NUCLEAR ENERGY INSTITUTE

Alexander Marion
DIRECTOR
ENGINEERING DEPARTMENT
NUCLEAR GENERATION DIVISION

August 21, 2001

Dr. Brian W. Sheron
Associate Director for Project Licensing and Technical Analysis
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Mail Stop O5-E7
Washington, DC 20555-0001

SUBJECT: Generic Information for Use by Licensees in Response to NRC Bulletin 2001-01

PROJECT NUMBER: 689

Dear Dr. Sheron:

Enclosed is EPRI Report TP-1006284, *PWR Materials Reliability Program Response to NRC Bulletin 2001-01 (MRP-48).* This report contains generic information that some licensees may reference in their response to Bulletin 2001-01, *Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles.* Proprietary and non-proprietary versions of the report are enclosed along with an affidavit requesting that the NRC withhold the proprietary version from public distribution. The data contained in this report is the most accurate available at this time, but it is possible that changes may occur in the future.

If you have any questions about the enclosures, please contact Kurt Cozens at 202-739-8085, koc@nei.org, or me.

Sincerely,

Alex Marion

Alexander Marion

KOC/avw
Enclosures

c: Mr. Jack R. Strosnider, U.S. Nuclear Regulatory Commission
Mr. Jacob I. Zimmerman, U.S. Nuclear Regulatory Commission

D046

EPRI

August 17, 2001

Document Control Clerk
U.S. Nuclear Regulatory Commission
11555 Rockville Pike
Washington, DC 20555

Subject: "PWR Material Reliability Program Response to NRC Bulletin 2001-01," EPRI Report TP10016284, August 2001

Gentlemen:

This is a request under 10CFR2.790(a)(4) that the NRC withhold from public disclosure the information identified in the enclosed affidavit consisting of EPRI owned Proprietary Information identified above (the "Report"). Copies of the Report and the affidavit in support of this request are enclosed.

EPRI desires to disclose the Report in confidence to the NRC as a means of exchanging information with the NRC staff for the purpose of supporting generic regulatory improvements related to the management of the MRP Alloy 82/182 weld integrity. EPRI welcomes any discussion with the NRC regarding the Report that the NRC desires to conduct.

The Report is for the NRC's internal use and may be used only for the purposes for which it is disclosed by EPRI. The report should not be otherwise used or disclosed to any person outside the NRC without prior written permission from EPRI.

If you have any questions about the legal aspects of this request for withholding, please do not hesitate to contact me at (650) 855-2997. Questions on the contents of the Report should be directed to Mr. Al McIlree of EPRI at (650) 855-2092.

Sincerely,

Theodore U. Marston, Ph.D.
Vice President & Chief Nuclear Officer

Enclosures

c: Licensing

EPRI

AFFIDAVIT

RE: "PWR Material Reliability Program Response to NRC Bulletin 2001-01," EPRI Report TP10016284, August 2001

I, THEODORE U. MARSTON, being duly sworn, depose and state as follows:

I am a Vice President at the Electric Power Research Institute ("EPRI") and I have been specifically delegated responsibility for the report listed above that is sought under this affidavit to be withheld (the "Report") and authorized to apply for their withholding on behalf of EPRI. This affidavit is submitted to the Nuclear Regulatory Commission ("NRC") pursuant to 10 CFR 2.790 (a)(4) based on the fact that the Report consists of trade secrets of EPRI and that the NRC will receive the Report from EPRI under privilege and in confidence.

The basis for withholding such Report from the public is set forth below:

(i) The Report has been held in confidence by EPRI, its owner. All those accepting copies of the Report must agree to preserve the confidentiality of the Report.

(ii) The Report is a type customarily held in confidence by EPRI and there is a rational basis therefor. The Report is a type, which EPRI considers as a trade secret(s) and is held in confidence by EPRI because to disclose it would prevent EPRI from licensing the Report at fees, which would allow EPRI to recover its investment. If consultants and/or other businesses providing services in the electric/nuclear power industry were able to publicly obtain the Report, they would be able to use it commercially for profit and avoid spending the large amount of money that EPRI was required to spend in preparation of the Report. The rational basis that EPRI has for classifying this/these Report(s) as a trade secrets is justified by the Uniform Trade Secrets Act, which California adopted in 1984 and which has been adopted by over twenty states. The Uniform Trade Secrets Act defines a "trade secret" as follows:

> "Trade secret" means information, including a formula, pattern, compilation, program, device, method, technique, or process, that:
>
> (1) Derives independent economic value, actual or potential, from not being generally known to the public or to other persons who can obtain economic value from its disclosure or use; and
>
> (2) Is the subject of efforts that are reasonable under the circumstances to maintain its secrecy.

(iii) The Report will be transmitted to the NRC in confidence.

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(iv) The Report is not available in public sources. EPRI developed the Report only after making a determination that the Report was not available from public sources. It required a large expenditure of dollars for EPRI to develop the Report. In addition, EPRI was required to use a large amount of time of EPRI employees. The money spent, plus the value of EPRI's staff time in preparing the Report, show that the Report is highly valuable to EPRI. Finally, the Report was developed only after a long period of effort of at least several months.

(v) A public disclosure of the Report would be highly likely to cause substantial harm to EPRI's competitive position and the ability of EPRI to license the Report both domestically and internationally. The Report can only be acquired and/or duplicated by others using an equivalent investment of time and effort.

I have read the foregoing and the matters stated therein are true and correct to the best of my knowledge, information and belief. I make this affidavit under penalty of perjury under the laws of the United States of America and under the laws of the State of California.

Executed at 3412 Hillview Avenue, Palo Alto, being the premises and place of business of the Electric Power Research Institute:

August 17, 2001

Theodore U. Marston

Subscribed and sworn before me this day: August 17, 2001

Nichole Alexandra Edraos, Notary Public

NICHOLE ALEXANDRA EDRAOS
Comm. # 1295461
NOTARY PUBLIC - CALIFORNIA
Santa Clara County
My Comm. Expires Feb. 28, 2005

EPRI

PWR Materials Reliability Program Response to NRC Bulletin 2001-01 (MRP-48NP)

1006284-NP

Final Report, August 2001

LICENSED MATERIAL

NON-PROPRIETARY INFORMATION

PWR Materials Reliability Program Response to NRC Bulletin 2001-01 (MRP-48NP)

1006284-NP

Final Report, August 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

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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 Response to NRC Bulletin 2001-01 (MRP-48NP), EPRI, Palo Alto, CA: 2001. 1006284-NP.

REPORT SUMMARY

Background

Between November 2000 and April 2001 leaks were discovered from 15 reactor vessel top head CRDM nozzles at Oconee 1, 2 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 to have come from the annulus between the nozzles and holes in the vessel head. The leaks were traced to predominantly axial PWSCC cracks initiating on the outside surface of the nozzle wall below the J-groove weld. Three of the leaking nozzles at Oconee 3 and one of the leaking nozzles at Oconee 2 had circumferential cracks propagating from the OD of the nozzle above the J-groove welds. On August 3, 2001, the USNRC issued NRC Bulletin 2001-01, *Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles*, requesting that utilities submit their inspection plans to address RPV head penetration PWSCC.

Objective

The objective of this report is to provide information to support utility responses to NRC Bulletin 2001-01.

Approach

All PWR plants in the US have been ranked for the potential for PWSCC of reactor top head nozzles. The ranking has been based on the plant operating time adjusted for differences in reactor vessel head operating temperature using an activation energy model. On this basis, the three Oconee units and ANO-1 are the four highest ranked plants in the United States. Using the ranking, all PWR plants in the United States can be grouped into four categories as defined in the NRC bulletin. These are 1) plants with known leaks and cracks, 2) plants with less than 5 EFPY to reach the same time at temperature as Oconee 3 at the time that the leaks were discovered, 3) plants 5-30 EFPY relative to Oconee 3, and 4) plants >30 EFPY relative to Oconee 3. In addition to the plant ranking, NRC comments regarding applicable regulations were reviewed.

Results

The report contains the plant rankings using the time-at-temperature model, provides the supplementary information requested by paragraphs 1.a and 1.b of NRC Bulletin 2001-01, and provides comments regarding applicable regulatory requirements.

EPRI Perspective

As a consequence of leaks at the three Oconee plants and ANO-1, the industry, acting through the PWR Materials Reliability Program, is providing information to assist utilities in developing

responses to NRC Bulletin 2001-01. Other supporting information is provided in the interim safety assessment (MRP-44, Part 2—EPRI TP-1001491, Part 2) and in the response to NRC questions on the interim safety assessment (MRP 2001-050).

Keywords

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

ABSTRACT

This report was produced by the PWR Materials Reliability Program (MRP) to support individual utility responses to NRC Bulletin 2001-01. Following a background section, this report provides the latest plant rankings for all 69 domestic operating PWRs based on the time-at-temperature model, provides the supplementary information requested by paragraphs 1.a and 1.b of NRC Bulletin 2001-01, and provides comments regarding applicable regulatory requirements.

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1 BACKGROUND

This section briefly describes the industry experience that has led to the issuance of NRC Bulletin 2001-01.

1.1 PWSCC of Reactor Pressure Vessel (RPV) Top Head Nozzles

Report MRP-44, Part 2 [1] provides extensive background information regarding primary water stress corrosion cracking (PWSCC) of RPV top head nozzles. The following is a brief overview of this experience.

Reactor vessel top heads in pressurized water reactor (PWR) plants have a number of penetrations that are used for various purposes including control rod drive mechanism (CRDM) nozzles,[1] instrument nozzles, head vent nozzles and thermocouple nozzles. Figure 1-1 shows a typical reactor vessel top head arrangement for a plant designed by Babcock & Wilcox (B&W). The plants designed by Westinghouse and Combustion Engineering have similar top head configurations. Figure 1-2 shows a typical CRDM nozzle that is installed into a hole in the vessel head, typically with a small interference fit, and then welded to the inside surface of the head by a partial penetration Alloy 182 J-groove weld.

CRDM nozzles in several plants have experienced PWSCC of the Alloy 600 nozzle base material and Alloy 182 weld material. The following is a brief chronology of selected key events:

- In 1991, a small leak [<1 liter/hr (0.004 gpm)] was discovered from a CRDM nozzle at Bugey 3 in France. The leak was traced to an axial crack in the nozzle that had initiated on the inside surface of the nozzle at the elevation of the J-groove weld and then propagated through the nozzle wall thickness. Water was discovered leaking from the annulus between the nozzle and hole in the vessel head during a hydrostatic test. Laboratory examination showed a small [3 mm (0.12 inch) long × 2.25 mm (0.09 inch) deep] circumferentially oriented indication above the weld on the outside surface of the nozzle near the through-wall axial crack. There was no evidence that these conditions represented an immediate safety problem.

[1] Throughout 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. Further information regarding the nozzle designs and vessel head arrangements are provided in Table 2-3 of this report and Appendix A of MRP-44, Part 2 [1].

- As a result of the Bugey 3 experience, many plants worldwide have inspected the inside surfaces of their CRDM nozzles for PWSCC. These inspections showed that about 6.5% of nozzles in EdF plants had axial cracks on the nozzle ID surface while only about 1.25% of inspected nozzles in other plants had axial cracks on the ID surface. Most of these cracks were shallow, and none had resulted in leaks. Eddy current examinations of the ID surfaces of CRDM nozzles in seven plants in the United Sates (Point Beach 1, Oconee 2, Cook 2, Palisades, North Anna 1, Millstone 2, and Ginna) showed that axial cracks were initiating at a much slower rate than in EdF plants.
- In 1993/94 all three NSSS Owners Groups submitted safety assessments to the USNRC in response to the Bugey 3 leak [2–7]. The analyses demonstrated that CRDM nozzles are capable of accommodating long through-wall axial flaws, and that the reactor head can accommodate 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.
- In 1994, NUREG/CR-6245, *Assessment of Pressurized Water Reactor Control Rod Drive Mechanism Nozzle Cracking*, was issued [8]. This study concluded that the axial PWSCC cracks reported to date did not represent an immediate safety problem.
- In 1997, the NRC issued Generic Letter 97-01 [9] requesting the industry to respond formally to PWSCC of Alloy 600 RPV top head nozzles. In response to this Generic Letter, the industry developed predictive models for ID surface initiated PWSCC and provided the NRC with plant rankings and planned inspections of the ID surface of nozzles for the types of cracking that had been observed at other plants [10].
- Between November 2000 and April 2001, leaks were discovered from a total of 15 CRDM nozzles at four Babcock & Wilcox designed plants:
  - Oconee 1 (one leaking nozzle),
  - Oconee 2 (four leaking nozzles),
  - Oconee 3 (nine leaking nozzles), and
  - ANO-1 (one leaking nozzle).

  In addition, five of the eight smaller diameter thermocouple nozzles at Oconee 1 were discovered to have leaks. All of these leaks were first detected during visual inspections of the top surface of the vessel heads for boric acid crystal deposits (see Figure 1-3). In all cases the quantity of boric acid crystals at each nozzle location was small (<1 in^3). Additional findings were as follows:

  - Destructive examinations of several specimens from cracked Oconee 1 and 3 nozzles showed that the leaks were the result of primary water stress corrosion cracking (PWSCC).
  - Non-destructive examinations of the leaking CRDM nozzles showed that most of the cracks originated on the outside surface of the nozzles below the J-groove weld, were axially oriented, and propagated primarily in the nozzle base material to an elevation above the top of the J-groove weld where leakage could then pass through the annulus to the top of the head where it was detected by visual inspection. In some cases the cracks

initiated in the weld metal or propagated into the weld metal, and in a few cases the cracks propagated through the nozzle wall thickness to the inside surface.

- In addition to the predominantly axial cracks, several nozzles had cracks on the outside surface of the nozzle approximately following the weld contour above or below the J-groove weld. Four of these nozzles (three in Oconee 3 and one in Oconee 2) were found to have cracks approximately following the weld contour just above the J-groove weld. Two of the nozzles had relatively short and shallow cracks. Two of these nozzles had cracks either through-wall or essentially through-wall over an arc length of about 165° around the nozzle centered approximately about the nozzle uphill side. Cracks which follow the weld contour are a greater concern than axial cracks in that they raise the potential for a nozzle to be ejected if the crack extends more than about 92% around the nozzle circumference.
- Seventeen (17) additional non-leaking Oconee 1 and nine (9) non-leaking Oconee 3 CRDM nozzles were inspected by eddy current, ultrasonic testing, or eddy current and ultrasonic testing to assess the extent of condition of non-leaking nozzles in the vessel head. No significant cracking was found in any of these additional nozzles.
- The root cause evaluation showed that the observed axial cracks posed no safety concern other than allowing leakage to occur. As reported in Paragraph 5.2 of MRP-44, Part 2 [1], the remaining ligament in the two Oconee 3 nozzles with large cracks following the J-groove weld contour was 2.2 times the ligament required to meet code requirements (and about 7 times the ligament that would hold the applied load on a limit basis).

The recent experience at Oconee and ANO-1 differs from previous industry experience in that the cracking appears to initiate primarily on the outside surface of the nozzle below the weld rather than on the nozzle ID surface, and four of the nozzles have developed flaws approximately following the contour of the top of the J-groove weld.

Laboratory tests of specimens removed from Oconee 3 showed that they had a significant through-thickness hardness gradient with the outside surface being harder than the inside surface. The yield strength measured on a tensile specimen taken from the outer third of the wall thickness of one Oconee 3 CRDM nozzle was 67 ksi. This is higher than the reported yield strength of 49.5 ksi on the nozzle material certification and higher than the maximum reported yield strength of 64 ksi for all other CRDM nozzles in PWR plants in the United States. In summary, the cracks at Oconee and ANO-1 appear different from previous experience.

1.2 NRC Bulletin 2001-01

The USNRC issued Bulletin 2001-01, *Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles* [11], addressing the CRDM nozzle leaks at Oconee and ANO-1 on August 3, 2001. This bulletin requests that utilities provide plant design information, rankings relative to Oconee 3, previous inspection results, plans for future inspections, the bases for how the planned inspections will meet regulatory requirements, and a commitment to provide the results of any inspections performed during the next refueling outage. The bulletin requests that plants be grouped into four categories, the first being plants with known CRDM nozzle leaks and cracks and the other three representing increasing periods of time until the plants are predicted to reach the same time at temperature as Oconee 3. Oconee 3 had the greatest number of leaking nozzles and the most significant cracks following the weld contour above the J-groove weld.

1.3 Utility Response to NRC Bulletin 2001-01

The nuclear industry, including utility licensees, Nuclear Energy Institute (NEI), and the Electric Power Research Institute (EPRI), has established the Materials Reliability Program (MRP) to address generic issues relating to materials in PWR environments such as PWSCC of Alloy 600 materials. The MRP has been tasked by the industry to support the industry response to the recent RPV top head nozzle PWSCC experience. Information in this document has been prepared by the MRP to assist utilities in developing their responses to NRC Bulletin 2001-01.

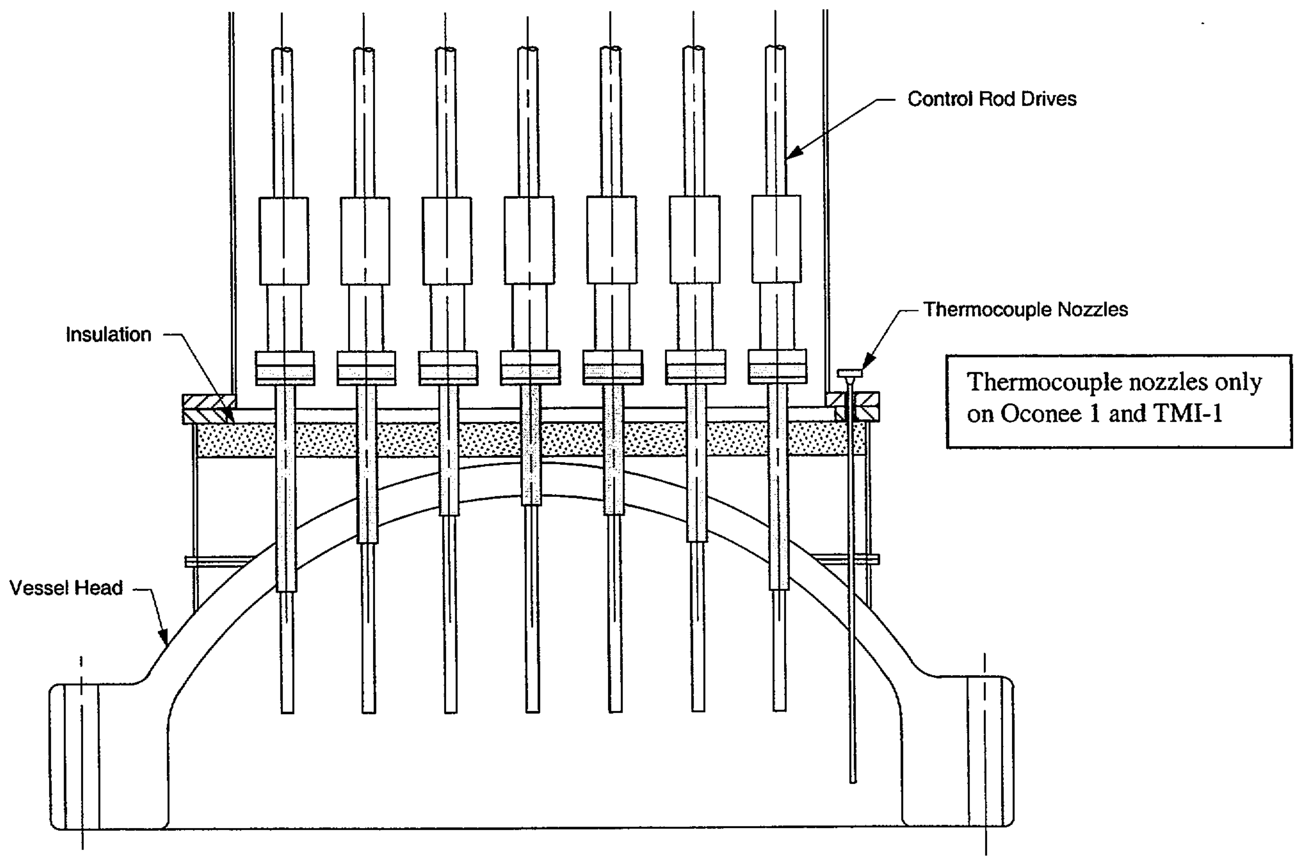

Figure 1-1
Typical Reactor Vessel Head – Oconee 1 (Babcock & Wilcox Design)

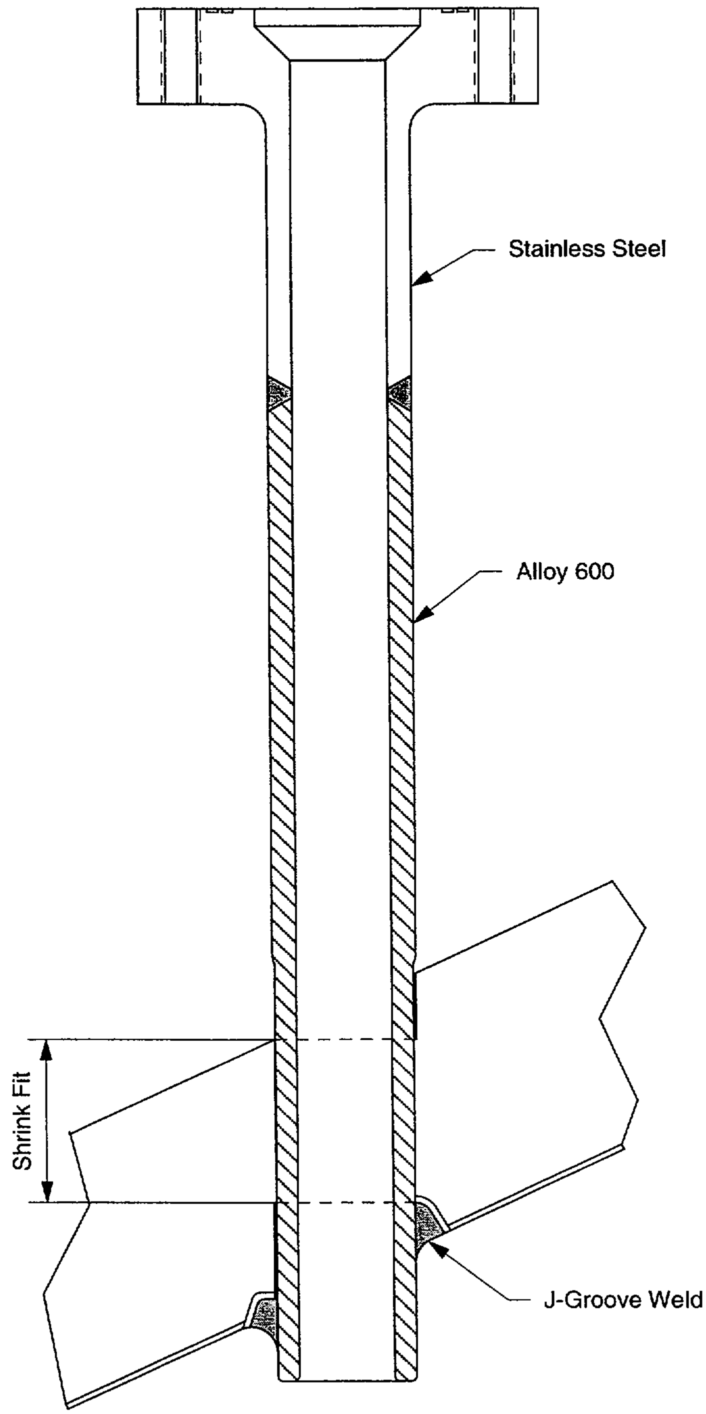

Figure 1-2
Typical CRDM Nozzle (Babcock & Wilcox Design)

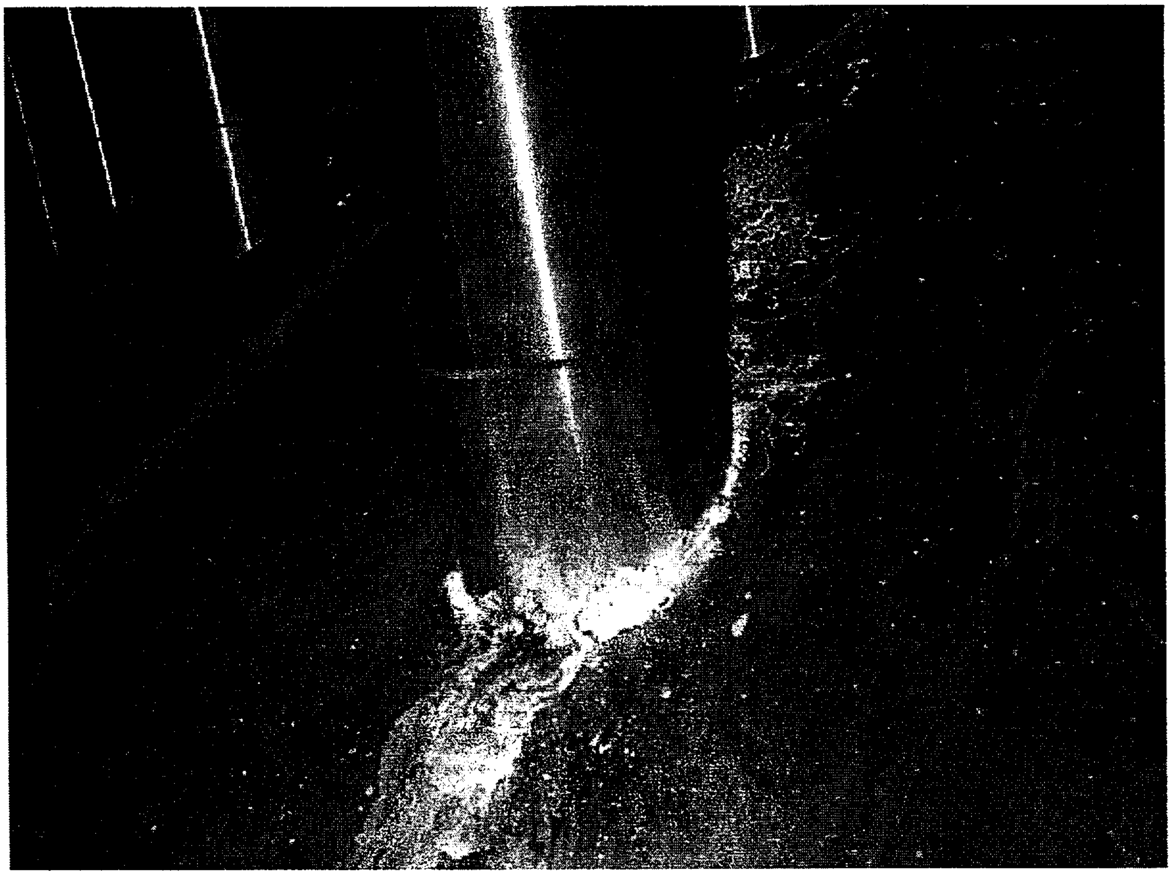

Figure 1-3
Leaking CRDM Nozzle at Oconee 3

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2
PLANT PWSCC RANKING AND SUPPORTING INFORMATION

This section describes the time-at-temperature model developed to rank plants and provides the most recent rankings. The section also provides some of the other data requested by NRC Bulletin 2001-01 for all domestic operating PWR plants, specifically Items 1.a. and 1.b. along with the basic RPV head insulation type and configuration and the basic inspection history. This information is compiled in this document for the convenience of the licensees and the NRC.

2.1 PWSCC Rankings

Plants have been ranked for the potential for RPV top head nozzle PWSCC using a time-at-temperature model. The methodology is the same as was described previously in MRP-44, Part 2 [1]. However, the plant rankings presented here are based on the best available inputs as of August 21, 2001.

2.1.1 Time-at-Temperature Model

Since stress corrosion cracking (SCC) of Alloy 600 nozzle material and Alloy 182 weld metal is sensitive to temperature, the current MRP 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 for 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 predicted value of 10 EFPYs would reach an equivalent degradation time as Oconee 3 after 10 EFPYs of additional operation at the current vessel head temperature.

2.1.2 Total Effective Full Power Years

The first step in the simplified plant ranking methodology was 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. The model is based on the EFPYs for each plant through February 2001.

2.1.3 Head Temperature History

The second step in the time-at-temperature ranking methodology was 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 97-01 [9]. Because of thermal-hydraulic differences between reactor designs, some plants operate with a head temperature close to the hot leg temperature, while some plants 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.

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 using the expression:

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2.1.4 Temperature-Adjusted Degradation Time

The third step in the time-at-temperature calculation was to calculate the plant 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:

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2.1.5 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:

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In addition, the effect of any reported significant planned future head temperature changes (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 approach of equation 2.2.

2.1.6 Plant Groupings

The number of EFPYs for each plant to reach the Oconee 3 time at temperature are provided in Table 2-1. Using this information, each PWR unit may be assigned to the four groups specified by NRC Bulletin 2001-01. These were

- Plants with known leaks or cracks
- Plants with less than 5 EFPY remaining relative to Oconee 3
- Plants with 5 to 30 EFPY remaining relative to Oconee 3
- Plants with more than 30 EFPY remaining relative to Oconee 3

Note that the histogram groups listed in Table 2-1 refer to the eight ranking categories previously defined by the MRP in report MRP-44, Part 2 [1].

2.2 Other Requested Information

In addition to the plant rankings, Tables 2-1 through 2-3 provide some additional plant-specific data requested by NRC Bulletin 2001-01. This additional information includes the following:

- The NSSS supplier

- The nozzle material supplier
- The vessel head fabricator
- The specified interference fit of the CRDM or CEDM nozzles into the hole in the vessel head
- The basic type of insulation on the head and insulation configuration
- The date of the next scheduled refueling outage
- Methods, dates and results of any previous visual leak inspections or NDE inspections of the nozzle inside surface
- Head temperature changes over plant life
- Reference to head maps in MRP-44, Part 2 [1], Appendix A
- Spacing between adjacent nozzles
- Number and size of each type of nozzle

2.3 Verification of Data

The data in Tables 2-1 through 2-3 are the best available as of August 21, 2001. However, as the industry performs further work on this issue and analyses are refined, it is likely that some changes may occur. The MRP will inform the NRC if these changes result in a plant changing category.

Table 2-1
Key Parameters and Ranking (Revised August 21, 2001[1])

		Design and Fabrication					*Operating Time and Temperature*							*Previous Inspection Status*				
Rank	Unit Name	NSSS Design	Nozzle Material Supplier[2]	Head Fabricator[3]	Design Diametral Nozzle Interference Fit (mils)	Insulation Type and Config.	EFPYs thru Feb. 2001	Head Temp. Range Over Life (°F)	Current Head Temp. (°F)	EFPYs Norm. to 600°F[4]	Remain. EFPYs to Reach Oconee 3 from 3/1/01[4]	Histogram Group (EFPYs)[4]	Next Scheduled Refueling Outage Date	Bare-Metal Visual or ID NDE	Date	Full / Partial	Result	Comments

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Table 2-1
Key Parameters and Ranking (Revised August 21, 2001[1]) (continued)

		Design and Fabrication					*Operating Time and Temperature*							*Previous Inspection Status*				
Rank	Unit Name	NSSS Design	Nozzle Material Supplier[2]	Head Fabricator[3]	Design Diametral Nozzle Interference Fit (mils)	Insulation Type and Config.	EFPYs thru Feb. 2001	Head Temp. Range Over Life (°F)	Current Head Temp. (°F)	EFPYs Norm. to 600°F[4]	Remain. EFPYs to Reach Oconee 3 from 3/1/01[4]	Histogram Group (EFPYs)[4]	Next Scheduled Refueling Outage Date	Bare-Metal Visual or ID NDE	Date	Full / Partial	Result	Comments

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Table 2-1
Key Parameters and Ranking (Revised August 21, 2001[1]) (continued)

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Table 2-1
Key Parameters and Ranking (Revised August 21, 2001[1]) (continued)

		Design and Fabrication					Operating Time and Temperature							Previous Inspection Status				
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NOTES:

[1]Corrections and revisions to this table since July 31, 2001, submittal to the NRC (letter from Marion of NEI to Sheron of NRC, dated July 31, 2001):

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[2]Key for Material Suppliers:
B = B&W Tubular Products
H = Huntington
S = Sandvik
SS = Standard Steel
W = Westinghouse (Huntington)
CL = C.L. Imphy
A = Aubert et Duval

[3]Key for Head Fabricators:
BW = B&W
CBI = Chicago Bridge & Iron
CE = Combustion Engineering
RDM = Rotterdam Dockyard
CL = C.L. Imphy

[4]Calculated using a thermal activation energy of 50 kcal/mole.

Table 2-2
Supplemental Data – Vessel Head Temperature History (Revised August 21, 2001)

Rank	Name	NSSS Design	Current Head Temp. (°F)	*Period #1*		*Period #2*		*Period #3*		*Period #4*	
				EFPYs at Temp.	Head Temp. (°F)	EFPYs at Temp.	Head Temp. (°F)	EFPYs at Temp.	Head Temp. (°F)	EFPYs at Temp.	Head Temp. (°F)

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Table 2-2
Supplemental Data – Vessel Head Temperature History (Revised August 21, 2001) (continued)

Rank	Name	NSSS Design	Current Head Temp. (°F)	*Period #1*		*Period #2*		*Period #3*		*Period #4*	
				EFPYs at Temp.	Head Temp. (°F)	EFPYs at Temp.	Head Temp. (°F)	EFPYs at Temp.	Head Temp. (°F)	EFPYs at Temp.	Head Temp. (°F)

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Table 2-3
Supplemental Data – Head Arrangement and Nozzle Information (Revised August 21, 2001)

Rank	Unit Name	NSSS Design	Head Map Figure in MRP-44, Part 2[1]	*Minimum Distance[2] Between CRDM/CEDM Nozzle and Adjacent VHP Nozzle of Type...*						*J-Groove Type VHP Nozzles[3]*																					*Other VHP Nozzles[3]*								
				CRDM/CEDM Nozzle (in)	ICI Nozzle (in)	Thermocouple Nozzle (in)	Auxiliary Head Adapter (in)	De-Gas Line Nozzle (in)	Intern. Support Housing (in)	*CRDM Nozzles[4]*			*CEDM Nozzles[4]*			*ICI Nozzles*			*J-Groove Head Vent Nozzles*			*B&W Thermocouple Nozzles*			*J-Groove Auxiliary Head Adapters*			*De-Gas Line Nozzles*			*Butt Weld Head Vent Nozzles[5]*			*Internals Support Housings[6]*			*Auxiliary Head Adapters[7]*		
										Number	OD (in)	ID (in)	Number	OD (in)	ID (in)	Number	OD (in)	ID (in)	Number	OD (in)	ID (in)	Number	OD (in)	ID (in)	Number	OD (in)	ID (in)	Number	OD (in)	ID (in)	Number	OD (in)	ID (in)	Number	OD (in)	ID (in)	Number	OD (in)	ID (in)

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MRP/EPRI Proprietary Information

Table 2-3
Supplemental Data – Head Arrangement and Nozzle Information (Revised August 21, 2001) (continued)

Rank	Unit Name	NSSS Design	Head Map Figure in MRP-44, Part 2[1]	*Minimum Distance[2] Between CRDM/CEDM Nozzle and Adjacent VHP Nozzle of Type...*						*J-Groove Type VHP Nozzles[3]*																					*Other VHP Nozzles[3]*								
				CRDM/CEDM Nozzle (in)	ICI Nozzle (in)	Thermocouple Nozzle (in)	Auxiliary Head Adapter (in)	De-Gas Line Nozzle (in)	Intern. Support Housing (in)	*CRDM Nozzles[4]*			*CEDM Nozzles[4]*			*ICI Nozzles*			*J-Groove Head Vent Nozzles*			*B&W Thermocouple Nozzles*			*J-Groove Auxiliary Head Adapters*			*De-Gas Line Nozzles*			*Butt Weld Head Vent Nozzles[5]*			*Internals Support Housings[6]*			*Auxiliary Head Adapters[7]*		
										Number	OD (in)	ID (in)	Number	OD (in)	ID (in)	Number	OD (in)	ID (in)	Number	OD (in)	ID (in)	Number	OD (in)	ID (in)	Number	OD (in)	ID (in)	Number	OD (in)	ID (in)	Number	OD (in)	ID (in)	Number	OD (in)	ID (in)	Number	OD (in)	ID (in)

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MRP/EPRI Proprietary Information

NOTES:

[1]Head map figure number in *PWR Materials Reliability Project Interim Alloy 600 Safety Assessments for US PWR Plants (MRP-44), Part 2: Reactor Vessel Top Head Penetrations*, EPRI TP-1001491, Part 2, May 2001 [1].

[2]Minimum nominal nozzle centerline-to-centerline lateral distance.

[3]The basic designs of the VHP nozzles are shown in Figures A-9 through A-16 of MRP-44, Part 2. The nozzle material is Alloy 600 for all J-groove type VHP nozzles. The "Butt Weld" head vent, internals support housing, and "Butt Weld" auxiliary head adapter nozzles comprise an Alloy 600 pipe joined to the head as shown in Figures A-12, A-14 and A-15 of MRP-44, Part 2. The nozzle dimensions listed are nominal dimensions.

[4]Not all CRDM and CEDM nozzles are used for control rod (element) drive shafts. Some CRDM nozzles are empty (spares) or are used for part-length shafts, thermocouple instrumentation or the reactor vessel level instrumentation system, and some CEDM nozzles house heated junction thermocouple instrumentation.

[5]OD and ID dimension are for the Alloy 600 pipe welded to the top of the Alloy 600 extension (see "Butt Weld" design in Figure A-12 of MRP-44, Part 2).

[6]OD and ID dimension are for the Alloy 600 pipe section directly above the field weld (see Figure A-14 of MRP-44, Part 2).

[7]OD and ID dimension are for the Alloy 600 pipe section directly above the field weld (see "Butt Weld" design in Figure A-15 of MRP-44, Part 2).

3 REGULATORY REQUIREMENTS

The NRC Bulletin 2001-01 section entitled Applicable Regulatory Requirements cites the following regulatory requirements as providing the basis for the bulletin assessment:

- Appendix A to 10 CFR Part 50, *General Design Criteria for Nuclear Power Plants*
  - Criteria 14 - *Reactor Coolant Pressure Boundary*
  - Criteria 31 - *Fracture Prevention of Reactor Coolant Boundary, and*
  - Criteria 32 - *Inspection of Reactor Pressure Coolant Pressure Boundary*
- Plant Technical Specifications
- 10 CFR 50.55a, Codes and Standards, which incorporates by reference Section XI, *Rules for Inservice Inspection of Nuclear Power Plant Components*, of the *ASME Boiler and Pressure Vessel Code*
- Appendix B of 10 CFR Part 50, *Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants*, Criteria V, IX, and XVI

This section discusses how the cited regulatory requirements affect plant decisions relating to addressing NRC Bulletin 2001-01 requested actions and regulatory compliance.

3.1 Design Requirements: 10CFR § 50, Appendix A – General Design Criteria

The Bulletin states:

> *"The applicable GDC include GDC 14, GDC 31, and GDC 32. GDC 14 specifies that the reactor coolant pressure boundary (RCPB) have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture; the presence of cracked and leaking VHP nozzles is not consistent with this GDC. GDC 31 specifies that the probability of rapidly propagating fracture of the RCPB be minimized; the presence of cracked and leaking VHP nozzles is not consistent with this GDC. GDC 32 specifies that components which are part of the RCPB have the capability of being periodically inspected to assess their structural and leak tight integrity; inspection practices that do not permit reliable detection of VHP nozzle cracking are not consistent with this GDC."*

The three referenced design criteria state the following:

- Criterion 14 – Reactor Coolant Pressure Boundary

  "The reactor coolant pressure boundary shall be designed, fabricated, erected and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture."

- Criterion 31 – Fracture Prevention of Reactor Coolant Pressure Boundary

  "The reactor coolant pressure boundary shall be designed with sufficient margin to assure that when stressed under operating, maintenance, testing, and postulated accident conditions (1) the boundary behaves in a non-brittle manner, and (2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures and other conditions of the boundary material under operating, maintenance, testing and postulated accident conditions and the uncertainties in determining (1) material properties, (2) the effects of irradiation on material properties, (3) residual, steady state and transient thermal stresses, and (4) flaw sizes."

- Criterion 32 – Inspection of Reactor Coolant Pressure Boundary

  "Components which are part of the reactor coolant pressure boundary shall be designed to permit (1) periodic inspection and testing of important areas and features to assess their structural and leak tight integrity, and (2) an appropriate material surveillance program for the reactor pressure boundary."

During licensing of the currently operating plants, licensees demonstrated that the design of the reactor coolant pressure boundary meets these requirements or those in the proposed Appendix A, "General Design Criteria for Nuclear Power Plant Construction Permits," as published in the Federal Register on July 11, 1967. Although the criteria of the proposed Appendix A are different, they convey similar intent. The following information discusses application of the design criteria for the cracking of RPV top head nozzles:

- Pressurized water reactors licensed both before and after issuance of Appendix A to Part 50 (1971) complied with these criteria in part by: 1) selecting Alloy 600 or other austenitic materials with excellent corrosion resistance and extremely high fracture toughness, for reactor coolant pressure boundary materials, and 2) following ASME Codes and Standards and other applicable requirements for fabrication, erection, and testing of the pressure boundary parts. NRC reviews of operating license submittals subsequent to issuance of Appendix A included evaluating designs for compliance with the General Design Criteria. The SRPs (standard review plans) in effect at the time of licensing do not address the selection of Alloy 600. They only require that ASME code requirements be satisfied.

- Although stress corrosion cracking of primary coolant system penetrations was not originally anticipated during plant design, it has occurred in the RPV top head nozzles at some plants. The suitability of the originally selected materials has been confirmed. The robustness of the design has been demonstrated by the small amounts of the leakage that has occurred and by the fact that none of the cracks in Alloy 600 reactor coolant pressure boundary materials has rapidly propagated or resulted in catastrophic failure or gross rupture. It should be noted that the proposed Appendix A was written in terms of extremely low probability of gross rupture or significant leakage throughout the design life.

- ASME requirements for the J-groove CRDM welds are for a visual examination of 25% for leakage during pressure testing. The component was designed for that inspection. Additionally, NDE and direct visual examination may be performed for some plants using specialized robotic tools to minimize personnel exposure.

As described above, the requirements established for design, fracture toughness, and inspectability in GDC 14, 31, and 32 respectively were satisfied during a plant's initial licensing review, and continue to be satisfied during operation, even in the presence of a potential for stress corrosion cracking of the RPV top head penetrations.

3.2 Operating Requirements: 10 C.F.R. § 50.36 - Plant Technical Specifications

The Bulletin states:

> *"Plant technical specifications pertain to the issue of VHP nozzle cracking insofar as they require no through-wall reactor coolant system leakage."*

Title 10 of the Code of Federal Regulations, Part 50.36 (10CFR 50.36) contains requirements for Plant Technical Specifications. Paragraphs 2 and 3 of 10CFR Part 50.36 are particularly relevant:

- 10CFR 50.36 (2) Limiting Conditions for Operation

  "Limiting conditions for operation are the lowest functional capability or performance levels of equipment required for safe operation of the facility. When a limiting condition for operation of a nuclear reactor is not met, the licensee shall shut down the reactor or follow any remedial action permitted by the technical specifications until the condition can be met.

  A technical specification limiting condition for operation of a nuclear reactor must be established for each item meeting one of the following criteria:

  Criterion 3: A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

  Criterion 4: A structure, system or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety."

- 10 CFR 50.36 (3) Surveillance Requirements

  "Surveillance requirements are requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions will be met."

The reactor coolant pressure boundary provides one of the critical physical barriers that guard against the uncontrolled release of radioactivity. Therefore, plant technical specifications generally include a requirement and associated action statements addressing reactor coolant pressure boundary leakage. The limits for PWR reactor coolant pressure boundary leakage are typically stated in terms of the amount of leakage, *e.g.,* 1 gallon per minute for unidentified leakage; 5-10 gpm for identified leakage; and no leakage from a non-isolable fault in the reactor coolant system pressure boundary.

Regulatory Guide 1.45, "RCPB Leakage Detection Systems," requires a leakage detection system sensitivity that can detect a leak rate of 1 gpm in less than one hour. Plants meet this criterion. Most leaks from reactor coolant system Alloy 600 penetrations have been well below the sensitivity of on-line leakage detection systems. Plants have evaluated this condition and have determined that the appropriate inspections are bare-metal visual inspections of the reactor head for boric acid deposits during plant shutdowns or NDE examinations of the CRDM nozzles. If leakage or unacceptable indications are found, then the defect must be repaired before the plant goes back on line. If through-wall pressure boundary leaks of CRDM nozzles increase to the point where they are picked up by the on-line leak detection systems, then the leak must be evaluated per the specified acceptance criteria, and the plant be shut down if it is a pressure boundary fault.

3.3 Inspection Requirements: 10 C.F.R. § 50.55a and ASME Section XI

The Bulletin states:

> *NRC regulations at 10 CFR 50.55a state that ASME Class 1 components (which include VHP nozzles) must meet the requirements of Section XI of the ASME Boiler and Pressure Vessel Code. Table IWA-2500-1 [IWB-2500-1[2]] of Section XI of the ASME Code provides examination requirements for VHP nozzles and references IWB-3522 for acceptance standards. IWB-3522.1(c) and (d) specify that conditions requiring correction include the detection of leakage from insulated components and discoloration or accumulated residues on the surfaces of components, insulation, or floor areas which may reveal evidence of borated water leakage, with leakage defined as "the through-wall leakage that penetrates the pressure retaining membrane." Therefore, 10 CFR 50.55a, through its reference to the ASME Code, does not permit through-wall cracking of VHP nozzles.*
>
> *For through-wall leakage identified by visual examinations in accordance with the ASME Code, acceptance standards for the identified degradation are provided in IWB-3142.*

[2] An erratum appears to exist in the Bulletin. Table IWA-2500-1 is cited, but does not exist. It appears that the citation should have been IWB-2500-1.

Specifically, supplemental examination (by surface or volumetric examination), corrective measures or repairs, analytical evaluation, and replacement provide methods for determining the acceptability of degraded components."

Title 10 of the Code of Federal Regulations, Part 50.55a requires that inservice inspection and testing be performed per the requirements of the ASME Boiler and Pressure Vessel Code, Section XI, *Inservice Inspection of Nuclear Plant Components.* Section XI contains applicable rules for examination, evaluation and repair of code class components, including the reactor coolant pressure boundary.

Requirements for partial penetration welds attaching CRDM nozzles to the reactor vessel head are contained in Table IWB-2500-1, Examination Category B-E, *Pressure Retaining Partial Penetration Welds in Vessels*, Items Numbers: B4.10, *Partial Penetration Welds*; B4.11, *Vessel Nozzles*; B4.12, *CRDM Nozzles*; and B4.13, *Instrumentation Nozzles.* The Code requires a VT-2 "visual examination" of 25% of the CRDM nozzles from the external surface. Since the head is insulated, and the nozzles do not represent a bolted flange, paragraph IWA-5242(b) permits these inspections to be performed with the insulation left in place.

All plants perform visual inspections for evidence of leakage by examining the RPV top head surface, or the insulation, per the requirements of NRC Generic Letter 88-05, *Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants.*

Some plants have conducted inspections beyond those required by Section XI and NRC Generic Letter 88-05. These inspections have included visual examinations of 100% of the bare metal surfaces of the reactor head, eddy current and liquid penetrant surface examinations and volumetric examinations of the nozzles. These supplemental inspections, coupled with evaluations of the cracking found, are considered to have provided a defense-in-depth approach for investigating and resolving this issue. Additional work is underway for developing alternative inspection and analysis tools.

The acceptance standard for the visual examination is found in paragraph IWA-5250, *Corrective Measures.* Paragraph IWA-5250 requires repair or replacement of the affected part if a through-wall leak is found and requires an assessment of damage, if any, associated with corrosion of steel components by boric acid. No plant has returned to service after finding a leak from a RPV top head nozzle without first having repaired the nozzle.

Flaws identified by nondestructive examination (NDE) methods, which are beyond current requirements, are evaluated in accordance with the flaw evaluation rules for austenitic piping contained in the Section XI of the ASME Code. This approach has been accepted by the NRC. Any flaw not meeting requirements for the intended service period would be repaired before returning it to service.

Repairs to RPV top head nozzles have been performed in accordance with Section XI requirements, NRC-approved ASME Code Case requirements, or an alternative repair or replacement method approved by the NRC.

Licensees comply with these ASME Code requirements through implementation of the plant's inservice inspection program. If a VT-2 examination detects the conditions described by IWB-3522.1(c) and (d), then corrective actions per IWB-3142 would be performed in accordance with the plant's corrective action program. No new plant actions are necessary to satisfy the cited regulatory criteria.

3.4 Quality Assurance Requirements: 10 C.F.R. § 50, Appendix B

The Bulletin states:

> *Criterion IX of Appendix B to 10 CFR Part 50 states that special processes, including nondestructive testing, shall be controlled and accomplished by qualified personnel using. BL 2001-01 qualified procedures in accordance with applicable codes, standards, specifications, criteria, and other special requirements. Within the context of providing assurance of the structural integrity of VHP nozzles, special requirements for visual examination would generally require the use of a qualified visual examination method. Such a method is one that a plant-specific analysis has demonstrated will result in sufficient leakage to the RPV head surface for a through-wall crack in a VHP nozzle, and that the resultant leakage provides a detectable deposit on the RPV head. The analysis would have to consider, for example, the as-built configuration of the VHPs and the capability to reliably detect and accurately characterize the source of the leakage, considering the presence of insulation, preexisting deposits on the RPV head, and other factors that could interfere with the detection of leakage. Similarly, special requirements for volumetric examination would generally require the use of a qualified volumetric examination method, for example, one that has a demonstrated capability to reliably detect cracking on the OD of the VHP nozzle above the J-groove weld.*

Criterion IX is a forward-looking requirement such that if inspections are performed they must be controlled and accomplished by qualified personnel. No action is required by a licensee to satisfy this criterion, unless a new inspection is proposed. However, if the bulletin response identifies a new inspection then the response should identify how Criterion IX is satisfied

The Bulletin further states:

> *"Criterion V of Appendix B to 10 CFR Part 50 states that activities affecting quality shall be prescribed by documented instructions, procedures, or drawings, of a type appropriate to the circumstances and shall be accomplished in accordance with these instructions, procedures, or drawings. Criterion V further states that instructions, procedures, or drawings shall include appropriate quantitative or qualitative acceptance criteria for determining that important activities have been satisfactorily accomplished. Visual and volumetric examinations of VHP nozzles are activities that should be documented in accordance with these requirements."*

Criterion V is also a forward-looking criterion that applies should the bulletin response identify new inspections. It does not establish criteria for when or if inspections should be performed. If new inspections are performed, they will meet Criterion V.

The last Appendix B criterion cited in the bulletin is:

> *"Criterion XVI of Appendix B to 10 CFR Part 50 states that measures shall be established to assure that conditions adverse to quality are promptly identified and corrected. For significant conditions adverse to quality, the measures taken shall include root cause determination and corrective action to preclude repetition of the adverse conditions. For cracking of VHP nozzles, the root cause determination is important to understanding the nature of the degradation present and the required actions to mitigate future cracking. These actions could include proactive inspections and repair of degraded VHP nozzles."*

Criterion XVI has two attributes that should be considered by licensees in their response to the Bulletin.

The first attribute is *that measures shall be established to assure that conditions adverse to quality are promptly identified and corrected.* This criterion infers a licensee's responsibility to be aware of industry experience, and has been interpreted in this manner in most plant's corrective action programs. A licensee should determine if an industry experience applies to its plant and what, if any, corrective actions are appropriate. This approach is consistent with the NRC's generic communication process for an Information Notice, which reports industry experience, but does not require a response to the NRC. Licensees are expected to evaluate the applicability of the occurrence to their plant and document the plant specific assessment for possible NRC review during inspections.

Criterion XVI provides the objectives and goals of the corrective action program, but licensees are responsible for determining a specific process to accomplish these goals and objectives. With regard to the bulletin response, Criterion XVI does not provide specific guidance as to what is an appropriate response, but rather, the licensee is responsible for determining actions necessary to maintain public health and safety. That is, the licensee must justify its actions for addressing the stress corrosion cracking of vessel head penetrations. Furthermore, the regulatory criteria of 10 CFR 50.109(a)(7), provides supporting evidence when it states that *if there are two or more ways to achieve compliance . . . then ordinarily the applicant or licensee is free to choose the way which best suits its purposes.*

The second attribute of Criterion XVI that should be considered is that for *significant conditions adverse to quality, the measures taken shall include root cause determination and corrective action to preclude repetition of the adverse conditions.* The bulletin suggests that for cracking of vessel head penetrations, the root cause determination is important in understanding the nature of the degradation and the required actions to mitigate future cracking. As part of its corrective action program, a licensee, through its own efforts or as part of an industry effort, would determine the cause of cracks in the vessel head penetration, if they are detected. However, if no known cracks in the heads are identified through reasonable quality assurance measures or inspection and monitoring programs, this criterion would not require specific action on the part of a licensee for remaining in compliance with the regulation.

In summary, the industry approach to inspection, monitoring, cause determination, and resolution of the identified CRDM nozzle cracking is consistent with the performance-based objectives of Appendix B.

4 REFERENCES

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

2. *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.

3. *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.

4. *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.

5. *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.

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

7. *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.

8. *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.

9. *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.

10. "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.

11. *NRC Bulletin 2001-01: Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles*, US Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, August 3, 2001.

12. 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.

13. *Response to NRC Review Comments Transmitted by Letter Dated June 22, 2001, to the Nuclear Energy Institute Relating to MRP-44.* Report MRP 2001-050, dated June 29, 2001.