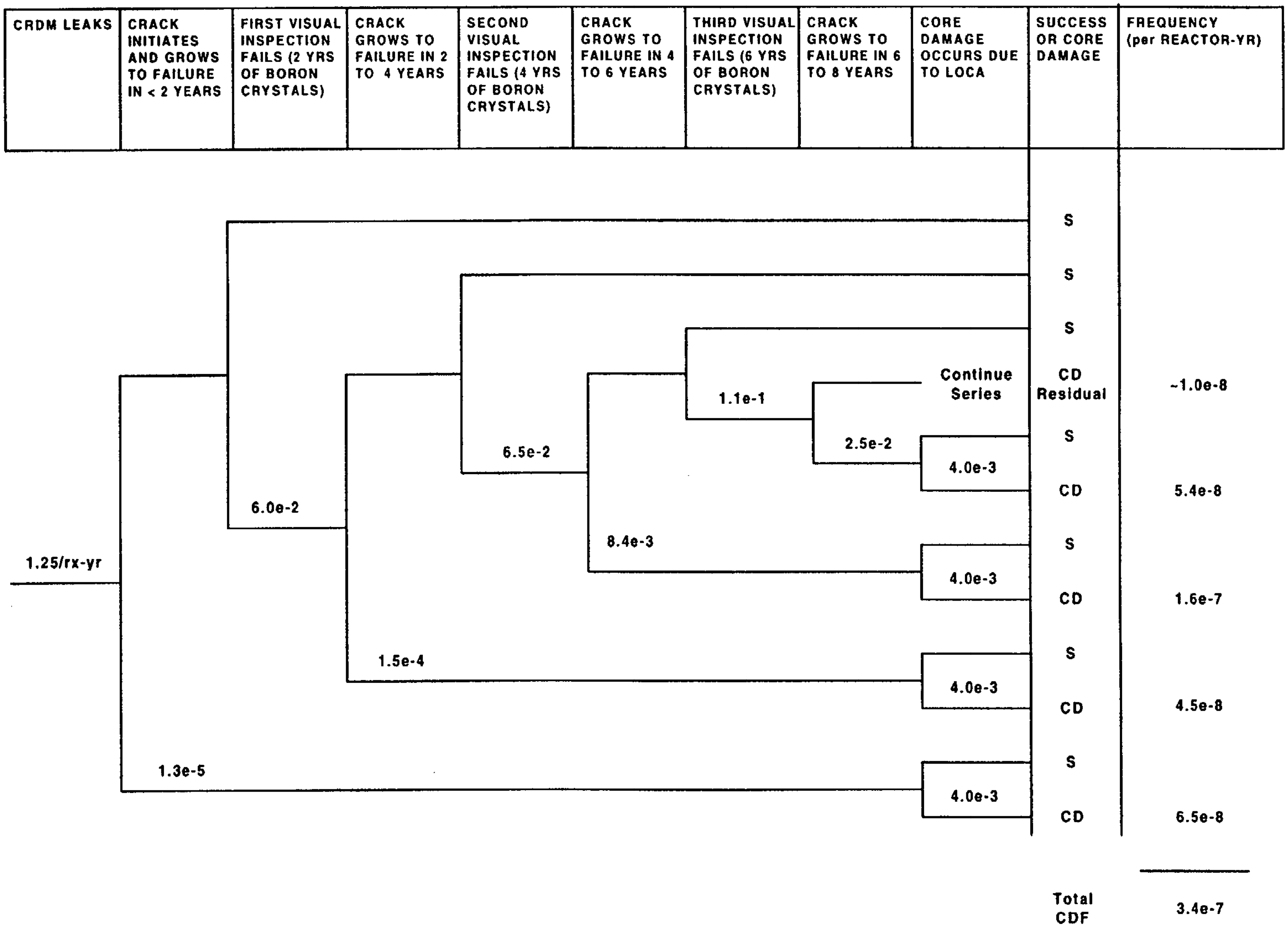


Figure 6. Crack Growth Rate Assumed in Monte Carlo Simulation

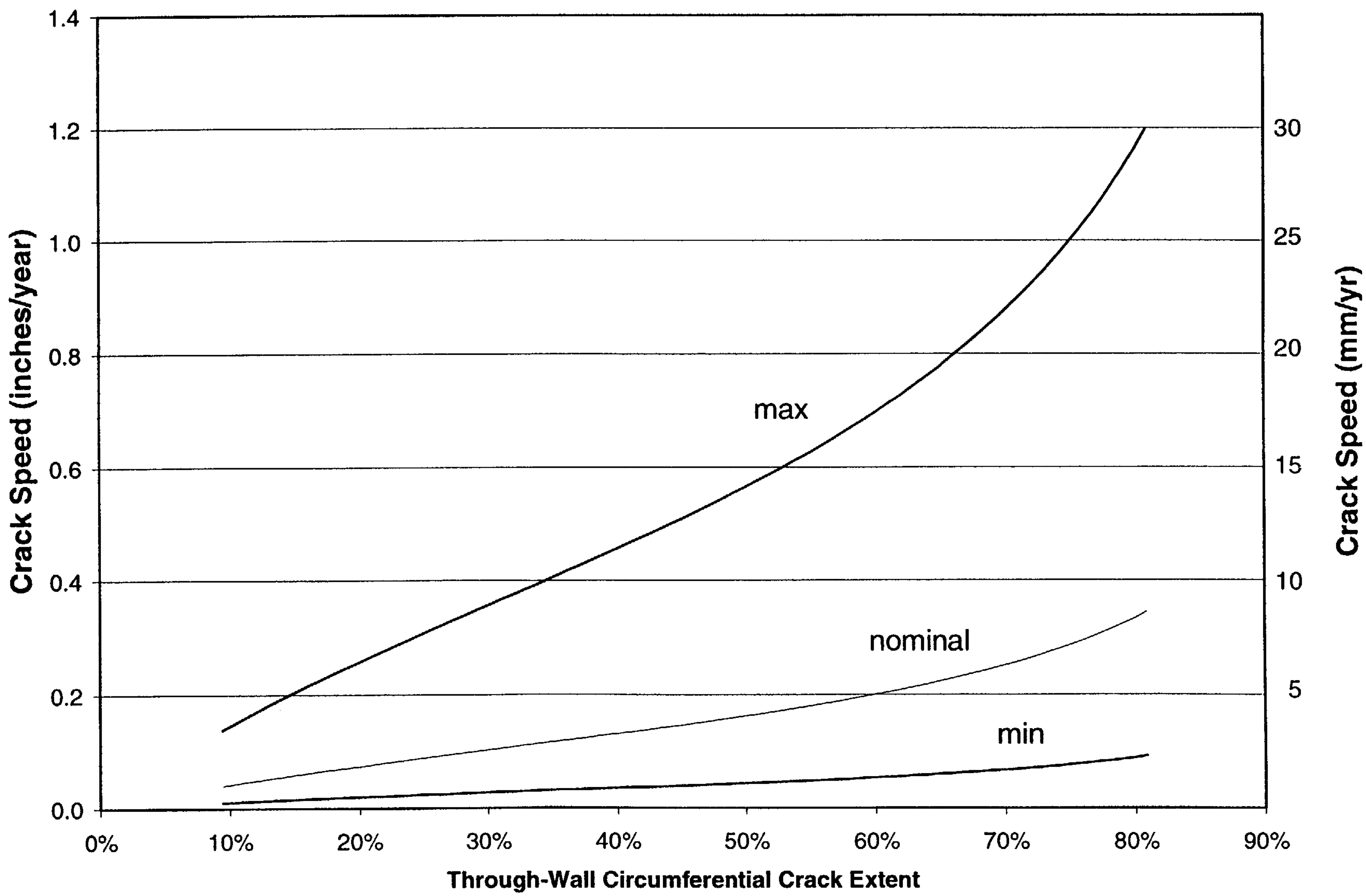


Figure 7. Probability of Through-Wall Crack versus Time after Initiation of Outside Diameter PWSCC

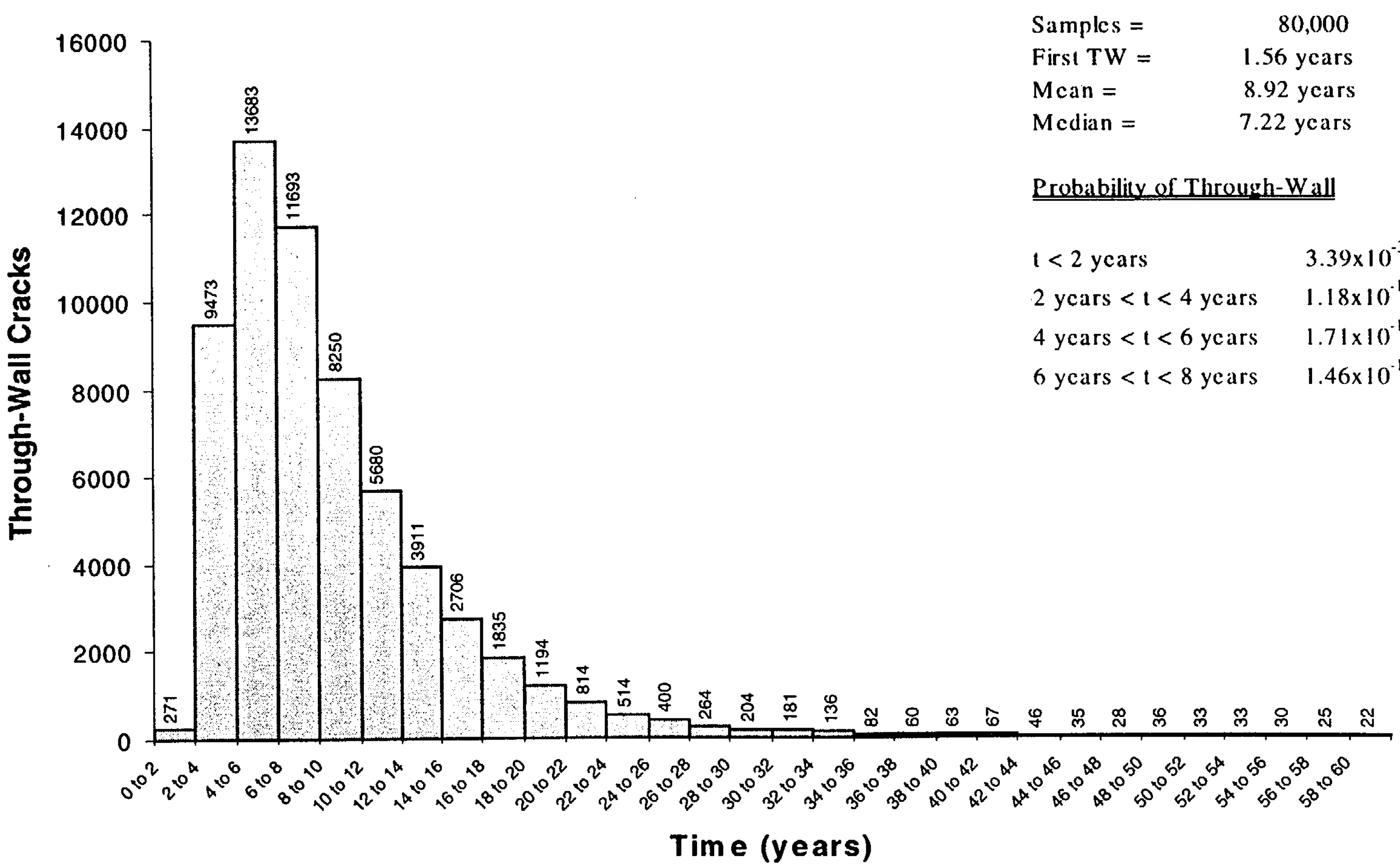
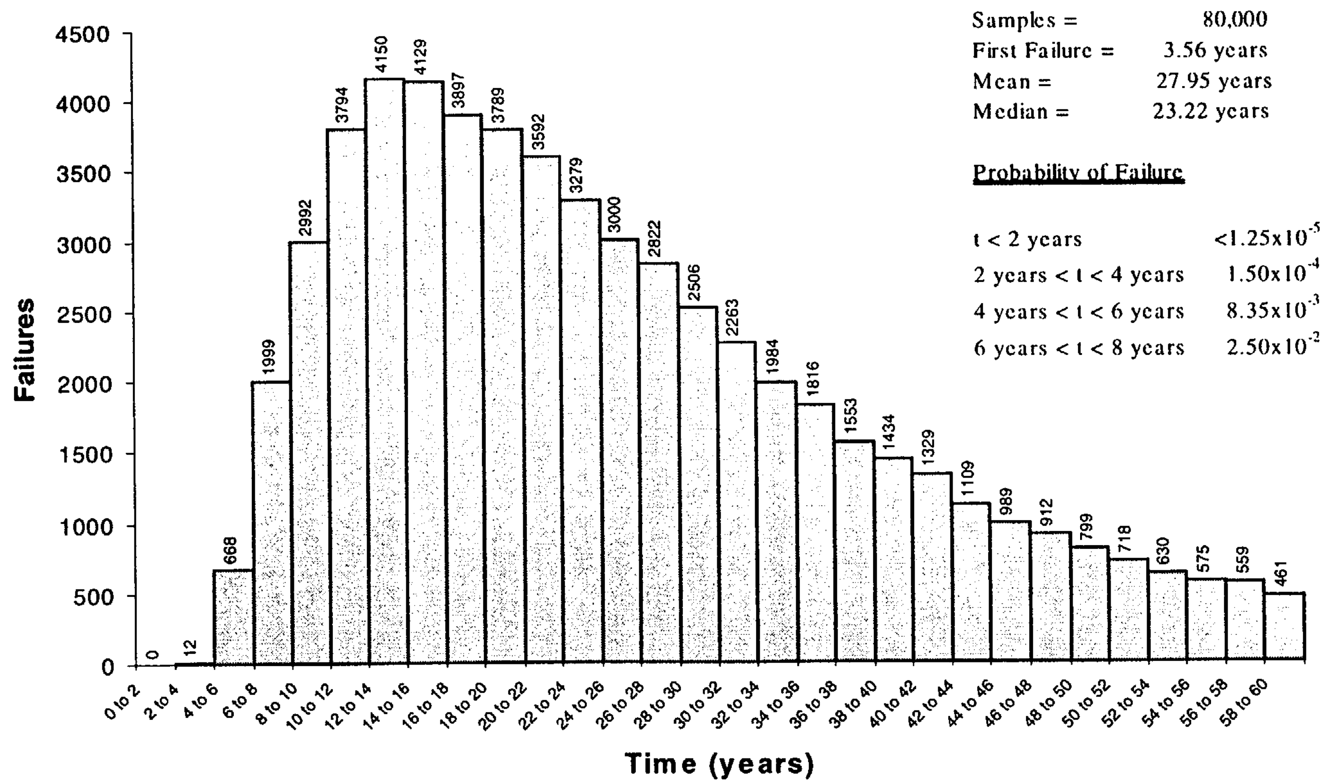


Figure 8. Probability of Net-Section Failure versus Time after Initiation of Outside Diameter PWSCC



Appendix A

Leakage Assessments

The leakage assessments due to various postulated flaws and due to lack of fusion in the CRDM nozzle/J-groove weld region are addressed in this section. As a result of stress analyses of the B&W-design CRDM nozzles, it has been previously demonstrated that during normal operation steady state conditions an annular gap develops (above the CRDM weld to the RV head) between the CRDM nozzle and the RV head penetration (Reference 3). Of particular interest is the prediction of a radial gap in a previously interference-fit region. The prediction of this radial gap during steady state operating conditions is utilized in the assessment of leakage rates through the CRDM nozzle/head annulus.

The radial gaps are different for the two types of CRDM nozzles that were evaluated by a detailed stress analysis, the center nozzle design and the outermost nozzle design. For the center nozzle, the initial interference-fit between the nozzle and the head separates to form a 0.003 inch maximum radial gap above the weld during steady state conditions. The average radial gap is 0.0016 inch and the minimum radial gap is 0.001 inch as illustrated in Figure A-1. For the outermost nozzle, the radial clearance in the initial interference fit region is approximately 0.001-inch minimum during steady state conditions as depicted in Figure A-2. However, a major portion of the periphery of the CRDM nozzle/RV head penetration shows a radial clearance of at least 0.002 inch and the maximum radial gap is about 0.003 inch.

A.1 Axial Flaws in CRDM Nozzle Above the J-Groove Weld

Leakage assessments for postulated through-wall axial flaws in the CRDM nozzle above the J-groove weld were previously addressed in Reference 3 and summarized below.

Reactor coolant system (RCS) leakage rates through postulated CRDM nozzle cracks and the annulus clearances between the nozzle and reactor vessel head were predicted by a parametric analysis. Both the crack lengths and annulus clearances were varied. Because of the high pressure-high energy conditions in the RCS, the sub-cooled coolant saturates, flashes, and then chokes at the exit of either the crack or annulus.

Leakage rates were obtained through an iterative process using the Homogeneous Equilibrium Model (HEM) critical flow tables and by solving single and two-phase pressure loss correlations. Since the flow chokes at either the exit of the crack (i.e., crack/annulus interface) or at the exit of the annulus (i.e., top of the penetration) for any given crack length and annulus clearance, both possibilities were considered in the analysis.

Therefore, the flow through the crack and annulus clearance was broken into two separate leakage flow paths to account for the two possibilities: (1) single and two-phase flow through the crack with choking at the exit of the crack, and (2) single phase flow through the crack and single and two-phase flow through the clearance annulus, with choking at the exit of the annulus.

For the first path with flow choking at the exit of the crack, the downstream leakage paths were calculated. The path with the lesser flow rate was considered to have the actual flow rate. Because of the choking properties of the flow, the greater flow rate was not possible. Thus, if the flow rate through the path with choking at the exit of the crack is less than that through the crack and the annulus, then the flow rate through the crack and the annulus is limited by choking at the exit of the crack.

In crack limited problems, the flow chokes at the crack exit. The pressure just upstream of the exit is assumed to be the exit pressure. Using this pressure, the RCS enthalpy, and the HEM tables, a trial critical mass flux is established. This flow rate is used in crack pressure loss calculations to determine a new value for the exit pressure. When the assumed and calculated values of the exit pressure agree, the solution has converged and the crack limited flow rate is established. The crack pressure loss calculations are divided into two calculations: sub-cooled flow and two-phase flow.

The results of the analysis show that for annulus clearances greater than 0.0001 inch and crack lengths less than 3 inches, the limiting factor is the size of the crack, while in cracks longer than 3 inches, the flow does not reach saturation conditions in the crack and therefore chokes at the exit of the annulus. For an annulus clearance less than 0.0008 inch, the flow rate will not exceed 1 gpm regardless of crack size. Likewise, for a crack length of 2 inches and shorter, the leakage flow rate will not exceed 1 gpm regardless of annulus clearance.

For a crack length of 2 inches and a maximum annulus clearance of 0.003 inch, the leakage flow rate was determined to be 0.559 gpm. However, it was demonstrated that as the crack extends from 2 to 3 inches in length, the flow rate would approach and exceed the leak detection capability rate of 1 gpm for annulus clearances of 0.001 inch and greater.

In addition, an independent leakage assessment was also performed as documented in Reference 38 and summarized below.

The objective of the report was to demonstrate that sufficient leakage of primary coolant, beyond the 1 gpm leak detection capability per Regulatory Guide 1.45 requirements, is feasible if a PWSCC indication of sufficient size occurs in the CRDM nozzle. The evaluation was based on applicable industry leak test data to the CRDM nozzle/head annulus (subsequently written as "CRDM annulus") gap. An inventory of experimental data on two-phase critical flow experiments were reviewed to help identify those that are applicable to the problem of predicting leakage rates through the CRDM nozzle and the annulus between the nozzle and the RV penetration.

Only the most pertinent data from the literature of experimental investigations were considered in the assessment of leakage rate through the CRDM annulus. The experiments were determined to be pertinent based on review against key thermal-hydraulic parameters for the evaluation of leakage through the CRDM nozzle/closure head annulus.

The pertinent data were identified in the work of Agostinelli, et al., Amos and Schrock, and Matsushima, et al. (see Reference 38 for these citings). The data from the first two references, when related to the CRDM problem predicted leakage rates greater than 1 gpm. The data from the third reference, when related to the CRDM problem, corresponded to a leakage rate of 0.6 gpm. However, the experiment was based on a stagnation pressure of only 975 psi and the stagnation pressure associated with the CRDM nozzle is 2250 psi. Accounting for the higher stagnation pressure should result in a predicted leakage rate greater than 1 gpm. Therefore, it is concluded in the report that, based on the plant's leak detection capability of 1 gpm within an hour per Regulatory Guide 1.45, the leakage through the CRDM annulus (under the conditions discussed in the report) will be detectable. Furthermore, it should be noted that the prediction of the leak rates given above were conservatively determined using the crack opening area of the CRDM annulus corresponding to a radial gap of only 1 mil. The report also concludes that, should a CRDM nozzle have a through-wall crack, a leak rate of 0.04 gpm to less than 1 gpm will result in significant accumulation of boric acid crystals.

A.2 Axial Flaws Within the J-Groove Weld

Flaws should grow axially through the J-groove weld due to the nature of the stresses in the J-groove weld. For a PWSCC-type crack, it may break the surface as a very tight or pinhole-type crack in the annulus region. These types of cracks would result in a low leakage as has, for example, been observed during the visual inspection of CRDM nozzle number 21 at ONS-1 in December 2000 (Reference 17) and at ONS-3 in February 2001. The maximum amount of boric acid crystals observed around the base of the ONS-1 CRDM nozzle number 21 was approximately 0.5 in^3, signifying a very low leakage rate through the crack. Only small quantities of boric acid crystals were present on the ONS-2, ONS-3, and ANO-1 RV heads, as well.

A.3 External Circumferential Flaw in CRDM Nozzle

An assessment of external circumferential crack growth in the CRDM nozzle above the J-groove weld was addressed in Reference 7. If it is postulated that a circumferential crack propagates through-wall and grows circumferentially along the weld-nozzle interface region, the potential safety concern is detachment of the upper nozzle from the lower nozzle section and its ejection from the closure head. However, detection of leakage prior to tube failure is predicted to occur.

Based on a limit load analysis of the CRDM nozzle geometry, the net section limit ligament is less than 25%. Postulating that a large portion of the nozzle cross-section contains a through-wall circumferential crack, there is ample room for leakage to occur before approaching the net section limit ligament. This will allow sufficient leakage of steam through this large crack to be detectable, thereby providing ample warning to prevent the failure of the nozzle. The flow rates were predicted (without consideration of potential "leak-plugging" in a narrow annulus) for a six-inch circumferential through-wall crack (nearly 50% of the circumferential extent, as observed in nozzle number 56 at ONS-3). For annulus clearances of 0.001 inch, 0.0016 inch and 0.002 inch (to cover the ranges of the predicted clearances during normal steady state operation for the center nozzle to the outermost nozzle), the leakage rates were determined to be 0.4 gpm, 0.8 gpm and 1.2 gpm, respectively.

A.4 Lack of Weld Fusion Areas in the J-Groove Weld

The allowable lack of weld fusion areas in the J-groove weld of the B&WOG plants was addressed in Reference 8. Framatome ANP performed an inspection of the nozzle-to-vessel head welds in a section of Midland Unit 1, which is typical of the B&WOG plants. The inspections revealed that the majority of the indications were located at the CRDM nozzle-to-weld interface, and all indications were less than 2 inches (51 mm) long. Most of the indications detected in the Midland welds are believed to be slag inclusions, with a fewer number of areas indicating lack of fusion of the weld zone. The two areas of concern for the lack of fusion are the CRDM nozzle-to-weld interface and the head-to-weld interface. Both these areas were evaluated to determine the minimum weld area required to meet the ASME Code primary shear stress limits (i.e., allowable lack of fusion size) and to determine the weld area required to limit the shear stress to the shear flow stress (i.e., critical lack of fusion size). It was demonstrated that the CRDM nozzle-to-weld interface was more limiting. The results showed that approximately 67% (corresponding to 8.4 inches of circumferential extent) of the total weld area may be unfused and still meet the ASME Code shear stress limit. Similarly, using the Tresca shear flow stress criteria, it was shown that 85% (corresponding to 10.6 inches of circumferential extent) of the total weld area may be unfused and still have sufficient strength to prevent a catastrophic failure.

The allowable lack of fusion size and indeed the critical lack of fusion size have significant circumferential crack lengths such that sufficient leakage can be demonstrated for these types of cracks.

A leakage assessment for this postulated circumferential crack in the weld was performed using the methodology very similar to that described in Section A.1. The only difference is that only one leakage path is considered which represents the annulus. The flow is assumed to choke at the exit of the annulus.

The crack lengths required to achieve the leak detection capability rate of 1 gpm were determined for annulus clearances of 0.0016 inches and 0.002 inches. These annulus clearances correspond to the average radial gaps during steady state normal operating conditions for the center and outermost CRDM nozzles, respectively. It was determined that a crack length of 7.5 inches in the center nozzle (annulus of 0.0016 inch) is required to achieve a 1 gpm leak rate. Similarly, it was determined that a crack length of 5.0 inches in the outermost nozzle (annulus of 0.002 inch) is required to achieve a 1 gpm leak rate.

Figure A-1. Radial Clearance for Center Nozzle

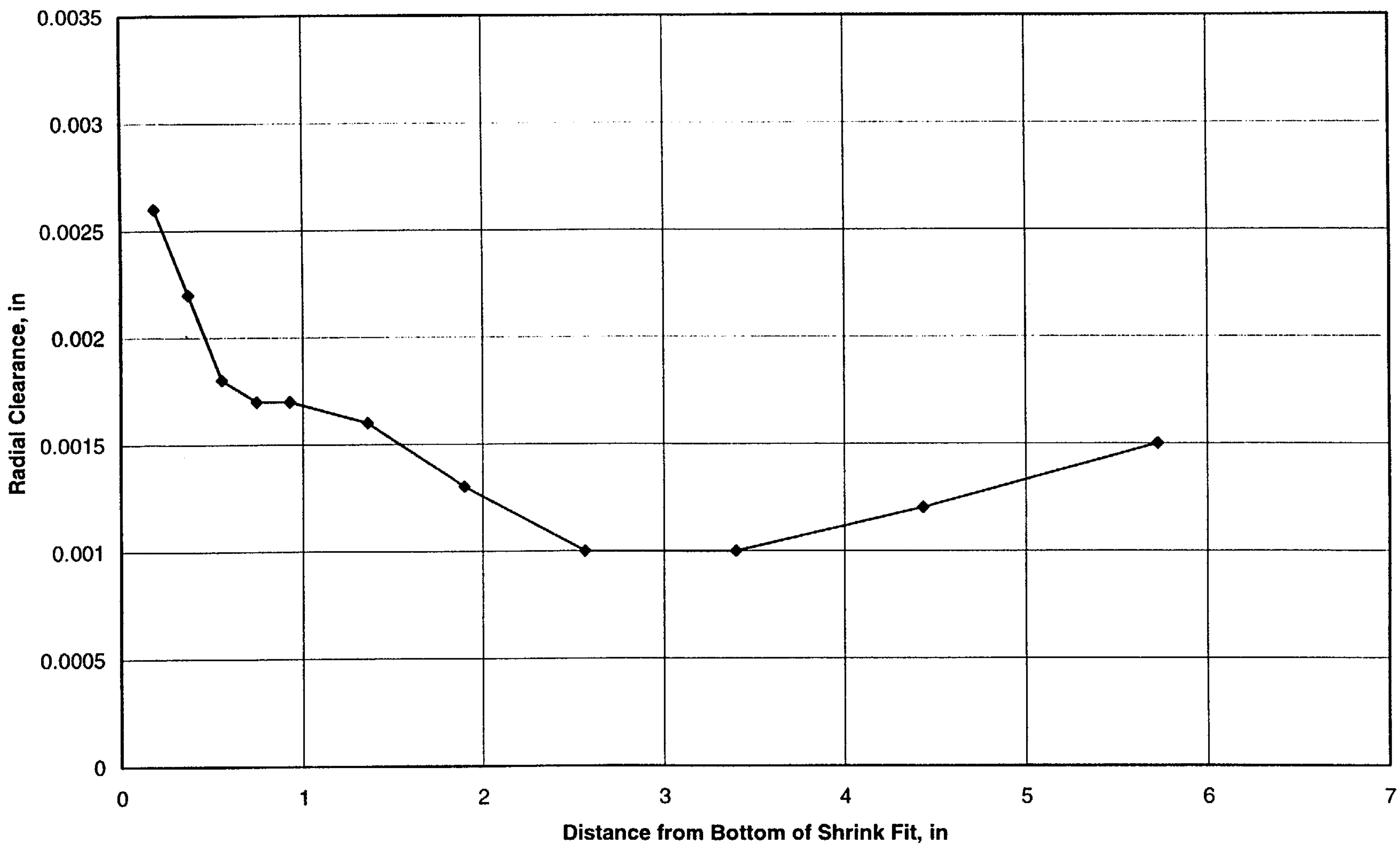


Figure A-2. Radial Clearance for Outermost Nozzle