Materials Reliability Program: Reactor Vessel Head Nozzle and Weld Safety Assessment for B&W Plants (MRP-103 NP)

1009402

Final Report, April 2004

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REPORT SUMMARY

Background

PWSCC of Alloy 600 materials, first observed in steam generator tubes, was found to be occurring in PWR reactor vessel upper head penetrations in 1991, after a leak was observed in a French plant. Cracking was observed in a few PWRs in the USA in the 1990s, but no severe cases were found until fall of 2000, when a number of leaks were identified. This safety evaluation was prepared to demonstrate that PWRs of Babcock & Wilcox (B&W) design can continue to operate safely.

Objectives

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 Alloy 600 reactor vessel (RV) closure head penetration nozzles.

Approach

A safety assessment was performed to address the potential for PWSCC cracking of RV closure head penetration nozzles and welds at the B&WOG plants. The assessment included new work and reviews of prior work in the areas of stress analysis, flaw growth, leakage assessment, wastage assessment, loose parts assessment, safety analysis review, and risk and collateral damage assessment. The assessment considers both axially and circumferentially oriented flaws. Boric acid corrosion concerns were addressed for a variety of conditions and leakage rates potentially assumed to occur.

Results

It is concluded that if PWSCC cracking occurs, flaws would predominantly be oriented in a longitudinal, or axial, plane, and as such would not promote catastrophic failure of the nozzle by ejection. However, circumferential flaws also have a potential to occur above the J-groove weld once leakage reaches the RV closure head penetration annulus. A short, isolated circumferential flaw above the weld on the outside surface of the nozzle would take more than 10 years to grow through-wall, while a long circumferential flaw (where multiple flaws have joined) could grow from the outside surface to the inside surface in about 3.5 years. It has also been demonstrated that even if a 180° through-wall circumferential crack would form in the nozzle wall above the weld, it would take more than 18 years to grow another 90°, which would still not compromise the structural integrity of the nozzle to the point that the nozzle would fail by ejection. It was determined that boric acid corrosion wastage of the RV closure head penetration 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 visual

inspection of the RV closure head area. 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 other plant safety analyses already completed to support current plant operation. In addition, it has been demonstrated through risk analysis that the risk from potentially undetected CRDM nozzle cracks is "small" per the guidelines of Regulatory Guide 1.174. The estimated core damage frequency due to OD PWSCC of the CRDM nozzles is 1.36×10^{-6} 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.

EPRI Perspective

This document addresses the assumed presence of PWSCC in either the nozzle base materials or the partial penetration (or "J-groove") welds used in their attachment to the RV closure head. 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 attachment welds. This assessment utilizes and builds upon the existing safety evaluations performed for CRDM nozzle PWSCC. The results in this document will be combined with probabilistic assessments of the likelihood of occurrence of the various failure modes to produce an overall safety assessment of the RPV top head nozzle cracking issue.

Keywords

Primary water stress corrosion cracking
PWSCC
Stress corrosion
Boric acid corrosion
Alloy 600
Alloy 82/182
Alloy 690
Alloy 52/152
CRDM nozzle
CEDM nozzle
J-groove weld
Reactor vessel head
Reactor vessel closure head
Reactor vessel closure head
ICI
Safety Assessment
Circumferential cracking

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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 Alloy 600 reactor vessel (RV) closure head penetration nozzles. This document addresses the assumed presence of PWSCC in either the nozzle base materials or the partial penetration (or "J-groove") welds used in their attachment to the RV closure head. This safety assessment applies to the RV closure heads for the following nuclear stations:

Plant[a]	Owner
Davis-Besse (D-B)	FirstEnergy 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)	Progress Energy-Florida
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; the results of additional stress, structural, flaw tolerance and fracture mechanics analysis; and industry events applicable to the B&W-design occurring since late 2000; the objective of this document is met. In addition, the results of a review of the existing safety analyses and a risk and collateral damage assessment (Sections 8 and 9) show that defense in depth is assured.

2
INTRODUCTION

Cracking was first observed in a CRDM nozzle at the French pressurized water reactor (PWR) Bugey Unit 3 in 1991. Between 1991 and 1996, the U.S. nuclear industry developed safety assessments (References 1-3) and several utilities 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 closure head penetration nozzles at B&WOG plants. [It is noted that the B&WOG plants have two types of RV closure head penetration nozzles, which consist of CRDM nozzles at all the plants and thermocouple nozzles at ONS-1 and TMI-1 only. [a]]

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 closure 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 closure 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, access to the RV closure head is available to assess the structural and leak tight integrity of the RV closure 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 in late 2000, the circumferentially-oriented flaw indications revealed at ONS-3, ONS-2, and ANO-1 in 2001, and RV closure head wastage discovered at D-B in the spring of 2002 have introduced new concerns that must be addressed. This document provides a bounding safety assessment to address the potential severity of these concerns at the 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

[a] The Alloy 600 thermocouple nozzles were removed or replaced with Alloy 690 materials at ONS-1 during EOC-19 and TMI-1 during EOC-13 outages.

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 leakage sufficient to corrode the RV closure head), and that within this time, the leak would be detected during a visual inspection of the RV closure head area. Thus, the potential for cracking of CRDM nozzles would 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 and the weld (Reference 7). In this case, the initiation and propagation of the axial crack preceded exposure of the nozzle outer surface in the annulus (above the weld) to leaking reactor coolant. An addendum to the B&WOG safety evaluation was prepared to address this concern in December 1993 (Reference 8). 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 nondestructive examination (utilizing ultrasonic testing) data from several sources to determine whether similar manufacturing defects typically exist. First, the data from Ringhals Units 2 and 4 and data from a cancelled Westinghouse designed reactor, Shearon Harris, were acquired from the Westinghouse Owners Group (WOG). Second, the B&WOG performed an inspection of the RV closure head from Midland Unit 1, which was a cancelled nuclear station fabricated and designed by B&W.

Another addendum to the B&WOG safety evaluation was prepared to analyze these data (Reference 9). 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 B&PV 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 RV closure head and the ASME B&PV 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.

In 1998, the industry formed the Materials Reliability Program (MRP) to address Alloy 600 and other materials issues. The Owners Groups and the MRP have since worked together to develop appropriate responses to the CRDM nozzle leakage concerns.

The NRC has since issued a number of information notices, bulletins, an order, and an issue summary regarding PWSCC of CRDM nozzles and the RV closure head degradation seen at D-B (References 10 – 18). Each of the bulletins has required the B&WOG member utilities to prepare written responses and submit them to the NRC in accordance with the provisions of 10 CFR 50.54(f).

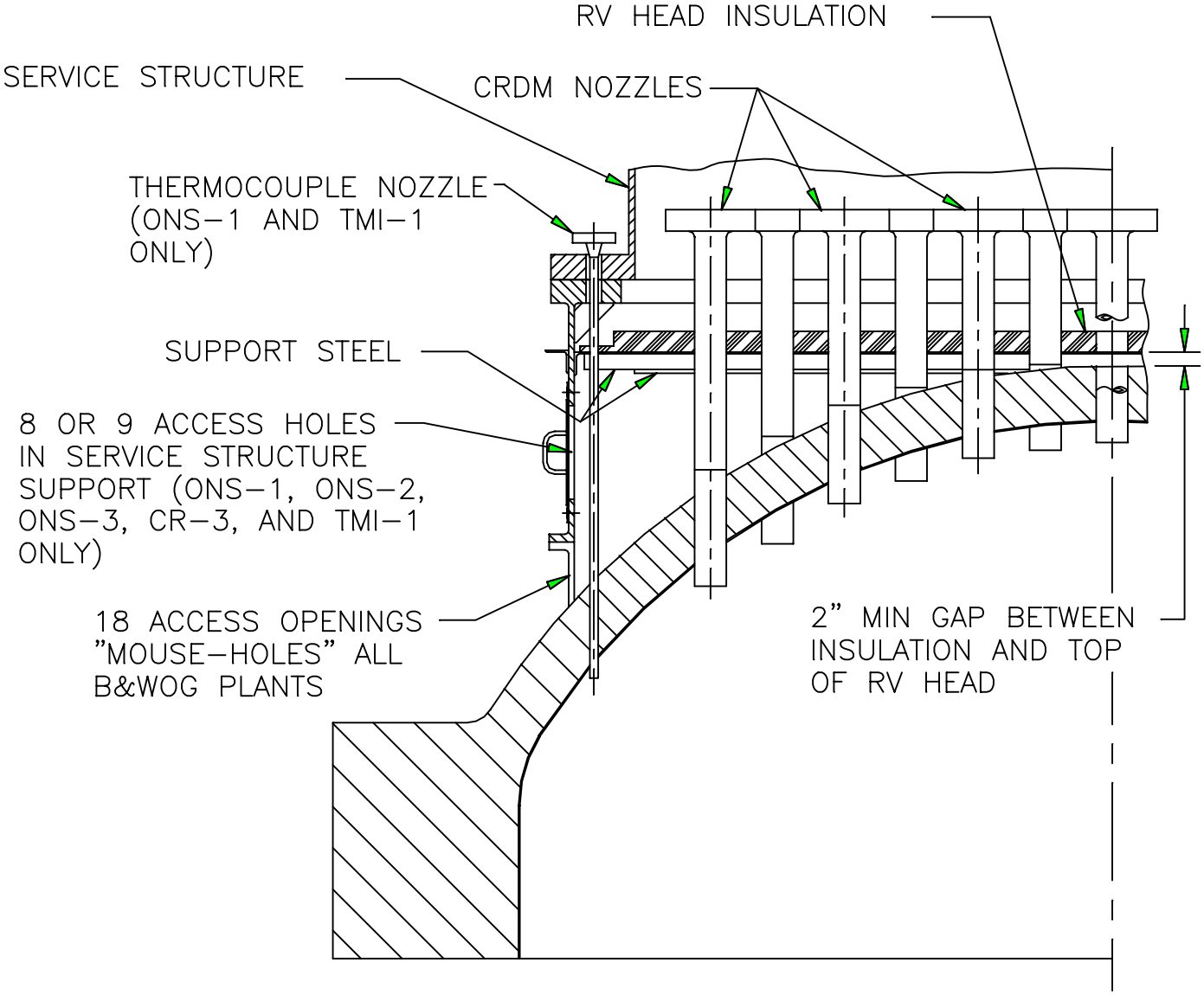
2.2 B&WOG Plant Inspections

The B&WOG utilities have included plans to visually inspect the CRDM nozzle area during their outages to determine if leakage is observed on top of the RV closure head, which would indicate through-wall cracking has occurred. In addition, walk-down inspections have been implemented in response to NRC Generic Letter 88-05 (Reference 19) at each of the B&WOG plants. The walk-down inspections include an enhanced visual inspection of the gasket area and RV closure head during every refueling outage (a period between inspections of 18 or 24 months). The B&W RV 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 closure head (see Figure 2-1). If any leaks or boric acid crystal deposits are noted during inspection of the RV closure head area, an evaluation of the source of the leak and the extent of any wastage is performed. This program has shown to generally be effective, as evidenced at ONS and ANO-1. However, management and programmatic issues may affect the capability to address such leakage in a timely manner (e.g., D-B [Reference 20]). 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.

NRC Bulletin 2001-01, “Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles” (Reference 11) was issued in August 2001, which requested plant-specific information regarding the structural integrity of the RV closure 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 21-25).

NRC Bulletin 2002-01, "Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity" (Reference 13) was issued In March 2002. This Bulletin requested additional plant-specific information regarding plant inspections and compliance with applicable regulatory requirements. This was followed in August 2002 with NRC Bulletin 2002-02, "Reactor Pressure Vessel Head and Vessel Head Penetration Nozzle Inspection Programs" (Reference 15), and then in February 2003 by an NRC Order (Reference 17), which established inspection requirements for reactor pressure vessel closure heads at PWRs. Again, responses were prepared by each of the B&WOG member utilities to each of these Bulletins. Reference 18 contains a summary of the NRC staff's review of the responses to Bulletin 2002-01.



Note: The Alloy 600 thermocouple nozzles were removed or replaced with Alloy 690 materials at ONS-1 during EOC-19 and TMI-1 during EOC-13 outages.

Figure 2-1
Side View Schematic of Original B&W-Design Reactor Vessel Closure Head, CRDM Nozzles, Thermocouple Nozzles, and Insulation.

3
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 closure head, the stainless steel cladding in the closure head, and the Alloy 182 weld material used for the partial penetration (J-groove) weld and butter between the nozzle and closure head. The purpose of this analysis was to determine the preferential direction for nozzle inside surface 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 (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 RV closure head, the center nozzle was analyzed using a two-dimensional model. Since the outer hillside nozzle penetrates the closure 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 closure head):

a. Shrink fit of the nozzle within the closure head during installation (0.0010 inch nominal diametric interference).

b. Simulated welding of the nozzle to the closure head (heatup of the weld material to 2470 ^{o}F and cooldown to develop residual stresses).

c. Cold hydrostatic testing of the completed closure head assembly at a pressure of 3125 psig.

d. Steady state operation at a temperature of 600 ^{o}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 the original Alloy 600 RV closure head penetration nozzles installed at B&W-design plants range from 31 ksi to 64 ksi (see References 3 and 5). 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

The 1993 stress analysis showed that at most locations the inside surface hoop stress is higher than the axial stress. 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. Thus 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 on the nozzle outside surface would also promote the propagation of outside diameter (OD) initiated circumferential cracks above the weld.

In the weld region, hoop stresses are about two times the axial stress at the same location. Therefore any PWSCC cracking that were to develop in the weld should be oriented in an axial plane (relative to the nozzle).

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 inside surface 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 26), 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 closure head (see Section 4.0), the flaw would continue to grow into the Alloy 600 CRDM nozzle, as seen at ONS-1 (Reference 27).

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 (0.020 in) 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 closure head.

3.3.2 Circumferential OD CRDM Flaws

Since the nozzle OD surface hoop stresses in the vicinity of the weld are about two times the surface axial stresses; flaws originating at this location should be oriented in an axial plane. Development of an axial leak path through the weld to the annulus between the nozzle and RV closure head would however, expose the outside surface of the nozzle above the weld 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 closure 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 (0.020 in) in depth, initiates immediately after the plant returns to service. Using 0.5 mm (0.020 in) 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 closure head, it is believed that 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), based on satisfaction of primary stress limits using a safety factor of 3 (Reference 28).

Alternate estimates of circumferential crack growth have also been performed using stress intensity factors calculated by Dominion Engineering, Inc., that include the effects of crack growth-induced stress relaxation. These results show that it would take 23.4 years for a downhill-centered 180° circumferential through-wall crack to grow another 90° around the nozzle (to 270°). A 180° circumferential crack centered on the uphill side would take 18.9 years to grow to 270° (Reference 29).

Lack of fusion 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 9, there may be up to 67% lack of fusion between the nozzle and weld before the ASME B&PV Code primary shear stress limits are violated. It has been calculated that it would take two years for a 0.25-inch long 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 26). Based on operating experience (e.g., 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 at ONS-3 at the nozzle 34 location. It was located in the weld material and spiraled from a distance of 1.125 inches 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
FLAW GROWTH INTO THE RV CLOSURE HEAD

A crack, propagating through the J-groove weld by PWSCC, will eventually grow to the RV closure 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 several plants (e.g., ONS-1 [Reference 27]), 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 30). SCC of steels containing up to 5% chromium is most frequently observed in caustic and nitrate solutions and in media containing hydrogen sulfide (References 31 and 32). 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 33). 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 34 and 35) and by recent events with PWSCC of Alloy 600 weld materials at ONS-1 and VC Summer (References 36 and 37). Although a PWSCC-initiated flaw may continue to propagate by fatigue crack growth into the low alloy steel closure head, this is considered to be insignificant over several operating cycles (i.e., until a RV closure head is replaced) 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. General corrosion in this crevice-like area could potentially plug with build-up of corrosion products and cease. However, based on the data in Reference 38, a conservative corrosion rate in stagnant PWR environment conditions is 1-3 mils/year.

5
LEAKAGE ASSESSMENT

The B&WOG had performed leakage assessments for various potential leak scenarios expected prior to the recent (late 2000 through 2003) leak events at the B&WOG plants. The results from these previous assessments are documented in detail in Appendix A. The recent experience, however, indicates that the actual leak rates are apparently very low based on the amount of boric acid crystals observed on the RV closure head in the vicinity of leaking nozzles. It was estimated that approximately 0.5 in^3 was present around CRDM nozzle number 21 at ONS-1 when leakage was observed. 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 postulated that a small leak and narrow annulus can lead to "leak plugging" by the deposition of boron compounds or 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, actual leakage is anticipated to be minimal until a long axial flaw (i.e., approximately the length of the RV closure head penetration) develops above the weld or the annular gap widens from corrosion wastage of the low alloy steel RV closure head material (see Section 6).

6
WASTAGE ASSESSMENT

The purpose of this section is to assess the potential damage that can occur to the RV closure 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 RV closure head as a result of exiting boric acid crystals and borated steam condensing on the closure 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 RV closure 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 38.

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 closure head J-groove welds.

It was determined in Reference 3 that this type of leakage would lead to corrosion of the RV closure head penetration, at a maximum volumetric metal loss rate of 1.07 in^3/yr. This volumetric rate is based on testing in which a maximum linear rate of 2 ipy was observed and did not assume that the conditions present at Davis-Besse would exist. 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 closure head, that safe operation of the plant would not be affected as a result of this level of corrosion of the RV closure head penetration.

For the case of the conditions observed at Davis-Besse (Reference 20) and available test and field data (Reference 38), wastage could be hypothesized to progress in the following manner:

a. Leakage initiated in the annulus region near the CRDM nozzle #3 crack (~1991-1994 timeframe).

b. Minimal wastage occurred within the annulus as the borated water leaked to the RV closure head surface where the initial corrosion most likely occurred. This occurred over a period of approximately 2 years (1996-1998).

c. Corrosion propagated outward from nozzle #3 toward nozzle #11 and down into the annulus from the RV closure head OD surface at the rate of 1.07 in^3/yr. This occurred over a period of approximately 1 year (1998-1999). [The actual profile (length, width, and depth) of this wastage is unknown, but could be postulated to be similar to one of the three profiles used in the 1993 CRDM nozzle safety assessment boric acid corrosion discussions.]

d. The CRDM nozzle through-wall crack was long enough (i.e., length above the weld) and the annular gap was now wide enough for a sufficient leak rate to cause erosion-corrosion to occur at a rate of approximately 11 ipy (Reference 38). This occurred over a period of approximately 0.5 year (in 1999).

e. Erosion-corrosion ended as the width of the cavity (i.e., length from nozzle # 3 toward nozzle #11) expanded to the point that the cavity was able to fill with a slurry of borated water and corrosion (iron oxide, etc.). Continued wastage then occurred at a significantly slower uniform rate (approximately 3 ipy). This occurred over the remaining time of operation, i.e., 2.5 years (1999-2002).

Leakage occurring from CRDM nozzle flanges, located above the RV closure head and insulation, may have contributed to the confusion at Davis-Besse about the source and timing of the deposits on the closure head. Although many assumptions are utilized in developing this hypothesis, it is reasonable and supportable based on the most representative test data and field experience available.

Finally in the 1993 safety assessment, 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 closure head area. It should be noted that this minimum six-year period represents corrosion of the RV closure head at the rate of 1.07 in^3/yr, which would occur when a sufficient leakage rate had been realized (steps a-c above). Thus, the potential for cracking of CRDM nozzles and RV closure 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 closure head J-groove weld leakage at ONS-1, ONS-2, ONS-3, ANO-1, TMI-1 and CR-3. In all cases, only minimal corrosion (wastage) was observed. However, it must be realized that the conditions and wastage observed at Davis-Besse, which apparently occurred as a result of human performance factors, did challenge these conclusions (Reference 20).

7
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 closure 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 39). The potential transport of fragments originating at the RV closure 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 7-1). 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 7-2 and 7-3). It has been calculated that there is a 25% chance 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 closure head and the fact that low cross flow velocities in this region would tend to allow debris to fall vertically (if all of the CRDM nozzles are attached). If fragments enter the column weldments, they may be stopped on one of the control rod guide tube brazements where a relatively small fragment (<0.75 in.) could prevent the complete length of the control rod from inserting if it became trapped between a brazement and the control rod spider.

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 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.

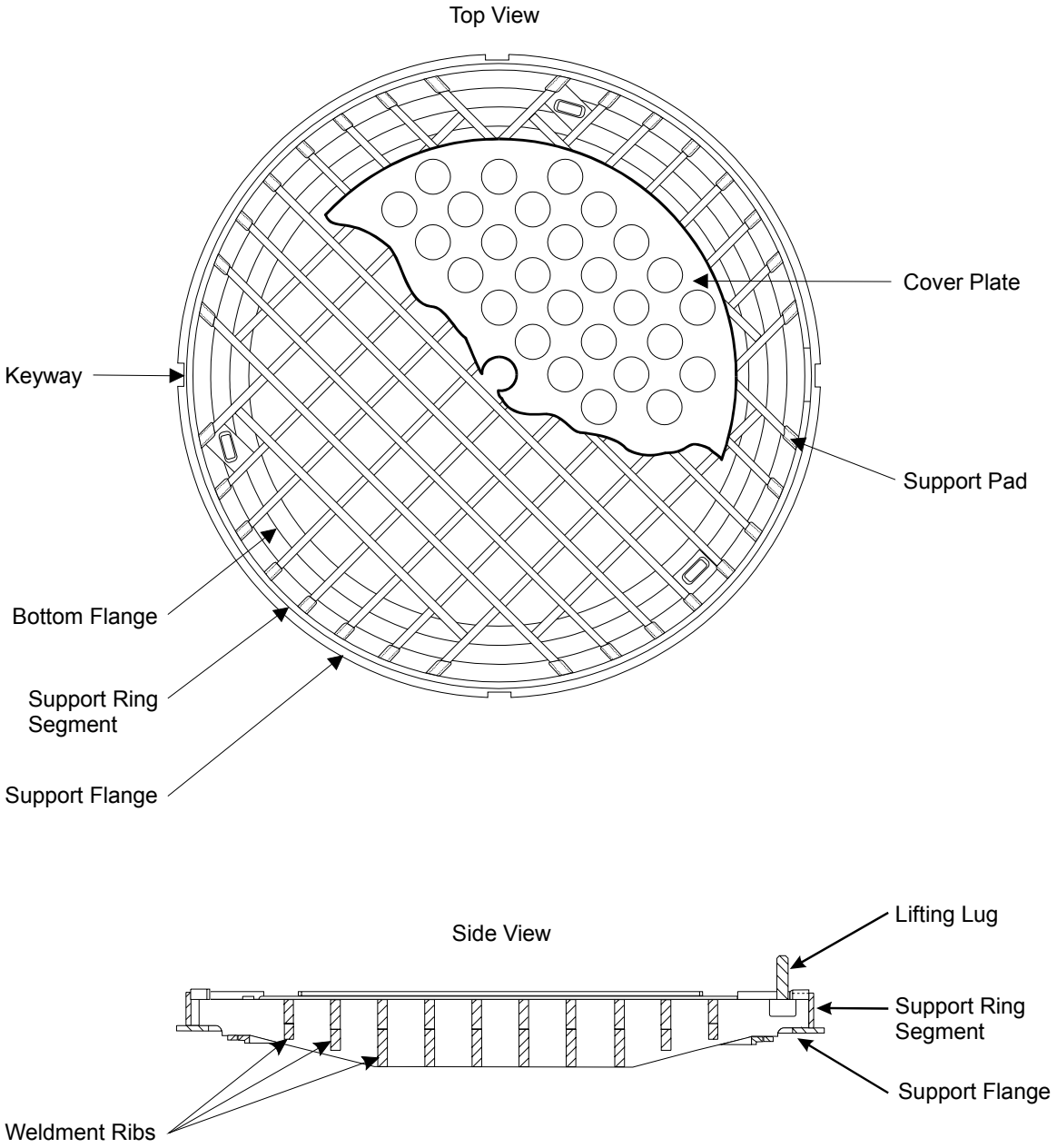


Figure 7-1
Plenum Cover Assembly

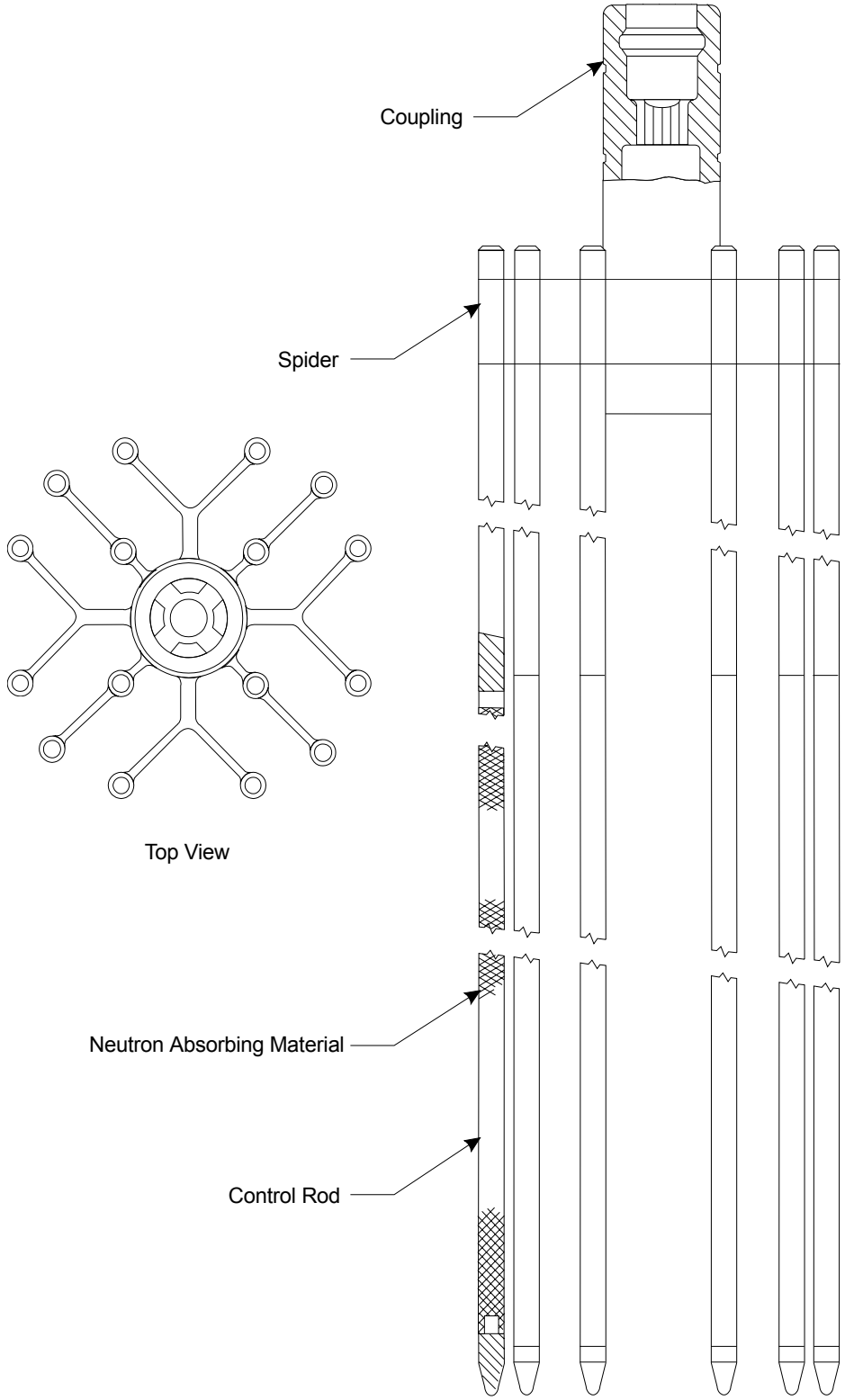


Figure 7-2
Control Rod Spider Assembly

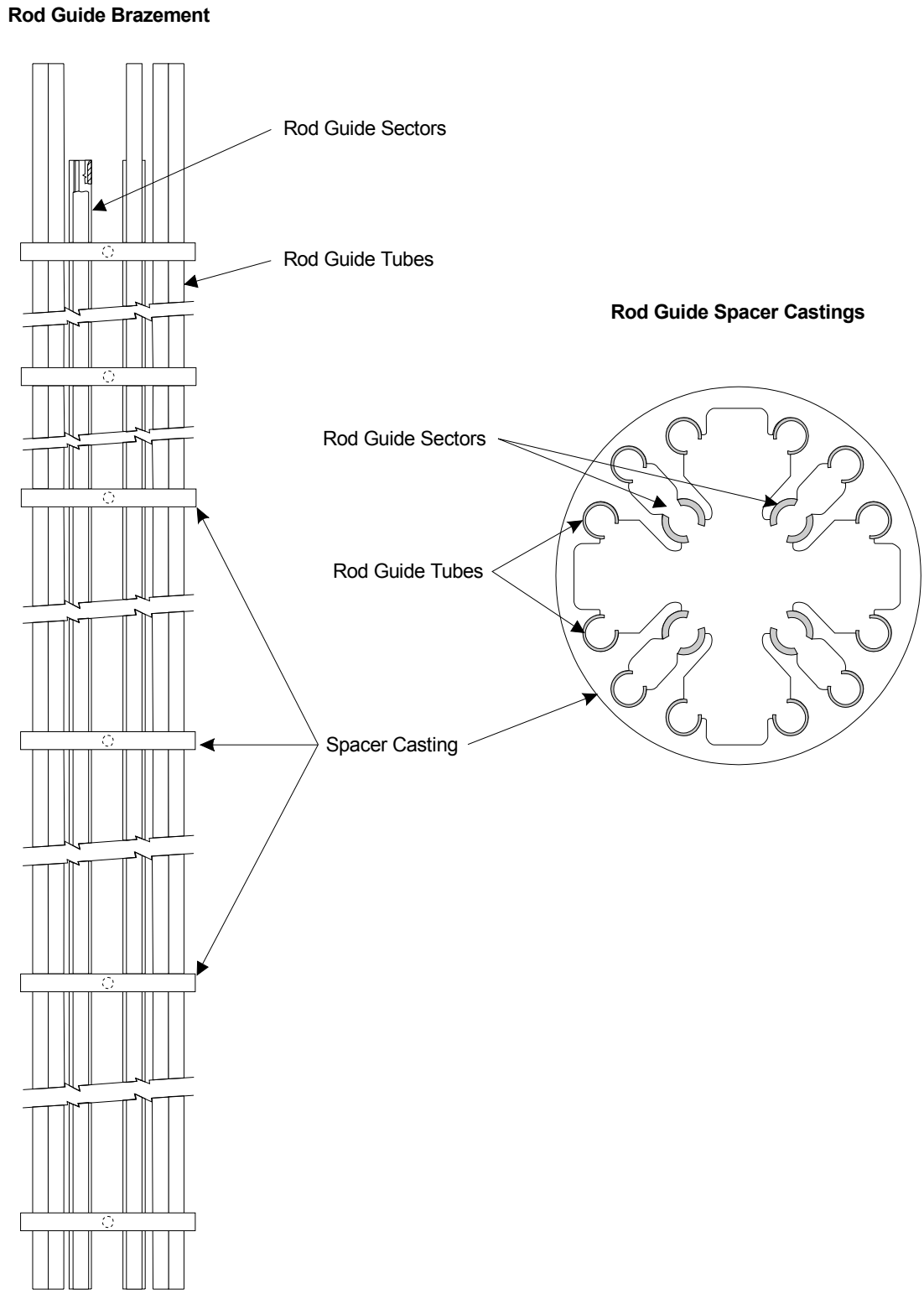


Figure 7-3
Control Rod Guide Brazement Assembly

8
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.

Loss of coolant accidents (LOCA) analyses and other plant safety analyses are performed to justify that nuclear power plants can be safely shut down following postulated accidents. Although these analyses do not specifically consider failure (i.e., complete severance) of a CRDM nozzle, they consider events that have more limiting consequences. LOCA analyses typically postulate breaks in RCS pipes. The break areas considered range from those that the plant normal makeup capacity can keep up with, up to and including a double-ended guillotine break of the hot leg. Acceptable core cooling must be demonstrated in the short term as well as the long term. Plant 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 B&W design 177-FA plants.

As described in the previous sections, once a crack initiates, it is estimated that it may take up to six years for it to propagate through the CRDM nozzle wall thickness and begin to leak. It will take additional time for the leakage to reach a detectable rate. A rigorous visual inspection program will assure detection of minor leaks that grow at slow rates. 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 resulting from boric acid corrosion of the RV closure 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 other plant safety analyses.

Loss-of-Coolant Accident Safety Analyses

Plant LOCA analyses do not specifically analyze the potential failure of the RV 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% shutdown margin exists at hot zero power conditions.

Technical Specifications require plant shutdown if any pressure boundary leak or unidentified leak rates of 1 gpm or greater are detected. If the leak rate is higher than 1 gpm 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.

Other Safety Analyses

Other plant 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% 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
RISK AND COLLATERAL DAMAGE ASSESSMENT FOR CRDM NOZZLE CRACKS

This section provides a risk analysis and collateral damage assessment 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 at ONS.

9.1 Risk Assessment

The risk evaluation (detailed in Appendix B) addresses the period from 2002 through RV closure head replacement (~2005). This analysis assumes a fuel cycle length of two years, which is conservative for plants with 18-month cycles for the same time period (because the 18-month plants will have more frequent inspections over the same four-year period). The risk estimate addresses the risk from potential CRDM nozzle failure occurring before vessel closure head replacements, which are tentatively scheduled to take place in B&WOG plants approximately one or two fuel cycles from the time of this analysis (2002).

The estimated core damage frequency compares favorably to the risk acceptance guidelines contained in Regulatory Guide 1.174 (Reference 40) for core damage frequency. Although Regulatory Guide 1.174 addresses permanent changes to the licensing basis, and the issue here is temporarily (until RV closure head replacement), it still provides a useful guideline for acceptability of the incremental risk. Per these guidelines, the risk of operation with potentially undiscovered CRDM nozzle cracks is categorized as “small” over the next two fuel cycles.

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 RV 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.

In addition, the public health risk associated with the CRDM nozzle cracking is correspondingly small. For example, the conditional population dose for a medium break LOCA core damage accident from a typical B&WOG plant (ONS) probabilistic risk assessment (PRA) is 1.1×10^4

person-rem (Reference 41). The estimated core damage frequency of 1.36×10^{-6} per reactor-year corresponds to a public health risk of only 1.5×10^{-2} 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 inspections, the nozzle cracking does not significantly increase the core damage frequency that is estimated in PRAs for B&WOG plants. Furthermore, the consequences of a CRDM nozzle failure are less severe than the LOCAs assumed in the FSAR analyses. 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.

9.2 Collateral Damage Assessment

Collateral damage caused by CRDM nozzle failure (such as from missiles or loose parts) could potentially affect other CRDMs or engineered safeguards and degrade plant response to the LOCA. This possibility is also discussed in Appendix B (Section B.9) as it relates to event mitigation and risk.

It is concluded in Appendix B that with respect to emergency core cooling system (ECCS) performance, the conditional core damage probability (CCDP) for a medium LOCA is appropriate or conservative for the postulated LOCA involving a CRDM nozzle penetration. In addition, it is concluded that the potential increase in CCDP over the LOCA CCDP as a consequence of collateral damage that may be caused by ejection of a CRDM housing is insignificant. Therefore, the CCDP for a typical LOCA is representative of the risk from CRDM nozzle failure.

10
SUMMARY AND CONCLUSIONS

A safety assessment has been performed to address the potential for PWSCC cracking of RV closure 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 closure heads. This safety assessment utilizes and builds upon the existing safety evaluations performed for CRDM nozzle PWSCC (References 3, 8, and 9).

It is concluded that if PWSCC cracking occurs, flaws would predominantly be oriented in a longitudinal, or axial, plane, and as such would not promote catastrophic failure of the nozzle by ejection.

However, circumferential flaws also have a potential to occur above the J-groove weld once leakage reaches the RV closure head penetration annulus. A short, isolated circumferential flaw above the weld on the outside surface of the nozzle would take more than 10 years to grow through-wall, while a long circumferential flaw (where multiple flaws have joined) could grow from the outside surface to the inside surface in about 3.5 years. It has also been demonstrated that even if a 180° through-wall circumferential crack would form in the nozzle wall above the weld, it would take more than 18 years to grow another 90°, which would still not compromise the structural integrity of the nozzle 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 RV closure 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. 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 RV closure 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 2% wrap-around flaw to grow to an allowable 67% flaw size.

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 B&PV 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 closure head that will provide sufficient leak paths for a fairly large volume of steam to escape thereby providing leak detection.

Boric acid corrosion concerns were addressed for a variety of conditions and leakage rates potentially assumed to occur. It was determined that corrosion wastage of the RV closure head penetration 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 visual inspection of the RV closure head area. These calculations have been justified by the detection of boric acid crystal deposits around CRDM and thermocouple nozzles and the subsequent identification of RV closure head J-groove weld leakage at ONS-1, ONS-2, ONS-3, ANO-1, TMI-1, and CR-3. In all cases, only minimal corrosion (wastage) was observed. However, it must be realized that the conditions and wastage observed at Davis-Besse, which apparently occurred as a result of human performance factors, did challenge these conclusions.

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 the 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, it was 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 other plant safety analyses already completed to support current plant operation.

In addition, it has been demonstrated through risk analysis that the risk from potentially undetected CRDM nozzle cracks is “small” per the guidelines of Regulatory Guide 1.174. The estimated core damage frequency due to OD PWSCC of the CRDM nozzles is 1.36×10^{-6} 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 and/or volumetric inspections of the RV closure head will be sufficient to minimize public risk. The 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 closure 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 closure 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 was and remains very low (Reference 5).

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

5. Visual and/or volumetric inspections of the RV closure head will permit the plant owner to discover signs of CRDM nozzle leakage before there is a significant likelihood of total failure of a CRDM nozzle due to PWSCC.

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

7. All B&WOG utilities continue to perform nondestructive examinations (e.g., visual inspections and ultrasonic examinations) of the RV closure head and CRDM nozzles in accordance with their respective Generic Letter 88-05 responses, Bulletin 2001-01 responses, and NRC Order EA-03-009.

8. 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.

9. All of the B&WOG plant owners (except for D-B) have ordered replacement RV closure heads, containing materials that are more PWSCC resistant (Alloy 690 nozzles and Alloy 52 weld metal), which are expected to be installed by 2005. Davis-Besse replaced their RV

closure head with one from a canceled sister plant in 2002 that has CRDM nozzles fabricated out of Alloy 600.

11
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A
LEAKAGE ASSESSMENTS

The leakage assessments due to various postulated flaws 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 closure head) between the CRDM nozzle and the RV closure head penetration (Reference A1). 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/ closure 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 closure 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 closure 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 A1 and are summarized below.

Reactor coolant system (RCS) leakage rates through postulated CRDM nozzle cracks and the annulus clearances between the nozzle and RV closure 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 A2 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/closure 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 A2 for these citations). 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 A3) 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 closure 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 A4. 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 References

A1) "Safety Evaluation for B&W-Designed Reactor Vessel Head Control Rod Drive Mechanism Nozzle Cracking," BAW-10190P (B&W Owners Group Proprietary), May 1993.

A2) "Leakage Assessment Through CRDM Nozzle and Closure Head," BAW-2213 (B&W Owners Group Proprietary), June 1994.

A3) "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.

A4) "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.

Figure A-1. Radial Clearance for Center Nozzle

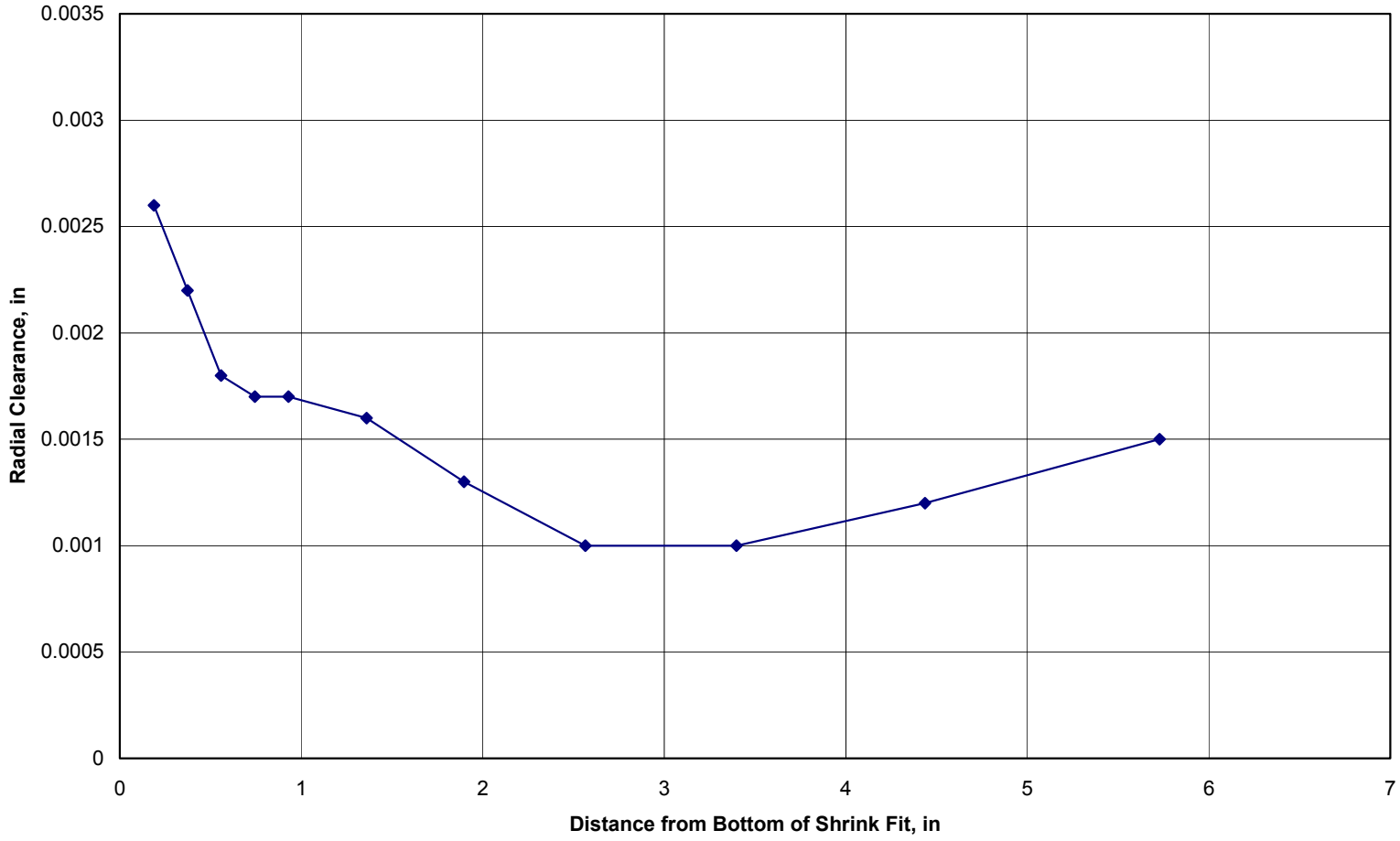

Figure A-1
Radial Clearance for Center Nozzle

Figure A-2. Radial Clearance for Outermost Nozzle

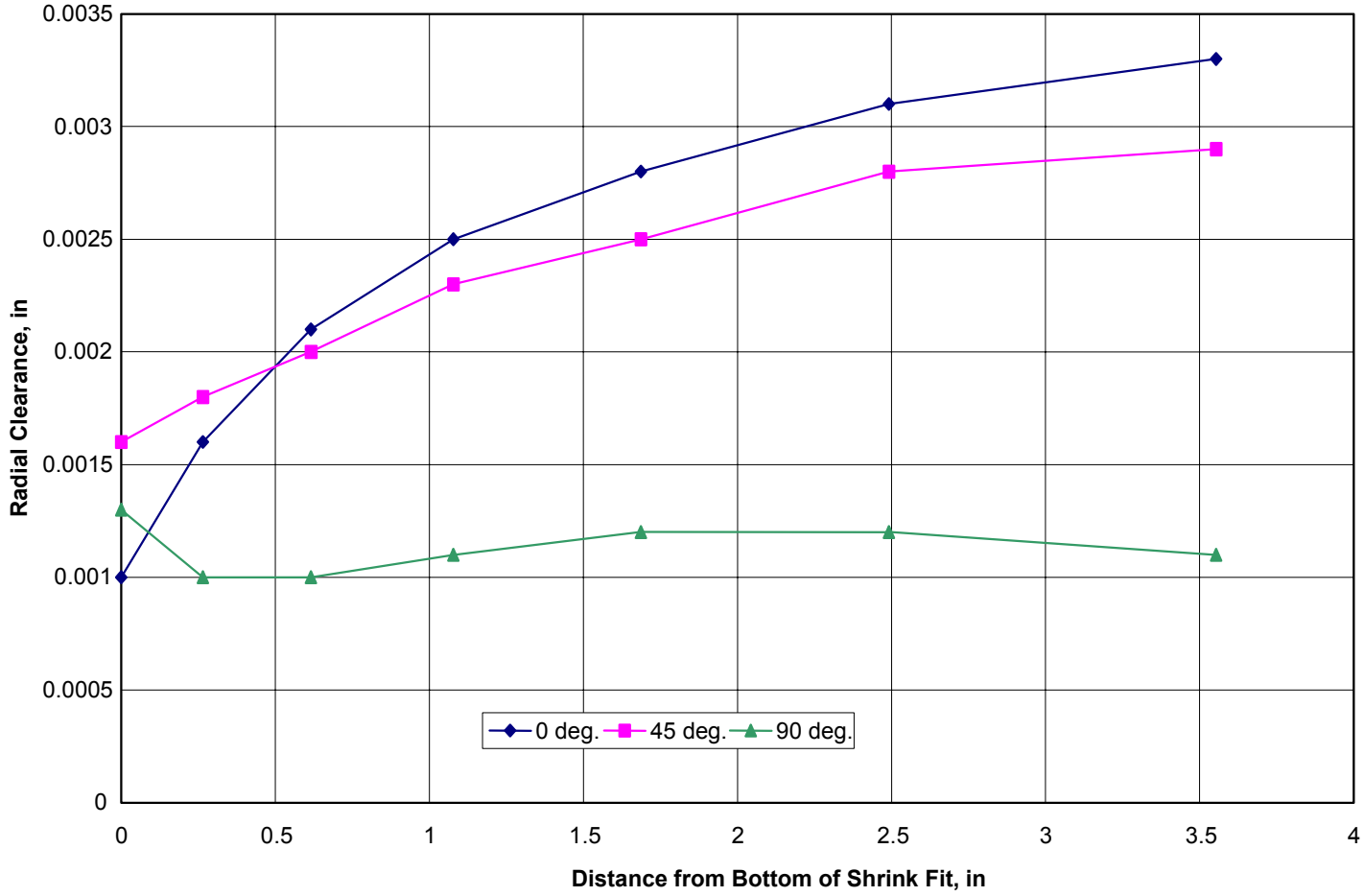

Figure A-2
Radial Clearance for Outermost Nozzle

B
RISK AND COLLATERAL DAMAGE ASSESSMENT DETAILS

B.0 Risk Assessment for CRDM Nozzle Cracks

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B.1 Potential Risks from CRDM Nozzle Cracking

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B.2 Identification of CRDM Nozzle Cracks that are a Risk Concern

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B.3 OD Circumferential Crack Risk

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B.4 Frequency of Weld or Nozzle Leak

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B.5 Probability that CRDM Nozzle Leakage is Undetected

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B.5.1 Reactor Vessel Closure Head Visual Inspections

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B.5.2 Estimate of Human Error Probability for Visual Inspections

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B.5.3 Probability that Circumferential Crack is Undetected by Volumetric Examination

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B.6 Probability of OD Crack Initiation

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B.7 Crack Growth to Failure of CRDM Nozzle

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B.8 Probability of Core Damage

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B.9 Collateral Damage (Reference B16)

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B.9.1 Effect of Collateral Damage on ECCS

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B.9.2 Effect of Collateral Damage on Reactivity Control

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B.9.3 Collateral Damage Conclusion

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B.10 Risk Analysis Results for OD PWSCC

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B.11 References

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Figure B-1
Event Tree for Frequency of Core Damage from Outside Diameter PWSCC in B&WOG CRDM Nozzles

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Figure B-2
CRDM Nozzle Leak Probability for Bounding Plant (worst B&WOG heat) versus EFPY with Weibull Shape Parameter = 3 and Scale Parameter = 35.6

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Figure B-3
Crack Growth Rate vs. Crack Length from "Heat 69"

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Figure B-4
Stress Intensity Factor (KI) vs. Crack Half-Angle

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Figure B-5
Probability of Net Section Failure vs. Time after Initiation of Circumferential Crack due to Outside Diameter PWSCC

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Figure B-6
Spreadsheet for Core Damage Frequency from CRDM Nozzle Circumferential Cracking