Technical Basis for RPV Head CRDM Nozzle Inspection Interval H. B. Robinson Steam Electric Plant, Unit No. 2

R-3515-00-1-NP
Revision 0

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The last revision number to reflect any changes for each section of the report is shown in the Table of Contents. The last revision numbers to reflect any changes for tables and figures are shown in the List of Tables and the List of Figures. Changes made in the latest revision, except for Rev. 0 and revisions which change the report in its entirety, are indicated by a double line in the right hand margin as shown here.

NON-PROPRIETARY REPORT

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1. Introduction

Several PWR plants in the United States have experienced cracks in control rod drive mechanism (CRDM) nozzles and J-groove welds, and some of these plants have experienced primary coolant leaks from through-thickness cracks in the nozzles or welds. In a few cases, these cracks and leaks have led to significant consequences, including:

- Large through-wall circumferential cracks above the J-groove weld at Oconee 3
- Significant boric acid wastage of the vessel head at Davis-Besse
- Cracks in a large percentage of the welds at North Anna 2

In each of these cases, utilities have replaced the vessel head or are planning to replace the vessel head as soon as possible. As a result of the above experience, the NRC has issued three bulletins and one order:

- NRC Bulletin 2001-01, *Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles* (1)
- NRC Bulletin 2002-01, *Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity* (2)
- NRC Bulletin 2002-02, *Reactor Pressure Vessel Head Penetration Nozzle Inspection Programs* (3)
- NRC Order EA-03-009, *Order Establishing Interim Inspection Requirements for Reactor Pressure Vessel Heads at Pressurized Water Reactors* (4)

For plants such as H. B. Robinson 2 (hereafter called Robinson) that are in the "High Susceptibility" category based on operating time and head temperature, the requirements in Order EA-03-009 are that the following inspections must be performed every refueling outage:

(a) Bare metal visual examination of the RPV head surface (including 360° around each RPV head penetration nozzle, AND

(b) Either:
   (i) Ultrasonic testing of each RPV head penetration nozzle (i.e., nozzle base material) from two (2) inches above the J-groove weld to the bottom of the nozzle and an assessment to determine if leakage has occurred into the interference zone, OR

   (ii) Eddy current testing or dye penetrant testing of the wetted surface of each J-groove weld and RPV head penetration nozzle base material to at least two (2) inches above the J-groove weld.

Bare metal visual inspections were performed of the Robinson RPV head nozzles during the spring 2001 (RO-20) and fall 2002 (RO-21) refueling outages. Each RPV head nozzle and J-groove weld was inspected by eddy current examination during the fall 2002 refueling outage. In addition to the above surface inspections, the 17 open penetration nozzles were also inspected ultrasonically from the nozzle ID during the fall 2002 outage. These open nozzle UT inspections confirmed the absence of cracks and also were used to interrogate the J-groove weld zone and to determine if leakage had occurred into the interference zone between the nozzle OD surface and the ID of the hole in the vessel head. There were no reportable indications of PWSCC in any of these inspections.

The purpose of this document is to provide the technical basis for a relaxation of the order requirements to permit the next non-visual NDE inspection specified by NRC Order EA-03-009 to be deferred by one operating cycle until the refueling outage scheduled for fall 2005. Provision for relaxation of the order requirements is provided in paragraph IV.F of the order. The primary bases for requesting the relaxation at Robinson are as follows:

- Visual and non-destructive inspections performed to date have shown no reportable PWSCC indications.
- Experience to date with vessels fabricated by Combustion Engineering and with Huntington Alloys nozzle material has been excellent. Only a few cracked nozzles have been detected at one plant, and there have been no leaks and no circumferential cracks above the J-groove welds.
- The technical evaluations performed for Robinson demonstrate that the requisite levels of safety, as defined by Regulatory Guide 1.174 (5), will be met by a bare metal visual (BMV) inspection during the RO-22 outage, scheduled for spring 2004, and a combination of a BMV inspection and a non-visual NDE inspection of each CRDM nozzle during the RO-23 outage, scheduled for fall 2005.

The technical basis is provided in the following sections of this report:

- Section 2 – provides a summary of the work performed and conclusions,
- Section 3 – provides analysis requirements,
- Section 4 – provides background technical information including a description of the head, nozzle design, materials, etc.,
- Section 5 – provides a summary of recent Robinson inspections,

- Section 6 – provides predictions of the time to crack initiation,
- Section 7 – provides deterministic evaluations of the required reinspection interval to prevent nozzle ejection or significant head boric acid wastage,
- Section 8 – provides probabilistic evaluations of the effect of a one operating cycle deferral of the next non-visual inspection on the core damage frequency based on the potential failure modes of nozzle ejection and significant head boric acid wastage, and also calculations of the probability of penetration leakage,
- Section 9 – provides references, and
- Appendices – provide supporting analyses.

2. Summary and Conclusions

The following is a summary of the work performed and the conclusions regarding the effect of a one operating cycle deferral of the next non-visual inspection of the Robinson RPV head penetrations.

2.1 Robinson Inspection Status

Robinson modified the RPV top head insulation during RO-20 to replace the original reflective metal insulation with batt type insulation that is removable to permit bare metal visual inspections. Bare metal visual inspections were performed during RO-20 and RO-21. Both of these inspections showed the presence of boric acid crystals on the top surface of the vessel head, but these deposits were conclusively traced to leaks from CRDM canopy seal welds. In both cases the deposits were removed, the head was inspected for evidence of leakage from CRDM nozzle annulus regions and corrosion, and the insulation reinstalled. As discussed in Section 5, these inspections showed no evidence of leakage from the CRDM annulus regions.

During RO-21 all of the nozzles and J-groove welds were nondestructively examined for evidence of cracks. The examinations and results were as follows:

- All of the nozzles were examined on the OD surface below the J-groove weld by EDDY current testing (ECT). These examinations showed no reportable flaws.

- All of the nozzles were examined on the J-groove weld surface by ECT. These examinations showed no reportable flaws.

- The 52 nozzles with thermal sleeves were examined on the inside surface by gap scanner ECT. These examinations showed that seven of the 52 nozzles had shallow craze-type surface indications. Industry experience at Ringhals, Oconee 2 and elsewhere has shown that these types of flaws do not tend to grow significantly in depth.

- The 17 open nozzles were examined by ultrasonic testing (axial, circumferential and 0° transducers) and by rotating coil ECT. These inspections showed no indications that were attributed to PWSCC. These supplemental inspections did reveal craze-type surface indications in nine nozzles, parent tube indications in seven nozzles, and weld interface indications in two nozzles. Only one of these indications, a parent tube indication 0.28" long and less than 0.06" depth, 2.84" above the weld in nozzle 47, was of reportable size. This indication is well outside the high stress region of the nozzle and was attributed to a fabrication-related defect.

In summary, the BMV, ECT, and UT inspections, including weld surface ECT inspections, performed on each CRDM nozzle during RO-21 in fall 2002 showed no indications attributed to PWSCC.

2.2 Time to Predicted CRDM Nozzle Cracks and Leaks

Because the previous Robinson inspections revealed no CRDM nozzle PWSCC, the deterministic and probabilistic predictions in Sections 7 and 8 are based on the conservative assumption that one nozzle tube and one J-groove weld initiated cracking at the applicable flaw detectability limit immediately upon restart from the RO-21 outage in fall 2002 when both BMV and non-visual examinations were performed. Growth of these flaws is calculated using the results of stress intensity factor calculations and crack growth rates based on statistical evaluations performed by the Materials Reliability Program (MRP) using the worldwide set of available laboratory crack growth rate data for Alloy 600 base metal and Alloy 182 weld metal in the primary water environment. These calculations show only one or two cracked nozzles at the time of the RO-23 outage, scheduled for fall 2005, even for an aggressive upper bound Weibull slope of 6.0. The probabilistic analyses in Section 8 include a wide statistical distribution of Weibull slopes, ranging from 1.5 to 6.0.

2.3 Deterministic Evaluations

The deterministic evaluations support a one cycle deferral of the next non-visual reactor vessel head penetration inspection. The calculations in Section 7 show that it would take at least 7.2 years for a 30° through-wall circumferential flaw located above the J-groove weld to grow to the limiting flaw size of 284° based on a factor of 2.7 on the limit load. The deterministic evaluation for the potential for boric acid wastage shows that bare metal visual (BMV) examinations performed during every refueling outage at Robinson preclude rapid boric acid wastage of the low-alloy steel material of the head. In addition, the crack growth calculations for axial nozzle cracks and for weld cracks in Section 7 show that the greatest concern for leakage is through cracks in the J-groove attachment weld.

2.4 Probabilistic Evaluations

The base case probabilistic fracture mechanics (PFM) evaluation shows a maximum increment in core damage frequency (CDF) of 1.0×10^{-7} per year. This is one order of magnitude lower than the 1×10^{-6} criterion recommended by Reg. Guide 1.174 (5) for risk-informed decision making. In addition, a detailed sensitivity study shows that the conclusion that the effect on CDF is insignificant is robust and is not overly dependent on the set of input assumptions. Furthermore, the base case PFM calculations show a maximum probability of leakage just under 6% per year, a quite reasonable result considering the conservatisms inherent in the PFM inputs. Finally, a probabilistic wastage model similar to that presented in MRP-75 (9) shows that the potential for boric acid corrosion of the low-alloy steel head material—given the BMV examination that is scheduled for the spring 2004 refueling outage—has an insignificant effect on CDF in comparison to the value of 1.0×10^{-7} per year calculated on the basis of the PFM simulation.

A risk-informed basis exists for deferring the next non-visual RPV head inspection by one operating cycle on the basis of deterministic and probabilistic results. The deterministic evaluations in Section 7 show that net section collapse due to nozzle ejection or head/cladding rupture due to boric acid wastage are unlikely. The probabilistic evaluations in Section 8 confirm the conclusions of the deterministic evaluations using Monte Carlo simulations in combination with the conditional core damage probability (CCDP) values for Robinson LOCAs and the ΔCDF criterion of 1×10^{-6} per year recommended by Reg. Guide 1.174 (5).

3. Analysis Inputs

This section provides analysis inputs used in performing the calculations.

3.1 Dimensions

Reactor vessel head dimensions were taken from the vessel design report and drawings (50). Many of these dimensions were previously documented in Tables III-1, III-2, III-3 and A-1 of DEI Report R-3510-00-1, Revision 0, *Reactor Vessel Bolting Evaluations – HB Robinson 2 Nuclear Power Plant* (6).

3.2 Materials and Fabrication

Nozzle material heat and material certification information was provided by Progress Energy (59). The probable nozzle material processing was established by Mr. Jim England, formerly the INCO Alloys Tube and Pipe Project Manager (Attachment 4-1). Vessel fabrication information was obtained from Westinghouse (12, 57).

3.3 Inspection Status

The Robinson vessel inspection status was taken from Progress Energy reports and from the Westinghouse report of non-destructive examinations of the nozzles and welds during the fall 2002 refueling outage (18, 19). The compilation of summary industry inspection results and statistics is based on ongoing work by DEI for the MRP and was updated through the spring 2003 outage season for this report. Many industry documents, including plant submittals to the NRC, were used to compile these data, which are provided for information only, since the evaluations presented in this report do not use the industry data as a direct input.[1]

3.4 Time to Cracks and Leaks

Since Robinson was fabricated by Combustion Engineering using nozzle material from Huntington Alloys, Robinson has no reported PWSCC defects, and there are only three Huntington Alloys nozzles with cracking in other plants fabricated by Combustion Engineering, it has been conservatively assumed that Robinson developed one detectable crack in the nozzle base metal and one detectable crack in a J-groove weld immediately after completion of the fall 2002 refueling outage. The time to a leak is calculated from this assumption and deterministic or probabilistic crack growth analyses.

3.5 Limit Flaw Sizes

Limit flaw sizes have been calculated using standard fracture mechanics models, the margin of safety specified by Section XI of the ASME Boiler and Pressure Vessel Code for

[1] There are two exceptions to this statement. The first is that the fraction of leaking nozzles that lead to above-weld tube circumferential cracks is used as an analysis input as described in Section 8. The second exception is that inspection data were used to assess the probability of detection (POD) for bare metal visual (BMV) inspections. Conservatism was introduced for the BMV POD analysis as described in Section 8.2. Sensitivity cases presented in Section 8.4 show that the conclusions of the probabilistic evaluations of this report are not reliant on these two inputs.

evaluation of flaws by analysis (49) and the material flow stress, which is assumed to be midway between the tensile and yield strengths at temperature.

3.6 Crack Growth Rate Models

Crack growth rate calculations were performed using the methodology of the current NRC guidance for accepting actual flaws for continued service (7). This methodology included calculation of crack tip stress intensity factors and crack growth rates in nozzle base metal. Crack growth rates in weld metal have been established based on available data (as described in paragraph 7.2) as 4.2 times the crack growth rate in the nozzle base material. Crack tip stress intensity factors for through-wall circumferential flaws above the J-groove weld elevation have been taken from MRP calculations for a Westinghouse reactor vessel that were presented to the NRC on June 12, 2003 (8, 57).

3.7 Boric Acid Corrosion

The potential for rapid boric acid corrosion has been evaluated using the basic methodology presented in MRP-75 (9), except that the backup inspection for the outage where NDE is not performed is a bare metal visual inspection with the insulation removed instead of a supplemental visual inspection with the insulation left in place. The allowable corrosion volume in the vessel head is calculated assuming that the stresses in the remaining material must meet ASME Code primary membrane and bending requirements.

3.8 Probabilistic Analysis

Probabilistic analyses of the risk of net section collapse, collapse due to boric acid wastage, and core damage are performed using a methodology similar to that described to the NRC in MRP-75 (9). However, the analysis presented in this study includes use of parameters specific to Robinson and explicit modeling of crack initiation and growth in the base and weld metal leading to leakage, as well as other differences described in detail in Section 8. Values for the conditional core damage probability (CCDP), or Birnbaum probability, specific to Robinson loss of coolant accidents (LOCAs) (54, 55, 57) were used in combination with the calculated initiating event frequencies (nozzle ejection and head/cladding rupture) to conservatively determine the change in core damage frequency associated with deferral of the next non-visual head inspection.

3.9 Acceptance Criteria

NRC Regulatory Guide 1.174 (5) is used to provide the basis for the acceptance criterion for the calculated change in core damage frequency resulting from the one outage deferral.

4. Background Technical Information

The following is a description of the Robinson head, including design, materials and fabrication information relevant to establishing reinspection intervals.

4.1 Head and Nozzle Configuration

Figures 4-1 and 4-2 are isometric views of the Robinson vessel head and major parts of the head service structure. Figure 4-3 is a close-up of the top of the vessel head. Figure 4-4 is a head map plan showing the CRDM nozzle layout, and Figure 4-5 is a corresponding elevation view of the head.

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The head and flange are fabricated from low-alloy steel materials. The head inside radius is [ ] and the nominal head thickness is [ ]([ ] base metal thickness plus [ ] clad thickness).

The head has 69 four-inch nozzles and one head vent nozzle. As shown in Figure 4-4, the 69 four-inch diameter nozzles are used for the following purposes:

- 45 Control Rod Drive Mechanisms
- 7 Part Length Control Rod Drive Mechanisms
- 1 Part Length Drive Removed and Capped
- 10 Capped Dummy Can Assemblies
- 1 Capped Dummy Can Assembly (RVLIS Installation)
- 5 Thermocouple Columns

The control rod drive mechanisms are surrounded by upper and lower CRDM air cooling baffles. The lower baffle is fabricated in three 120° sectors that are bolted together by flanged seam joints. The upper baffle is comprised of an assembly of 12 baffle former sections, each with vertical walls and a horizontal plate that connects to the top of the lower baffle. A cooling duct is connected to each of the lower sectors, and cooling air is pulled down through the drive assemblies and exhausted out through the ducts.

As shown in Figure 4-3, a thin shroud ring obscures a small portion of the upper head surface outside the central region where the CRDM nozzles penetrate the head. The presence of this ring, which rests on the three lifting truncheons, precludes bare metal inspection of the total upper head surface, but the ring does not reduce the effectiveness of a bare metal visual examination as there is no interference with the visibility of the intersections between the head upper surface and the CRDM nozzles and the adjacent surface area.

As discussed in greater detail in Section 5, the Robinson head was originally provided with reflective metal insulation. However, the reflective metal insulation was removed during the spring 2001 refueling outage (RO-20) and replaced with flexible batt insulation that can be removed to perform bare metal inspections.

The head is lifted by a rig consisting of three vertical legs that are attached to lugs on the vessel head and a ring shaped upper strongback. The upper strongback is tied to the foundation by tie rods and is fitted with plates that transmit lateral seismic loads from the tops of the CRDM pressure tubes to the strongback. Removable insulation panels fit over the vessel flange and studs.

A concrete missile shield is positioned over the top of the assembled vessel head. The missile shield is intended to limit the amount of vertical travel of a potentially ejected CRDM to 4.97 feet (10).

4.2 Nozzle Materials and Material Processing

The Robinson CRDM nozzles were fabricated from nine heats of material supplied by Huntington Alloy Products Division of the International Nickel Company, Inc. The material heat numbers and key chemical and physical properties from the material certifications are given in Table 4-1. Table 5-1 provides a tabulation of the heat of material used at each nozzle location (59).

According to the material certifications, the materials are all SB-167 Alloy 600 supplied in the pickled and annealed condition.

Mr. Jim England, formerly the Tube and Pipe Product Manager at INCO Alloys, has reviewed the available certification information and other data available at Huntington Alloys, Inc. to establish how the Robinson materials were likely to have been processed. Mr. England's report is provided as Attachment 4-1. The main conclusions of this study were:

- The nozzles were produced by hot extrusion.
- Extruded material was heat treated in a gas fired continuous roller hearth furnace at 1,725°F (25-40 minutes at temperature) followed by air cooling at about 800°F/hr.
- Annealed tubes were descaled using standard pickling practices for Inconel.
- The descaled tubes were probably roll straightened on a large two roller rotary straightener.
- Chemical and mechanical properties meet specified requirements. DEI understands that the mechanical properties were measured after the straightening operation.
- The range of yield strengths 35.5 ksi to 57.5 ksi could be the result of the rotary straightening operation. It was reported that cold work produced by roll straightening can increase material yield strength 5-15 ksi without affecting the material tensile strength, grain size or microstructure. (DEI Note: The higher

yield strength produced in some nozzles by roll straightening can result in higher operating condition stresses as a result of weld shrinkage.)

- Grain size is estimated to be ASTM 2.5 to 3.5.

- Carbide distribution is expected to be mixed intra- and inter- granular with intergranular carbides predominant.

In summary, the tubes were produced using a process that is expected to produce large grain size and predominantly intergranular carbides. This material would be expected to have lower PWSCC susceptibility than material with small grain size and predominantly intragranular carbides.

Table 4-2 is a listing of other plants that have the same heats of material as Robinson. This table was produced using the compilation of heat numbers submitted by the industry to the NRC (11). No problems have been reported for any of these plants or heats of materials. However, it should be noted that Robinson has only a few nozzles from these heats and the other plants have significantly less time at temperature than Robinson. Given these two factors, the absence of reported problems with these heats in other plants is not surprising.

4.3 Nozzle Fabrication History

Section 6 of Westinghouse WCAP-16110-P (12, 57) provides a review of the Robinson vessel head fabrication process. The following is a very brief summary of the high points:

- Penetration holes were rough machined in the vessel head.

- J-groove weld preparations were machined into the vessel head using a "Pogo-stick" tool (Figure 6-3 of WCAP-16110-P (12, 57)), ground flat at the edge of the hole to obtain the correct contour, and then examined using the magnetic particle method. The key factor here is that the preparation dimensions were controlled as opposed to J-groove dimensions that might have resulted from hand grinding.

- The J-groove weld preparations were buttered to a nominal thickness of ¼" using Alloy 182 electrodes with the vessel held at a 250°F preheat temperature. The completed buttering was ground and examined by liquid penetrant. WCAP-16110-P (12, 57) reports that the buttering at 10 nozzle locations required repair and the balance were acceptable.

- After application and inspection of buttering, the closure head was heat treated at 1150±25°F for a minimum of 10-1/2 hours. Heatup and cooldown rates above 600°F were held to 100°F/hour. The J-groove weld buttering was inspected by liquid penetrant after completion of heat treatment.

- The bores in the vessel head were machined to final dimensions.

- The CRDM housings were match fit to the holes in the vessel head to produce the minimum interference fit, were cooled to -88°F in an acetone and dry ice bath, and inserted into the holes in the vessel head to the specified depth. The nozzles were then tack-welded to the buttering.

  Some of the nozzles were polished on the OD surface prior to insertion to achieve the specified interference fit.

- The CRDM nozzles were welded into the vessel head using 1/8" and 5/32" diameter Alloy 182 shielded metal arc electrodes. The welds were visually inspected at every pass, and liquid penetrant examined at the root pass and every ¼" of deposit depth after grinding. There are no records of any repairs having been made.

In summary, other than the repairs to buttering at 10 nozzle locations, there is little information in WCAP-16110-P (12, 57) to suggest fabrication related problems with the Robinson nozzles.

It was determined during the RO-21 non-destructive examinations that the fillet between the nozzle and the head ID on the downhill side of the nozzle is larger than shown on the fabrication drawings. The effect of this larger fillet is addressed in Appendix A.

4.4 Operating Time and Temperature

Per Table 2-2 of MRP-48, *PWR Materials Reliability Program Response to NRC Bulletin 2001-01* (13), Robinson was reported to have 20.56 EFPYs (February 2001); the uprated head temperature is now 599.7°F (14). It is assumed that Robinson had 22.1 EFPYs at the time of the fall 2002 refueling outage (RO-21).

4.5 Head Vent Nozzle

Most PWR plants have head vent nozzles consisting of pieces of Alloy 600 pipe/tube welded into the vessel head using a J-groove weld. Robinson, however, has a large alloy steel head vent penetration that is welded in and stress relieved with the vessel. This configuration does not include a J-groove weld, and there is no high stress Alloy 600 region within the head thickness. Therefore, the requirements of NRC Order EA-03-009, which specifies inspection scope in relation to the J-groove weld location, do not apply to the Robinson head vent nozzle design.

In summary, there is nothing related to the design, materials or fabrication that would suggest the potential for a high risk of PWSCC, with the possible exception of increased yield strength of some nozzles produced during roll straightening and grinding of the OD surface of some nozzles to achieve the specified fitup.

Table 4-1
Summary of Robinson Material Heats and Properties

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Heat	Count	Spec	Cert Date	OD	ID	Carbon	Yield	Tensile	Hardness	Elongation	Reduction of Area	Condition
	69											

Table 4-2
Robinson Heats Used in Other PWR Plants

Unit	Heats Shared with Robinson	NSSS Design	Head Fabricator	Current Head Temp. (°F)	Date of Last Inspection	Approx. EDYs at Last Inspection	Inspections Performed	Inspection Results
Conn Yankee		W	Not avail.	Not avail.	Note 1	Note 1	No BMV or NDE (Note 1)	No known leakage
Diablo Canyon 1		W	CE	589	May-2002	9.1	BMV	No detected leakage
Indian Point 2		W	CE	586	Oct-2002	8.0	BMV, 92/97 UT, 56/97 ET	No detected cracking
Salem 1		W	CE	595	Oct-2002	11.9	BMV	No detected leakage
Robinson		W	CE	600	Oct-2002	20.3	BMV, ET, weld ET + 17/69 UT	No detected cracking

Notes:
(1) Connecticut Yankee permanently shutdown in November 1996 after about 29 calendar years and 21 EFPYs of operation.

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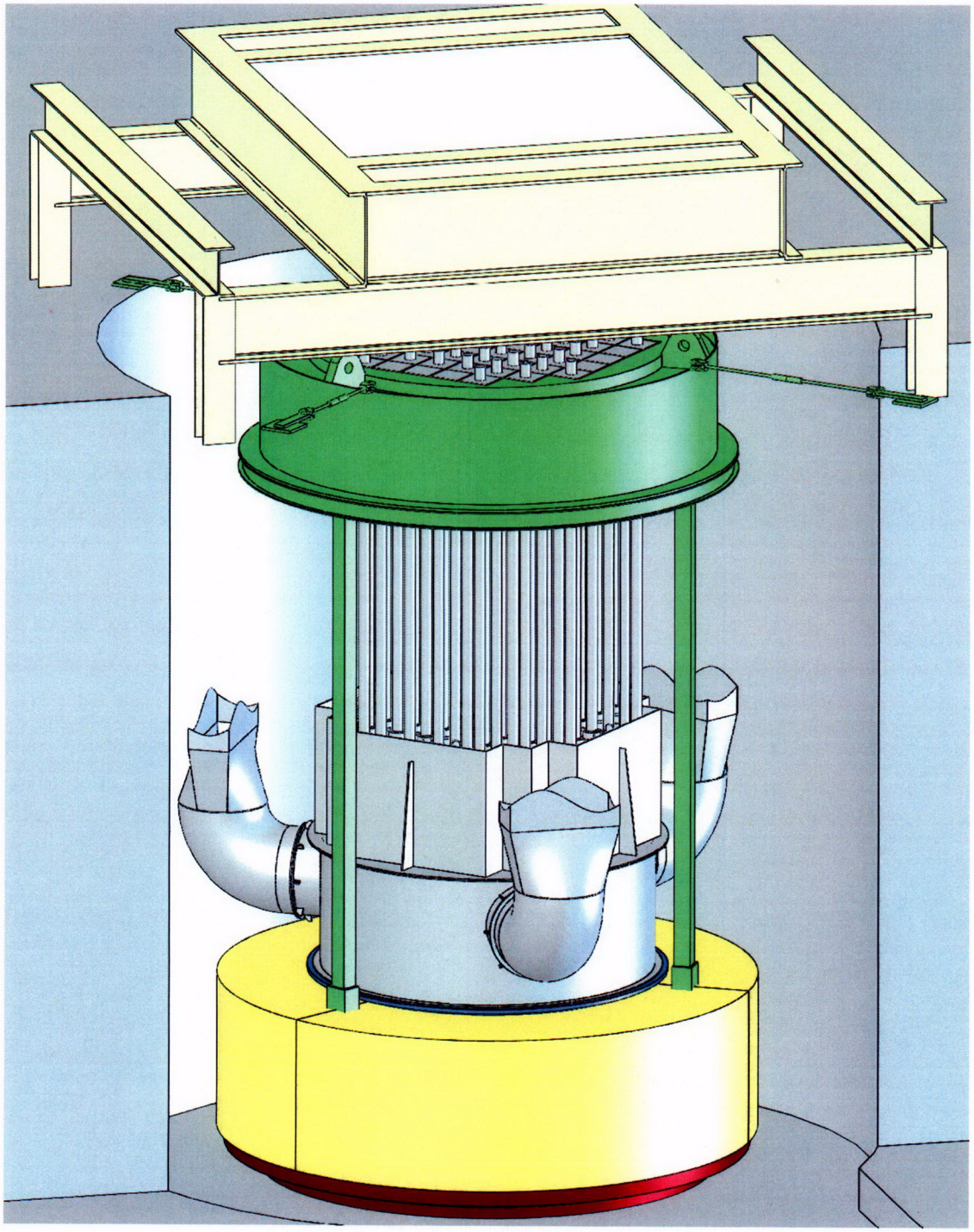

Figure 4-1
Robinson Reactor Vessel Head – Assembled Isometric

C01

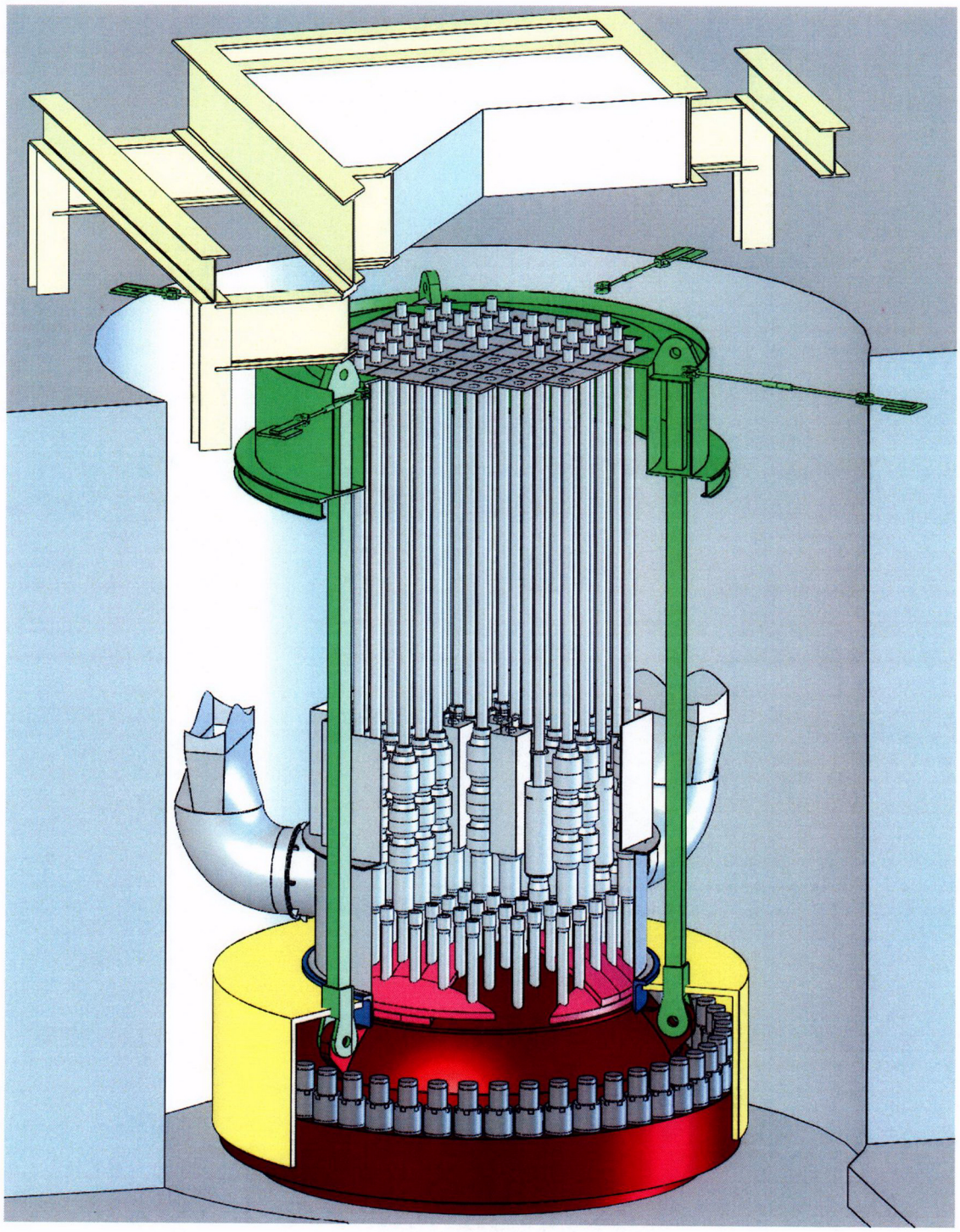

Figure 4-2
Robinson Reactor Vessel Head – Cutaway Isometric

C02

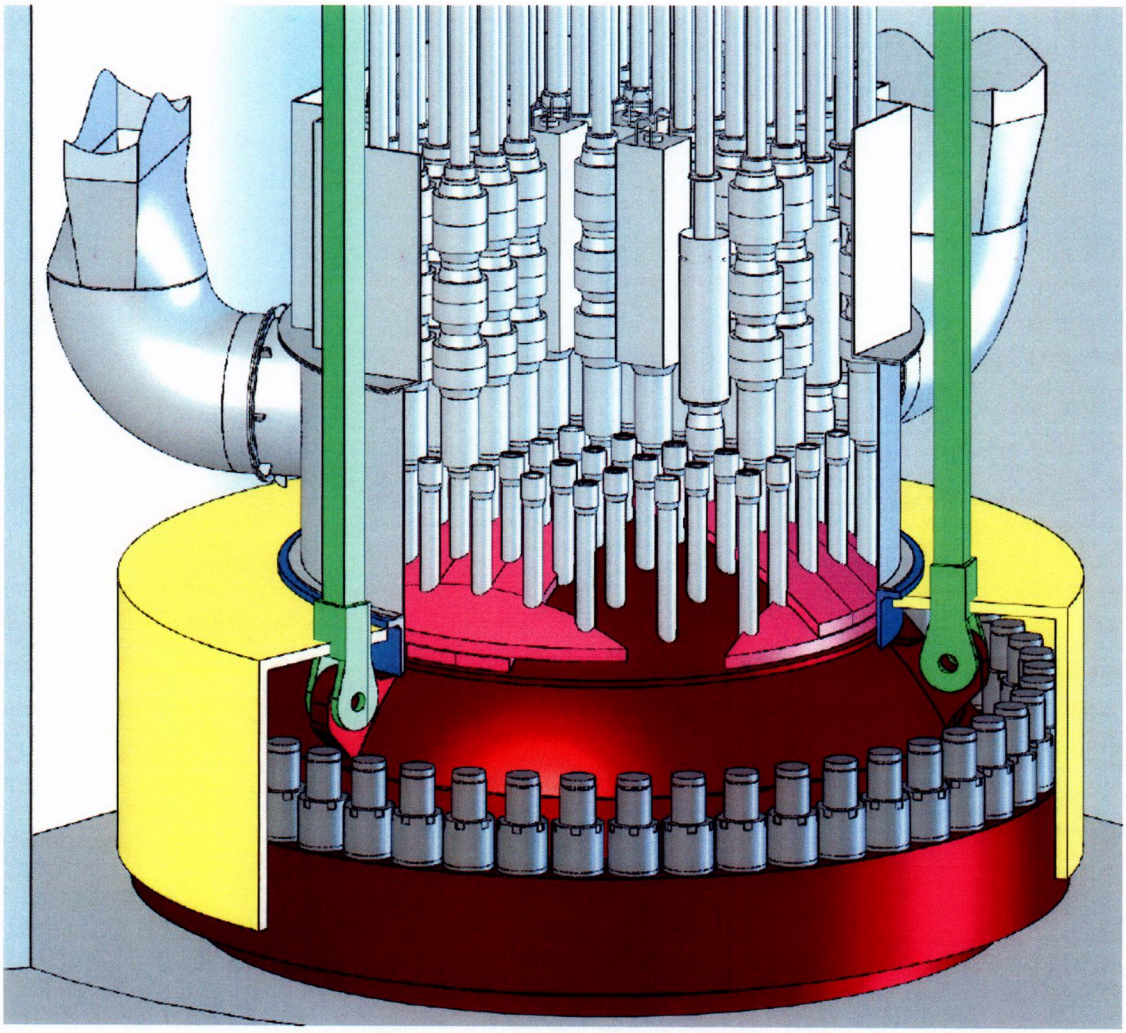

Figure 4-3
Robinson Reactor Vessel Head – Close-up Cutaway Isometric

C03

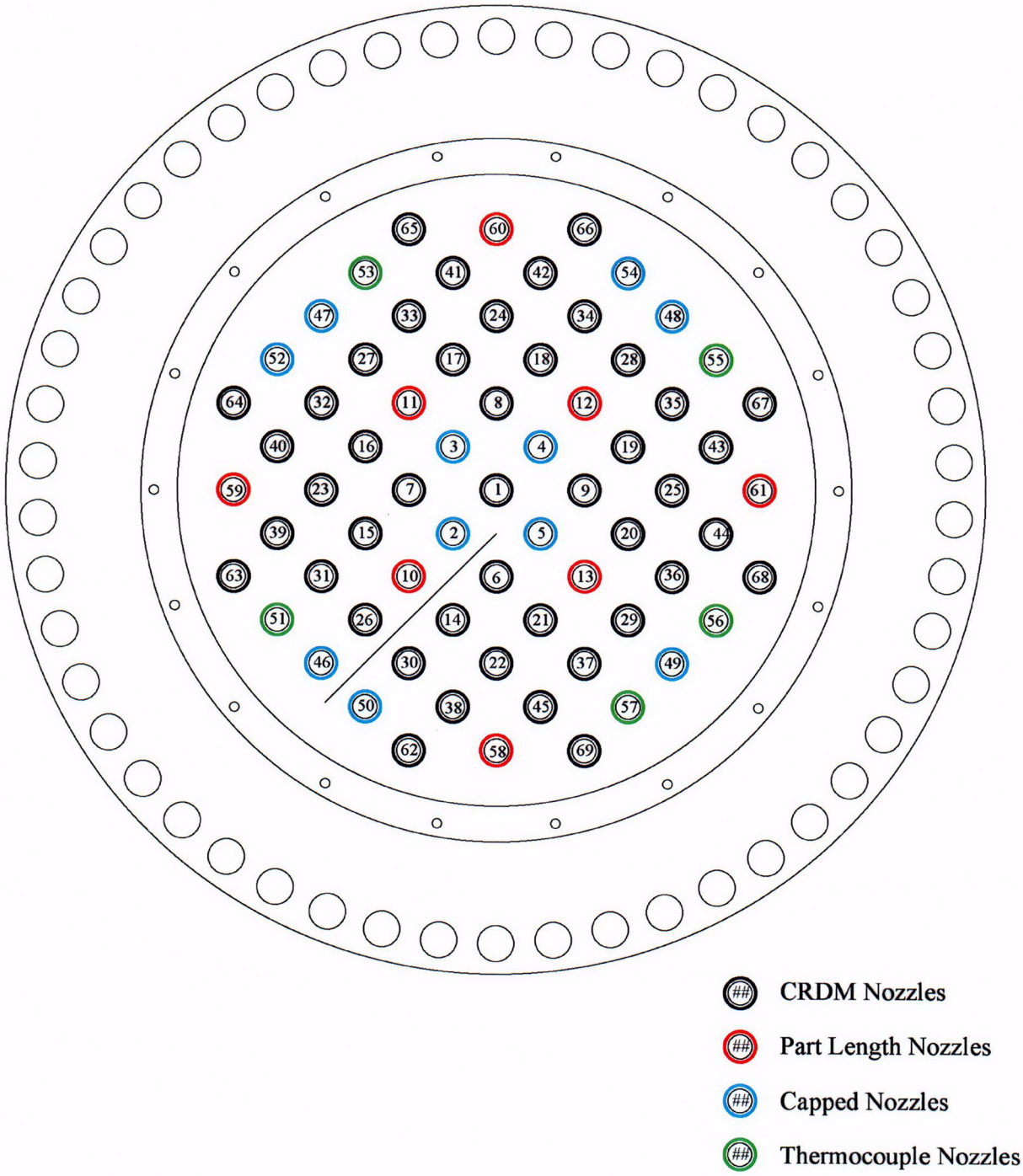

Figure 4-4
Robinson Reactor Vessel Head – Plan

C04

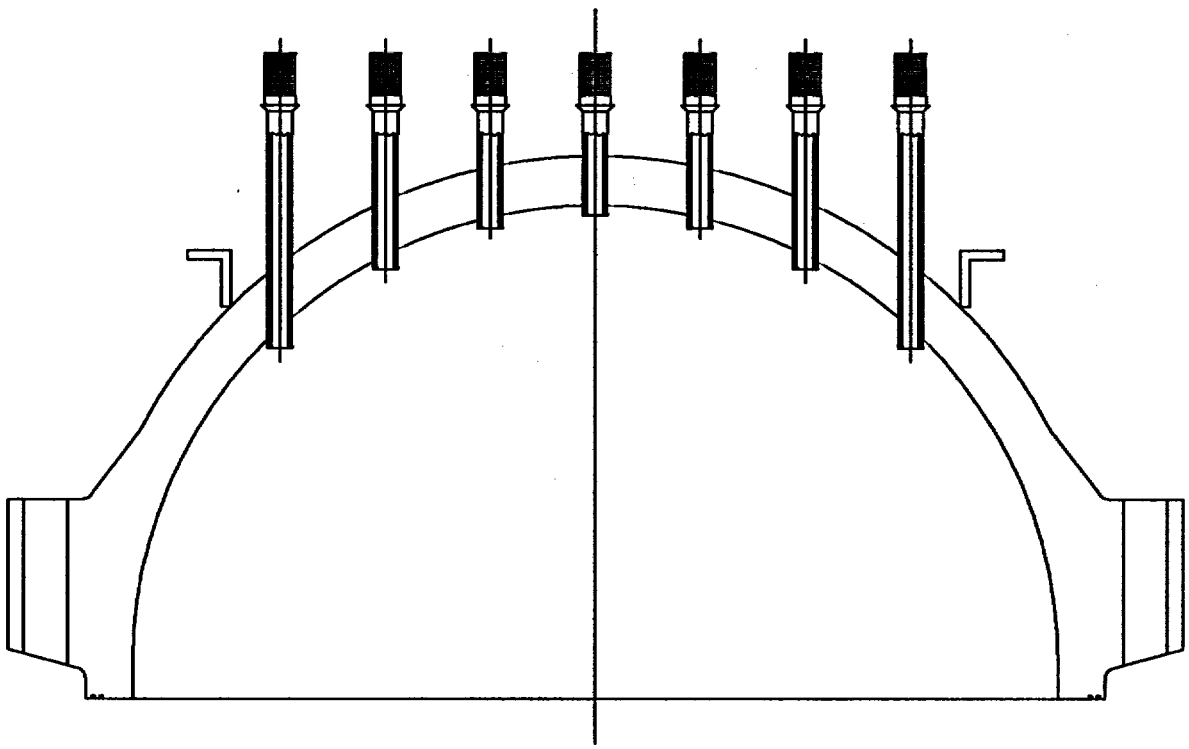

Figure 4-5
Robinson Reactor Vessel Head – Elevation

Process Review for Inconel® Alloy 600

Extruded annealed average wall tubes.
Material produced on INCO ALLOYS orders:
B00758(Customer No. 43-75526),
B18560(Customer No. 44-76759),
B76191(Customer No. 46-73822), and
B93367(Customer No. 46-75258)
Mill certifications for the materials involved are dated from 7-13-1963 through 4-4-1967

®Inconel is a trademark of Special Metals Companies

Review prepared by:

*Jim England**
England Associates

*Retired (1993) INCO Alloys Inc. Tube & Pipe Product Manager

Review Date: May 2003

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[Attachment 4-1 is proprietary to Dominion Engineering, Inc., and has been removed from this non-proprietary version of DEI Report R-3515-00-1.]

5. Robinson Inspection Results

The Robinson RPV head has been subjected to two bare metal visual (BMV) inspections and one non-destructive examination of all CRDM nozzles and J-groove welds. The following is a summary of the inspection status as of summer 2003.

5.1 Pressure Tests and Inspections During RO-18, RO-19 and RO-20

The Robinson response to NRC Bulletin 2001-01 (15) provided a description of pressure tests and inspections performed of the RPV head during RO-18, RO-19 and RO-20. This work was performed in accordance with Engineering Surveillance Test (EST) – 083, *Inservice Inspection Pressure Testing of Reactor Coolant System (Refueling Interval)*. The tests and inspection results were reported as follows:

a. A VT-2 visual examination was performed of bolted connections at the start of each refueling outage. The following components /areas were specifically identified for inspection:
   - Canopy seal welds
   - Penetration tube surfaces
   - Penetration tube/head insulation interface – particularly the outer three rows
   - Around the inside of the control rod drive mechanism (CRDM) cooling duct shrouds
   - Conoseal bolting (five places)

   The procedure specifies "*view as much of the above items as possible from all accessible areas.*" In the event that leakage had been observed during these visual examinations, plant engineering personnel would have reviewed the examination results in accordance with plant procedures prepared to address implementation of NRC Generic Letter 88-05.

   It should be noted that no bare metal visual inspections were performed prior to RO-20 due to the presence of the reflective metal insulation.

b. A VT-2 visual examination was performed during the RCS inservice leakage testing at the end of each refueling outage. The examination areas specifically included the control rod drive housing area and the reactor vessel stud area.

No indications of VHP leakage were detected during these examinations.

5.2 Modification to Permit Bare Metal Visual Inspection (RO-20)

During RO-20, the originally installed reflective type insulation was removed from the vessel head and replaced with removable batt type insulation. Access to the top of the vessel head was achieved by removing the CRDM Air Cooling Baffle upper and lower chambers. The lower chamber is comprised of three 120° sectors, each containing a cooling air outlet duct opening, which are joined by flanged connections. The upper

chamber is comprised of twelve (12) baffle former sections, each with vertical walls and a horizontal plate that connects to the top of the lower chamber. The upper chamber sections are assembled with the bolts loose and the bolts are then tightened to the final specified torque.

As shown in Figure 5-1, the replacement insulation consists of approximately 2 inch thick insulation sewn into batts with notches cut out at each CRDM nozzle location. A first layer of batts is installed on the head followed by a second layer at a 90° angle. The overlapping batts provide complete coverage of the head with gaps limited to a small annulus around each nozzle.

5.3 Visual Inspections During RO-20

As reported in the Robinson response to NRC Bulletin 2001-01 (15), the head was subjected to VT-2 visual inspections four separate times during RO-20.

- The first VT-2 inspection was performed at the start of the outage with insulation in place. This inspection showed evidence of leakage from a canopy seal weld.
- A second VT-2 inspection was performed with the lower shroud partially removed and several sections of reflective metal insulation removed. This inspection allowed further assessment of the leakage and head condition.
- A third VT-2 inspection was performed after removal of boric acid residue to assess whether there had been any corrosion of the vessel head.
- A fourth bare metal VT-2 inspection was performed after removal of the insulation. It was reported in a supplemental response regarding NRC Bulletin 2001-01 that this inspection represented a "qualified visual examination" per NRC Bulletin 2001-01.

Individuals performing the last three inspections had been briefed on the VHP leakage issue at Oconee 3.

The vessel head was found to be free from any significant amount of wastage resulting from boric acid leakage. The head was cleaned sufficiently to assess any corrosion and to highlight any new boric acid leaks. Attachment II to the Robinson submittal of information requested by NRC Bulletin 2002-01 stated that the "visual examination by VT-2 qualified inspectors identified scattered areas of light to medium rust, with no evidence of metal loss or pitting detected" (16).

5.4 Repeat Bare Metal Visual Inspection (RO-21)

The air cooling baffles and insulation strips were removed again during RO-21, and the exposed head subjected to a VT-2 bare metal visual inspection with the responsible engineer present during the examinations. As a prerequisite, the examination team was required to review the EPRI report on RPV head visual examinations (17). The inspection

results were transmitted to the NRC (18). Small amounts of loose boric acid crystals were found on the back side of nozzle #50 resulting from spray from a canopy seal leak. It was reported that the boric acid had the appearance of having "fallen" to the position next to the nozzle, rather than of having extruded up from the annulus between the nozzle and vessel head. This loose boric acid was vacuumed up to permit examination for evidence of leakage from the annulus under the loose deposits.

The bare metal visual inspection of the clean head during RO-21 did not show any evidence of leakage from a CRDM nozzle.

Corrective action to stop the leakage consisted of weld repairs to canopy seals on nozzles 10, 14 and 30, and installation of a mechanical clamp assembly on nozzle 50.

5.5 Non-Destructive Examination of Nozzles and Welds (RO-21)

All 69 CRDM nozzles and welds were nondestructively examined during RO-21. The approach used was to eddy current examine the critical "wetted surfaces" on the ID and OD of the nozzles and J-groove welds. For open penetration tubes, the eddy current examinations were supplemented by ultrasonic testing. Details of the examination method were provided to the NRC. The inspection results are summarized in Table 5-1, and the main findings were as follows:

- There were no recordable or reportable indications in any of the J-groove welds or on the penetration OD surfaces below the welds.

- ECT examination of the nozzle ID surfaces showed that seven nozzles had craze-type indications below the weld. Industry experience at Oconee and Ringhals has shown that these craze indications do not tend to grow significantly in depth. A discussion of this Oconee and Ringhals experience is included in paragraph 8.3.

- TOFD ultrasonic examinations of the open nozzles showed that nine nozzles had shallow indications at the interface between the nozzle and weld. Since these indications did not appear on the weld or OD surface ECT examinations, it was determined that they were fabrication related and do not represent service-related degradation.

- Nozzle 47 was found by TOFD ultrasonics to have a recordable indication 0.28 inches long and less than 0.060 inches deep ($\ell/a = 4.7$). However, as shown in Figure 5-2, this indication was located about 2.84 inches above the top of the weld and is clearly not associated with PWSCC adjacent to the J-groove weld. An engineering evaluation showed that this indication was most likely a scratch or other surface anomaly and that it was not likely to grow since it was in a location where the crack tip stress intensity factor is below the threshold for growth by PWSCC.

In summary, the non-destructive examination of all nozzles and welds did not identify any reportable indications attributed to PWSCC.

5.6 Other Indicators of Possible RPV Head Nozzle Leakage

Several other provisions have been made to locate small leaks, including leaks from CRDM nozzles. These provisions were described to the NRC in the response to an NRC request for additional information relative to Bulletin 2002-01 (19). These provisions are as follows:

- The plant Boric Acid Corrosion Control Program.

- Walkdowns at the start and end of each refueling outage per Operations Surveillance Test OST-052, *RCS Leakage Test and Examination Prior to Startup Following and Opening of the Primary System (Refueling and/or Startup Interval).*

- Inspections per OST-053, *Inspections for Reactor Coolant System Leakage (Prior to and Following Cooldown) (Refueling Interval).*

- During power operation, an RCS leakage surveillance is performed every 72 hours in accordance with OST-051, *Reactor Coolant System Leakage Evaluation (Every 72 hours during steady state operation and within 12 hours after reaching steady state operation).*

  Indication of leakage is provided by the following:

  a) Reactor coolant drain tank level
  b) Pressurizer relief tank level
  c) Accumulator pressure and level
  d) Containment air particulate and noble gas monitors
  e) Containment sump level
  f) Component cooling water radiation monitor
  g) Increasing charging pump flow rate compared to RCS inventory changes
  h) Unscheduled increases in reactor makeup water usage

  Investigations are commenced if the identified leak rate exceeds 0.3 gpm or the unidentified leak rate exceeds 0.2 gpm.

- Sensitivity to leaks is increased by maintaining the rate of leakage very low. As was reported to the NRC, the average unidentified leak rate after RO-21 was 0.03 gpm.

In summary, Robinson is aware of the need to identify small leaks and has established procedures to ensure that leaks are detected at a small size.

5.7 Probability of Detection (POD) by Non-Destructive Examinations

Performance of probabilistic evaluations of the likelihood of flaw growth as a function of time after the last inspection that can cause through-wall leaks requires that estimates be developed of the starting sizes of the flaws that could have been undetected after the inspection. This requires that sizes of the flaws that were possibly not detected by the NDE examinations be estimated. To develop such estimates, POD curves are required for each of the flaw types that affect CRDMs. The flaw types of concern are:

- ID axial flaws in the nozzle base material
- OD circumferential flaws in the nozzle base material
- OD axial flaws in the nozzle base material (initiating below the J-groove weld)
- Flaws of any orientation at the weld wetted surface

The inspections performed at the last refueling outage included blade or rotating probe ECT inspections of the nozzle ID surface, and Grooveman ECT inspections of the nozzle OD below the weld and of the weld surface. The organization that performed the ECT, Wesdyne, provided the following estimates regarding the detectability of flaws by the ECT techniques employed at Robinson (20, 57):

- ECT of nozzle ID for axial or circumferential flaws, and of nozzle OD for axial or circumferential flaws: >90% POD for flaws of 0.04 inches depth or more.

- ECT of weld surface for flaws of any orientation: >90% POD for flaws of 0.160 inches length or more. This figure of 0.160 inches matches the detectability limit for ECT examination of weld surface flaws that has been reported by the Material Reliability Program based on the EPRI NDE Center's demonstration program for reactor vessel head penetration inspections (24, 57).

POD curves are often fit to a log-logistic equation of the following type (21):

$$POD(a) = \frac{\exp(\alpha + \beta \ln(a))}{1 + \exp(\alpha + \beta \ln(a))} \qquad \text{[Eq. 5-1]}$$

where

a = size of flaw, expressed as a fraction of some specified dimension
α = adjustable parameter selected to fit data
β = adjustable parameter selected to fit data

Since there are two unknown parameters in the above POD equation, two points on the curve need to be determined in order to define the POD curve. The information provided by Wesdyne for flaw sizes at which a 90% POD was expected provided one of the two points. The second point was determined by assuming that a POD of 95% would apply for a crack size equal to 1.2 times the size for the 90% POD. This assumption is based on typical behavior of the POD curves for steam generator tube inspections using similar

probes. The POD was conservatively assumed to remain at 95% for sizes greater than 1.2 times the size for 90% POD.

For example, for ECT inspections of the J-groove weld surface, the first point in the POD curve vs. length of flaw is 90% at a length of 0.160 inches. It is assumed that a POD of 95% will be achieved at a flaw length of 1.2 times this length, i.e., at 0.2 inches. The POD is assumed to remain constant at 0.95 for greater lengths. This is considered to be conservative, since the POD actually is expected to increase as the crack length increases.

The resulting POD curves are shown in Figures 5-3 and 5-4. Figure 5-3 also shows a POD curve developed using data from an EPRI ETSS for Plus Point probe inspections for axial PWSCC in steam generator tubes (22). As shown on Figure 5-3, the POD curve developed from the ETSS is above that developed for the CRDM inspections. Even though the frequencies and other details of the steam generator inspection are somewhat different than used at Robinson, the fact that the POD curve used at Robinson for nozzle base material falls below that developed for steam generator tubes provides confidence that the Robinson curve is conservative and does not result in too high PODs.

Table 5-1
Summary of RO-21 CRDM Nozzle Inspection Results

					OD ECT Exams		Gap Scanner	Open Housing Scanner UT and ECT				
Nozzle No.	Application	Heat	Yield (ksi)	Carbon (%)	Weld	Tube	ECT	Axial TOFD	Circ TOFD	2.25 Mhz 0°	5.0 Mhz (0°)	ECT
1	CRDM				NDD	NDD	NDD					
2	Capped				NDD	NDD		NDD	NDD	NDD	NDD	CC
3	Capped				NDD	NDD		NDD	NDD	NDD	NDD	NDD
4	Capped				NDD	NDD		NDD	NDD	NDD	NDD	CC
5	Capped				NDD	NDD		NDD	WII	NDD	NDD	NDD
6	CRDM				NDD	NDD	CC					
7	CRDM				NDD	NDD	NDD					
8	CRDM				NDD	NDD	NDD					
9	CRDM				NDD	NDD	NDD					
10	P/L				NDD	NDD	NDD					
11	Capped				NDD	NDD		NDD	NDD	NDD	NDD	NDD
12	P/L				NDD	NDD	NDD					
13	P/L				NDD	NDD	NDD					
14	CRDM				NDD	NDD	CC					
15	CRDM				NDD	NDD	NDD					
16	CRDM				NDD	NDD	NDD					
17	CRDM				NDD	NDD	NDD					
18	CRDM				NDD	NDD	NDD					
19	CRDM				NDD	NDD	NDD					
20	CRDM				NDD	NDD	NDD					
21	CRDM				NDD	NDD	NDD					
22	CRDM				NDD	NDD	NDD					
23	CRDM				NDD	NDD	NDD					
24	CRDM				NDD	NDD	NDD					
25	CRDM				NDD	NDD	NDD					
26	CRDM				NDD	NDD	NDD					
27	CRDM				NDD	NDD	NDD					
28	CRDM				NDD	NDD	NDD					
29	CRDM				NDD	NDD	NDD					
30	CRDM				NDD	NDD	NDD					
31	CRDM				NDD	NDD	NDD					
32	CRDM				NDD	NDD	NDD					
33	CRDM				NDD	NDD	NDD					
34	CRDM				NDD	NDD	CC					
35	CRDM				NDD	NDD	NDD					
36	CRDM				NDD	NDD	NDD					
37	CRDM				NDD	NDD	NDD					
38	CRDM				NDD	NDD	NDD					
39	CRDM				NDD	NDD	NDD					
40	CRDM				NDD	NDD	NDD					
41	CRDM				NDD	NDD	NDD					
42	CRDM				NDD	NDD	NDD					
43	CRDM				NDD	NDD	CC					
44	CRDM				NDD	NDD	CC					
45	CRDM				NDD	NDD	CC					
46	Capped				NDD	NDD		NDD	PTI	NDD	NDD	NDD
47	Capped				NDD	NDD		PTI	PTI	NDD	NDD	CC
48	Capped				NDD	NDD		LCG	WII	NDD	NDD	NDD
49	Capped				NDD	NDD		NDD	NDD	NDD	NDD	CC
50	Capped				NDD	NDD		NDD	NDD	NDD	NDD	NDD
51	T/C				NDD	NDD		NDD	NDD	NDD	NDD	CC
52	Capped				NDD	NDD		NDD	PTI	NDD	NDD	CC
53	T/C				NDD	NDD		NDD	PTI	NDD	NDD	CC
54	Capped				NDD	NDD		NDD	NDD	NDD	NDD	CC
55	T/C				NDD	NDD		NDD	PTI	NDD	NDD	CC
56	T/C				NDD	NDD		NDD	PTI	NDD	NDD	NDD
57	T/C				NDD	NDD		NDD	PTI	NDD	NDD	NDD
58	P/L				NDD	NDD	NDD					
59	P/L				NDD	NDD	NDD					
60	P/L				NDD	NDD	NDD					
61	P/L				NDD	NDD	NDD					
62	CRDM				NDD	NDD	NDD					
63	CRDM				NDD	NDD	NDD					
64	CRDM				NDD	NDD	NDD					
65	CRDM				NDD	NDD	NDD					
66	CRDM				NDD	NDD	NDD					
67	CRDM				NDD	NDD	CC					
68	CRDM				NDD	NDD	NDD					
69	CRDM				NDD	NDD	NDD					

NDD = No Detectable Defects
LCG = Loss of Coupling-Geometry
WII = Weld Interface Indication
PTI = Parent Tube Indication
CC = Craze Cracking

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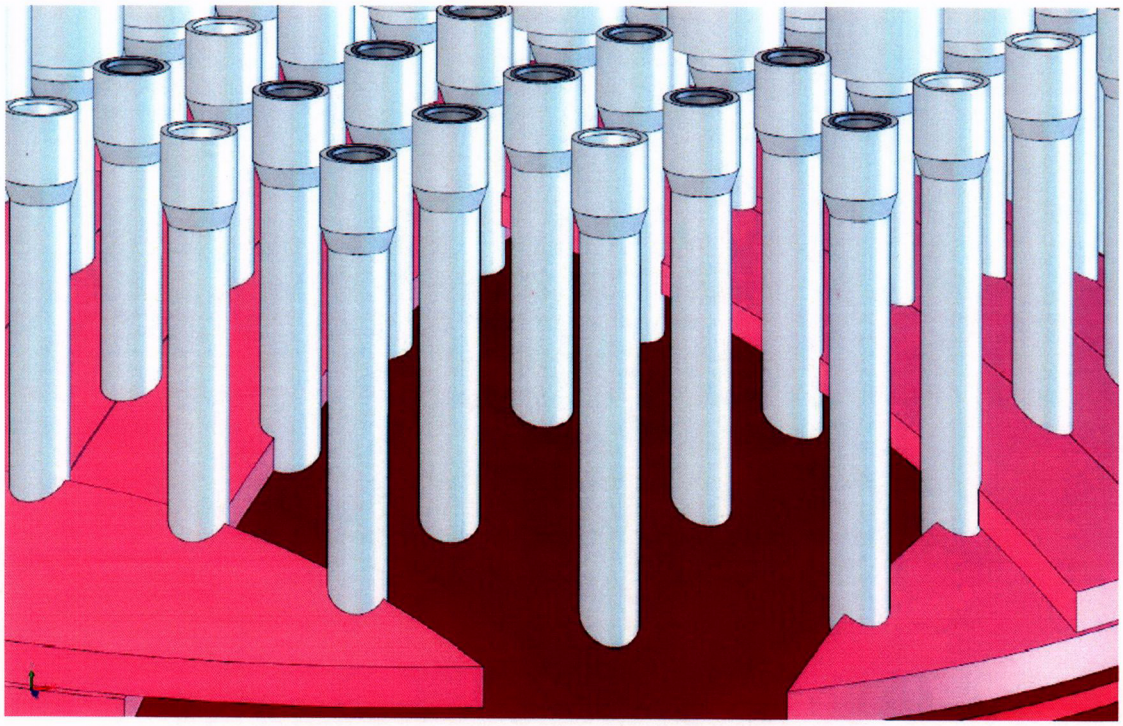

Figure 5-1
Cutaway View of Robinson Head Showing New Batt Type Insulation

C05

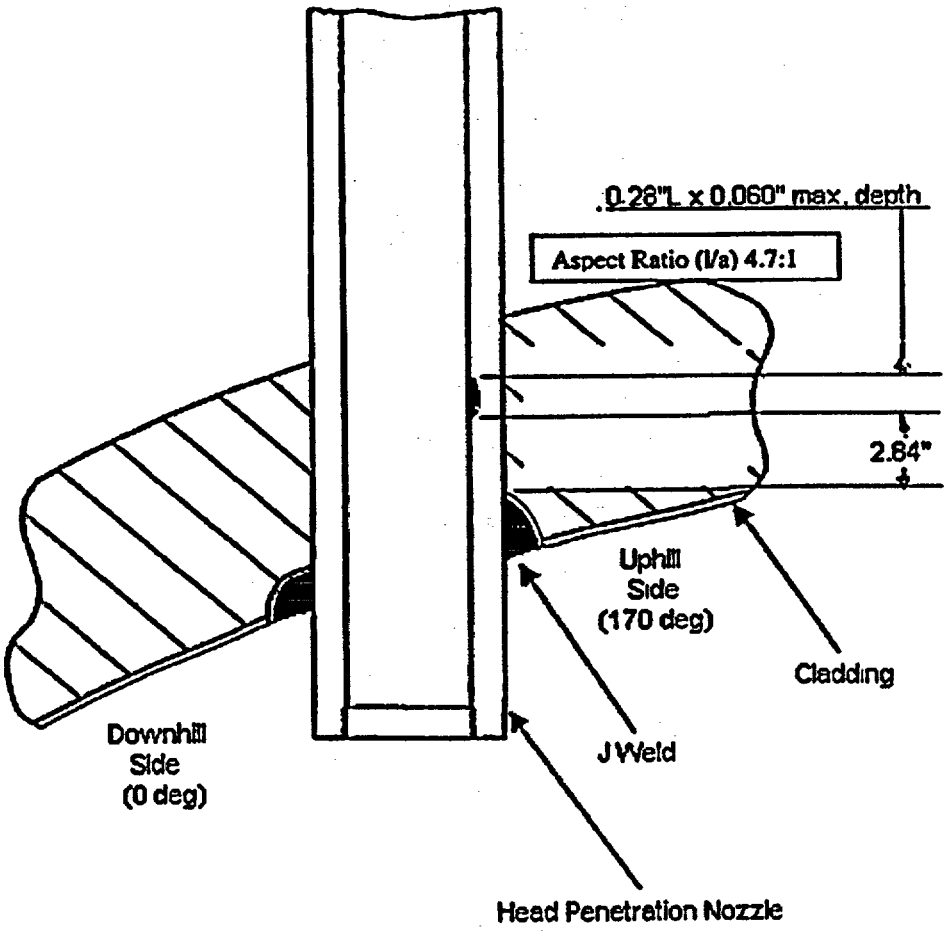

Figure 5-2
Location of Reportable Indication in Nozzle #47

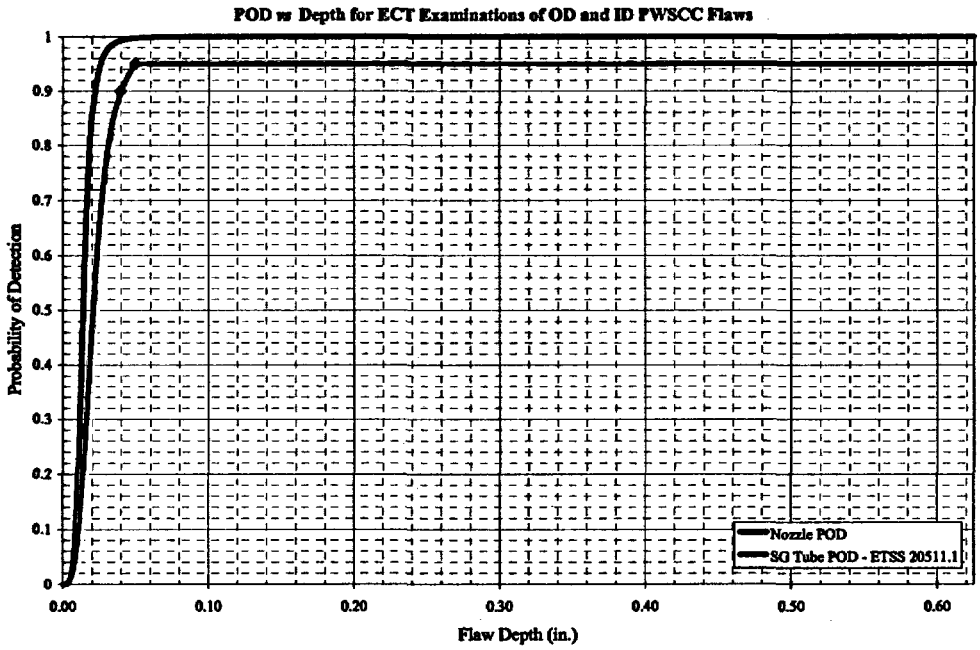

Figure 5-3
Probability of Detection by ECT of PWSCC on CRDM Nozzle Surfaces

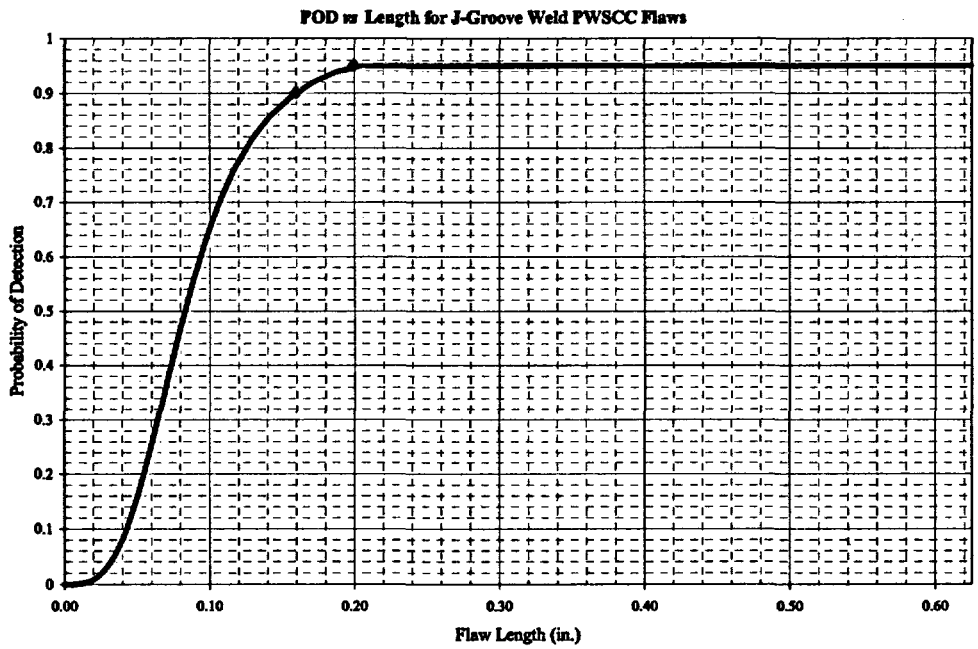

Figure 5-4
Probability of Detection by ECT of PWSCC in J-Groove Welds

6. Time to Crack Initiation

As discussed in Section 5, no reportable PWSCC cracks were detected in the Robinson nozzles or J-groove welds at RO-21 in fall 2002. This section provides predictions of future cracking in Robinson nozzles and welds that are used as input to the probabilistic analysis supporting deferral of non-destructive examination by one outage.

6.1 Industry Experience with PWSCC of Alloy 600 Nozzles

PWSCC of Alloy 600 reactor coolant system penetrations has been a significant concern for PWR plants worldwide since the mid 1980s. Table 6-1 is an outline of the initial incidents at each degradation location. PWSCC of Alloy 600 reactor pressure vessel (RPV) head nozzles has been an industry concern since the early 1990s. The following is a brief chronology of the major milestones:

- The first leak was discovered from a CRDM nozzle at Bugey 3 in 1991. This led to an extended outage at Bugey 3 and inspection of CRDM nozzles in all EdF PWR plants. The presence of cracks in the nozzles on many heads led to a program to replace all of the RPV heads in EdF plants with new heads fabricated using Alloy 690 nozzle materials and Alloy 52/152 J-groove welds.

- Cracks discovered in the CRDM nozzles, and fabrication related defects in J-groove welds, led to the replacement of RPV heads at Ringhals 2 and 4.

- Eddy current (ECT) inspections of the inside surfaces of CRDM nozzles of two RPV heads in the United States (Oconee 2 and Cook 2) showed one nozzle at DC Cook with a deep crack (50% through wall) that was weld repaired and several nozzles at Oconee 2 with shallow "craze" type cracking that did not require repairs.

- A leaking CRDM nozzle was discovered at Oconee 1 in December 2000. The leak was traced to a crack primarily in the J-groove weld that was subsequently weld repaired. Over the next 15 months, leaks were discovered from CRDM nozzles at all seven B&W design plants. Two nozzles were discovered to have circumferential cracks in the nozzle wall above the weld that extended about 160° around the nozzle circumference and through-wall in some locations. The leaking nozzles were all repaired. One Oconee head has been replaced and the other two Oconee heads are scheduled for replacement within a year.

- Bare metal visual inspections of the heads on other plants with high predicted potential for PWSCC based on time at temperature showed that two of the heads (North Anna 2 and Surry 1) had leaking nozzles which were repaired. These leaks were determined to have been through cracks in the J-groove welds.

In September 2002, North Anna 2 discovered leaks from several additional CRDM nozzles and subsequent nondestructive examination showed that 63 of the 65 CRDM nozzles had indications in the welds with many requiring repair. There were small circumferential cracks in the walls of two nozzles near the top of the welds without the presence of leakage on the top surface of the vessel head. The North Anna 2 head was replaced, and replacement heads have been ordered for the other three Rotterdam Dockyard vessels at Surry and North Anna.

- In March 2002, a leak from a PWSCC crack in a CRDM nozzle at Davis-Besse was discovered to have corroded a large cavity in the low-alloy steel vessel head that penetrated to the stainless steel clad. The Davis-Besse head has been replaced.

As a result of the above incidents, the NRC issued three bulletins and an order requiring plants to first perform bare metal visual inspections and then nondestructive examinations of the nozzles and possibly welds of the higher susceptibility plants.

Table 6-2.a summarizes the status of RPV head inspections in the United States through the spring 2003 refueling outages. This table was compiled from documents prepared from utility responses to requests for information in the NRC bulletins and from other information compiled by the EPRI Materials Reliability Program (MRP). The data show the inspection status of every RPV nozzle in every PWR head in the United States. The plants are listed in the order of decreasing Effective Degradation Years (EDYs).[2] The plants are separated into three groups corresponding to High Susceptibility, Moderate Susceptibility, and Low Susceptibility categories as defined by the NRC based on time at temperature.

Tables 6-2.b and 6-2.c are subsets of the data in Table 6-2.a for vessels fabricated by Combustion Engineering (Table 6-2.b) and for vessels not fabricated by Combustion Engineering (Table 6-2.c). These data highlight the fact that fact that significant problems to date have been concentrated on vessels not fabricated by Combustion Engineering.

Figure 6-1 is a comparison of the inspection status for all CRDM head nozzles in the U.S. industry. These data again highlight that experience with reactor vessels fabricated by Combustion Engineering and tubes fabricated by Huntington has been good.

6.2 Experience with RPV Head Nozzle PWSCC at Robinson

As described in Section 5, all of the nozzles in the Robinson head have been inspected by ECT and/or UT, and all of the J-groove welds have been inspected by ECT. These inspections have not detected any reportable PWSCC indications. The lines for Robinson in Tables 6-2.a and 6-2.b show the Robinson inspection results in the context of the other industry experience.

[2] Effective Degradation Years are the Effective Full Power Years for the plant adjusted to a common 600°F head temperature. Plants which have head temperatures less than 600°F, accumulate EDYs at a slower rate than EFPYs.

6.3 Experience with RPV Head Nozzle PWSCC in Other Vessels Fabricated by CE

As indicated in Table 6-2.b, 40 PWR vessels in the United States were fabricated by Combustion Engineering or had the fabrication completed by Combustion Engineering. As shown in this table, only 9 nozzles in vessels fabricated by Combustion Engineering have required repairs. Interestingly, the 4 nozzles requiring repair at Beaver Valley 1 were from B&W Tubular Products Heat M3935, which experienced leaks at Oconee 3 and Davis-Besse.

In summary, there is no evidence to date of significant problems with vessels fabricated by Combustion Engineering with Huntington nozzle material with the exception of the three repairable defects at Millstone 2.

Tables 6-3 through 6-5 present the results of a review of the materials and fabrication practices used to construct the Millstone 2 head, in comparison to the corresponding parameters for the Robinson head, and are provided for information only (58). The most significant differences are the greater time at temperature for Robinson and the somewhat smaller nozzle wall thickness and greater design weld sizes for Millstone 2. The differences in wall thickness and weld size for Millstone 2 are in the direction that would be expected to increase relative PWSCC susceptibility. Although it is recognized that the actual sizes of the Robinson J-groove welds are somewhat larger than its design drawings indicate, the J-groove welds at Millstone 2, which were also fabricated by Combustion Engineering, are also likely to be similarly larger than design. In addition, the nozzle heats for Millstone 2 were reported to be water quenched rather than air cooled, as was reported for Robinson. Air cooling is generally preferable to water quenching because a faster cooling rate after annealing tends to produce a poorer microstructure from a PWSCC standpoint, with less grain boundary carbide decoration. However, overall the materials and fabrication practices for the two heads are similar, so that there is no clear reason to suspect one head to be significantly more susceptible to PWSCC than the other.

6.4 Robinson CRDM Nozzle Predictive Model

Predictions for cracked nozzles at Robinson are based on the Weibull method as described by Abernethy (27). Since there has been no PWSCC to date at Robinson, and very little in other Combustion Engineering vessels, it will be conservatively assumed that one Robinson nozzle developed an NDE detectable flaw as soon as the plant started up after RO-21 in fall 2002. The fraction of nozzles cracked would then be:

$$F = \frac{i - 0.3}{N + 0.4} \qquad \text{[Eq. 6-1]}$$

where

F	=	fraction of nozzles cracked
i	=	number of cracked nozzles (1)
N	=	total number of nozzles (69)

$$F = \frac{1 - 0.3}{69 + 0.4} = 0.01009$$

The characteristic time (θ) for Weibull predictions is determined by the fraction failed (F) at time of the inspection (t), and the Weibull slope (b).

$$\theta = t_{21}\left[-\ln(1-F)\right]^{-\frac{1}{b}} \qquad \text{[Eq. 6-2]}$$

where

θ	=	characteristic time (EFPYs)
t_{21}	=	EFPY at end of RO-21 = 22.1 years (13)
b	=	Weibull slope = 3 (based on PWSCC experience—see Section 8)

$$\theta = 22.1\left[-\ln(1-0.01009)\right]^{-\frac{1}{3}} = 102 \text{ EFPY} \qquad \text{[Eq. 6-3]}$$

Figure 6-2 is a plot of the above equation relative to data compiled by the EPRI Materials Reliability Program (MRP) for other vessel heads with significant PWSCC and leaks. The data plotted have been adjusted to a reference head temperature of 600°F using an activation energy of 50 kcal/mole to be consistent with the standard MRP reporting criteria. The data are consistent with a significantly longer time to PWSCC for the Combustion Engineering fabricated plants as illustrated in a different way in Figure 6-1. Note that probabilistic analyses in Section 8 are performed using a range of Weibull slopes shown in Figure 6-2 rather than only the nominal value of 3.

The fraction of cracked nozzles at a future time (t) is given by the expression

$$F = 1 - e^{-\left(\frac{t}{\theta}\right)^b} \qquad \text{[Eq. 6-4]}$$

where

F	=	number of cracked nozzles at time, t
t	=	time at which fraction of cracked nozzles is predicted (EFPY)
θ	=	102 EFPY

Figure 6-3 is a projection for the number of cracked nozzles at future outages based on the above model and assuming a refueling outage every 18 months of operation. These Weibull calculations are used as inputs for the Section 8 probabilistic predictions.

The above calculations demonstrate that even conservative assumptions regarding the timing of the first initiation of PWSCC results in at most two predicted cracked nozzles over the next two operating cycles.

Table 6-1
Chronology of Key Leading Events Relating to PWSCC of Alloy 600 Type Materials in Non-Steam Generator Tubing PWR Plant Applications (For Information Only)*

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* Only the initial occurrence of each type is reported.

Table 6-2.a
Summary of Key Parameters Related to RPV Head Nozzle PWSCC – All Vessels (For Information Only)

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Table 6-2.b
Summary of Key Parameters Related to RPV Head Nozzle PWSCC – Vessels Fabricated by Combustion Engineering (For Information Only)

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Table 6-2.c
Summary of Key Parameters Related to RPV Head Nozzle PWSCC – Vessels Not Fabricated by Combustion Engineering (For Information Only)

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Table 6-3
Comparison of Robinson and Millstone 2 Key Parameters (For Information Only)

Parameter	*Robinson 2 CRDM Nozzles*	*Millstone 2 CEDM Nozzles*
NSSS Supplier	Westinghouse	CE
Vessel Fabricator	CE	CE
Nozzle Material Supplier	Huntington	Huntington
Material Spec.	SB-167	SB-167
No. CRDM or CEDM Nozzles	69	69
Head Temp. History (°F)	598.0 - 599.7	586.9 - 593.9
Approx. EDYs (600°F)	22.5	11.2
Date of EDYs	June 2003	February 2002 (when 3 cracked CEDM nozzles were detected)
Design Diametral Interference Fit (mils)	0.0 - 3.0	0.0 - 3.0
Nozzle Angle Range (°)	0.0 - 46.0	0.0 - 42.5
Nozzle Yield Strength Range (ksi)	35.5 - 57.5	
Avg. Nozzle Yield Strength (ksi)	48.4	
Nominal Nozzle ID (in)	2.750	
Nominal Nozzle OD (in)	4.000	
Nom. Nozzle Wall Thickness (in)	0.625	
Ratio of Mean Radius to Wall Thickness	2.70	

NOTES:

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Table 6-4
Comparison of Nozzle Heat Data for Robinson and Millstone 2 (For Information Only)

Unit (see Note 1)	*Heat No.*	*No. Nozzles*	*Nozzle Type*	*Nozzle Material Supplier*	*Mat'l Spec.*	*Carbon Content (wt%)*	*Yield Strength (ksi)*	*Ultimate Tensile Strength (ksi)*	*Heat Treatment (Temp. / Time / Cooling)*	*Angle Range (°)*

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NOTES:

Table 6-5
Comparison of Nozzle and Weld Geometry for Robinson and Millstone 2 (For Information Only)

						J-Weld Geometry				*Max. Nozzle ID Stress (Note 3)*	
Unit	*Nozzle No.*	*Nozzle Type*	*Heat No.*	*Incidence Angle (°)*	*Yield Strength (ksi)*	*Uphill Side Cross Section Area, A_{up} (in^2)*	*Downhill Side Cross Section Area, A_{down} (in^2)*	*Average Area, $A = (A_{up}+A_{down})/2$ (in^2)*	*Area Ratio, $R = A_{up}/A_{down}$*	*Hoop Stress (ksi)*	*Axial Stress (ksi)*

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NOTES:

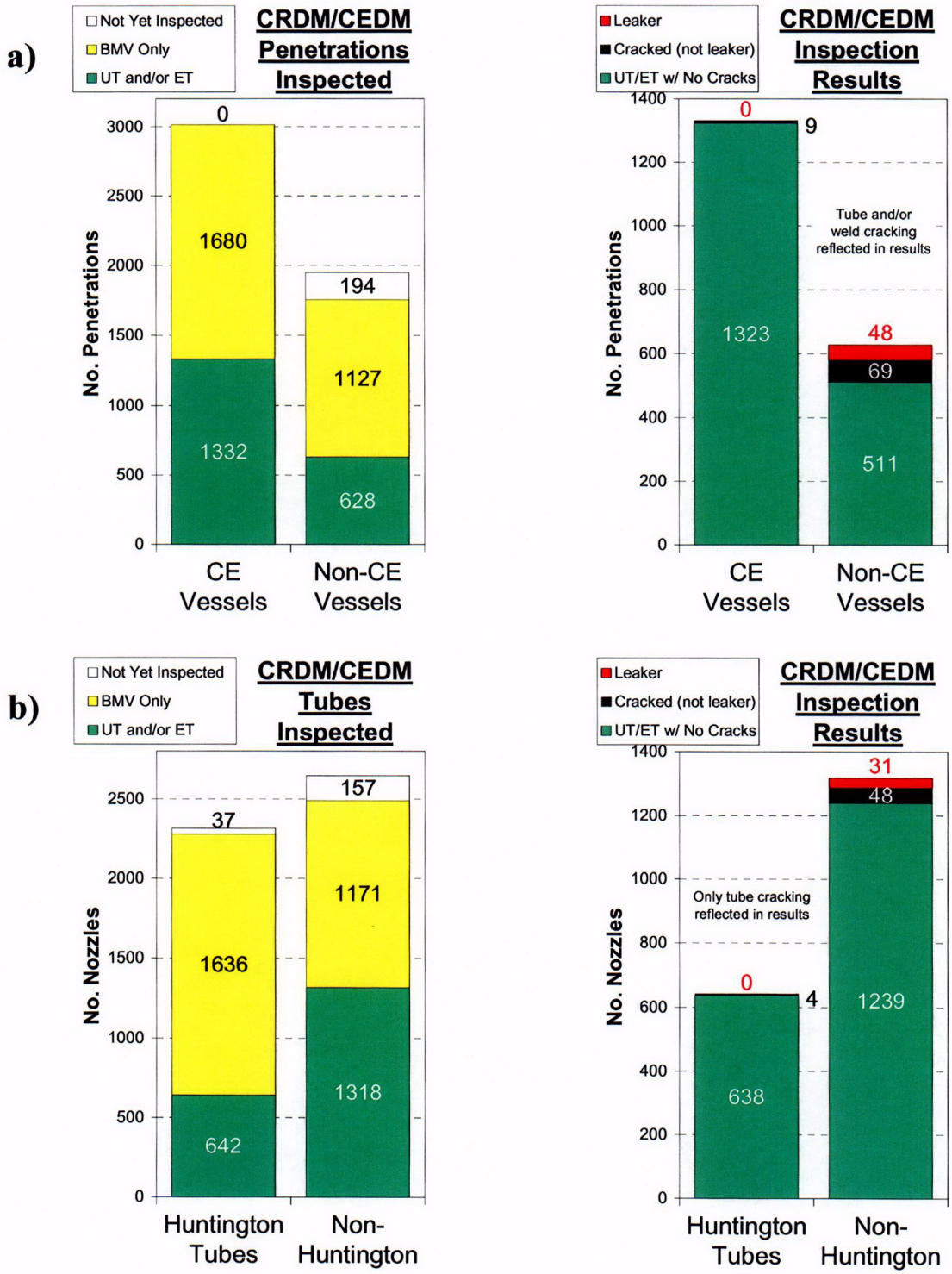

Figure 6-1
Summary Statistics for Industry Inspection Results through Spring 2003: a) Comparison for Vessels Fabricated by Combustion Engineering; b) Comparison for Nozzle Material Supplied by Huntington (For Information Only)

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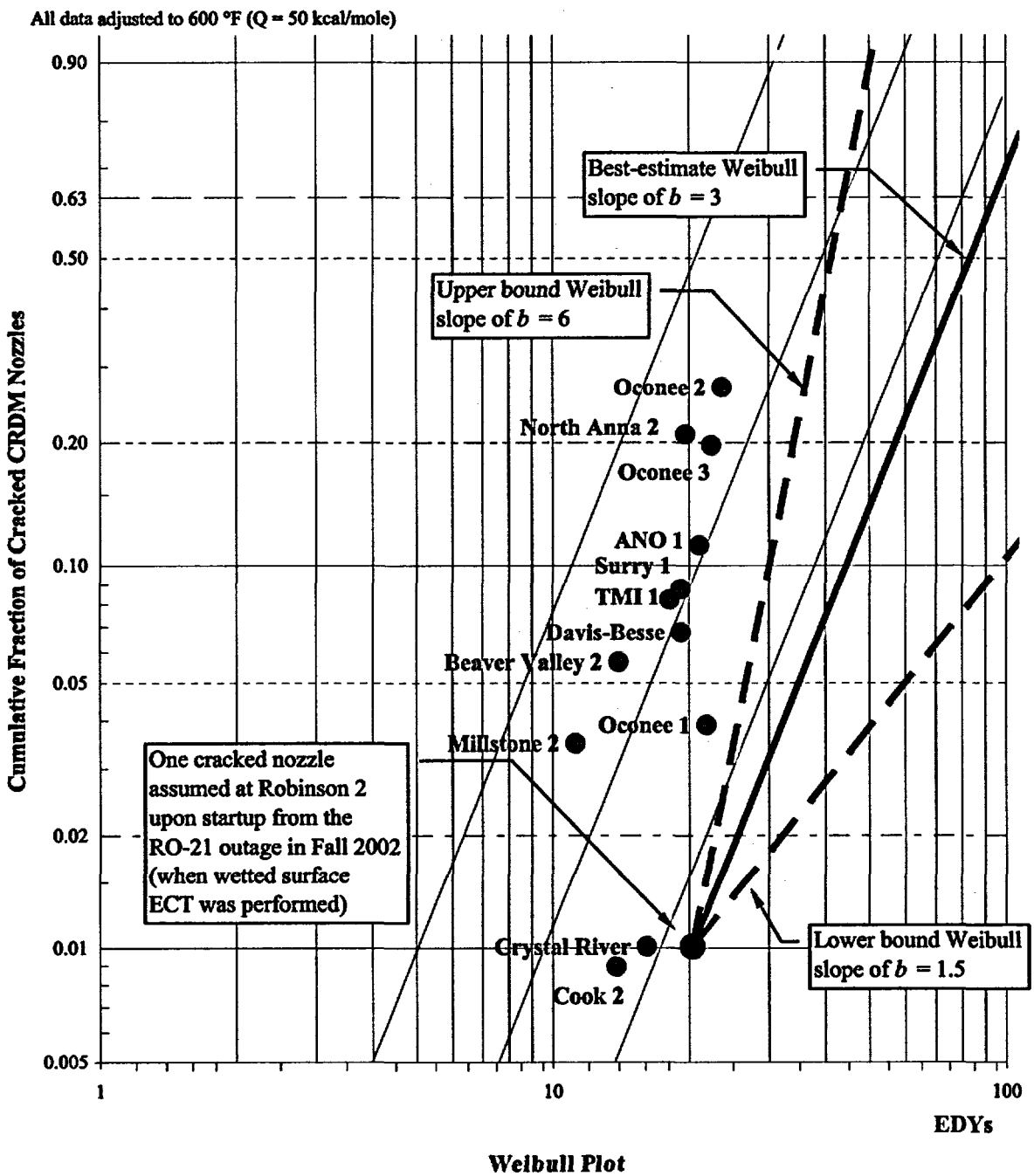

Figure 6-2
Predicted Weibull Curve for Robinson Relative to Other Plants with Reported PWSCC

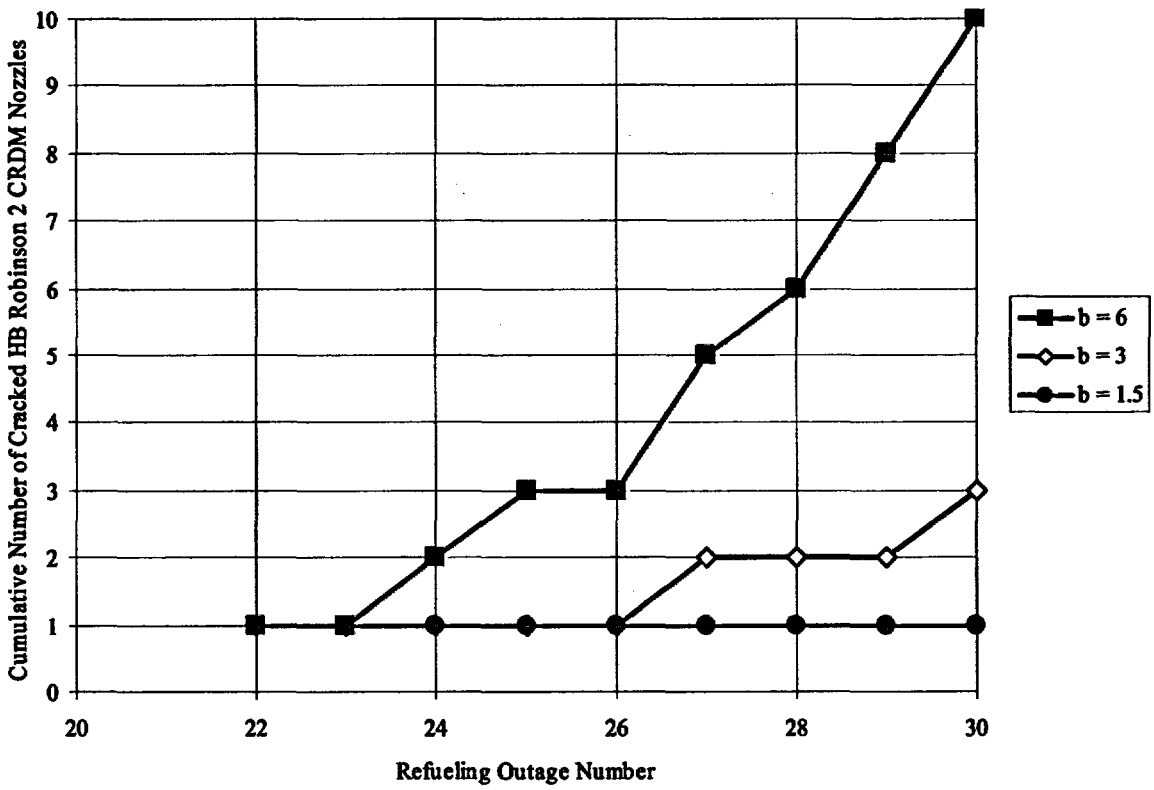

Figure 6-3
Predicted Number of Robinson Nozzles with Reportable PWSCC Cracks

7. Deterministic Evaluations

The purpose of this section is to provide deterministic calculations that demonstrate why there is a low risk of nozzle ejection or head rupture caused by boric acid corrosion. The risk will be assessed quantitatively in probabilistic analyses in Section 8.

The deterministic analysis addresses five main factors in assuring a low risk of nozzle ejection or head rupture. These factors are:

- The non-destructive examinations of the nozzle and wetted surface of the J-groove weld during the RO-21 outage will ensure that the nozzle and weld are free of significant defects at the time of the inspection.
- Time will be required for cracks to initiate in the nozzle or J-groove weld and grow to the point where they result in a leak. It will be assumed for purposes of the deterministic analysis that a crack initiates and grows in the J-groove weld, which is predicted to have a higher crack growth rate than the nozzle base material.
- Once a leak has occurred, it will be conservatively assumed that a 30° through-wall circumferential crack exists in the nozzle above the J-groove weld. The time for this crack to grow from 30° to a limiting arc length of 284° (see Appendix E)[3] will be determined based on fracture mechanics analyses and base metal crack growth rates recommended by the NRC in its current guidance.
- Once a leak has occurred, it will be conservatively assumed that it is of an axial length above the J-groove weld that will result in detectable leakage (0.5" based on field experience) (9). The time for this crack to grow from 0.5" to a length that will result in sufficient leakage for the head temperature to be reduced to the point where leakage can concentrate on the vessel head and cause significant boric acid corrosion will be determined.
- Bare metal visual inspections will be performed every refueling outage to ensure that the maximum time that a crack can grow beyond the point of a visually detectable leak is 18 months (1.5 EFPYs).

7.1 Size Flaws Escaping Detection by Nondestructive Examination

For purposes of the deterministic evaluation, it is assumed that any cracks remaining in the nozzles or J-groove welds after non-destructive examinations have a size at the ECT detectability limit. Based on work in Section 5 and additional input provided by Wesdyne (25, 57), this size is a depth of 0.04 inches (1.0 mm) for an ID or OD tube crack and a

[3] The 284° limiting arc length is conservatively based on a factor of 2.7 on the design pressure as described in Appendix E.

length of 0.12–0.16 inches (3.0–4.1 mm) for a weld crack. For the deterministic calculations of weld crack growth, the initial crack depth is based on the length to depth aspect ratio of either 2:1 or 1:1 based on experience with actual weld cracking (e.g., V.C. Summer and Ringhals) (25, 57). Therefore, a set of four initial depth cases are assumed for the deterministic weld crack growth calculations: 0.12/2 = 0.06 inches, 0.16/2 = 0.08 inches, 0.12/1 = 0.12 inches, and 0.16/1 = 0.16 inches.

7.2 Time for Crack to Grow Through J-Groove Weld

Figure 7-1 shows the shortest growth path for a crack initiating in a J-groove weld for the 27° nozzle on the uphill side, which is shown in Appendix B to be the limiting case. This minimum length from the weld surface to the nozzle OD annulus is approximately 0.94 inches. Subtracting 0.16 inches for the most conservative initial depth leaves a growth length of 0.78 inches.

The EPRI-MRP expert panel on crack growth rates is currently producing a statistical evaluation of the worldwide set of latest available laboratory data for Alloys 182 and 82, but the results are not expected to be available until later in 2003. Therefore, for these calculations it was assumed that the crack growth rate for the weld metal is a factor of 5 higher than the modified Scott equation based on the work presented in MRP-21 (36). This choice for the weld crack growth rate curve is supported by a review of the crack growth rate data for Alloy 182 published since MRP-21 was released (37, 38, 39), which indicates that the use of 5 times the Scott equation for Alloy 182 crack growth still appears to be a conservative assumption. Five times the Scott curve is equivalent to 4.18 times the MRP-55 (35) deterministic rate for the base metal:

$$\frac{da}{dt} = 5\times\left(2.23\times10^{-12}\right)\left(K-9\right)^{1.16} = 4.18\times\left(2.67\times10^{-12}\right)\left(K-9\right)^{1.16} \qquad \text{[Eq. 7-1]}$$

where,

$\frac{da}{dt}$ is in m/s, at 325°C

K is the crack tip stress intensity factor, MPa√m

The crack tip stress intensity factor (K) is calculated assuming an edge crack in a uniform stress field equal to the average weld hoop stress on the shortest path through the J-groove weld as calculated in the finite-element analyses described in Appendix A. The weld length and the average hoop stress are calculated for the four penetration geometries (0.0°, 9.3°, 27° and 46°) and for the uphill and downhill sides (see Appendix B, paragraph B.3). The limiting case is the 27° nozzle on the uphill side, which corresponds to an average hoop stress of about 63 ksi along the assumed crack path. The stress intensity factor expression assumed for this limiting case is the following:

$$K = Y\sigma\sqrt{\pi a} = 1.12\left(63.4ksi\right)\sqrt{\pi a} \qquad \text{[Eq. 7-2]}$$

The value for the geometry factor, *Y*, of 1.12 corresponds to a long edge crack and generally is a conservative assumption for cracks at a free surface (23). In reality, the geometry factor will depend on the aspect ratio of the surface crack, the distance to the free surface ahead of the crack, and the size of the plastic zone at the crack tip. Because plant experience with actual weld cracks indicates a length to depth aspect ratio in the range of 1:1 to 2:1 (25, 57) and because the aspect ratio for deep weld cracks is naturally limited by the weld geometry (see Figure 7-1), the value of 1.12 is judged to be conservative. Note that the assumed stress of 63 ksi in the stress intensity factor expression based on the welding residual stress finite-element analyses is close to the average of the reported Alloy 182 yield strength (47 ksi) and ultimate tensile strength (83 ksi) at a temperature of 600°F (60).

Figure 7-2 shows the crack depth as a function of time for the four assumed initial flaw depths and the uphill side of the 27° nozzle at the head operating temperature of 599.7°F. The calculations suggest that an undetected crack could grow through the weld thickness in 0.5 to 0.7 years. In other words, with nondestructive examinations every 3 years, it would not be possible to ensure on a conservative deterministic basis that there would be no leaks. However, the probabilistic analysis in Section 8 shows that the probability of a leak is low.[4]

It should be noted that the relatively short predicted time to a possible leak for a crack in the J-groove weld is in marked contrast to the data in Appendix B for ID flaws propagating through the nozzle wall. For an initial 0.04" deep flaw, the calculations of Appendix B show that 3.5 to 5.1 years would be required to produce a leak. This longer time results from a combination of slower crack growth rates in the base metal and a smaller initial flaw size. Similarly, the calculations in Appendix B show that an axial flaw initiating on the nozzle OD below the J-groove weld requires a time period of from 5.4 to 10 years to cause leakage.

In summary, the potential for high weld crack growth rates and poorer weld inspection sensitivity dominate the calculated time to a predicted leak.

7.3 Time for Circumferential Crack Above J-Groove Weld to Grow From 30° to 284°

Paragraph B.4 of Appendix B is an analysis of the time required for a 30° through-wall circumferential crack above the J-groove weld to grow to the limiting flaw size of 284° based on a factor of 2.7 on the limit load. The deterministic results in Figure 7-3 for the limiting cases of the circumferential crack originating on the downhill side show that it would take 7.2-10.3 years for this growth to occur, considerably longer than the 3 years of operation between the RO-21 non-visual inspection and an RO-23 follow-up inspection in fall 2005. Note that as described in Appendix B, the deterministic calculation of circumferential crack growth includes a multiplicative factor of 2 increasing the MRP-55 (35) crack growth rate for Alloy 600 base metal to account for current uncertainties in the

[4] Figures B3-1.a and B3-1.b in Appendix B show crack depths as a function of time for flaws in the weld for all four representative geometries on the uphill side and downhill side. These curves were used for the probabilistic evaluations of Section 8 and assumed an initial flaw length of 0.16 inches (4.1 mm) with a length to depth aspect ratio of 1.5 (initial depth of 2.67 mm).

exact chemical environment that exists in the nozzle OD annulus for a leaking penetration. This factor is recommended by MRP-55 and reflected in Figure 7-3.

As a check, the time to grow from 30° to 165° in Figure 7-3 is 5.0-7.2 years. This is considered to be in reasonable agreement with the roughly 10 years between the time that the large flaws were discovered at Oconee 3 and the time that the first Oconee 3 leak is predicted to have occurred as indicated in Figure 6-2 assuming a Weibull slope of 3.

7.4 Time for Axial Crack Above J-Groove Weld to Grow to Size Supporting Significant BAC

Most axial cracks have been discovered by very small amounts of boric acid crystal deposits around the nozzle when the extension of axial cracks above the J-groove weld is in the range of 0.25-0.60 inches (9). These cracks and leaks have not resulted in significant boric acid wastage (e.g., > 1 in^3).

Calculations in MRP-75 (9) show that a precondition of high boric acid corrosion (BAC) rates is to have leakage rates high enough to lower the metal temperature to the range where concentration of liquid can occur over a significant area of low-alloy steel surface. Analyses in MRP-75 show that high corrosion rates are expected to begin at leak rates of about 0.1 gpm. The Davis-Besse plant experience indicates that leak rates of 0.15 gpm represent axial crack extension to about 1.3 inches above the top of the J-groove weld (9), and Figure 7-4 (9) shows the resulting empirical fit of leak rate to crack extension.

The final step in the deterministic calculation for wastage is to link the crack extension required for the leak rate to increase to 0.1 gpm with time using a crack growth calculation. The result from MRP-75 (9) is shown in Figure 7-5. This figure shows that even a supplemental visual inspection of the RPV head without removing the insulation is adequate to ensure that leakage would be detected with sufficient advance warning that high corrosion rates could be avoided. As demonstrated in Section 8, bare metal visual inspections will provide significantly more margin.

7.5 Role of Bare Metal Visual Inspection Every Outage in Preventing Ejection or Rupture

While the initial non-visual nondestructive examination during the RO-21 outage provides the first line of defense against significant PWSCC cracks, leaks, ejection or rupture, bare metal visual inspections every outage provide additional assurance that the risk of ejection or rupture is extremely low. This risk is quantified in the Section 8 probabilistic analyses.

In summary, the deterministic analyses have demonstrated that a high degree of assurance against nozzle ejection or vessel rupture are provided by a BMV inspection during the RO-22 outage and a combination of a BMV inspection and a non-visual NDE inspection of each CRDM nozzle during the RO-23 outage. It will also provide a low risk of leaks from PWSCC cracks propagating in the nozzle wall. The greatest risk of a leak is from a through-wall crack in the J-groove weld. However, the probabilistic analyses in Section 8 indicate that this risk is low.

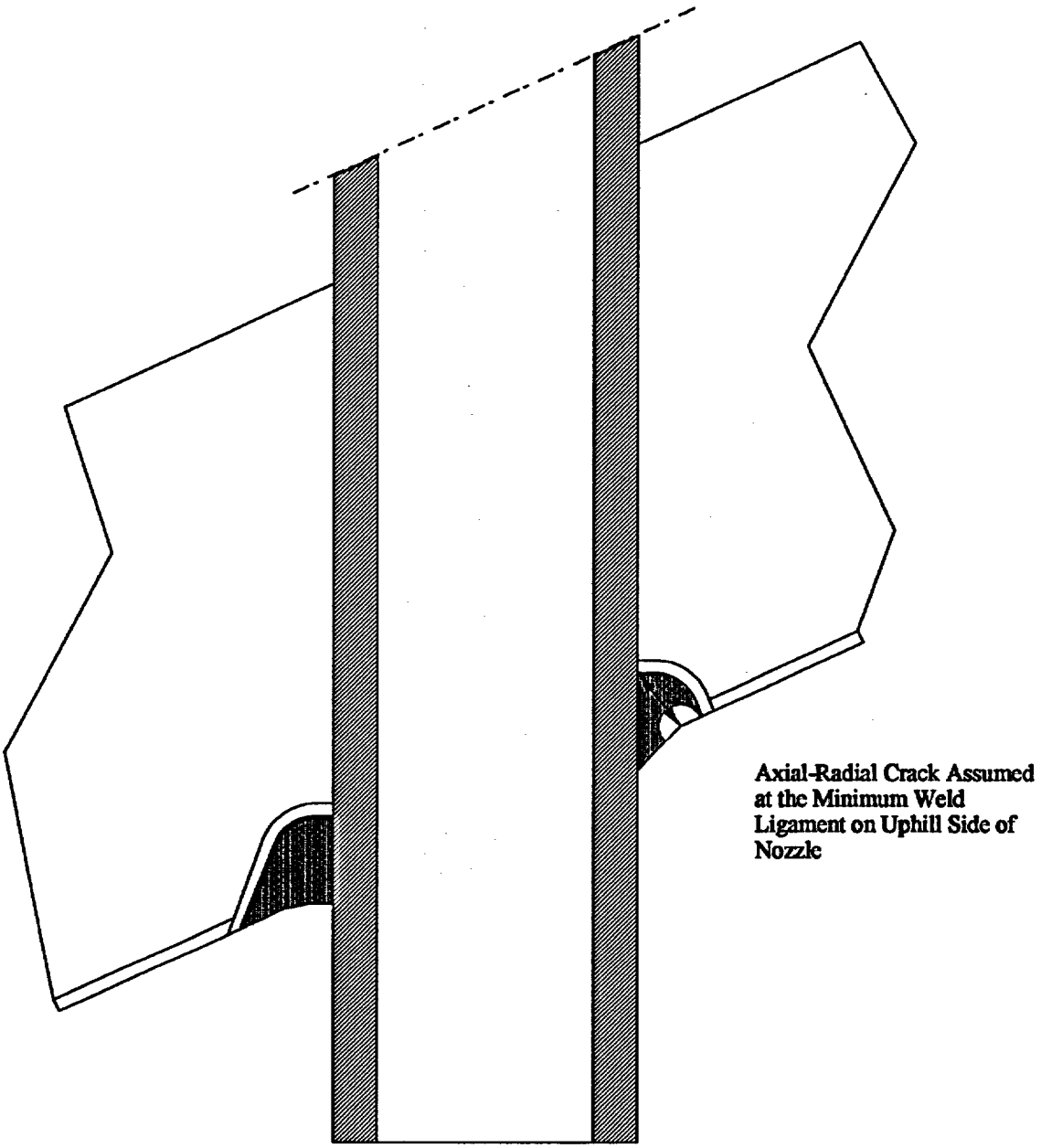

Figure 7-1
Crack Growth Path Through J-Groove Weld (27.1° Nozzle, Uphill Side)

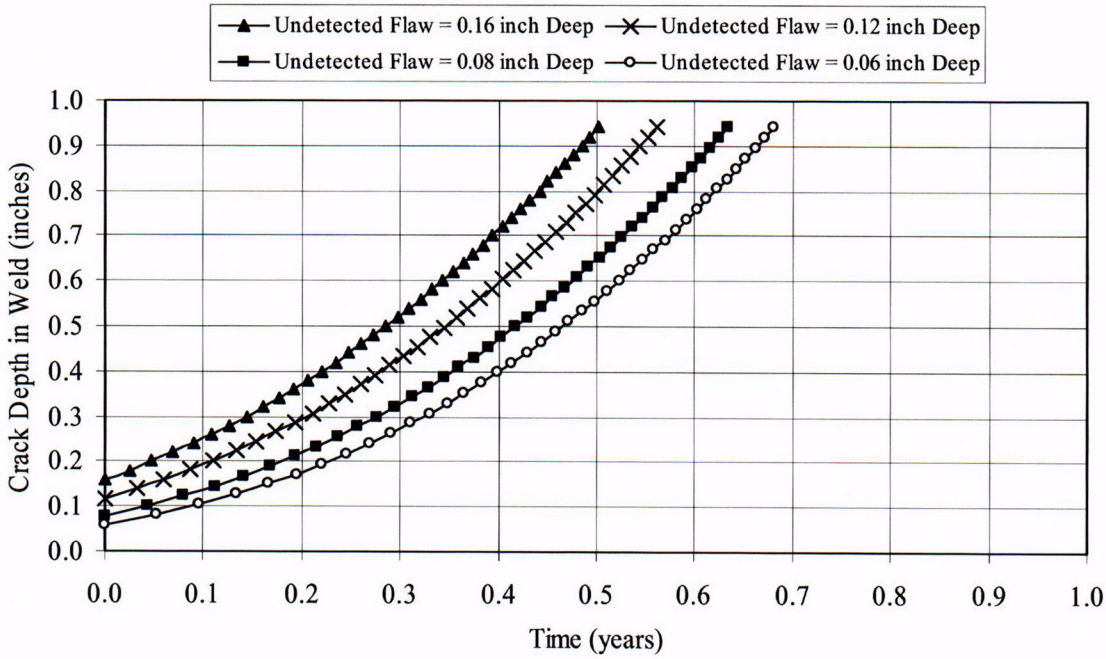

Figure 7-2
Predicted Crack Growth for Flaws in Weld of Four Initial Sizes (27.1° nozzle, Uphill Side)

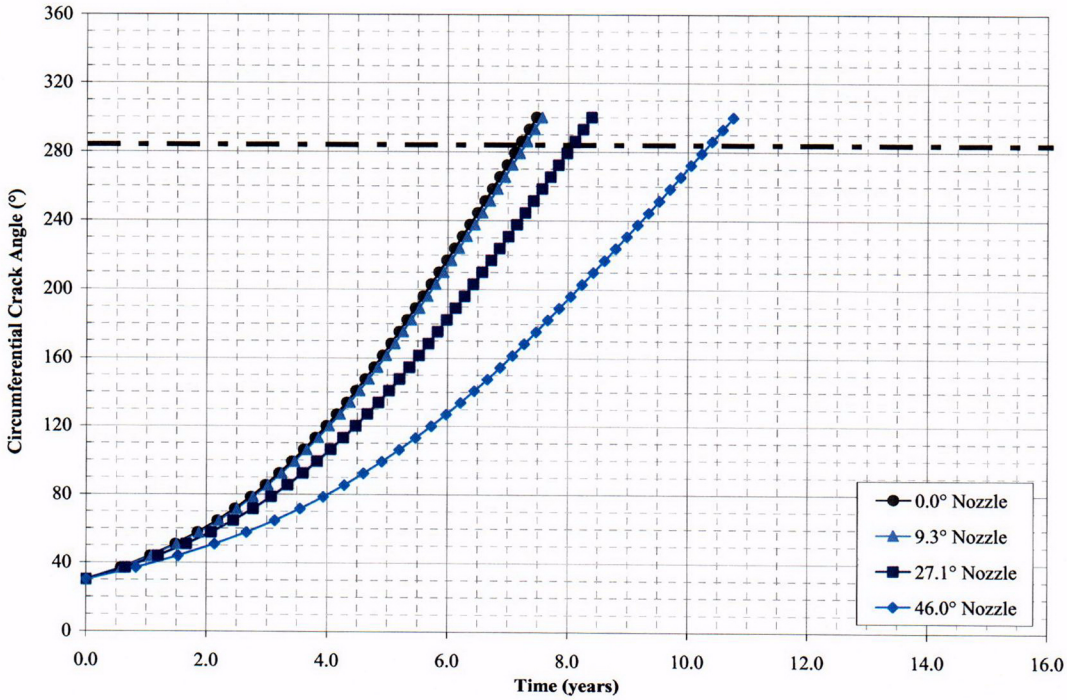

Figure 7-3
Time for 30° Through-Wall Circumferential Crack Above J-groove to Grow to 284° Limit

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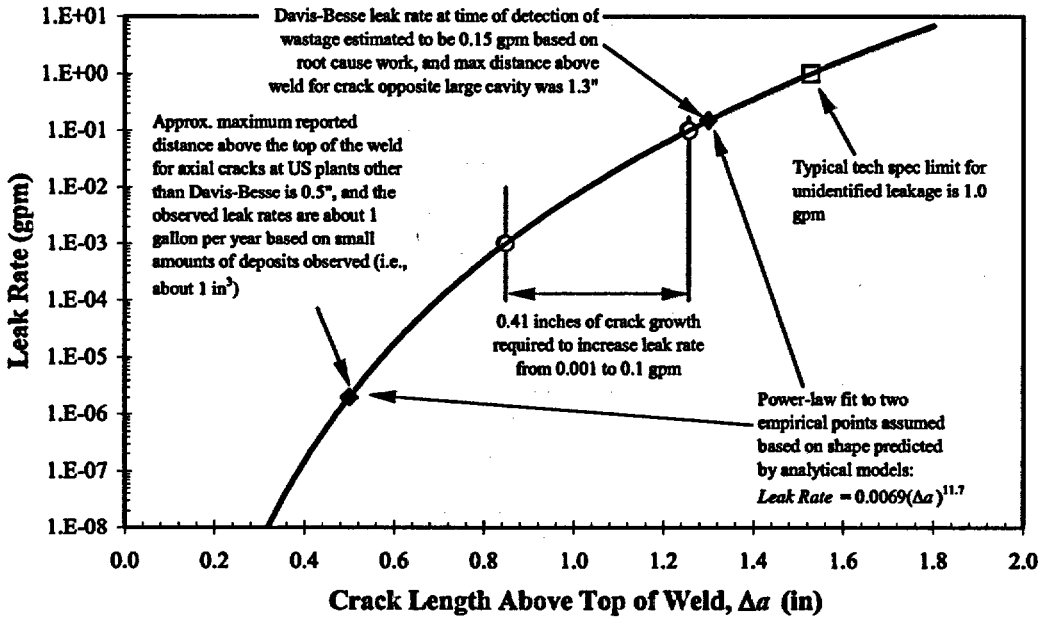

Figure 7-4
Time for Undetected Weld Defect to Growth Through Weld (MRP-75) (9)

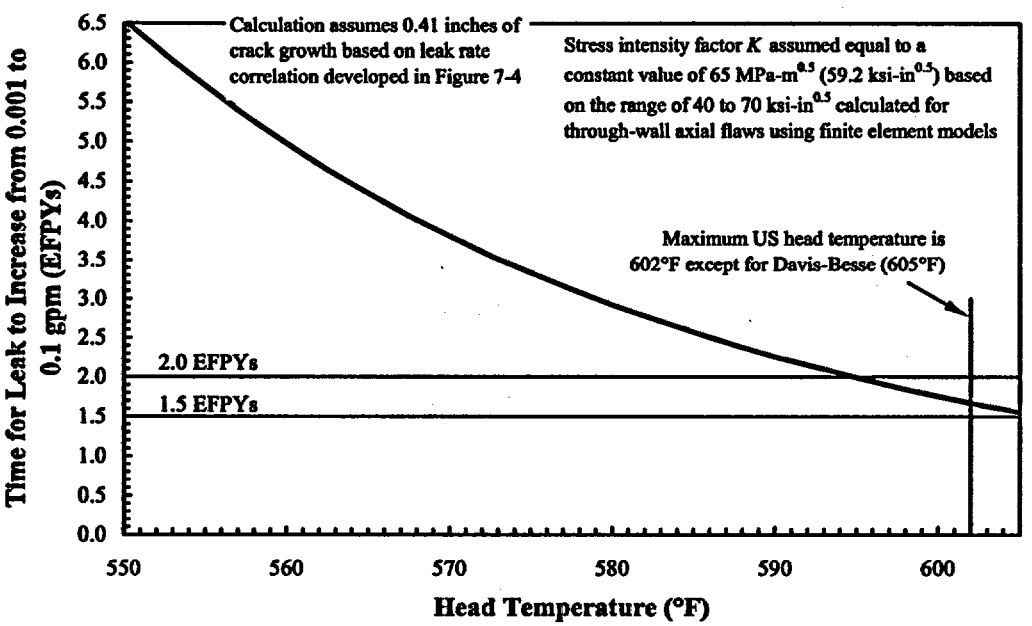

Figure 7-5
Reinspection Interval to Ensure that Supplemental Visual Inspection will Detect Leak Prior to High BAC Corrosion Rates Occurring (MRP-75) (9)

8. Probabilistic Evaluations

This section presents the methodology and results of probabilistic risk evaluations of the two main failure modes of potential concern: nozzle ejection and cladding blowout or head rupture due to wastage. The probabilistic evaluations allow the re-inspection interval for non-visual NDE inspections of the Robinson closure head nozzles to be set on a risk-informed basis, with Reg. Guide 1.174 (5) providing the basis for the criterion for judging that deferral of the next NDE inspection by one outage has a negligible impact on nuclear safety. As will be shown below using two Monte Carlo calculations that conservatively simulate the processes of nozzle ejection and boric acid wastage, deferral of the next inspection by one outage results in less than a 1×10^{-6} per reactor year increase in core damage frequency (CDF). This level of reliability is defined by Reg. Guide 1.174 to provide a risk-informed basis for concluding that a negligible impact on nuclear safety results from the inspection deferral.

The first three subsections below show that the inspection deferral does not result in a significant increase in CDF, on the basis of the nozzle ejection failure mode. This is shown on the basis of a probabilistic fracture mechanics (PFM) base case simulation using appropriately conservative inputs in combination with a sensitivity study that shows that the conclusions are not significantly affected when the potential uncertainties in the driving input assumptions are considered. The PFM model results also show that there is a reasonably low probability of pressure boundary leakage given deferral of the non-visual inspection. Following the PFM model description and results, a probabilistic wastage model similar to the model presented in Appendices C and D of MRP-75 (9) is presented that indicates that the potential for boric acid wastage—given the bare metal visual (BMV) inspection that is scheduled for the next refueling outage in spring 2004—has a negligible impact on the CDF increments calculated to account for the possibility of nozzle ejection.

8.1 Probabilistic Fracture Mechanics (PFM) Model Description

Although cracking in RPV closure head nozzles has been observed to be predominantly axial, there have been some reported instances of circumferential cracking in the region just above the top of the J-groove weld. If such a circumferential flaw in the CRDM nozzle grows to become a through-wall circumferential flaw having a circumferential extent greater than 330° (see Appendix E), then net section collapse could occur, leading to a

pressure boundary break having a diameter of about 2.75 inches. At Robinson, this size break is conservatively categorized as a small-break LOCA.

Therefore, the basic approach presented here is to calculate the increase in CDF during each of the three years between the time of the wetted surface ECT inspection in fall 2002 (RO-21) and the refueling outage scheduled for fall 2005 (RO-23), given deferral of the next non-visual inspection from RO-22 in spring 2004 to RO-23. It is conservatively assumed that not deferring the next inspection corresponds to a zero probability of core damage,[5] so the increase in CDF given deferral due to nozzle ejection is:

$$\Delta CDF_{ejection} = P_{ejection} \times CCDP_{SBLOCA} \qquad \text{[Eq. 8-1]}$$

where $P_{ejection}$ is the probability per reactor year of the initiating event of nozzle ejection due to net section collapse of the nozzle base metal cross section and $CCDP_{SBLOCA}$ is the conditional core damage probability, or Birnbaum probability, for a small-break loss of coolant accident (LOCA). [

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[5] The CDF due to nozzle ejection without deferring the next non-visual inspection could be subtracted from the CDF with the deferral to determine the increment in CDF. However, no credit is taken in the probabilistic evaluations presented here for such a subtraction; the increment in CDF is just taken as the initiating event frequency multiplied by the CCDP.

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8.2 PFM Model Inputs

The two pages of Table 8-1 list the complete set of PFM model inputs. The selection of these inputs is described below.

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8.3 Industry Experience for Surface Craze Indications

As noted in Section 5, NDE inspections performed during RO-21 showed that several nozzles had craze type cracking on the ID surface.

Two plants have performed repeat inspections to assess potential growth of multiple shallow axial flaws sometimes referred to as "craze cracks." These are Ringhals 2 and Oconee 2. Details and conclusions from these inspections are as follows:

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Conclusions Regarding Craze Indications

Multiple shallow axial cracks in close proximity, also called craze cracks, have been discovered at several plants, and their behavior has been studied by repeat inspections over several years at three plants. There is no evidence to suggest that these cracks begin to grow rapidly in depth, and none of the leaks has been attributed to a nozzle with known craze cracks.

8.4 PFM Model Results

The results of the PFM calculations are shown in tabular form in Table 8-6. The first row in this table is the set of results for the base case set of inputs described above. The 16 rows below show the input variable that is changed for each sensitivity case along with the corresponding results to the right. The results are shown on a per-year basis for each of the

three years between the time of the RO-21 ECT and UT inspections in fall 2002 and the time of the RO-23 refueling outage in fall 2005, assuming deferral of the RO-22 non-visual inspections. The columns to the right of the PFM model case definitions show the probability of leakage, the probability of nozzle ejection, and the increment in core-damage frequency (CDF) resulting from multiplication of the probability of nozzle ejection and the CCDP Birnbaum probability for a small-break LOCA (54, 57).

The per-year increases in CDF in the right portion of this table are also shown graphically in Figure 8-11. The maximum value for ΔCDF for the base case is 1.0×10^{-7} per year, one order of magnitude less than the criterion recommended by Reg. Guide 1.174 (5) for defining an acceptable change. In addition, the PFM model produced acceptable results for the 16 various sensitivity cases. As discussed above, these cases were generally designed to probe the dependence of the PFM conclusions on the model assumptions and inputs, so more conservative inputs were chosen for these cases. The sensitivity results verify the robustness of the base case results by showing that the results are not overly dependent upon the precise set of assumptions and inputs. [

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The ΔCDF results in Table 8-6 should be viewed in light of the conservatisms discussed above that have been built into the base case set of inputs. One particularly important conservatism is in the selection of the stress intensity factors (8, 57) that are assumed to drive above-weld circumferential crack growth around the nozzle circumference. These stress intensity factors are conservative for several reasons. First, the stress intensity factors used are based on "enveloping stresses" perpendicular to the crack plane that correspond to a meandering crack plane through the highest stress locations above the weld. Second, as shown in Appendix B, the assumed stress intensity factors are very large compared to other values presented in the industry. Third, stress intensity factors for an outer peripheral penetration geometry are applied for all Robinson nozzle angles even though the stress intensity factors are expected to be significantly smaller for lower incidence angles based on the Robinson stress calculations presented in Appendix A.

The base case and sensitivity cases also generally show a low probability of leakage. The maximum value for the base case is just under 6% per year, a quite reasonable result considering the conservatisms inherent in the PFM inputs, in particular, the selection of the nominal Weibull curves assuming one cracked tube and one cracked weld immediately upon restart from the RO-21 outage in fall 2002, the use of rather high estimates of the stress intensity factor driving weld crack growth, and the use of the minimum distance for weld crack growth from the surface of the weld to the nozzle OD annulus.

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8.5 Probabilistic Wastage Model and Results

A probabilistic wastage model similar to the MRP probabilistic wastage model (9) was used to verify that the potential for boric acid wastage—given the bare metal visual (BMV) inspection that is scheduled for the next refueling outage in spring 2004—has a negligible impact on the CDF increments calculated to account for the possibility of nozzle ejection. The model used is identical in structure to the MRP model described in detail in Appendices C and D of MRP-75 (9), but several inputs have been modified for the Robinson-specific evaluation.

As described in MRP-75, the probabilistic wastage model is a single nozzle Monte Carlo model for which an axial tube crack to the nozzle OD annulus above the J-groove weld root produces an increasing leak rate as the crack grows to a greater elevation above the weld root. The simulation assumes that the modeled nozzle has a leak at the beginning of the simulation. The leak rate through this nozzle is then tied in the model to the boric acid wastage rate on an assumed active wastage front area. Finally, the low-alloy steel wastage volume is integrated over time and then compared to the wastage volume that would exhaust the ASME Code margins on primary membrane and primary membrane plus bending stress limits in the head. Appendix D of the report shows that 300 in^3 of low-alloy steel material can be lost via boric acid wastage without causing the Code stresses to be exceeded. Appendix D evaluates the allowable wastage volume assuming that material is lost between adjacent nozzles or around a single nozzle.

Table 8-7 shows the full set of model inputs and identifies the differences between the set of inputs used for Robinson and the set of inputs used to generate the MRP-75 results. These differences are summarized as follows:

- The Robinson model uses the current Robinson head temperature of 599.7°F (14) and a conservative operating cycle length of 1.5 EFPYs.

- An inspection sensitivity appropriate for detection of boron deposits via bare metal visual (BMV) inspection—rather than supplemental visual (SV) inspection—was applied for Robinson.

- For the Robinson evaluation, more conservative wastage rates were assumed for leak rates exceeding the critical value for rapid wastage, including an upper bound wastage rate of 7 inches per year.

- Considerably higher stress intensity factors were conservatively chosen to drive the axial crack growth in the model for the Robinson evaluation.

- The same crack growth rate assumptions used for the Robinson PFM model were applied to the probabilistic wastage model for Robinson including the use of a factor to model within-heat variability.

- The Robinson simulation is terminated after 25.1 EFPYs of operation, which corresponds to the time of the RO-23 non-visual inspection, scheduled for fall 2005. Thus, the simulation assumes that the modeled nozzle may be leaking for the entire 25.1 EFPYs of operation at Robinson.

The results of the Robinson probabilistic wastage calculation are shown in Figure 8-12, which is the cumulative distribution of wastage cavity sizes at the time that the wastage cavity is detected. Usually, the simulation results in detection via a BMV examination although for 2.5% of the Monte Carlo trials detection is via the leak rate exceeding 1.0 gpm, which conservatively would trigger a plant shutdown and a search for the leak source. The results show that there is an extremely small probability that the wastage cavity could exceed the 300 in^3 allowable volume.

For example, for a probability level of 1.0×10^{-5} the model calculates a wastage cavity size of 126 in^3. Given the maximum base case probability of leakage of 5.9% per year from the PFM evaluation, the probability level of 1.0×10^{-5} can conservatively be used to calculate a limiting CDF increment using the following relationship:

$$\Delta CDF_{wastage} = P_{leakage} \times P_{cavity>allow} \times CCDP_{LBLOCA} \qquad \text{[Eq. 8-2]}$$

where the CCDP Birnbaum probability is taken for a large-break LOCA since this value (55) is greater than that for small- and medium-break LOCAs at Robinson (54, 57). Then,

$$\Delta CDF_{wastage} < 0.059 \times 1.0\times10^{-5} \times 2.93\times10^{-2} = 1.7\times10^{-8} \text{ per year} \qquad \text{[Eq. 8-3]}$$

Since this value is about 6 times lower than the maximum base case value for the CDF increment due to nozzle ejection, the potential for boric acid wastage does not have an effect on the conclusion that deferring the next non-visual inspection by one cycle at Robinson produces an acceptably small change in CDF.

Note that the ΔCDF calculation presented here conservatively assumes that the large-break LOCA occurs at a wastage cavity size of 126 in^3, which is considerably smaller than the 300 in^3 value calculated in Appendix D for maintaining ASME Code stress limits. Finally, because of the relatively low probability of leakage and the limited number of predicted cracked nozzles over the three-year evaluation period (see Figure 6-3), the potential effect of multiple leaking nozzles on the calculation of ΔCDF due to wastage is insignificant.

8.6 Other Potential Concerns

This subsection briefly addresses some additional potential concerns related to the probabilistic evaluations of nozzle ejection and boric acid wastage presented above. These

are the potential for collateral damage following nozzle ejection or head/cladding rupture, the potential for loose parts generation, and the effect on the large early release frequency (LERF).

The calculations of the increment in core damage frequency presented above use the CCDP Birnbaum probabilities for the representative Robinson small-break LOCA in the case of nozzle ejection. This is appropriate because control rod ejection is an analyzed event covered by the Birnbaum probability for a small-break LOCA and because the potential for collateral damage is not expected to significantly affect the Birnbaum probability. MRP evaluations that are in progress are expected to conclude that the potential for collateral damage, for example to other CRDM housings preventing rod insertion, has an insignificant effect on the CCDP values for the representative small-break LOCA. This conclusion results from the favorable location of the break on top of the reactor vessel head, which does not interrupt flow from the emergency core cooling system (ECCS), and from the reactor shutdown margin that is expected to be sufficiently large to tolerate a limited number of adjacent control rod failures. A similar conclusion applies for potential breaks caused by boric acid wastage. For a potential large-break LOCA (> 13" break (54, 57)) caused by a very large wastage volume, there may be more of a concern for collateral damage to adjacent CRDM housings, but the faster depressurization, voiding within the core, and subsequent injection of borated ECCS water are expected to shut down the fission reaction without any need for rod insertion. Lastly, it is noted that the potential for sump clogging following a LOCA is currently being evaluated by Robinson as part of its response to NRC Bulletin 2003-01 (56), which was issued on June 9, 2003. This evaluation will include the insulation on top of the reactor vessel head as a potential source of debris.

A potential safety concern given CRDM nozzle cracking in addition to nozzle ejection and boric acid corrosion is the generation of loose parts. Loose parts may either be captured by a drive rod or released to the flow in the upper plenum of the reactor vessel. Captured loose parts have the potential to prevent a control rod to drop, while non-captured loose parts have the potential to prevent a control rod to drop or to damage fuel pins, steam generator tubes, the steam generator tubesheet, or the bottom reactor vessel area. Therefore, generation of non-captured loose parts is more of a potential concern. Because of the presence of guide funnels at the bottom of most of the CRDM nozzles at Robinson, the generation of non-captured parts due to cracking is most plausible for the 17 open design CRDM nozzles at Robinson. A non-captured loose part would require either a 360° below-weld circumferential crack or multiple below-weld axial and circumferential cracks in one of these 17 open nozzles. However, because nozzle ejection is always assumed to produce a LOCA with a 2.02×10^{-2} probability of core damage, the potential effect of loose parts generation is judged to have an insignificant effect on nuclear safety in comparison to the process of nozzle ejection modeled in the PFM simulation.

The final additional potential concern related to the probabilistic evaluations is the effect on LERF given nozzle ejection or head rupture due to boric acid wastage. The initiating events that contribute materially to LERF are ones that either involve containment bypass or which involve failure of systems important to containment heat removal. Because of

this observation and because none of the Robinson LOCAs (small, medium, or large) in its design basis produce any significant contributions to LERF, the increment in core damage frequency (CDF) and not LERF is the proper risk parameter for evaluation.

8.7 Overall Probabilistic Results

The base case probabilistic fracture mechanics (PFM) evaluation shows a maximum increment in CDF of 1.0×10^{-7} per year. This is one order of magnitude lower than the 1×10^{-6} criterion recommended by Reg. Guide 1.174 (5) for risk-informed decision making. In addition, a detailed sensitivity study shows that the conclusion that the effect on CDF is insignificant is robust and is not overly dependent on the set of input assumptions. Furthermore, the base case PFM calculations show a maximum probability of leakage just under 6% per year, a reasonable result considering the conservatisms inherent in the PFM inputs. Finally, a probabilistic wastage model similar to that presented in MRP-75 (9) shows that the potential for boric acid corrosion of the low-alloy steel head material—given the BMV examination that is scheduled for the spring 2004 refueling outage—has an insignificant effect on CDF in comparison to the value of 1.0×10^{-7} per year calculated on the basis of the PFM simulation.

Therefore, a risk-informed basis exists for deferring the next non-visual RPV head inspection by one operating cycle on the basis of deterministic and probabilistic results. The deterministic evaluations in Section 7 show that net section collapse due to nozzle ejection or head/cladding rupture due to boric acid wastage are unlikely. The probabilistic evaluations above confirm the conclusions of the deterministic evaluations using Monte Carlo simulations in combination with the conditional core damage probability (CCDP) values for Robinson LOCAs and the ΔCDF criterion of 1×10^{-6} per year recommended by Reg. Guide 1.174 (5).

Table 8-1
Input Distributions Used in the Monte Carlo Probabilistic Fracture Mechanics (PFM) Model

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Table 8-1 (cont'd)
Input Distributions Used in the Monte Carlo PFM Model

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Table 8-2
Typical Values of Weibull Slopes for Steam Generator Tube PWSCC Based on Plant Data (31)

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Type of PWSCC	Number of Plants	Median	Average	Standard Deviation

Table 8-3
Summary of Circumferential Cracks Detected Above or Near Top of J-Groove Weld (For Information Only)

Unit	NSSS Design	Nozzle ID	Nozzle Angle (°)	Inspection Results							
				Date	Approx. EDYs	OD/ ID	Axial Location	Circ. Angle (°)	UH/DH Side	Depth (in)	TW Depth (%)
Crystal River 3	B&W	32	26.2	Oct-01	16.2	OD	above weld	91	DH	0.29	47%
Davis-Besse	B&W	2	8.0	Mar-02	19.2	OD	above weld	34	DH	0.31	50%
North Anna 2	W	15	19.8	Sep-02	19.7	OD	≥1.12" below root	5	DH	0.23	36%
		41	33.1			OD	≥0.52" below root	46	DH	0.10	16%
		54	38.6			OD	≥0.04" below root	79	UH	0.23	36%
						OD	≥0.28" below root	32	DH	0.16	25%
		59	40.0			OD	≥0.31" below root	76	DH	0.15	24%
						OD	≥0.32" below root	50	UH	0.15	24%
		65	42.6			OD	≥0.32" below root	72	DH	0.15	24%
						OD	≥0.20" below root	30	UH	0.08	12%
		67	42.6			OD	≥0.80" below root	44	DH	0.09	15%
Oconee 2	B&W	18	18.2	Apr-01	22.2	OD	above weld	36	DH	0.07	11%
Oconee 3	B&W	11	16.2	Feb-01	21.7	OD	over weld	153	DH	0.36	57%
						OD	over weld	113	UH	0.25	40%
		23	23.2			OD	above weld	66	DH	0.22	35%
		50	35.1			OD	above weld	165	UH	0.62	pin holes
		56	35.1			OD	above weld	165	UH/DH	0.62	100%
		2	8.0	Nov-01	22.5	OD	above weld	48	DH	0.18	29%
		26	24.7			OD	over weld	44	DH	0.07	11%

Table 8-4
Summary of Industry Experience Regarding Circumferential Nozzle Cracking and Leakage (For Information Only)

No.	Unit	NSSS Supplier	*Number of Nozzles on Head*				*A. Leak OR Circ. Crack above or near Top of J-Groove Weld*		*B. Circ. Crack above or near Top of J-Groove Weld*		*Ratio B/A*
			CRDM	CEDM	ICI	*Total*	No. Nozzles	*Percentage*	No. Nozzles	*Percentage*	
1	ANO 1	B&W	69			69	1	1.4%	0	0.0%	0.00
2	Crystal River 3	B&W	69			69	1	1.4%	1	1.4%	1.00
3	Davis-Besse	B&W	69			69	3	4.3%	1	1.4%	0.33
4	North Anna 1	W	65			65	1	1.5%	0	0.0%	0.00
5	North Anna 2	W	65			65	14	21.5%	6	9.2%	0.43
6	Oconee 1	B&W	69			69	2	2.9%	0	0.0%	0.00
7	Oconee 2	B&W	69			69	11	15.9%	1	1.4%	0.09
8	Oconee 3	B&W	69			69	14	20.3%	6	8.7%	0.43
9	Surry 1	W	65			65	2	3.1%	0	0.0%	0.00
10	TMI 1	B&W	69			69	5	7.2%	0	0.0%	0.00
	Totals (All U.S. Units)		3871	1090	94	5055	54	1.1%	15	0.3%	0.28
										Average	0.23

Table 8-5
Evaluation of Within-Heat Crack Growth Rate Variability Using MRP-55 Database of Laboratory Data

Heat Rank	Material Heat	No. of Data Points	Log Mean Power-Law Constant α_{heat} at 325°C (617°F)[1]			
			α_{heat}	$\alpha_{heat}/\alpha_{MRP\text{-}55}$	STD DEV[2] $\ln(\alpha_i/\alpha_{heat})$	EXP(2*STDDEV)[3]

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Notes:
[1]Assuming form $CGR = \alpha(K - 9)^{1.16}$ where CGR is expressed in m/s and K is expressed in MPa√m
[2]Population standard deviation in log-space for each heat.
[3]Two standard deviation units expressed in terms of a factor on the log-mean α_{heat} value.

Table 8-6
Results of the PFM Calculations Including Sensitivity Study

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Table 8-7
Inputs to the Monte Carlo Probabilistic Wastage Model

Quantity / Description	*Symbol*	*Nominal Value*	*Units*	*Statistical Distribution*	*Parameter 1* Triangular = *c* Log-triang = c	*Parameter 2 (lower bound)* Triangular = *a* Log-triang = *a* Uniform = *a*	*Parameter 3 (upper bound)* Triangular = *b* Log-triang = *b* Uniform = *b*	*Comments*
Operating Cycle Length	$\Delta EFPY$	1.5	EFPYs	Constant				100% availability assumed between BMV inspections at each RFO
Head Temperature	T_{head}	599.7	°F	Triangular	599.7	594.7	604.7	MRP-75 probabilistic wastage evaluations assumed 602°F
Fraction of Fuel Cycle Completed When Leak Begins	f_{t0}	0.5	–	Uniform		0.0	1.0	Same assumption as MRP-75, Rev. 1, Appendix D
Stress Intensity Factor Driving Axial Crack Growth Above Top of Weld	K	80	MPa√m	Triangular	80	60	100	Higher K values than assumed in MRP-75 (55, 65, 75 MPa√m) are conservatively assumed
Crack Growth Rate Power Law Coefficient × 10^{13}	A_{ref}	19.0	(m/s) × (MPa√m)$^{-1.16}$	Log-triang	19.0	1.07	92.9	Same CGR distribution assumed for the Robinson PFM model
Within Heat Crack Growth Rate Variability Multiplier	f_{wh}	1.00	–	Log-triang	1.00	1/3.12 = 0.3205	3.12	Same within heat distribution assumed for the Robinson PFM model
Crack Growth Rate Activation Energy	Q_g	31.0	kcal/mol	Triangular	31.0	27.0	35.0	Same Q_g distribution assumed for the Robinson PFM model
Leak Rate for Crack Extending 0.5" Above Top of Weld	LR_0	2.0E-06	gpm	Log-triang	2.0E-06	1.0E-06	1.0E-04	Same assumption as MRP-75, Rev. 1, Appendix D
Leak Rate for Crack Extending 1.3" Above Top of Weld	LR_1	0.15	gpm	Log-triang	0.15	0.001	1.0	Same assumption as MRP-75, Rev. 1, Appendix D
Leak Rate Yielding Wastage Rate WR_{low}	LR_{low}	0.001	gpm	Log-triang	0.001	0.0001	0.01	Same assumption as MRP-75, Rev. 1, Appendix D
Critical Leak Rate Yielding Upper Shelf Rapid Corrosion Rate WR_{crit}	LR_{crit}	0.10	gpm	Log-triang	0.10	0.02	0.20	Same assumption as MRP-75, Rev. 1, Appendix D
Wastage Rate at Leakage Rate LR_{low}	WR_{low}	0.072	in/yr	Triangular	0.072	0.010	0.250	Same assumption as MRP-75, Rev. 1, Appendix D
Upper-Shelf Wastage Rate for Leak Rates Greater than LR_{crit}	WR_{crit}	2.5	in/yr	Triangular	2.5	1.0	7.0	Higher wastage rates than assumed in MRP-75 (0.75, 1.5, 5.0 in/yr) are assumed
BMV Detection Sensitivity for Boric Acid Crystal Release	BAC_{det}	10	in^3	Triangular	10	5	20	~0.5 in^3 or more of boron deposits at the periphery of the nozzle expected to be detectable

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Figure 8-1
Simplified Process for PFM Simulation Model for Robinson 2 Plant

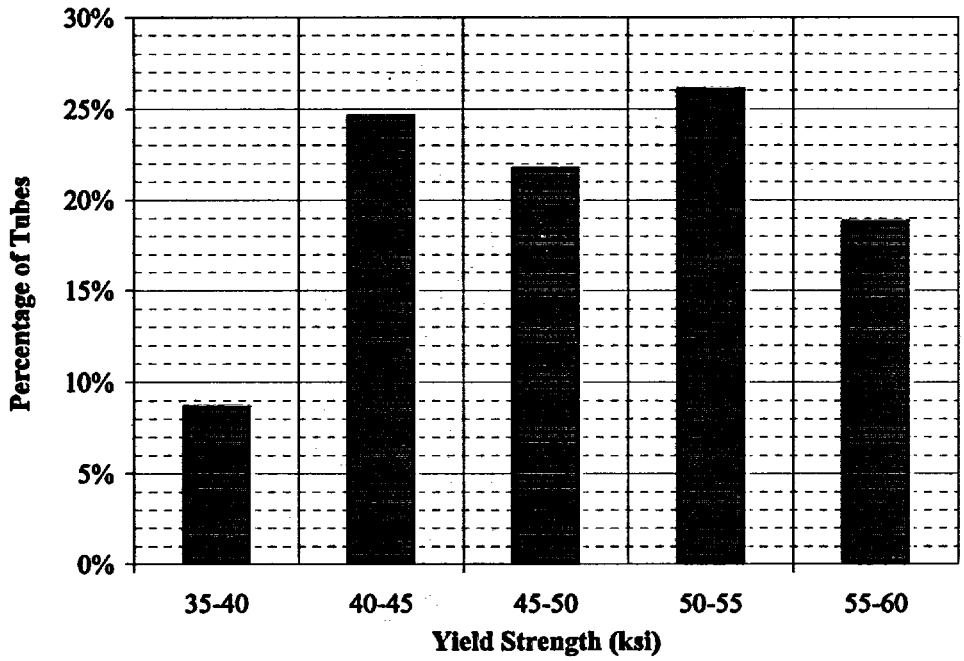

Figure 8-2
Distribution of Robinson Yield Strengths for the 69 Nozzles Based on Heat CMTRs

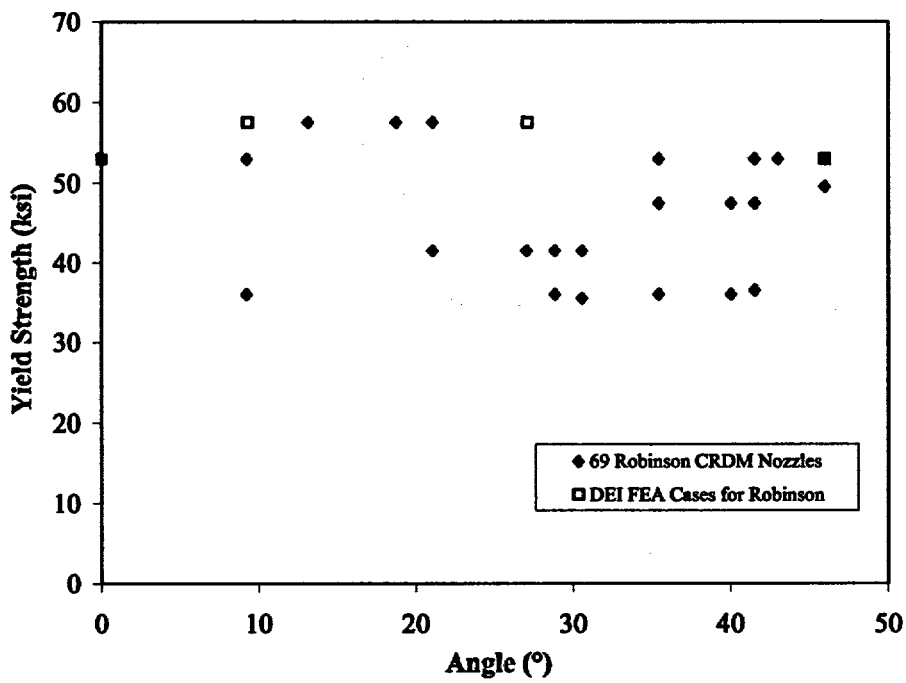

Figure 8-3
Robinson Yield Strength versus Nozzle Incidence Angle

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Figure 8-4
Distribution of Time-to-PWSCC for Steam Generator Hot Leg Kiss Roll Expansion Transitions (26)

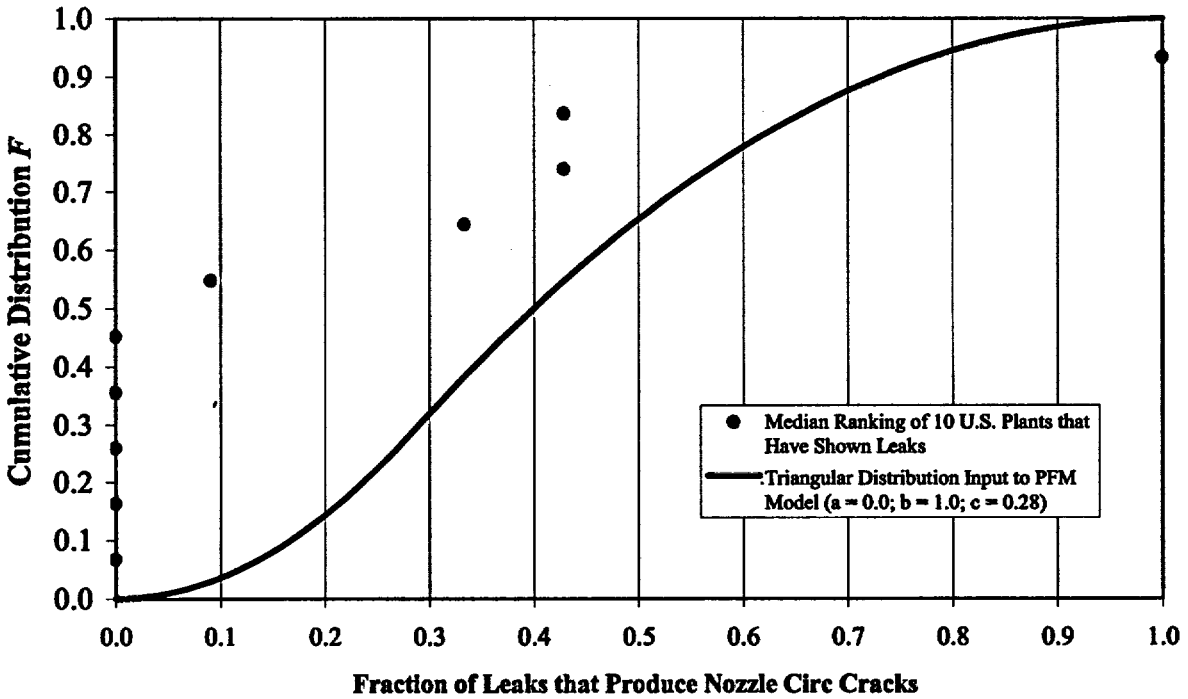

Figure 8-5
Conservative Selection of Input Distribution to Describe the Probability that Leakage (or Weld Cracking) Leads to Circumferential Tube Cracking Above or Near the Top of the J-Groove Weld (For Information Only)

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Figure 8-6
Median Ranking Cumulative Distribution for the 158 Individual Laboratory Crack Growth Rate Data Points in the Screened MRP-55 Database

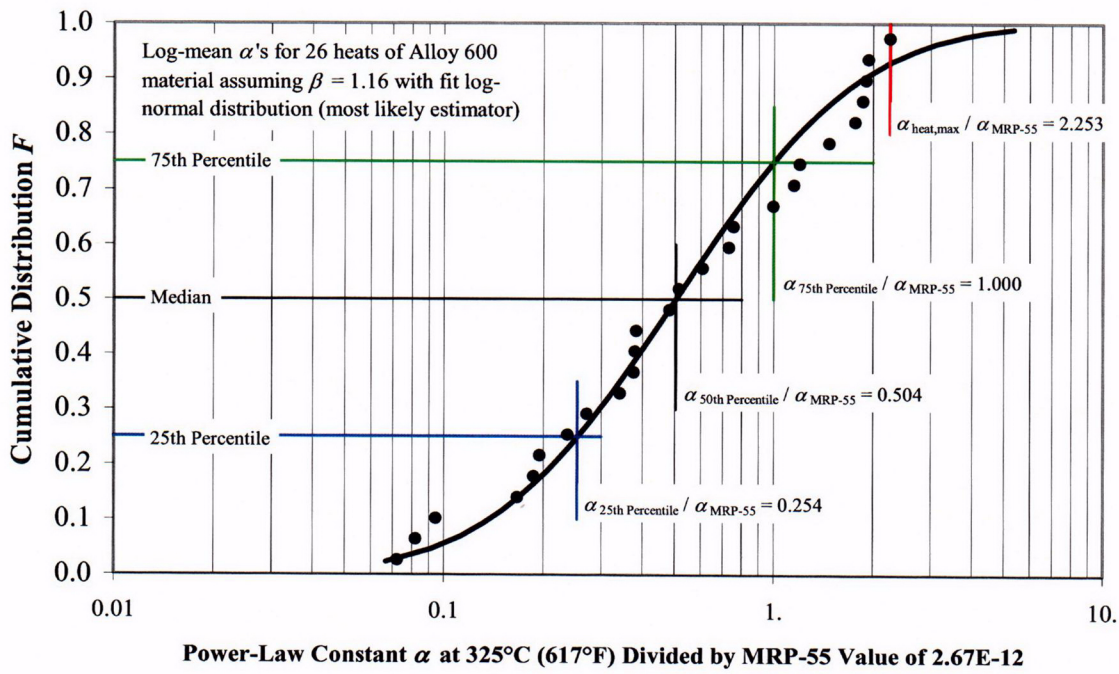

Figure 8-7
Log-Normal Distribution Fit on a Heat Basis to Laboratory Crack Growth Rate Data Presented in MRP-55

C08

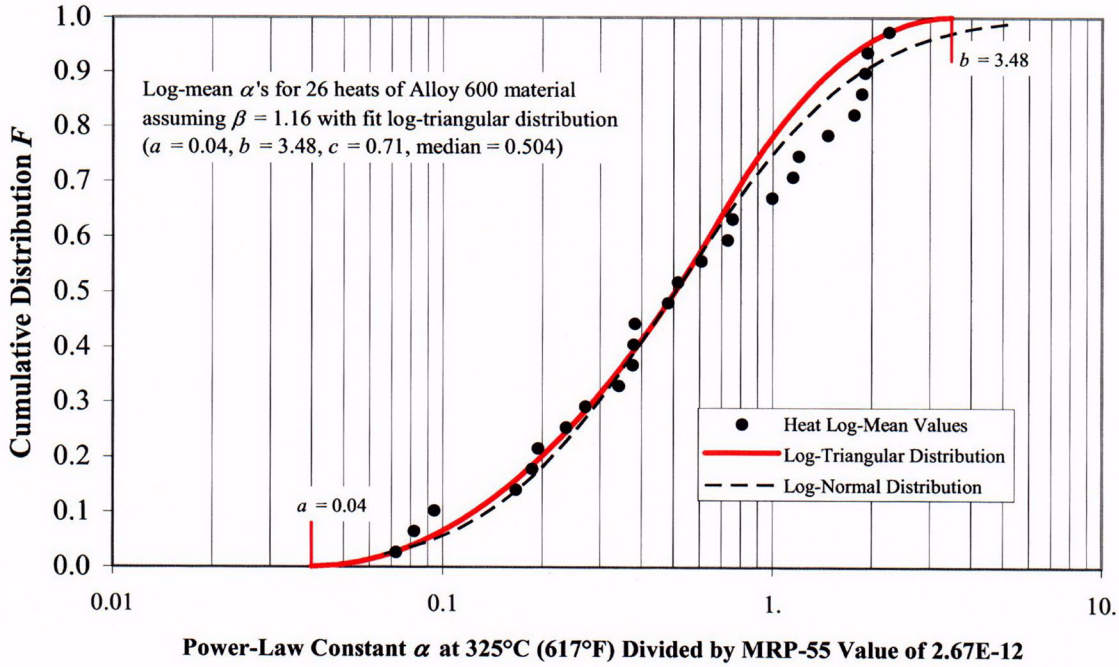

Figure 8-8
Log-Triangular Distribution Fit on a Heat Basis to Laboratory Crack Growth Rate Data Presented in MRP-55

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Figure 8-9
Selection of Distribution of Within-Heat Variability Multiplicative Factor Based on Screened MRP-55 Laboratory Database

C09

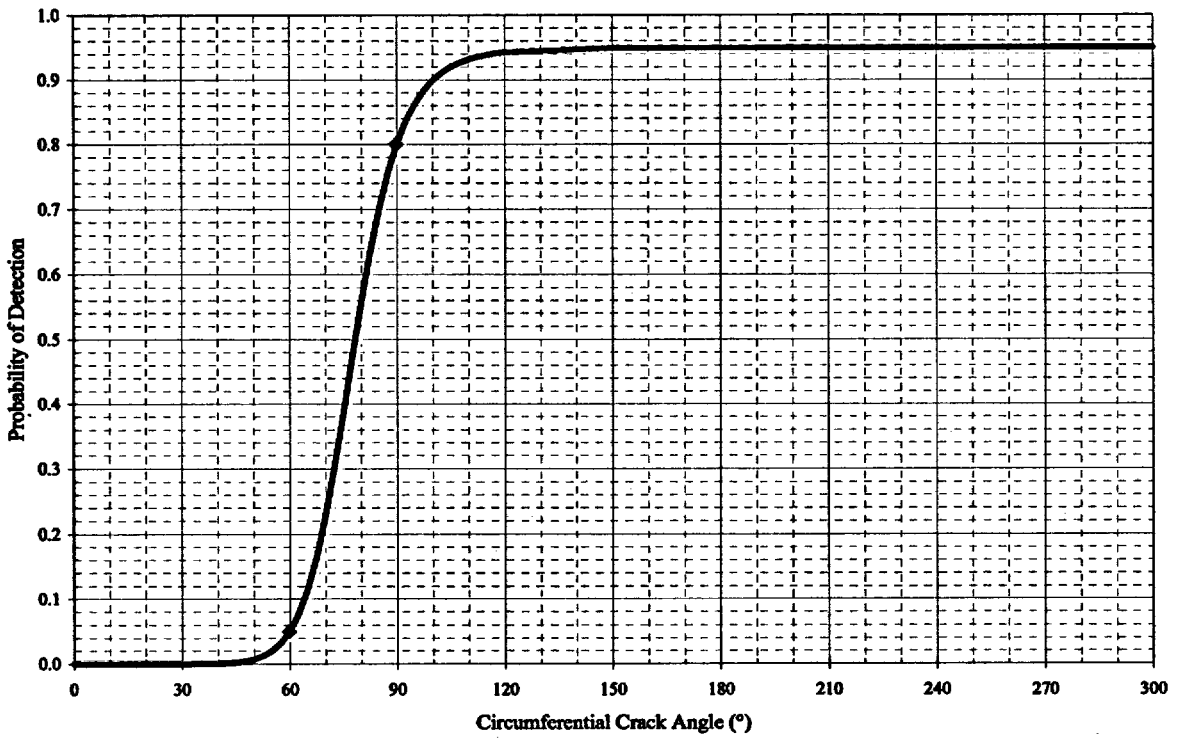

Figure 8-10
Assumed Probability of Detection for Above-Weld Circumferential Tube Cracks Given ECT Inspection of Tube ID

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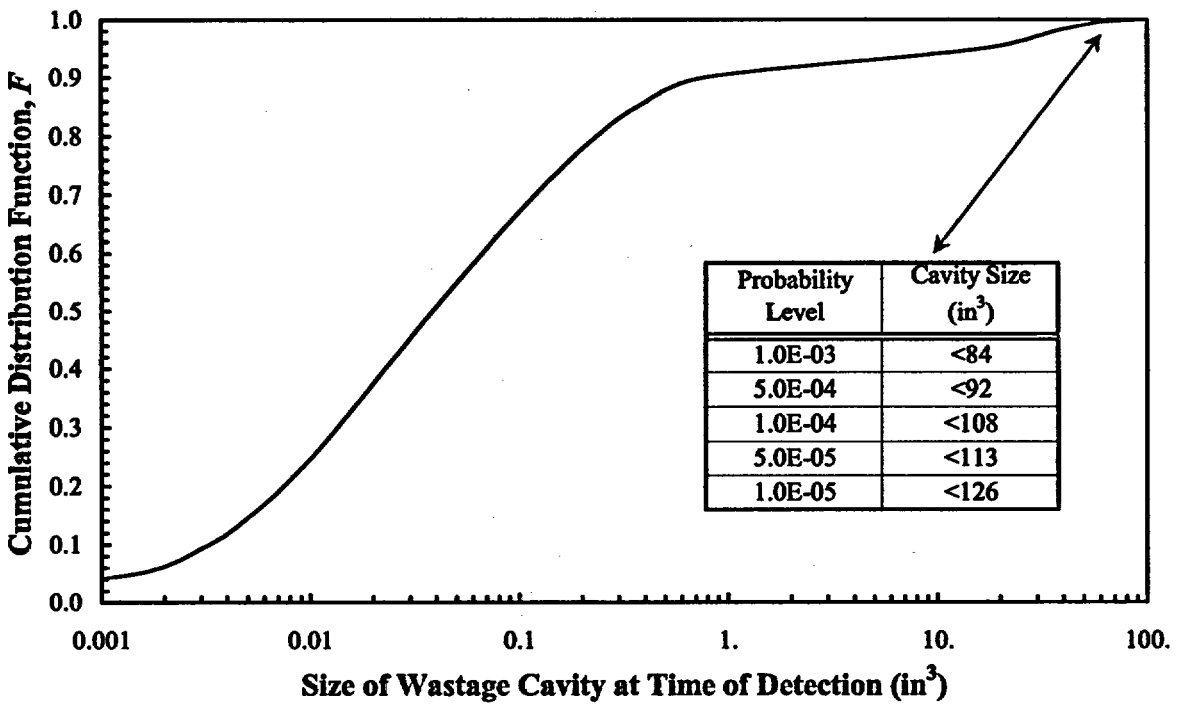
Figure 8-11
Results of the PFM Calculations Including Sensitivity Study (See Table 8-6 for Definitions of the Sensitivity Cases)



Probability Level	Cavity Size (in^3)
1.0E-03	<84
5.0E-04	<92
1.0E-04	<108
5.0E-05	<113
1.0E-05	<126

Figure 8-12
Results of the Probabilistic Wastage Calculation