PWR Materials Reliability Program Interim Alloy 600 Safety Assessments for US PWR Plants (MRP-44NP)

Part 2: Reactor Vessel Top Head Penetrations

TP-1001491-NP, Part 2

Interim Report, May 2001

EPRI Project Manager
A. R. McIlree

EPRI • 3412 Hillview Avenue, Palo Alto, California 94304 • PO Box 10412, Palo Alto, California 94303 • USA
800.313.3774 • 650.855.2121 • askepri@epri.com • www.epri.com

DISCLAIMER OF WARRANTIES AND LIMITATION OF LIABILITIES

ORGANIZATION(S) THAT PREPARED THIS DOCUMENT

PWR Materials Reliability Program Alloy 600 Issues Task Group

CITATIONS

This report was produced through the cooperative effort of PWR utility, EPRI and NEI members of the MRP Alloy 600 Issues Task Group with support from the following contractors: Dominion Engineering, Inc., Framatome ANP, Structural Integrity Associates, and Westinghouse Electric Company LLC.

This report describes research sponsored by EPRI.

The report is a corporate document that should be cited in the literature in the following manner:

PWR Materials Reliability Program, Interim Alloy 600 Safety Assessments for US PWR Plants (MRP-44NP): Part 2: Reactor Vessel Top Head Penetrations, EPRI, Palo Alto, CA: 2001. TP-1001491-NP, Part 2.

REPORT SUMMARY

Background

Between November 2000 and March 2001 leaks were discovered from reactor vessel top head penetrations at Oconee 1 and 3 and Arkansas Nuclear One-1 (ANO-1). The leaks were discovered by visual inspections of the heads which showed small amounts of boric acid crystal deposits that were determined later to have come from the annulus between the nozzles and holes in the vessel head. Leaks occurred in one CRDM nozzle and five of eight thermocouple nozzles at Oconee 1, one CRDM nozzle at ANO-1, and nine CRDM nozzles at Oconee 3. The CRDM nozzle leaks were traced to predominantly axial PWSCC cracks initiating on the outside surface of the nozzle wall below the J-groove weld. The crack at Oconee 1 appeared to initiate on the surface of the J-groove weld. Two of the leaking nozzles at Oconee 3 had circumferential cracks propagating from the OD of the nozzle above the J-groove weld.

Objective

The objective of this report is to provide an interim safety assessment for primary water stress corrosion cracking (PWSCC) of Alloy 600 CRDM nozzles and related Alloy 182 J-groove welds in PWR plants. This work is an extension to the safety assessments submitted to the NRC in 1993/4.

Approach

This interim safety assessment addresses the background regarding leakage from CRDM nozzles and welds at Oconee 1, Oconee 3 and ANO-1, and provides: 1) a compilation of locations where Alloy 600 base materials and Alloy 182 weld materials are used on reactor vessel top heads, 2) a discussion of the safety assessment methodology, 3) the results of the interim safety assessments for the most important RPV head penetrations, 4) the basis for determining that there is no significant near-term impact on plant safety, and 5) near-term inspection recommendations. The safety assessment includes review of crack orientations and sizes, limiting flaw sizes, and the ability to detect leaks while there is still substantial remaining structural margin.

Results

This interim safety assessment demonstrates that axial PWSCC similar to that observed in the Oconee 1 and ANO-1 CRDM nozzles has no significant near-term impact on plant safety and is bounded by the previously submitted CRDM nozzle safety assessments (1993/4) [1–6].

This interim safety assessment also demonstrates that there is reasonable assurance that other PWR plants do not have circumferential cracking greater in extent than that permitted by the minimum structural margin. This conclusion is based on the following:

1. The Oconee and ANO-1 units are in the highest grouping of plants based on a comparison of effective time at temperature.

2. Leaks were discovered by visual inspections while there was significant structural margin remaining, and

3. Other plants in the highest grouping have performed comprehensive inspections and found no evidence of leakage.

EPRI Perspective

As a consequence of the Oconee 1 and 3 and ANO-1 CRDM nozzle leaks, and the reactor vessel outlet nozzle weld leak at VC Summer, the industry, acting though the EPRI PWR Materials Reliability Program, is providing safety assessments addressing Alloy 600 materials and Alloy 82/182 welds in PWR plants. This report, Part 2, provides results of the interim assessments for reactor vessel top head Alloy 600 nozzles and associated Alloy 182 J-groove welds. The interim assessments for Alloy 82/182 pipe butt welds was provided in Part 1 of this report in April 2001.

Keywords

Primary water stress corrosion cracking
PWSCC
Stress corrosion
Alloy 600
Alloy 82/182
CRDM nozzle
CEDM nozzle
J-groove weld

ABSTRACT

This interim safety assessment provides the integrated industry technical response to the issue of primary water stress corrosion cracking (PWSCC) of Alloy 600 reactor vessel head penetrations, and Alloy 182 J-groove welds. Emphasis is on evaluating the most important reactor vessel top head nozzle designs for each NSSS plant design. The interim report addresses the background regarding leakage from CRDM nozzles and welds at Oconee 1, Oconee 3 and ANO-1, provides a compilation of locations where Alloy 600 base materials and Alloy 182 weld materials are used on reactor vessel top heads, describes the safety assessment methodology and results for the most important locations, discusses the basis that there is no significant near-term impact on plant safety, and provides near-term inspection recommendations. Specifically:

- Section 1 describes the locations of Alloy 600 penetrations on reactor vessel top heads, the recent experience that has led to increased attention to these penetrations, and the industry response to this recent experience.
- Section 2 provides a brief chronological summary of events related to PWSCC and leaks in reactor vessel top head Alloy 600 nozzles and associated Alloy 182 J-groove welds.
- Section 3 provides a summary of reactor vessel head penetration designs, selection of the penetrations to be evaluated in this interim safety assessment, and an evaluation of the fits of these penetrations into the vessel head relative to the ability to detect leaks.
- Section 4 is a review of the original CRDM nozzle PWSCC assessment model developed in 1998, a supplemental approach to comparing plants on a time and head temperature basis, and a review of recently completed inspections of CRDM nozzles.
- Section 5 is an analysis of through-wall circumferential cracks above the J-groove weld to determine the margin that existed in the Oconee 3 CRDM nozzles at the time that the cracks were detected. This analysis also addresses through-wall circumferential cracks above the J-groove weld in CRDM nozzles in other plants and in CEDM and ICI nozzles, which have different dimensions.
- Section 6 provides the basis that there is no significant near-term impact on plant operation in the presence of potential CRDM nozzle PWSCC.
- Section 7 summarizes current recommendations, and describes near-term actions.

CONTENTS

LIST OF FIGURES

LIST OF TABLES

1
INTRODUCTION

This section describes the locations of Alloy 600 penetrations on reactor vessel top heads, the recent experience that has led to increased attention to these penetrations, and the industry response to this recent experience.

1.1 Purpose and Scope

The purpose of this report is to provide an interim safety assessment for primary water stress corrosion cracking (PWSCC) of Alloy 600 CRDM nozzles and related Alloy 182 J-groove welds in PWR plants. This work is an extension to the safety assessments submitted to the NRC in 1993/4 [1–6]. PWR plants will use this assessment to help determine an appropriate course of action in response to this issue.

1.2 Overview of Recent Vessel Top Head Penetration Leaks

Leaks were discovered from eleven CRDM nozzles in three PWR plants between November 2000 and March 2001[1]—one at Oconee 1, nine at Oconee 3, and one at Arkansas Nuclear One – Unit 1 (ANO-1). In addition, five of the eight smaller diameter (1.375" OD) thermocouple nozzles at Oconee 1 were discovered to have leaks. All of these leaks were first identified during visual inspections of the top surface of the vessel heads for boric acid crystal deposits. In all cases the quantity of boric acid crystals was small (<1 in^3) at each nozzle location.

Destructive examinations of several specimens from Oconee 1 and 3 showed that these leaks were the result of primary water stress corrosion cracking (PWSCC). At Oconee 1, the leak was from an axial/radial crack that initiated in the weld metal and then progressed through the J-groove weld and part-depth into the nozzle wall from the outside surface. At Oconee 3 the leaks were from cracks that initiated on the outside surface of the Alloy 600 base metal below the J-groove weld at the toe of the weld. Two of the Oconee 3 nozzles had deep circumferential cracks about 165° around the nozzle following the contour at the top of the J-groove weld. At ANO-1, the leak was from a part-depth axially oriented crack on the outside surface of the nozzle. This crack extended 1.3 inches above the J-groove weld. A short distance below the J-groove weld, the crack developed a "Y" branch with two short circumferentially oriented legs.

[1] Four leaking nozzles were reported at Oconee 2 on April 29, 2001. The root cause analysis of the Oconee 2 leaks had not been completed at the time this report was issued.

1.3 Alloy 600 Reactor Vessel Top Head Penetrations

Reactor vessel top heads in PWR plants have a number of penetrations that are used for various purposes including CRDM[2] nozzles, instrument nozzles, head vent nozzles and thermocouple nozzles. Figure 1-1 shows a typical reactor vessel head arrangement for a plant designed by Babcock & Wilcox (B&W). This interim safety assessment is focused on CRDM nozzles in B&W and Westinghouse designed plants and CEDM and ICI nozzles in Combustion Engineering designed plants as shown in Figure 1-2. These nozzles are installed into holes in the vessel head, typically with an interference fit, and are welded to the inside surface of the head by partial penetration J-groove welds. Section 3 and Appendix A describe head arrangements encountered in domestic PWR plants.

Content Deleted –
EPRI Proprietary Information

Figure 1-1
Typical Reactor Vessel Head – Oconee 1 (Babcock & Wilcox Design)

[2] Throughout the main body of this report, the term Control Rod Drive Mechanism (CRDM) nozzle is used as a generic description for control rod drive mechanism (CRDM) nozzles in Babcock & Wilcox and Westinghouse-designed plants as well as the Control Element Drive Mechanism (CEDM) nozzles and Incore Instrument (ICI) nozzles in Combustion Engineering-designed plants. The three types of nozzles are compared in Figure 1-2.

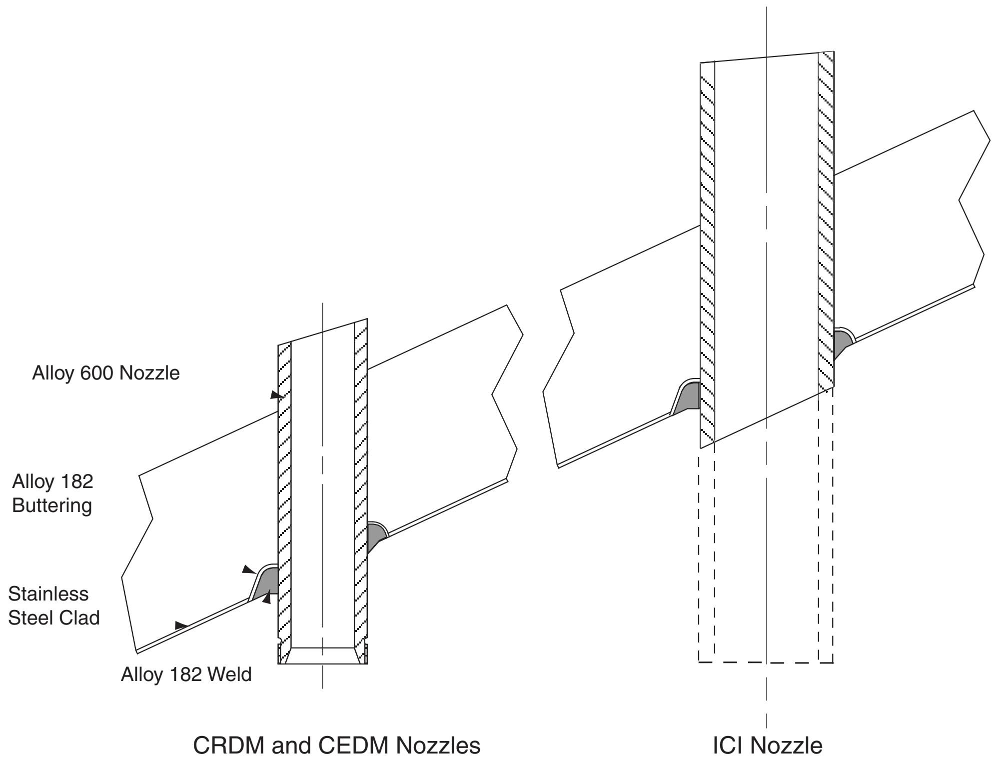

Figure 1-2
CRDM, CEDM and ICI Nozzles Evaluated in Interim Safety Assessment

1.4 Safety Assessment

Safety assessments for CRDM nozzles were submitted to the USNRC by the PWR NSSS Owners Groups in 1993/4 in response to the Bugey 3 CRDM nozzle leak [1–6]. The potential for PWSCC of nozzles in individual plants was addressed in the industry response to USNRC Generic Letter 97-01 [7].

This document provides an interim supplement to previous reports by addressing issues raised by the recent Oconee and ANO-1 experience, by the letter from the USNRC dated April 17, 2001 [8], and by the USNRC Information Notice 2001-05 issued April 30, 2001 [9].

This page is intentionally left blank.

2 BACKGROUND

This section provides a brief chronological summary of events related to PWSCC and leaks in reactor vessel top head Alloy 600 nozzles and associated Alloy 182 J-groove welds. Significant additional background information regarding cracking in the nozzle base material is provided in EPRI TR-103696, *PWSCC of Alloy 600 Materials in PWR Primary System Penetrations* [10] and NUREG/CR-6245, *Assessment of Pressurized Water Reactor Control Rod Drive Mechanism Nozzle Cracking* [11].

2.1 CRDM Nozzle Leak at Bugey 3 (1991)

During the ten-year outage primary system hydrostatic test at Bugey Unit 3 in France, a small amount of leakage [<1 liter/hr (0.004 gpm)] was discovered on the outside surface of the reactor vessel head. Investigation showed that the leak was from a through-wall crack in an outer row CRDM nozzle that had initiated at the nozzle inside surface. Subsequent inspections showed the presence of several part-depth axial cracks on the inside surface of the leaking nozzle near the elevation of the J-groove weld. Eddy current inspections of the other 65 CRDM nozzles in Bugey 3 showed part-depth axial cracks in one other outer row nozzle. Failure analysis confirmed that the crack which caused the leak was PWSCC and that susceptible material microstructure, stress concentration at a counterbore on the nozzle inside surface, high hardness of the cold worked machined surface, and high residual stresses induced in the nozzle during welding were significant contributing factors. Laboratory examinations associated with the Bugey 3 leak showed several features in addition to the axial cracks as shown in Figures 2-1.

Significant features in the leaking Bugey 3 CRDM nozzle include:

Content Deleted –
EPRI Proprietary Information

There was no evidence that any of the above conditions represented an immediate safety problem.

Content Deleted –
EPRI Proprietary Information

Figure 2-1
Locations of Cracks in Bugey 3 CRDM Nozzle 54 [10]
(cross section through leaking crack)

2.2 Axial CRDM Nozzle Cracks at Other PWR Plants (1991–Summer 2000)

Subsequent to the report of PWSCC in CRDM nozzles at Bugey 3, PWSCC of Alloy 600 base metal has been discovered in other PWR vessel heads worldwide. Until the recent leaks at Oconee 1, Oconee 3 and ANO-1, all of the cracks since Bugey 3 were discovered by eddy current and ultrasonic examinations from inside the nozzles using robotic manipulators. Table 2-1 is the most recent summary of the industry status regarding reactor vessel top head penetration PWSCC. These data [12] show that about 6.5% of all inspected nozzles in EDF plants have contained cracks and about 1.25% of all inspected nozzles in other plants worldwide have contained cracks greater than the minimum measurable depth of about 2 mm (0.08 inch) [13].

Table 2-1
Summary of Worldwide CRDM Nozzle PWSCC Experience as of February 2000 [12]

Content Deleted –
EPRI Proprietary Information

An overview of worldwide experience through the summer of 2000 shows the following:

Content Deleted –
EPRI Proprietary Information

- Eddy current inspections have been performed of the inside surface of CRDM nozzles at six plants in the US: Point Beach 1, Oconee 2, DC Cook 2, North Anna 1, Millstone 2 and Ginna. The major results of these inspections are as follows:
  - The most significant crack was 6.8 mm (0.27 inch) deep (43% through-wall) at DC Cook 2. This crack was partially ground out from the inside of the nozzle and weld repaired leaving a portion of the crack in place.
  - Millstone 2, Oconee 2 and Ginna detected shallow "craze type cracks" which are groups of shallow, predominantly axial, cracks less than the 2 mm (0.08 inch) ultrasonic depth sizing limit.
  - Palisades performed eddy current inspections of the incore instrument nozzles at the periphery of the vessel head, with no indications reported.

2.3 Initial Safety Assessments (1993/4)

All three NSSS Owners Groups submitted safety assessments to the USNRC in 1993/4 in response to the Bugey 3 leak [1–6]. The analyses demonstrated that CRDM nozzles are capable of accommodating long through-wall axial flaws, including the leakage resulting from these flaws. The analyses also demonstrated that the CRDM nozzles are capable of accommodating significant circumferential flaws above the J-groove weld.

2.4 PWSCC Rankings in Response to GL 97-01 (1998)

In response to Generic Letter 97-01, the industry prepared a ranking of plants for reactor vessel head nozzle PWSCC initiating on the ID surface of the Alloy 600 base metal [15]. The industry rankings were based on probabilistic crack initiation and growth models including factors such as operating time, head temperature, nozzle material strength and component geometry. The predictions were benchmarked against available inspection data. The final rankings were compiled into a histogram as shown in Figure 2-2. This histogram was submitted to the NRC in December 1998 [15].

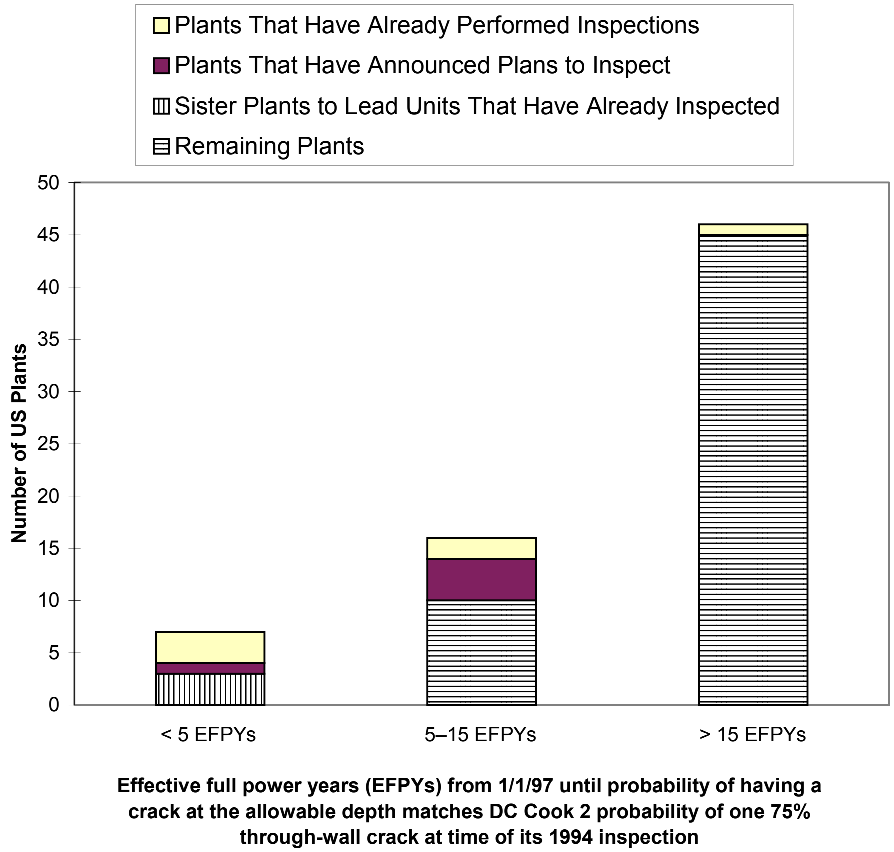

Figure 2-2
Industry Ranking Histogram for ID-Initiated Nozzle Base Metal Cracking

There are seven plants in the first group, including all three Oconee units. Utilities in the US have used the histogram to help determine when to perform inspections of the inside surfaces of nozzles for the type of cracking that has been observed in other plants worldwide.

Because the three Oconee units were recognized as being among the lead units, Duke Power performed a complete eddy current inspection of the inside surface of all Oconee 2 CRDM nozzles[3] from under the vessel head in 1994. Selected nozzles with shallow axial indications on the inside surface were reinspected in 1996 and again in 1999. No crack growth was detected. Inspections of Oconee 1 and 3 nozzles from under the head were deferred based on the findings from the Oconee 2 inspections. In addition to these inspections, Duke Power improved access to the top of the Oconee heads for visual inspections and cleaned the top surfaces of all three heads to remove boric acid crystal deposits left on the head from previous CRDM nozzle flange gasket leaks.

[3] Oconee 2 nozzles were ranked as slightly more susceptible than Oconee 1 and 3 nozzles due primarily to higher reported material yield strength.

In summary, the industry developed a histogram ranking model in 1998 which utilities have used to develop inspection plans with the focus being directed towards the highest ranking plants. These plans were communicated to the NRC in the NEI/MRP response to Generic Letter 97-01 [15].

2.5 CRDM Nozzle Leaks at Oconee 1, Oconee 3 and ANO-1 (2000–2001)

In December 2000, a visual inspection of the top surface of the Oconee 1 reactor vessel head indicated that at least five of the eight thermocouple nozzles, and one of the control rod drive mechanism (CRDM) nozzles, had developed small leaks similar to that shown in Figure 2-3.

Figure 2-3
Leaking CRDM Nozzle at Oconee 3

The CRDM and thermocouple nozzle leaks at Oconee 1 were evidenced by small quantities (less than 1 in^3 total volume) of boric acid crystal deposits at the locations where the nozzles penetrated the holes in the vessel head. Eddy current and ultrasonic inspections of the insides of the nozzles showed through-wall axial cracks in all eight thermocouple nozzles. Metallurgical examinations of samples showed the cracks to be PWSCC. The thermocouple nozzles were removed and the holes in the head plugged.

Inspections and tests of the leaking Oconee 1 CRDM nozzle showed an axial/radial PWSCC crack that appeared to initiate in the J-groove weld that attached the nozzle to the inside of the vessel head. This crack grew in a predominantly axial/radial direction through the J-groove weld and part-depth into the Alloy 600 CRDM nozzle base material from the outside surface. This was different from previous industry experience which showed that almost all flaws initiated on the inside surface of the nozzles. The crack in the nozzle and weld was removed and the nozzle was weld repaired using a temper bead process. The crack in the weld arrested when it reached the low-alloy steel vessel head material.

In February 2001, a visual inspection of the top surface of the Oconee 3 reactor vessel head indicated that nine CRDM nozzles had developed small leaks. Eddy current, ultrasonic, and liquid penetrant inspections of the leaking nozzles confirmed the presence of through- and partial through-wall axial cracks, predominantly originating on the outside surface of the nozzles below the J-groove weld. These inspections also showed several partial-depth circumferential cracks located on the outside surface of the nozzles below the welds.

As part of the repair effort, deep circumferential cracks were discovered on the outside surface of two of the leaking nozzles above the J-groove welds. These deep circumferential cracks followed the weld contour and extended up to about 165° around the nozzle. Duke Power provided further details regarding the Oconee 1 and 3 leaks to the NRC during a meeting on April 12, 2001 [16].

In addition to the leaking CRDM nozzles, nondestructive examinations were performed on 17 non-leaking Oconee 1 CRDM nozzles and 9 non-leaking Oconee 3 CRDM nozzles to assess the extent of the condition of nozzles in the head. No significant cracking was found in any of these non-leaking nozzles.

Content Deleted –
EPRI Proprietary Information

In summary, recent experience at Oconee and ANO-1 has shown several new characteristics regarding CRDM nozzle PWSCC. These include:

- The first leaking nozzles since Bugey 3 in 1991
- Detection of leakage by boric acid crystal deposits near nozzles (leaks at Bugey 3 were detected during a cold hydrostatic test)
- The first crack reported to have initiated in a CRDM nozzle Alloy 182 J-groove weld
- Cracks initiating predominantly on the outside of the nozzles below the weld as opposed to previous experience that showed that nearly all cracks initiate on the inside of the nozzles
- Circumferential cracks propagating inward from the outside of two nozzles above the J-groove weld

This page is intentionally left blank.

3
REACTOR VESSEL CLOSURE HEAD PENETRATION CONFIGURATIONS, FITS AND LEAKAGE DETECTABILITY

This section provides a summary of reactor vessel head penetration designs, selection of the penetrations to be evaluated in this interim safety assessment, and an evaluation of the fits of these penetrations into the vessel head relative to the ability to detect leaks.

3.1 Head Penetration Configurations

A key factor in the interim safety assessment is that through-wall PWSCC cracks in either the nozzle base metal or the J-groove weld are expected to produce leakage into the annulus region between the RPV head and the nozzle, and that such leakage can be detected by visual inspection of the top surface of the head. To that end, the industry has identified the locations and types of Alloy 600 reactor vessel head penetrations and associated Alloy 182 welds. The types of penetrations used are listed in Table 3-1. Details of the head and nozzle configurations are provided in Appendix A.

3.2 Selection of Penetrations for Evaluation in Interim Safety Assessment

The interim safety assessment focuses on the nozzles associated with CRDM, CEDM and ICI penetrations, since these are considered to be most important to safety based on size, temperature and function.

3.3 Fits of Selected Nozzles Into Vessel Head

The majority of CRDM nozzles are installed into the vessel heads with an interference fit at room temperature. The ranges of diametral interference fits are listed in Table 3-2. As shown, the plants fall into three categories:

Content Deleted –
EPRI Proprietary Information

Table 3-1
Types of Reactor Vessel Head Penetrations With J-Groove Welds

Head Temp	Type of Penetration	Westinghouse Design Plants	Combustion Engineering Design Plants	Babcock & Wilcox Design Plants
Hot Head Plants[1]	*Penetrations with Shrink Fit*			
	Control Rod Drive Mechanism (CRDM)	Some Plants	None	All Plants
	Control Element Drive Mechanism (CEDM)	None	All Plants	None
	Incore Instrument (ICI)	None	Most Plants	None
	Penetrations without Shrink Fit			
	Control Element Drive Mechanism (CEDM)	None	Two Plants	None
	Head Vent (J-groove weld)	Most Plants	All Plants	None
	Thermocouple	None	None	One Plant[2]
Cold Head Plants[1]	*Penetrations with Shrink Fit*			
	CRDM	Some Plants	None	None
	Auxiliary Head Adapters	Some Plants	None	None
	Penetrations without Shrink Fit			
	Head Vent (J-groove weld)	Most Plants	None	None
	De-Gas Line	Two Plants	None	None
	Internals Support Housing (full penetration weld)	Two Plants	None	None

[1] Hot head plants have a head temperature closer to the hot-leg temperature than the cold-leg temperature, and cold head plants have a head temperature near or at the cold-leg temperature.
[2] Thermocouple nozzles were plugged at Oconee 1 leaving one plant with thermocouple nozzles.

Table 3-2
Range of Specified CRDM and CEDM Nozzle Diametral Fits Into Vessel Heads

Content Deleted –
EPRI Proprietary Information

3.4 Ability to Detect Leakage Considering Interference Fits

The following is an assessment of the ability to detect leaks considering the source of the leakage from tight PWSCC cracks and leakage through the annulus between the nozzle and vessel shell. This work demonstrates that there is reasonable assurance that leaks from vessel top head penetrations can be detected. Recommendations regarding visual inspections to ensure that leakage is discovered are provided in Section 7.

3.4.1 Leakage Past PWSCC Cracks

Through-wall PWSCC cracks will normally result in leakage through the crack. This is supported as follows:

Content Deleted –
EPRI Proprietary Information

3.4.2 Leakage Through Nozzle to Vessel Shell Annulus

Leakage through PWSCC cracks will result in some leakage through the annulus between the nozzle and hole in the vessel shell. This is supported as follows:

Content Deleted –
EPRI Proprietary Information

Content Deleted –
EPRI Proprietary Information

3.5 Conclusions Regarding Leak Detection

Through-wall PWSCC cracks in either the nozzle base metal or the J-groove weld are expected to produce leakage into the annulus region between the RPV head and the nozzle. This leakage can be detected by visual inspection of the OD surface of the RPV head at the penetration location. The ability to detect small leaks should be enhanced based on insights gained from recent head penetration inspections (i.e., Oconee and ANO-1), and the inspection personnel should be sensitive to the small amounts of boric acid residue similar to that observed at Oconee and ANO-1. This position is supported by the following:

- Leaks caused by PWSCC in Alloy 600 CRDM nozzles, and numerous smaller diameter nozzles (without shrink fits) have been discovered by visual inspections for boric acid crystal deposits.
- Leaks were discovered at Oconee and ANO-1 by visual inspections of the top surface of the vessel head.
- The interference fits for Westinghouse- and Combustion Engineering-designed plants are not significantly different from those for B&W-designed plants.
- It is very difficult to develop a leak tight seal relying only upon a moderate interference fit between large section pressure vessel components. It is necessary to roll, hydraulically, or explosively expand nozzles into a shell to produce a leak free joint.

Guidance will be provided to utilities regarding inspection practices that will ensure that small leaks from PWSCC cracks will be detected.

Content Deleted –
EPRI Proprietary Information

Figure 3-1
Estimated Nozzle Interference Based on Measured Hole Diameter and Mean Nozzle Diameter for Two Typical Plants

This page is intentionally left blank.

4
TIME-AT-TEMPERATURE COMPARISONS AND PLANT INSPECTION STATUS

The following is a review of the original PWSCC histogram assessments developed in 1997 and 1998, a supplemental approach to comparing plants on a time and head temperature basis, and a review of recently completed inspections of nozzles from above and below the vessel heads.

4.1 Review of Original Histogram Predictions (1998)

The original assessment groups for inside surface initiated PWSCC (Figure 2-2) were developed in support of the industry response to Generic Letter 97-01 [15]. These assessment groups were based on probabilistic crack initiation and growth analyses including factors such as plant operating time, head temperature, nozzle material strength, nozzle material chemistry and processing parameters, and component geometry. Plants were grouped based on the predicted time for each to reach an equivalent probability of cracking as DC Cook 2 at the time Cook performed its original inspection and detected the only significant CRDM nozzle crack in the US prior to the recent Oconee and ANO-1 cracks. This original model, while focused on cracks developing from the inside of the penetrations, allowed the US nuclear industry to rank plants and establish a program of lead-plant, under-the-head robotic inspections for the detection of such cracks.

4.2 Time-Temperature Comparison

Because of the nature of the recently discovered cracks at Oconee 1 and 3 and ANO-1, the MRP has recognized that, while useful for managing ID cracking of the head penetrations, the models used for the original histogram were not established to predict outside diameter (OD) or weld cracking such as has recently been reported. Lacking specific information about the factors controlling the new forms of cracking, it was determined that the best way to initially focus the industry efforts was to concentrate on those factors that are well understood, i.e. operating time and operating temperature of the penetrations. Appendix B describes development of a simplified time and temperature comparison that was prepared by the MRP to assist in planning while the root cause evaluations are being completed and previous models are evaluated relative to recent experience. This simplified approach, reported in Table 4-1 and Figure 4-1, is similar to the histogram approach originally used for ID cracking, but groups plants according to the time (EFPY) required for each unit to reach an equivalent effective time at temperature as Oconee 3 at the time the above-weld circumferential cracks were discovered in February 2001.

The simplified comparison assumes that all plants are identical to Oconee 3, except for the operating time and head temperature. This simplified comparison again shows the Oconee units to be in the lead group of plants with the highest effective time at temperature. Also, ANO-1, which was in the "> 15 EFPY" category based on the original histogram for ID cracking, is now also in the group of plants with the highest time at temperature. Since the Oconee units are predicted to lead all other US PWRs on the basis of effective time at temperature, cracking in other units is expected to lag that at Oconee 3. However, to account for uncertainties in parameters other than time and head temperature, and the probabilistic nature of PWSCC, a margin of ten years is proposed for use when deciding the timing of head inspections, capable of detecting small amounts of leakage, at the other units.

4.3 Plant Inspection Status

In addition to showing a breakdown of plants based on effective time at temperature, Table 4-1 and Figure 4-1 also present information relative to the current industry inspection status. The seven B&W units are tabulated separately because they have a unique arrangement of the insulation on top of the head (see Figure 1-1) that allows the intersections of the nozzles and the top surface of the head to be visually inspected during each refueling outage. The inspection status of the Westinghouse- and Combustion Engineering-designed units is shown, indicating either under-the-insulation visual inspections (prior to or after the discovery of the leak at Oconee 1) or under-the-head nozzle ID eddy current inspections.

Content Deleted –
EPRI Proprietary Information

4.4 Histogram Details

Further information relative to the 25 units that are within 10 EFPY of Oconee 3 time-at-temperature equivalence are:

Content Deleted –
EPRI Proprietary Information

Content Deleted –
EPRI Proprietary Information

4.5 Summary

While detailed models have not been developed and benchmarked yet for the new cracking phenomena observed at Oconee and ANO-1, the MRP has developed a simplified approach to categorizing the US PWRs, and has developed interim recommendations for those units with near-term outages and relatively high effective times at temperature to perform visual inspections of their heads capable of detecting small amounts of leakage at their next refueling outage. Several of these units have already performed full or partial inspections with no reported signs of leakage, indicating that the cracking is not as severe at all units. Since the Oconee units lead the industry in effective time at temperature, and 10 EFPYs margin has been added to account for uncertainties when planning inspections, there is assurance that significant cracking at any of the US PWRs will be detected before there is any significant impact on plant safety.

Table 4-1
Summary of Plant Assessments Using Supplemental Time and Temperature Model and the Plant Inspection Status

Content Deleted –
EPRI Proprietary Information

Content Deleted –
EPRI Proprietary Information

Figure 4-1
Supplemental Time-Temperature Histogram for Addressing Head Nozzle Cracking

5 STRUCTURAL MARGIN FOR CIRCUMFERENTIAL CRACKS ABOVE J-GROOVE WELD

As previously reported in paragraph 2.3, safety analyses for axial and circumferential cracks in CRDM, CEDM and ICI nozzles were submitted to the NRC in 1993/4 [1-6]. In their review of the industry flaw acceptance criteria, the NRC stated that flaws more than 45° from the axial direction are considered to be circumferential flaws and that the criteria for acceptance of circumferential flaws would not be pre-approved [17].

The following is an analysis of through-wall circumferential cracks above the J-groove weld to determine the margin that existed in the Oconee 3 CRDM nozzles at the time that the cracks were detected. This analysis also addresses through-wall circumferential cracks above the J-groove weld in CRDM nozzles in other plants and in CEDM and ICI nozzles, which have different dimensions.

5.1 Required Ligament for Circumferential Cracks Above J-Groove Weld

The maximum extent of the two circumferential cracks at Oconee 3 (Nozzles 50 and 56) was about 165°. The remaining uncracked ligament was about 195°, or 54% of the nozzle circumference.

The minimum required ligament for structural integrity was calculated for the full range of CRDM, CEDM and ICI nozzle dimensions and minimum material strengths. Because of the tight fitting annulus and of the high ductility of the nozzle materials, bending loads on the nozzle at the top of the vessel head, including seismic moments, will not affect the required minimum ligament.

Therefore, the required ligament is that which will withstand three times the design pressure acting on the nozzle bore and the crack face at flow stress levels in the ligament. For large crack sizes the stress level may be calculated as the pressure force over the combined bore and crack face areas divided by the cross sectional area of the ligament (see equation [5.1]). For smaller size cracks the maximum permissible pressure is limited by the burst pressure in cracked tubes rather than axial stress in the remaining ligament.

$$P_{flow} = \sigma_{flow} \left[\frac{A_{wall}\left(1 - \frac{\theta}{360}\right)}{A_{bore} + A_{wall}\left(\frac{\theta}{360}\right)} \right] \qquad [5.1]$$

where P_{flow} = limit load pressure acting on the nozzle bore and crack face (ksi)
σ_{flow} = flow stress (ksi) = average of minimum yield and minimum ultimate tensile strengths at design temperature of 650°F
A_{bore} = cross-sectional area of nozzle bore in region above J-groove weld (in^2) $= \pi(ID)^2/4$
A_{wall} = cross-sectional area of nozzle in region above J-groove weld (in^2) $= \pi(OD)^2/4 - A_{bore}$
θ = circumferential angle of through-wall crack (deg)

Figure 5-1 shows the results of calculations for CRDM nozzles and the limiting CEDM and ICI nozzles. The maximum permissible crack size for limit load at three times the design pressure of 2500 psi varies from 179° for the limiting ICI nozzles, to 244° for the limiting CEDM nozzles, to 273° for CRDM nozzles.

Content Deleted –
EPRI Proprietary Information

Figure 5-1
Limit Load Pressure For Circumferential Flaw Angle and Limiting Nozzle Geometries

5.2 Structural Margin for Oconee 3 Cracks at Time of Detection

The maximum circumferential flaw above the J-groove weld in the Oconee 3 nozzles was 165°. This is significantly less than the calculated limit of 273° for a pressure of three times design pressure. Evaluated from a different perspective, the remaining ligament of 195° was 2.2 times the required ligament of 87°.

This page is intentionally left blank.

6 INTERIM SAFETY ASSESSMENT

This section provides the basis that there is no significant near-term impact on plant safety in the presence of potential CRDM nozzle PWSCC. The main points supporting this are:

Content Deleted –
EPRI Proprietary Information

- Leaks were found in the Oconee and ANO-1 vessel heads by routine visual inspections of the top surface of the vessel heads while the nozzles and welds were still well within required structural margins (see Section 5).

Content Deleted –
EPRI Proprietary Information

- Several other plants with long operating times and high head temperatures have already performed inspections of 1) the top surface of their vessel heads for leaks, or 2) the inside surfaces of the nozzles near the welds for cracks. In addition to the B&W units, top of the head inspections have been performed at eight plants subsequent to the leakage being discovered at Oconee 1. Individuals performing these inspections have been advised of the need to detect small amounts of leakage. There have been no significant findings in any of these inspections (see Section 4).
- The worst case of a CRDM nozzle ejection is an analyzed event in all plant FSARs (see Appendix C).

The three NSSS Owners Groups were asked to evaluate the need to address operator actions and training for scenarios involving rod ejection(s), small-, medium- and large-break LOCAs and rod-insertion failure(s). The results of this evaluation are provided in Appendix C, and can be summarized as follows:

- LOCA(s) resulting from head penetration failure(s) would be bounded by existing design basis analyses and therefore the core would remain covered by borated water which would provide adequate cooling. Core internals would remain in a coolable geometry, and the condition of the containment following the scenario(s) would not require implementation of the severe accident management guidelines (SAMGs).
- Existing EOPs provide guidance for the full range of LOCAs and include coverage for multiple events including reactivity excursions that might occur during the course of an accident. Existing guidelines provide adequate directions to mitigate the transient induced by one or more CRDM penetration failures.

- Existing EOPs also cover the reactivity insertion event during periods of highest rod worth, resulting in the same conclusion stated above.

7 MRP RECOMMENDATIONS AND FUTURE ACTIONS

Section 6 concludes that there is no significant near-term impact on safety for all operating PWR plants. However, the MRP recognizes that some enhancements may be beneficial and has made the following recommendations to operating PWR plants:

- All plants should continue with the regularly scheduled inspections of the top of the vessel head for boric acid deposits in accordance with licensee commitments made in response to Generic Letter 88-05 [18]. Furthermore, these inspections should be based on insights gained from head penetration inspections performed at Oconee and ANO-1.
  - This recommendation is consistent with the recommendation made by NEI in a letter to the Nuclear Strategic Issues Advisory Committee on April 26, 2001 [19].
- Plants ranked to be within 10 EFPYs of Oconee Unit 3 based on effective time at temperature (see the Figure 4-1 histogram) and having Fall 2001 outages should perform a visual inspection of the reactor vessel top head capable of detecting small amounts of leakage similar to that observed at Oconee and ANO-1.

The MRP will continue to proactively address this issue, including the following:

- Update inspection guidance and recommendations based on inspection results and evolution of inspection technology. The MRP plans to issue additional guidance addressing potential needs for the Spring and Fall 2002 refueling outages.
- Determine the capability and availability of improved inspection techniques, processes and tooling. Both domestic and international PWR operators, vendors and service supporters will be contacted to assess new information.
- Convene a panel of experts to assess available crack initiation and growth data and to develop recommendations regarding appropriate future laboratory tests.
- Conduct an MRP workshop on June 13 and 14, 2001, to advise the industry of the recent CRDM cracking events and to provide a forum for discussion of MRP recommendations. This workshop will provide useful input to licensees that will be performing head penetration inspections during the Fall 2001 or Spring 2002 outages.

This page is intentionally left blank.

8
REFERENCES

1. *Safety Evaluation for B&W-Design Reactor Vessel Head Control Rod Drive Mechanism Nozzle Cracking*, BAW-10190P (Proprietary), BAW-10190 (Non-Proprietary), B&W Nuclear Technologies, May and June 1993.

2. *External Circumferential Crack Growth Analysis for B&W Design Reactor Vessel Head Control Rod Drive Mechanism Nozzles*, BAW-10190P, Addendum 1 (Proprietary), December 1993 and BAW-10190, Rev. 1 (Non-proprietary), B&W Nuclear Technologies, January 1994.

3. *Safety Evaluation of the Potential for and Consequence of Reactor Vessel Head Penetration Alloy 600 ID Initiated Nozzle Cracking*, CEN-607, ABB Combustion Engineering Nuclear Operations, May 1993.

4. *Safety Evaluation of the Potential for and Consequence of Reactor Vessel Head Penetration Alloy 600 OD-Initiated Nozzle Cracking*, CEN-614, ABB Combustion Engineering Nuclear Operations, December 1993.

5. *Alloy 600 Reactor Vessel Head Adaptor Tube Cracking Safety Evaluation*, WCAP-13565, Rev. 1 (Proprietary and Non-Proprietary), Westinghouse Electric Corporation, February 1993.

6. *RV Closure Head Penetration Alloy 600 PWSCC (Phase 2)*, WCAP-13525, Addendum 1 (Proprietary), WCAP-13603, Addendum 1 (Non-Proprietary), Westinghouse Electric Corporation, December 1993.

7. *NRC Generic Letter 97-01: Degradation of Control Rod Drive Mechanism Nozzle and Other Vessel Closure Head Penetrations*, US Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, April 1, 1997.

8. "Issues to be Addressed in a Generic Justification for Continued Operation of PWRs," Letter from B.W. Sheron (US Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation) to A. Marion (Nuclear Energy Institute), dated April 17, 2001.

9. *NRC Information Notice 2001-05: Through-wall Circumferential Cracking of Reactor Pressure Vessel Head Control Rod Drive Mechanism Penetration Nozzles at Oconee Nuclear Station, Unit 3*, US Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, April 30, 2001.

10. *PWSCC of Alloy 600 Materials in PWR Primary System Penetrations*, EPRI, Palo Alto, CA: 1994. TR-103696.

11. *Assessment of Pressurized Water Reactor Control Rod Drive Mechanism Nozzle Cracking*, NUREG/CR-6245, US Nuclear Regulatory Commission, Office for Analysis and Evaluation of Operational Data, October 1994.

12. W. Bamford, D. Boyle and J. Duran. "An Update on World Inspection Experience for RV Head Penetrations," In *Proceedings: 2000 EPRI Workshop on PWSCC of Alloy 600 in PWRs (PWRMRP-27)*, EPRI, Palo Alto, CA: 2000. 1000873.

13. *Demonstration of Inspection Technology for Alloy 600 CRDM Head Penetrations*, EPRI, Palo Alto, CA: 1996. TR-106260.

14. Email from EDF to E.S. Hunt (Dominion Engineering, Inc.), dated April 9, 2001.

15. "Responses to NRC Requests for Additional Information on Generic Letter 97-01," Letter with Enclosures from D.J. Modeen (Nuclear Energy Institute) to G.C. Lainas (US Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation), dated December 11, 1998.

16. M.R. Robinson (Duke Power). "Oconee Unit 1 & Unit 3 Reactor Vessel Head Leakage: Cracking of RV Head Penetrations Due to Primary Water Stress Corrosion Cracking (PWSCC)," presented at NRC Meeting with the Nuclear Energy Institute (NEI) and the Electric Power Research Institute (EPRI) Materials Reliability Program (MRP) Regarding Control Rod Drive Mechanism (CRDM) Nozzle Circumferential Cracking Issues, Rockville, Maryland, April 12, 2001.

17. "Safety Evaluation for Potential Reactor Vessel Head Adaptor Tube Cracking," Letter from W.T. Russell (US Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation) to W. Rasin (Nuclear Management and Resources Council; now Nuclear Energy Institute), dated November 19, 1993.

18. *NRC Generic Letter 88-05: Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants*, US Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, March 17, 1988.

19. "Reactor Vessel Head Penetration Inspections," Letter from R.E. Beedle (Nuclear Energy Institute) to Nuclear Strategic Issues Advisory Committee, dated April 26, 2001.

20. *CHECWORKS™ PWR Vessel and Internals Application: RPV Head Nozzle Module, Version 1.1, User Guide*, EPRI, Palo Alto, CA: 1999. TR-103198-P8.

21. J.A. Gorman, R.A. Ogren, and J.P.N. Paine, "Correlation of Temperature with Steam Generator Tube Corrosion Experience," *Fifth International Symposium on Environmental Degradation of Materials in Nuclear Power Systems—Water Reactors*, E. Simonen and D. Cubicciotti, eds., American Nuclear Society, LaGrange Park, Illinois, 1992.

A
HEAD PENETRATION CONFIGURATIONS

This appendix identifies the locations and types of Alloy 600 reactor vessel head penetrations and the associated Alloy 182 J-groove weld configurations. This information was extracted from EPRI TR-103198-P8 [20].

A.1 Penetration Locations on Reactor Vessel Heads

Figures A-1 through A-8 show the locations of Alloy 600 penetrations on PWR reactor vessel heads in the US. The types of penetrations used and locations on the head vary by plant design and plant size.

A.2 Penetration Designs

Figures A-9 through A-16 show the different types of penetration designs. Most are attached to the underside of the vessel head by Alloy 182 J-groove welds. However, some nozzles are welded to stub tubes or weld built-up pads on the top surface of the vessel heads.

The CRDM, CEDM, ICI, J-groove type auxiliary head adapter and de-gas line nozzles are installed into the vessel head with a shrink (interference) fit prior to welding the nozzle to the underside of the head. The head vent nozzles are installed with a clearance fit.

Content Deleted –
EPRI Proprietary Information

Figure A-1
Penetration Locations—Westinghouse 2-Loop Plants

Content Deleted –
EPRI Proprietary Information

Figure A-2
Penetration Locations—Westinghouse 3-Loop Plants

Content Deleted –
EPRI Proprietary Information

Figure A-3
Penetration Locations—Westinghouse 4-Loop Plants Without Adapters

Content Deleted –
EPRI Proprietary Information

Figure A-4
Penetration Locations—Westinghouse 4-Loop Plants With Adapters

Content Deleted –
EPRI Proprietary Information

Figure A-5
Penetration Locations—Combustion Engineering Plants

Content Deleted –
EPRI Proprietary Information

Figure A-6
Penetration Locations—Combustion Engineering Plants

Content Deleted –
EPRI Proprietary Information

Figure A-7
Penetration Locations—Combustion Engineering Plants

Content Deleted –
EPRI Proprietary Information

Figure A-8
Penetration Locations—B&W Plants

Content Deleted –
EPRI Proprietary Information

Figure A-9
Penetration Designs—Control Rod Drive Mechanism (CRDM) Nozzles

Content Deleted –
EPRI Proprietary Information

Figure A-10
Penetration Designs—Control Element Drive Mechanism (CEDM) Nozzles

Content Deleted –
EPRI Proprietary Information

Figure A-11
Penetration Designs—Incore Instrument (ICI) Nozzles

Content Deleted –
EPRI Proprietary Information

Figure A-12
Penetration Designs—Head Vent Nozzles

**Content Deleted –
EPRI Proprietary Information**

**Figure A-13
Penetration Designs—Thermocouple Nozzles (Small Diameter ~1 inch)**

Content Deleted –
EPRI Proprietary Information

Figure A-14
Penetration Designs—Internals Support Housing Nozzles

Content Deleted –
EPRI Proprietary Information

Figure A-15
Penetration Designs—Auxiliary Head Adapter Nozzles

Content Deleted –
EPRI Proprietary Information

Figure A-16
Penetration Designs—De-Gas Line Nozzles

This page is intentionally left blank.

B TIME-TEMPERATURE COMPARISONS

Because OD-initiated nozzle cracking was found to play a large role at Oconee 3, the previous MRP histogram in Figure 2-2—which is based on probabilistic models of ID-initiated cracking—has been augmented by a simple ranking based only on *operating time* and *head temperature*. Other factors such as stress, microstructure, surface cold work, and head fabrication practices are not addressed by the simple ranking described here. The time-temperature histogram addresses OD-initiated base metal cracking and weld metal cracking, while the previous histogram remains valid for ID-initiated base metal cracking.

B.1 Simplified Time-Temperature Model

Since stress corrosion cracking (SCC) of Alloy 600 nozzle material and Alloy 182 weld metal is sensitive to temperature, the simple ranking model adjusts the operating time for each plant using its head temperature history and an activation energy appropriate to SCC initiation. (Initiation is a more important factor than crack growth for assessing plants since the time to crack initiation is longer than the time for crack growth.) The ranking for a particular plant is based on the number of effective full power years (EFPYs) of operation required for that plant to reach the same number of EFPYs as Oconee 3, normalized for any differences in head temperature. For example, a plant with a histogram value of 10 EFPYs would reach an equivalent degradation time as Oconee 3 after 10 EFPYs of additional operation. The histogram chart (see Table 4-1 and Figure 4-1) was constructed by assigning each of the 69 plants to one of eight bins as described below.

B.2 Total Effective Full Power Years

The first step in the simplified plant ranking methodology is to assign an operating time to each plant. Effective full power years (EFPYs) was selected as the measure of operating time because it reflects the effect of lower head temperatures during startups, shutdowns, and periods of reduced power operation. In order to respond to the Oconee 3 issue in a timely fashion, the EFPYs for each plant was calculated using the most recent power production table listed in *Nucleonics Week*:

Content Deleted –
EPRI Proprietary Information

Content Deleted –
EPRI Proprietary Information

Note that the use of EFPYs based on electrical power is a sufficiently accurate approximation of EFPYs based on reactor thermal power for the purpose of assigning plants to the histogram bins. In addition, note that some plants provided small corrections to the EFPY data calculated using equation [B.1], and this information has been incorporated into the histogram.

B.3 Head Temperature History

The second step in the time-at-temperature ranking methodology is to identify the current reactor closure head temperature at 100% power and any periods of past operation at significantly different temperatures. The three NSSS vendors previously determined the head temperatures as part of their work for the PWR NSSS Owners Groups, and the head temperature histories for all plants were compiled as part of the response to NRC Generic Letter (GL) 97-01. Because of thermal-hydraulic differences between reactor designs, some plants operate with a head temperature close to or somewhat below the hot leg temperature, while some plants are designed to have a small amount of internals bypass flow and operate with a head temperature closer to the cold leg temperature. Most but not all plants listed their head temperature history in the initial responses to GL 97-01. Because the head temperature changes with any changes in hot leg (or cold leg) temperature or bypass flow, the MRP surveyed all the plants in 2000 to determine whether there were any past or planned future changes in head temperature due to changes in primary temperatures or bypass flow. The temperature changes reported in the survey were considered in the current calculations.

For plants that have had prior head temperature changes, the operating time accumulated at the current head temperature through the end of February 2001 was calculated:

Content Deleted –
EPRI Proprietary Information

B.4 Temperature-Adjusted Degradation Time

The third step in the time-at-temperature calculations is to calculate the operating time normalized to a reference temperature of 600°F. The standard Arrhenius activation energy dependence on temperature is applied to each time period with a distinct head temperature:

**Content Deleted –
EPRI Proprietary Information**

B.5 Remaining Time to Reach Oconee 3 Degradation Time

The fourth step was to calculate the remaining time until the plant reaches the equivalent normalized operating time as Oconee 3 using the remaining margin in degradation time and the current head temperature to translate the margin back to EFPYs at the actual head temperature:

**Content Deleted –
EPRI Proprietary Information**

In addition, the effect of any reported significant planned changes in future head temperature (e.g., future conversion of head temperature to cold leg temperature) were also considered by breaking future operation into two time periods similar to the calculation of equation [B.3].

---

[4] 50 kcal/mole is an accepted industry standard activation energy for SCC initiation in primary water. For example, see [21].

B.6 Histogram Groupings

Finally, the 69 PWR units were placed into eight groupings. These groupings were chosen where significant steps occurred in the ΔEFPYs calculated:

Group 1: $\Delta EFPY_{histogram} \leq 3$
Group 2: $3 < \Delta EFPY_{histogram} \leq 6$
Group 3: $6 < \Delta EFPY_{histogram} \leq 10$
Group 4: $10 < \Delta EFPY_{histogram} \leq 15$
Group 5: $15 < \Delta EFPY_{histogram} \leq 20$
Group 6: $20 < \Delta EFPY_{histogram} \leq 30$
Group 7: $30 < \Delta EFPY_{histogram} \leq 50$
Group 8: $\Delta EFPY_{histogram} > 50$

B.7 Results and Inspection Status

The interim simplified histogram is given in Table 4-1 and Figure 4-1. The table and figure also give the inspection status for the plants in each group.

C
PWR OWNERS GROUPS RESPONSE TO NRC QUESTIONS

Following are the coordinated PWR Owners Groups responses to questions 1.d and 1.e from the Brian Sheron (USNRC) letter to Alex Marion (NEI) dated April 17, 2001 [8].

Question 1.d (This discussion should include, but not be limited to:)

MRP recommendations regarding any expanded operator actions/training on beyond design basis accident (DBA) scenarios involving rod ejection(s), small-, medium- and large-break loss of coolant accidents (LOCA), rod insertion failure(s), etc.

Response:

In the unlikely event that one or more control rod drive mechanism (CRDM) penetrations were to fail, the resultant transient would be similar to that of a hot leg break LOCA in the small to medium size range. Plants typically operate with the control rods either fully withdrawn or only slightly inserted such that there would be very little positive reactivity inserted as a result of control rod ejection during power operation. If the plant happened to be in a condition where the control rods were inserted and a reactivity excursion did occur, the size of the excursion would depend on the initial plant conditions, the number of rods ejected and the associated rod worth. Reactor power would initially be turned by doppler due to the rise in fuel temperature. Reactor shutdown would automatically follow by insertion of the remaining control rods by the reactor protection system. If the loss of inventory were to exceed the capacity of the normal make up system, the loss of reactor coolant system (RCS) inventory would result in a decrease in system pressure and an automatic actuation of the emergency core cooling system (ECCS). The ECCS would inject borated water into the RCS to compensate for the loss of inventory due to the LOCA. Injection of the borated water would also ensure that the reactor remained shutdown following the rod ejection. This event would be a beyond design basis accident if more control rods were to be ejected than were analyzed for. However, plant safety equipment and operator response for this scenario is within the realm of guidance provided in existing generic Emergency Response Guidelines. The resulting LOCA would be bounded by existing design basis analyses and therefore the core would remain covered by borated water which would provide adequate cooling. Core internals would remain in a coolable geometry. The condition of the RCS and containment following this accident would not require implementation of the severe accident management guidelines (SAMGs).

Emergency Operating Procedures at all domestic nuclear power plants are based on generic guidelines developed by a combination of the plants' reactor vendor and utility personnel. These generic guidelines were developed to be in compliance with NUREG-0737, Clarification of TMI Action Plan Requirements, Item I.C.1, Guidance for the Evaluation and Development of Procedures for Accidents and Transients. NUREG 0737, Item I.C.1 specified that multiple events, consequential failures, and operator errors of omission or commission should be considered in development of procedures for transients and accidents. Each vendor's generic guidelines were submitted in the 1980s to the USNRC for review and NRC Safety Evaluation Reports (SERs) were issued to each vendor. Utilities have all implemented plant specific EOPs based on vendor-specific generic guidelines and have had subsequent audits by the USNRC to assure compliance with the generic guidelines and the requirements of NUREG 0737. Operators are routinely trained on implementation of the EOPs on a plant specific simulator as a requirement for maintaining their NRC operating license. The generic guidelines developed by each reactor vendor are symptom-based and do not require event identification. Following a reactor trip the operators follow plant specific EOPs that will ensure that all safety functions are being addressed. Existing EOPs provide guidance for all ranges of LOCAs and include coverage for multiple events including reactivity excursions that would occur during the course of an accident. Therefore, no additional operator action recommendations are needed since existing guidelines provide adequate directions to mitigate the transient induced by one or more CRDM penetration failures.

Questions 1.e (This discussion should include, but not be limited to:)

MRP recommendations regarding increased operator actions during periods of highest rod worth (e.g., Startup, Cool-down and Hot Standby periods).

Response:

Plants typically operate with the controls rods either fully withdrawn or only slightly inserted such that there would be very little positive reactivity inserted as a result of control rod ejection during power operation. During other modes of operation, a CRDM penetration failure would result in a small to medium size range LOCA with a possible reactivity excursion similar to that discussed in Item 1d. The plant EOPs would be implemented following the subsequent auto or manual reactor trip due to the reactivity excursion or loss of system inventory. As discussed in Item 1d, the existing EOPs provide adequate directions to mitigate the transient. No additional operator guidance is needed to address this event should it occur during Startup, Cool-down or Hot Standby Periods. Therefore no additional operator training recommendations are needed.