October 2, 2002

Mr. Lew W. Myers
Chief Operating Officer
FirstEnergy Nuclear Operating Company
Davis-Besse Nuclear Power Station
5501 North State Route 2
Oak Harbor, OH 43449-9760

SUBJECT: DAVIS-BESSE NUCLEAR POWER STATION
NRC AUGMENTED INSPECTION TEAM FOLLOW-UP SPECIAL
INSPECTION REPORT NO. 50-346/02-08(DRS)

Dear Mr. Myers:

On March 12, 2002, the USNRC dispatched an Augmented Inspection Team (AIT) to the Davis-Besse site in accordance with USNRC Management Directive 8.3, “USNRC Incident Investigation Program.” The AIT was chartered to determine the facts and circumstances related to the significant degradation of the reactor vessel head pressure boundary material. The AIT developed a sequence of events, interviewed plant personnel, collected and analyzed factual information relevant to the degraded condition and conducted visual inspections of the reactor vessel head. The AIT results were summarized for you and your staff during a public exit meeting on April 5, 2002, and the AIT report was issued on May 3, 2002.

On May 15, 2002, USNRC began a special inspection focused on compliance with USNRC rules and regulations as they relate to the facts and circumstances associated with the degradation of the reactor pressure vessel head documented in the AIT report. On August 9, 2002, the USNRC completed this special inspection. The enclosed report documents the inspection findings which were discussed with you and other members of your staff on August 9, 2002.

Based on this special inspection, ten findings, some apparent violations with multiple examples, were identified and are documented in the enclosed report. Those findings include: operating the reactor with prohibited pressure boundary leakage; failure to take effective action to correct multiple identified safety concerns; inadequacies in the boric acid corrosion control procedure; failure to effectively implement the boric acid corrosion control procedure and the corrective action procedure; and multiple examples of inaccurate or incomplete information in letters to the USNRC or records required by the USNRC to be maintained onsite. Because the USNRC’s determination of the safety significance of the reactor vessel head degradation has not been finalized and several of these apparent violations remain under review by the USNRC, all of these findings are currently characterized as unresolved items in the enclosed report.

In accordance with 10 CFR Part 2.790 of the USNRC's "Rules of Practice," a copy of this letter and its enclosure will be available electronically for public inspection in the USNRC Public Document Room or from the Publicly Available Records (PARS) component of USNRC's document system (ADAMS). ADAMS is accessible from the USNRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

John A. Grobe, Chairman
Davis-Besse Oversight Panel

Docket No. 50-346
License No. NPF-3

Enclosure: USNRC Inspection Report
No. 50-346/02-08(DRS)

cc w/encl: B. Saunders, President - FENOC
Plant Manager
Manager - Regulatory Affairs
M. O'Reilly, FirstEnergy
Ohio State Liaison Officer
R. Owen, Ohio Department of Health
Public Utilities Commission of Ohio
President, Board of County Commissioners
Of Lucas County
President, Ottawa County Board of Commissioners
D. Lochbaum, Union of Concerned Scientists

In accordance with 10 CFR Part 2.790 of the USNRC's "Rules of Practice," a copy of this letter and its enclosure will be available electronically for public inspection in the USNRC Public Document Room or from the Publicly Available Records (PARS) component of USNRC's document system (ADAMS). ADAMS is accessible from the USNRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

John A. Grobe, Chairman
Davis-Besse Oversight Panel

Docket No. 50-346
License No. NPF

Enclosure: USNRC Inspection Report
No. 50-346/02-08(DRS)

cc w/encl: B. Saunders, President - FENOC
Plant Manager
Manager - Regulatory Affairs
M. O'Reilly, FirstEnergy
Ohio State Liaison Officer
R. Owen, Ohio Department of Health
Public Utilities Commission of Ohio
President, Board of County Commissioners
Of Lucas County
President, Ottawa County Board of Commissioners
D. Lochbaum, Union of Concerned Scientists

DOCUMENT NAME: G:DRS\ML022750524.wpd

To receive a copy of this document, indicate in the box: "C" = Copy without attachment/enclosure "E" = Copy with attachment/enclosure "N" = No copy

OFFICE	RIII	RIII	RIII	RIII
NAME	MFarber:sd	ZFalevits	JGrobe for CLipa	RPaul
DATE	10/01/02	10/01/02	10/01/02	10/01/02

OFFICE	RIII	RIII		
NAME	CWeil for BClayton	JGrobe		
DATE	10/02/02	10/02/02		

OFFICIAL RECORD COPY

ADAMS Distribution:
Chairman Meserve
Commissioner Dicus
Commissioner Diaz
Commissioner McGaffigan
Commissioner Merrifield
W. Travers, EDO
W. Kane, OEDO
S. Rosenberg, OEDO
B. Bonser, OEDO
OCIO
ACRS
S. Collins, NRR
B. Sheron, NRR
J. Zwolinski, NRR
R. Barrett, NRR
G. Holahan, NRR
B. Bateman, NRR
S. Bajwa, NRR
S. Long, NRR
F. Congel, OE
OPA
OCA
H. Miller, RI
L. Reyes, RII
E. Merschoff, RIV
AJM
DFT
SPS1
RidsNrrDipmIipb
GEG
HBC
CST1
C. Ariano (hard copy)
DRPIII
DRSIII
PLB1
JRK1
DB0350

U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket No: License No:	50-346 NPF-3
Report No:	50-346/02-08
Licensee:	FirstEnergy Nuclear Operating Company
Facility:	Davis-Besse Nuclear Power Station
Location:	5501 North State Route 2 Oak Harbor, OH 43449
Dates:	May 15 through August 9, 2002
Inspectors:	J. Gavula, Senior Reactor Inspector M. Farber, Senior Reactor Inspector J. Jacobson, Senior Mechanical Engineer
Approved by:	John A. Grobe, Chairman Davis-Besse Oversight Panel

SUMMARY OF FINDINGS

IR 05000346-02-08, on 05/15-08/09/2002, FirstEnergy Nuclear Operating Company, Davis-Besse Nuclear Power Station. Augmented Inspection Team Follow-up Special Inspection.

The report covers a special inspection, by three regional inspectors, that focused on compliance with USNRC rules and regulations as they relate to the facts and circumstances associated with the degradation of the reactor pressure vessel head. The USNRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 3, dated July 2000.

A. Inspector Identified Findings

Cornerstones: Initiating Events, Barrier Integrity

- Significance to be Determined (TBD). The inspectors identified an apparent violation of Technical Specification Limiting Condition for Operation for Reactor Coolant System Operational Leakage, paragraph 3.4.6.2, for operation of the plant with pressure boundary leakage from through-wall cracks in the reactor coolant system.

  This finding is more than minor because the pressure boundary leakage and resultant cavity in the reactor vessel head represented a loss of the design basis barrier integrity. The significance of this finding will be determined by the Significance Determination Process for the issue, which was begun following the Augmented Inspection Team activities (Section 4OA3.b.1).

- Significance to be Determined (TBD). The inspectors identified an apparent violation involving failure to take adequate corrective action for a continuing buildup of boric acid deposits on the reactor head.

  This finding is more than minor because the corrosion of the reactor head and the resulting cavity represented a significant loss of the design basis barrier integrity. The significance of this finding will be determined by the Significance Determination Process for the issue, which was begun following the Augmented Inspection Team activities (Section 4OA3.b.2.1).

- Significance to be Determined (TBD). The inspectors identified an apparent violation involving failure to take adequate corrective action for recurrent accumulations of boric acid on containment air cooler (CAC) fins. These accumulations resulted in reduced heat removal capability and reduced air flow through the cooler which was indicated by decreasing plenum pressure.

  This finding is more than minor because the corrosion of the reactor head and the resulting cavity represented a significant loss of the design basis barrier integrity. The significance of this finding will be determined by the Significance

Determination Process for the issue, which was begun following the Augmented Inspection Team activities (Section 4OA3.b.2.2).

- Significance to be Determined (TBD). The inspectors identified an apparent violation involving failure to take adequate corrective action for repeated clogging of radiation element filters although a sample of the filter deposits revealed iron oxides, radionuclides, and primary chemistry.

  This finding is more than minor because the corrosion of the reactor head and the resulting cavity represented a significant loss of the design basis barrier integrity. The significance of this finding will be determined by the Significance Determination Process for the issue, which was begun following the Augmented Inspection Team activities (Section 4OA3.b.2.3).

- Significance to be Determined (TBD). The inspectors identified an apparent violation involving the failure to follow the corrective action procedure and take timely corrective action for a condition adverse to quality, in that the licensee failed to implement a modification to permit complete inspection and cleaning of the reactor vessel head and CRDM nozzles.

  This finding is more than minor because the corrosion of the reactor head and the resulting cavity represented a significant loss of the design basis barrier integrity. The significance of this finding will be determined by the Significance Determination Process for the issue, which was begun following the Augmented Inspection Team activities (Section 4OA3.b.2.4).

- Significance to be Determined (TBD). The inspectors identified a finding involving failure to complete an identified corrective action for an adverse trend in RCS unidentified leakage.

  This finding is more than minor because the corrosion of the reactor head and the resulting cavity represented a significant loss of the design basis barrier integrity. The significance of this finding will be determined by the Significance Determination Process for the issue, which was begun following the Augmented Inspection Team activities (Section 4OA3.b.2.5).

- Significance to be Determined (TBD). The inspectors identified an apparent violation involving deficiencies in the licensee's Boric Acid Corrosion Control procedure, NG-EN-00324.

  This finding is more than minor because the corrosion of the reactor head and the resulting cavity represented a significant loss of the design basis barrier integrity. The significance of this finding will be determined by the Significance Determination Process for the issue, which was begun following the Augmented Inspection Team activities (Section 4OA3.b.3.1).

- Significance to be Determined (TBD). The inspectors identified an apparent violation involving multiple examples of failure to follow the boric acid corrosion control procedure.

This finding is more than minor because the corrosion of the reactor head and the resulting cavity represented a significant loss of the design basis barrier integrity. The significance of this finding will be determined by the Significance Determination Process for the issue, which was begun following the Augmented Inspection Team activities (Section 4OA3.b.3.2).

- Significance to be Determined (TBD). The inspectors identified an apparent violation involving two examples of failure to follow the station's corrective action program procedure.

  This finding is more than minor because the corrosion of the reactor head and the resulting cavity represented a significant loss of the design basis barrier integrity. The significance of this finding will be determined by the Significance Determination Process for the issue, which was begun following the Augmented Inspection Team activities (Section 4OA3.b.3.3).

- Significance to be Determined (TBD). The inspectors identified an apparent violation of 10 CFR 50.9 involving multiple examples of information provided to the Commission or required by the Commission's regulations to be maintained by the licensee that were not complete and accurate.

  Completeness and accuracy in the documents associated with this issue would have provided an earlier alert to licensee staff and the USNRC about the problems with control rod drive mechanism nozzle leakage or may have caused the USNRC to establish a different regulatory position concerning the urgency of inspections for the reactor pressure vessel head. The significance of this apparent violation requires additional review as specified in NUREG-1600, General Statement of Policy and Procedures for USNRC Enforcement (Section 4OA3.b.4).

B. Licensee Identified Findings

None

Report Details

4. OTHER ACTIVITIES (OA)

4OA3 Event Follow-up (93812)

Background

On March 6, 2002, Davis-Besse personnel notified the USNRC that reactor vessel head material, adjacent to a control rod drive nozzle, was significantly degraded. An Augmented Inspection Team (AIT) was chartered and dispatched to the site on March 12, 2002, to determine the facts and circumstances related to the reactor vessel head pressure boundary material degradation, and to identify any precursor indications of this condition. In accordance with USNRC procedures, the AIT charter did not include the verification of compliance with USNRC rules and regulations, nor the recommendation of enforcement actions. The AIT concluded its inspection on April 5, 2002, and issued USNRC Inspection Report 50-346/02-03 on May 3, 2002.

a. Inspection Scope

The purpose of this current inspection effort was to characterize any regulatory issues revealed during the AIT's activities. The inspection scope included a review of the AIT report, and also encompassed further reviews of licensee activities related to technical specification and procedural adequacy and compliance, and corrective action adequacy. This inspection was based on the facts and circumstances discussed in the AIT report and will not replicate chronologies or technical analyses unless needed to establish regulatory basis.

In addition to the information in the AIT report, subsequent questions were raised regarding completeness and accuracy of documents either required by the USNRC to be maintained by the licensee or submitted to the USNRC. Consequently, as licensee documents associated with this issue were reviewed for regulatory compliance, they were concurrently reviewed for completeness and accuracy. Because the risk significance of the reactor vessel head degradation has not been finalized, and several of these apparent violations remain under review by the USNRC, all of the findings will be classified as Unresolved Items in accordance with Manual Chapter 0612, "Power Reactor Inspection Reports."

b. Findings

b1. Technical Specification Reactor Coolant System (RCS) Operational Leakage

a. Introduction

The inspection identified an apparent violation, whose significance is yet to be determined, involving the Davis-Besse technical specification associated with operation of the plant with pressure boundary leakage from through-wall cracks in the RCS. This finding had a credible impact on safety and was characterized as an unresolved item (URI) pending USNRC determination of the significance of the apparent violation.

b. Description

On February 27 and March 5, 2002, the licensee notified the USNRC that their evaluation of ultrasonic test data, for axial indications in control rod drive mechanism nozzles Nos. 1, 2, and 3, confirmed that there was reactor pressure boundary leakage. Further investigation revealed a cavity adjacent to control rod drive penetration nozzle No. 3, approximately 5 to 7 inches long and 4 to 5 inches wide. Within this area, the 6.63 inch- thick low-alloy steel head had been corroded away, leaving only the stainless steel cladding layer on the inside of the reactor vessel head. Based on the length of the cracks, the amount of boric acid accumulation on the reactor vessel head, and the extensive corrosion of the reactor vessel head, it is clear that the unit operated well in excess of 36 hours with pressure boundary leakage.

At the time of plant shutdown, the unidentified primary coolant system leak rate was approximately 0.2 gpm, within the Technical Specification 3.4.6.2.b limit of 1.0 gpm. However, Technical Specification 3.4.6.2.a requires that primary coolant operational leakage shall be limited to “No PRESSURE BOUNDARY LEAKAGE” when in Modes 1-4. The associated action requires that the plant be placed in Hot Standby within 6 hours and in Cold Shutdown within the following 30 hours. Although the time the pressure boundary leakage began could not be precisely determined, it is clear that the leakage existed greater than the time frame that would have required plant shutdown.

c. Analysis

This issue represented a licensee performance deficiency because the licensee had multiple opportunities over a period of years to identify the leakage; consequently it was considered a finding. This finding is of more than minor safety significance because the pressure boundary leakage and resultant cavity in the reactor vessel head represented a loss of the design basis barrier integrity. Two cornerstones were impacted by this issue. The Barrier Integrity cornerstone was affected because the through-wall CRDM cracks compromised the reactor coolant pressure boundary and the Initiating Events cornerstone was impacted because cracking of the CRDM nozzles resulted in an increase in the likelihood of a loss of coolant accident (LOCA). The significance of this finding will be determined by the Significance Determination Process (SDP) for the issue, which was begun following the AIT activities.

d. Enforcement

Davis-Besse Technical Specification, Limiting Condition for Operation for Reactor Coolant System Operational Leakage, paragraph 3.4.6.2, states, in part, that RCS leakage shall be limited to no pressure boundary leakage, and that with any pressure boundary leakage, the unit is to be in Cold Shutdown within 36 hours. This issue is encompassed within the licensee root cause analysis, conducted for Condition Report (CR) 2002-01128. There is no current safety concern because the plant is presently shut down, cooled down, and defueled. Because the safety significance of the apparent violation has yet to be determined, the noncompliance will be classified as an unresolved item. This will be identified as URI 50-346/02-08-01, Reactor Operation with Pressure Boundary Leakage.

b.2 Corrective Action

b.2.1 Reactor Head Boric Acid Deposits

a. Introduction

The inspectors identified an apparent violation, whose significance is yet to be determined, involving failure to take adequate corrective action for a continuing buildup of boric acid deposits on the reactor head. This finding had a credible impact on safety and was characterized as a URI pending USNRC determination of the significance of the apparent violation.

b. Description

A series of Potential Condition Adverse to Quality Reports (PCAQR) and Condition Reports (CR) from 1990 through 2001 tracks recurrent identification of boric acid deposits on the reactor head. Section XI of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code IWA-5250, requires that the leakage source and areas of general corrosion be located when boric acid residues are detected on components. Section 4.3 of the AIT report (IR 50-346/02-03(DRS)) provides a chronology of reactor head inspections which identify boric acid deposits on the reactor head and how the licensee's engineering staff evaluated and dispositioned each occurrence.

Through much of the early operating years until the mid-1990s, CRDM flanges at Davis-Besse were prone to developing leaks during the operating cycle. This leakage was evidenced by boric acid deposits on the flange, the service structure, and on the reactor head (due to leakage which ran down the nozzles between the nozzles and the insulation). Beginning with the plant's sixth refueling outage (RFO) in 1990, the licensee began systematically correcting these leaks by replacing the flange gasket with a new design. As this program was implemented the frequency of CRDM flange leakage was reduced. By end of the tenth RFO in 1996, all the flanges had the redesigned gasket installed.

Beginning with the tenth RFO in 1996 and proceeding through the twelfth RFO in 2000, six PCAQRs and CRs documented the identification of boric acid deposits on the reactor head and the licensee engineering staff's disposition of the conditions. Each of these presented an opportunity to identify nozzle leakage. Collectively, they revealed a focus on CRDM flange leakage as the source of boric acid deposits despite evidence that the deposits must be from another source. For example, rust-colored deposits, indicative of iron, could not likely have come from the flanges which were stainless steel. Corrosion of the split ring nuts associated with the flanges would not have resulted in the quantity of corrosion products entrained in the deposits on the head. Finally, there were significant accumulations of boric acid on the head during operating cycles where CRDM flange leakage was non-existent or considered negligible.

c. Analysis

This issue represented a performance deficiency because the licensee failed to properly address, either individually or collectively, the continuing accumulation of large amounts of boric acid on the reactor head, a significant condition adverse to quality. This lack of adequate corrective action on the licensee's part, contributed to the failure to detect existing through-wall CRDM nozzle cracks.

This finding is more than minor because the corrosion of the reactor head and the resulting cavity represented a significant loss of the design basis barrier integrity. Two cornerstones were impacted by this issue. The Barrier Integrity cornerstone was affected because the through-wall CRDM cracks compromised the reactor coolant pressure boundary and the Initiating Events cornerstone was impacted because cracking of the CRDM nozzles resulted in an increase in the likelihood of a LOCA. The significance of this finding will be determined by the Significance Determination Process (SDP) for the issue, which was begun following the AIT activities.

d. Enforcement

10 CFR Part 50, Appendix B, Criterion XVI, states in part, that measures shall be taken to ensure that conditions adverse to quality such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and nonconformances are promptly identified and corrected. Criterion XVI also requires that for significant conditions adverse to quality, the measures shall assure that the cause of the condition is determined and that corrective actions are taken to preclude repetition. The failure to properly address the recurrent accumulation of boric acid deposits on the reactor head, a significant condition adverse to quality, contributed to the corrosion of the reactor head. There is no current safety concern because the plant is presently shut down, cooled down, and defueled. Because the safety significance of the apparent violation has yet to be determined, the noncompliance will be classified as an unresolved item. This will be identified as URI 50-346/02-08-02, Reactor Vessel Head Boric Acid Deposits.

b.2.2 Containment Air Cooler Deposits

a. Introduction

The inspectors identified an apparent violation whose significance is yet to be determined involving failure to take adequate corrective action for recurrent accumulations of boric acid on containment air cooler (CAC) fins. These accumulations resulted in reduced air flow through the cooler which was indicated by decreasing plenum pressure. This finding had a credible impact on safety and was characterized as an unresolved item pending USNRC determination of the significance of the apparent violation.

b. Description

The inspectors reviewed one PCAQR and three CRs spanning the period November 1998 through January 2001. Section 5.3 of the AIT report

(IR 50-346/02-03(DRS)) provides a chronology of CAC fouling and how the licensee's engineering staff evaluated and dispositioned each occurrence.

At the onset of the leak, CRDM nozzle leakage created an aerosol of steam, boric acid, and other contaminants in the air space above the head. The steam and aerosol particles were picked up by the service structure ventilation system intake and distributed throughout the containment. The CACs subsequently condensed the steam and the boric acid plated out on the cooler fins. Later, as the corrosion of the head progressed, the aerosol consisted of steam, boric acid, and corrosion particles. Consistent with this, the deposits on the cooler fins changed color from white to red-brown.

The licensee's attempts to address this phenomenon focused on maintaining operability of the coolers through frequent cleanings of the coolers. Of the four corrective action documents examined, only one considered the source of the boric acid deposits; however, no actions to investigate the source were prescribed. Of particular significance was the licensee's evaluation of the July 1999 appearance of rust-colored deposits. The licensee continued to attribute the boric acid deposits to CRDM flange leakage; the discoloration of the boric acid was attributed to migration of the surface corrosion on the CACs into the boric acid and the aging of the boric acid itself.

c. Analysis

This issue represented a performance deficiency because the licensee failed to properly address, either individually or collectively, the cause for the recurrent deposition of boric acid on CAC fins, nor the change in the color of the deposits, although the change was indicative of carbon steel corrosion. This lack of adequate corrective action on the licensee's part contributed to their failure to detect existing through-wall CRDM nozzle cracks and the reactor pressure vessel head corrosion.

This finding is more than minor because the corrosion of the reactor head and the resulting cavity represented a significant loss of the design basis barrier integrity. Two cornerstones were impacted by this issue. The Barrier Integrity cornerstone was affected because the through-wall CRDM cracks compromised the reactor coolant pressure boundary and the Initiating Events cornerstone was impacted because cracking of the CRDM nozzles resulted in an increase in the likelihood of a LOCA. The significance of this finding will be determined by the Significance Determination Process (SDP) for the issue, which was begun following the AIT activities.

d. Enforcement

10 CFR Part 50, Appendix B, Criterion XVI, states in part, that measures shall be taken to ensure that conditions adverse to quality such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and nonconformances are promptly identified and corrected. Criterion XVI also requires that for significant conditions adverse to quality, the measures shall assure that the cause of the condition is determined and that corrective actions are taken to preclude repetition. The failure to properly address the recurrent deposits of boric acid deposits on the CAC fins, a significant condition adverse to quality, contributed to the corrosion of the reactor head.

There is no current safety concern because the plant is presently shut down, cooled down, and defueled. Because the safety significance of the apparent violation has yet to be determined, the noncompliance will be classified as an unresolved item. This will be identified as URI 50-346/02-08-03, “Containment Air Cooler Boric Acid Deposits.”

b.2.3 Radiation Element Filter Deposits

a. Introduction

The inspectors identified an apparent violation, whose significance is yet to be determined, involving failure to take adequate corrective action for repeated clogging of radiation element filters, although a sample of the filter deposits revealed iron oxides, and radionuclides indicative of reactor coolant. This finding had a credible impact on safety and was characterized as a URI pending USNRC determination of the significance of the apparent violation.

b. Description

Starting in May 1999, yellowish-brown material began to accumulate on the radiation element filters, causing repetitive, degraded performance of the containment radiation monitors due to low flow. Condition Report 99-0882 issued on May 13, 1999, identified low flow conditions on radiation element 4597BA. The CR noted that the apparent cause was “boric acid particles collecting on the filter at a very high rate.” The licensee subsequently sent a sample of the material for analysis, and in August 1999, Southwest Research Institute’s Report No. 18-2321-190 concluded that the deposits on the filters were a powdery iron oxide and were likely corrosion products from an iron-based component within the system. The licensee initiated Condition Report 1999-1300 on May 23, 1999, to document this issue, and noted that Plant Engineering was to issue an Action Plan for the 12 RFO which would include containment walkdowns to identify possible sources of the rust particles. Sargent and Lundy was subsequently asked to review the report, and on November 5, 1999, their response letter stated:

> “The fineness of the iron oxide (assumed to be ferric oxide) particulate would indicate it probably was formed from a very small steam leak. The particulate was likely originally ferrous hydroxide in small condensed droplets of steam and was oxidized to ferric oxide in the air before it settled on the filters;” and “the iron oxide does not appear to be coming from the general corrosion of a bare metal surface in containment or from steam impingement on a metal surface.”

Although the licensee conducted containment entries at power to identify the source of the apparent steam leak, it was never identified. After the 12th RFO in May 2000, the radiation element filters continued to clog with corrosion products. CR 01-1110 was issued on April 23, 2001, to document continued clogging of the radiation monitor filters due to boric acid build-up. Corrective action was to move the sample point. CR 01-1822 dated July 23, 2001, documented increased frequency of monitor filter change outs again, due to boric acid clogging. Disposition of this CR was to continue to change the filters until an upcoming refueling outage could identify the source of RCS leakage. CR 01-2795, dated October 22, 2001, again identified a high frequency of filter clogging, noting that previous corrective actions were unsuccessful. Corrective action for this CR was to perform a temporary modification (TM 01-0019) to remove the filter cartridge.

c. Analysis

This issue represented a performance deficiency because the licensee failed to take appropriate corrective action (identify and repair the source of the RCS leakage) for a significant condition adverse to quality, in that, the filters clogged with a material indicative of RCS leakage and corrosion products and this continued for more than two years. It has subsequently been concluded that the material clogging the filters was from the ongoing corrosion of the reactor vessel head. While some containment walkdowns were conducted, the source of the RCS leakage was not identified. Furthermore, the corrective actions implemented for the CRs written on this problem appeared to focus on the operability of the radiation monitors and not the root cause (i.e., RCS leakage and corrosion). This lack of adequate corrective action on the licensee's part, contributed to their failure to detect existing through-wall CRDM nozzle cracks.

This finding is more than minor because it affected the Initiating Events cornerstone objective in that cracking of CRDM nozzles represented an increase in the likelihood of a LOCA. The Barrier Integrity cornerstone was also affected in that CRDM cracks resulted in leakage through the reactor coolant pressure boundary. The significance of this finding will be determined by the Significance Determination Process (SDP) for the issue, which was begun following the AIT activities.

d. Enforcement

10 CFR Part 50, Appendix B, Criterion XVI, requires, in part, that conditions adverse to quality be promptly identified and corrected. Criterion XVI also requires that for significant conditions adverse to quality, the measures shall assure that the cause of the condition is determined and that corrective actions are taken to preclude repetition. Contrary to this, the licensee failed to correct the condition identified in CR 99-0882, initiated on May 13, 1999. As of February 16, 2002, the source of the RCS leakage had not been identified and corrected. There is no current safety concern because the plant is presently shut down, cooled down, and defueled. Because the safety significance of this apparent violation has yet to be determined, it will be classified as an unresolved item. This will be identified as URI 50-346/02-08-04, Radiation Element Filters.

b.2.4 Service Structure Modification

a. Introduction

An apparent violation whose significance is yet to be determined, was identified for the failure to implement the corrective action procedure and take prompt corrective action for a condition adverse to quality. The licensee failed to implement a modification to permit complete inspection and cleaning of the reactor vessel head and CRDM nozzles. This finding had a credible impact on safety and was characterized as a URI pending USNRC determination of risk significance.

b. Description

USNRC Generic Letter 88-05 was issued on March 17, 1988, and notified licensees of the potential for boric acid degradation of carbon steel reactor pressure boundary components. On May 28, 1993, BAW issued document BAW 10190P, “Safety Evaluation for BAW Design Reactor Vessel Head Control Rod Drive Mechanism Nozzle Cracking.” This document stated that B&WOG utilities developed plans to visually inspect the CRDM nozzle area to determine if through wall cracking had occurred. If any leaks or boric acid crystal deposits were located, an evaluation of the source of the leak and the extent of any wastage was to be evaluated. The USNRC notified NUMARC of the results of the USNRC safety evaluation related to this subject on November 19, 1993. This safety evaluation concluded that there was no current safety concern for cracking of CRDM penetrations. The conclusion was predicated on the performance of the visual inspection activities requested in Generic Letter 88-05.

Modification 90-0012 was initiated during RFO 6 on March 21, 1990, to install multiple access ports in the service structure to permit cleaning and inspection of the reactor head. At this time, it was noted that boric acid had leaked down from the CRD flanges and accumulated on the head, thus subjecting the head to potential degradation. This modification was not performed during the next two refuel outages (RFO 7 and 8) and was subsequently voided on September 27, 1993, noting that the head had been successfully inspected with cameras and cleaned during the previous three refuel outages (utilizing existing small “mouse holes” holes in the service structure).

Modification 94-0025 was initiated on May 27, 1994, again to install access ports in the service structure to permit cleaning and inspection of the head. Reasons cited for the modification were that video inspections (through existing “mouse holes”) were difficult and not always adequate and that inspection and cleaning of 100 percent of the head could not be accomplished without performing the modification. This modification was not performed during RFO 9 (October 1, 1994) and a decision was made on June 15, 1995, to defer scheduling it for RFO 10 (April 8, 1996) “pending further industry information/investigation concerning actual benefit.” During a meeting of the Work Scope Committee (WSC) on February 20, 1997, a decision was made to reschedule the modification for RFO 12 (April 1, 2000). Subsequently, during a meeting of the WSC on September 17, 1998, the modification was again rescheduled, this time, for RFO 13 (February 16, 2002).

Potential Condition Adverse to Quality 96-0551 was initiated on April 21, 1996, to address the inability to comply with some of the provisions of NG-EN-00324 (Boric Acid Corrosion Control Program) with respect to inspection and cleaning of the head during RFO 10. This PCAQR stated, in part, “Since the boric acid deposits are not cleaned it is difficult to distinguish whether the deposits occurred because of the leaking flanges or the leaking CRDM. This situation represents an adverse trend with the potential for greater than marginal consequences.” The PCAQR further states, “The extent of the inspection was limited to approximately 50 to 60 percent of the head area because of the restrictions imposed by the location and size of mouse holes.” Modification 94-0025 was specified as the corrective action for this PCAQR.

c. Analysis

This issue represented a performance deficiency because the licensee failed to take corrective action (install the access port modification) for a condition adverse to quality. As noted above, from the information in Modification 94-0025 and PCAQR 96-0551, it is apparent that complete inspection and cleaning of the head could not be performed without installing the access ports. As of February 16, 2002, the modification had not been performed, the head had not been completely inspected, and the head had not been completely cleaned. This lack of action on the licensee's part, contributed to their failure to detect existing through-wall CRDM nozzle cracks.

This finding is more than minor because it affected the Initiating Events cornerstone objective in that cracking of CRDM nozzles represented an increase in the likelihood of a LOCA. The Barrier Integrity cornerstone was also affected in that CRDM cracks resulted in leakage through the reactor coolant pressure boundary. Furthermore, the failure to provide for adequate inspection and cleaning of the head was a contributing factor to the head degradation. The significance of this finding will be determined by the SDP for the issue, which was begun following the AIT activities.

d. Enforcement

10 CFR Part 50, Appendix B, Criterion V, requires, in part, that activities affecting quality be accomplished in accordance with written procedures. The licensee failed to follow its corrective action procedure and correct the condition identified on April 21, 1996 (inability to fully inspect the head and CRDM nozzles), in that, as of February 16, 2002, the corrective action (modification of the service structure) had not been accomplished. There is no current safety concern because the plant is presently shut down, cooled down, and defueled. The service structure has since been modified to permit complete inspection and cleaning of the head. Because the safety significance of this apparent violation has yet to be determined, it will be classified as an unresolved item. This will be identified as URI 50-346/02-08-05, Service Structure Modification Delay.

b.2.5 Reactor Coolant System Unidentified Leakage Trend

a. Introduction

The inspectors identified a finding whose significance is yet to be determined involving failure to follow the corrective action procedure and complete a prescribed corrective action for adverse trends in RCS unidentified leakage. This finding had a credible impact on safety and was characterized as an unresolved item pending USNRC determination of the significance of the apparent violation.

b. Description

In 1998, shortly after completing the eleventh RFO, the licensee identified a sharp rise in RCS unidentified leakage which had been relatively stable at 0.05 gpm. This was attributed to a temporary modification which bypassed a pressurizer relief valve drain line and allowed leakage past the relief valves to be vented directly into the containment atmosphere. This leakage collected in the normal sump and added to the unidentified leakage, which increased to a maximum of 0.8 gpm. During a mid-cycle outage in May of 1999, the licensee resolved this issue by installing new rupture disks and reconnecting the drain line. This resulted in a decrease in unidentified leakage. However, the unidentified leakage returned to levels between 0.15 and 0.25 gpm.

The inspectors reviewed a series of four CRs which demonstrated that the licensee was aware of the increase in unidentified leakage, and becoming increasingly concerned by their inability to identify the source. The engineering evaluation into the issue became more involved with each succeeding CR. CR 2001-2862 contained a detailed evaluation and a corrective action to develop a containment inspection plan for the forthcoming RFO. This inspection plan was completed and documented in the CR. It involved coordination of four actions: the Mode 3 (reactor shutdown, normal operating temperature and pressure) walkdown, mode 5 (cold shutdown) RCS walkdowns, Boric acid corrosion control walkdowns, and the ASME VT-2 examinations. The inspectors determined that proposed corrective actions associated with RCS unidentified leakage were adequate; however, a key action, the mode 3 walkdown was subsequently canceled. This significantly reduced the quality of the proposed corrective action to the point where it was no longer adequate.

c. <u>Analysis</u>

This issue represented a licensee performance deficiency because elimination of a key component of what was an adequate proposed corrective action rendered the proposal inadequate. Consequently, this was considered a finding. This finding was of more than minor safety significance because the corrosion of the reactor head and the resulting cavity represented a significant loss of the design basis barrier integrity. Two cornerstones were impacted by this issue. The Barrier Integrity cornerstone was affected because the through-wall CRDM cracks compromised the reactor coolant pressure boundary and the Initiating Events cornerstone was impacted because cracking of the CRDM nozzles resulted in an increase in the likelihood of a LOCA. The significance of this finding will be determined by the SDP for the issue, which was begun following the AIT activities.

d. <u>Enforcement</u>

The licensee failed to follow the corrective action procedure and implement an effective corrective action for adverse trends in RCS unidentified leakage. There is no current safety concern because the plant is presently shut down, cooled down, and defueled. Because the safety significance of this finding has yet to be determined, it will be classified as an unresolved item. This will be identified as URI 50-346/02-08-06, “Reactor Coolant System Unidentified Leakage Trend.”

b.3 Procedures

b.3.1 Procedures Not Appropriate to the Circumstances

a. Introduction

The inspectors identified an apparent violation whose significance is yet to be determined, involving deficiencies in the licensee's Boric Acid Corrosion Control procedure, NG-EN-00324. This finding had a credible impact on safety and was characterized as an unresolved item pending USNRC determination of the significance of the apparent violation.

b. Description

The inspectors reviewed the original and subsequent revisions, up to the date of discovery of the reactor head conditions, of the licensee's Boric Acid Corrosion Control Program, NG-EN-00324. The purpose of the review was to assess the adequacy of the procedure and determine whether any deficiencies may have contributed to the event. The inspectors noted the following weaknesses in the procedure:

a. The procedure had a clear focus on bolted, flanged connections. Seven of nine principal locations in Section 6.1.1, the definition of an RCS pressure boundary component in Section 4.9, and the definitions of minor (Section 4.2), moderate (Section 4.3), and substantial (Section 4.4) leakage all contained references to bolted connections.

b. In Section 6.3.4, the procedure directed preparation of a CR, repair tag, or work order if a "detailed inspection" was warranted, but guidance, specifications, or thresholds for initiating a "detailed inspection" were inadequate.

c. The inspectors determined that preparation of a repair tag or work order in lieu of a CR was inappropriate because it only addressed the symptom by fixing the leak rather than evaluate why the leak was occurring.

d. Qualifications for Plant Engineering staff conducting inspections and evaluations were not addressed. This resulted in inconsistencies in inspection techniques, observations, recording of results, and evaluations.

e. In section 6.3.1.f, the procedure exempted stainless steel or inconel components from further examination related to boric acid corrosion, unless the examination was during an ASME Section XI test which might require a bolting examination. However, there was industry experience dating back to 1990, including an USNRC Information Notice, identifying primary water stress corrosion cracking of stainless steel, due to boric acid attack, as a concern.

f. The procedure did not require maintenance of any documents, such as checklists or evaluations, although the procedure is quality related and Davis-Besse Supplemental Procedure Requirements/Guidance, NG-QS-00120 stated in Attachment 2, Section1.2.b that these procedures were used to assure safe operation.

The inspectors determined that these weaknesses collectively contributed to the corrosion of the reactor head, either through narrowing the scope of inspection or failing to provide adequate instruction for carrying out activities.

c. Analysis

This issue represented a licensee performance deficiency because the weaknesses in the procedure contributed to the failure, over a period of years, by the licensee's engineering staff to properly identify and evaluate the leaking CRDM nozzle and the expanding cavity in the reactor head. Consequently, this was considered a finding. This finding was of more than minor safety significance because the cavity in the reactor vessel head represented a loss of the design basis barrier integrity. The significance of this finding will be determined by the SDP for the issue, which was begun following the AIT activities.

d. Enforcement

10 CFR Part 50, Appendix B, Criterion V, states, in part, 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. NG-EN-00324, "Boric Acid Corrosion Control Program," Revisions 0 through 2, were classified as a quality procedure under the licensee's procedure administrative system and were not appropriate to the circumstances in that deficiencies in the procedure contributed to the failure to detect and address corrosion of the reactor head. There is no current safety concern because the plant is presently shut down, cooled down, and defueled. Because the safety significance of the apparent violation has yet to be determined, the noncompliance will be classified as an unresolved item. This will be identified as URI 50-346/02-08-07, "Inadequate Boric Acid Corrosion Control Program Procedure."

b.3.2 Failure to Follow Boric Acid Corrosion Control Program Procedure

a. Introduction

The inspectors identified an apparent violation whose significance is yet to be determined involving multiple examples of failure to follow the boric acid corrosion control procedure. This finding had a credible impact on safety and was characterized as an unresolved item pending USNRC determination of the significance of the apparent violation.

b. Description

The inspectors reviewed a series of PCAQRs and CRs that documented occurrences of the licensee's failure to adhere to the instructions in the Boric Acid Corrosion Control Program procedure.

- PCAQR 96-0551 (in RF010) recorded that a visual inspection of the head showed boric acid accumulation on the head and that CRDM nozzle No. 67 had rust-colored deposits where it penetrated the head. The PCAQR further records that inspection of CRDM nozzle No. 67 flange showed no signs of leakage during the operating cycle, signifying that the "present deposits were the result of leakage from previous operating cycles." The conclusion drawn from these two statements was that, as a minimum, boric acid deposits were not removed from the head, nor was the base metal inspected for corrosion during RF09 as directed by the boric acid corrosion control program procedure.

- PCAQRs 98-0649 and 98-0767 (in RF011) both recorded visual inspections of the reactor head using a video camera on April 17, 1998, and April 24, 1998, respectively. Both inspections revealed boric acid residue on the head. PCAQR 98-0649 focused on CRDM flange D-10 which was determined to have "minor" leakage, based on the amount of boric acid on the flange. A review of unidentified leakage was conducted and average unidentified leakage during the previous operating cycle was 0.05 gallons per minute. Based on this information, repair of flange D-10 was deferred. PCAQR 98-0767 recorded that "most of the head area was covered with an uneven layer or boric acid along with some lumps of boric acid. The color of the layer and the lumps varied from rust brown to white. The rust or brown color is an indication of the old boric acid deposits." The conclusion drawn from these two PCAQRs is that boric acid deposits were left on the head at the end of the tenth RFO and that the base metal under these deposits was not inspected as directed by the boric acid corrosion control program procedure.

- PCAQR 98-0767 (RF011) records that the reactor head was cleaned "as best we can." Later the PCAQR records that an inspection after the cleaning showed there were boric acid deposits left on the head after the cleaning. At the end of RF011, the base metal under these deposits was not inspected as directed by the boric acid corrosion control program procedure.

- CRs 2000-0781, 2000-0782, and 2000-1037 (RF012) were all written to document the extensive build-up of boric acid residue on the reactor head.

- CR 2000-0782 describes the conditions in detail and photographs of the head near the closure studs were included. Among the corrective actions specified in the condition report was cleaning boric acid deposits off the head in accordance with work order 00-001846-000. A video tape made after this cleaning showed that a thick layer of red/brown boric acid deposits remained around the nozzles near the center of the head. At the end of the twelfth RFO, the base metal under these deposits was not inspected as directed by the boric acid corrosion control program procedure.

c. Analysis

This issue represented a licensee performance deficiency because the recurrent failures, by the licensee's engineering staff, to remove boric acid deposits and inspect the base metal of the reactor head as directed by the boric acid corrosion control procedure, resulted in the perpetuation of the CRDM nozzle leak and the development of the expanding cavity in the reactor head. Consequently, this was considered a finding. This finding was of more than minor safety significance because the cavity in the reactor vessel head represented a loss of the design basis barrier integrity. The significance of this finding will be determined by the SDP for the issue, which was begun following the AIT activities.

d. Enforcement

10 CFR Part 50, Appendix B, Criterion V, states, in part, 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. The licensee's engineering staff failed, on multiple occasions, to remove boric acid deposits and inspect the base metal of the reactor head as directed by NG-EN-00324, Revision 2, "Boric Acid Corrosion Control Program." This issue is encompassed within the licensee root cause analysis, prescribed by CR 2002-01128. There is no current safety concern because the plant is presently shut down, cooled down, and defueled. Because the safety significance of the apparent violation has yet to be determined, the noncompliance will be classified as an unresolved item. This will be identified as URI 50-346/02-08-08, "Failure to Follow Boric Acid Corrosion Control Program Procedure."

b.3.3 Failure to Follow Corrective Action Program Procedure

a. Introduction

The inspectors identified an apparent violation whose significance is yet to be determined involving two examples of failure to follow the station's corrective action program procedure. This finding had a credible impact on safety and was characterized as an unresolved item pending USNRC determination of the significance of the apparent violation.

b. Description

CRs 2000-0782, and 2000-1037 (RF012) were written to document the extensive build-up of boric acid residue on the reactor head. CR 2000-0782 described the conditions in detail and photographs of the head near the closure studs were included. CR 2000-1037 described the analysis (deferring in part to CR 2000-0782) and the cleaning effort. In both CRs the extent and significance of the boric acid deposits, and that such build-up was recurrent, were clear. The quantity of boric acid deposits accumulated was highly unusual, extensive corrective actions were necessary, and an adverse repetitive trend existed.

Attachment 2, Categorization of Condition Report, to NG-NA-00702, Revision 3, "Corrective Action Program," provides guidance and examples for characterization of condition reports as significant, important, routine, or non-conditions adverse to quality. Among the examples of significant conditions are:

- Issues of collective significance that considered individually may not be significant, but as a whole indicate problems that warrant root cause investigation and corrective action to prevent recurrence.

- Substantial deviations, deficiencies in construction or design, damage such that extensive corrective actions are required.

- A repetitive or adverse trend exists.

During RF012, the engineering staff was aware of the continuing accumulation of boric acid on the reactor head; the problem had been documented in PCAQRs and CRs since 1996. Although the quantity of boric acid deposits was a substantial deviation from acceptable operating conditions for the reactor head, the history of reactor head boric acid deposits revealed a significant problem, and that a repetitive, adverse trend existed. Both condition reports were classified as routine.

c. Analysis

This issue represented a licensee performance deficiency. A proper characterization of condition reports 2000-0782 and 2000-1037 would have resulted in a formal root cause evaluation as prescribed by the licensee's Root Cause Analysis Reference Guide. Incorrectly characterizing these CRs as routine resulted in an apparent cause determination with no required corrective actions to prevent recurrence; an opportunity to identify the true nature of the leak and the growing cavity in the head was missed. Consequently, this was considered a finding. This finding was of more than minor safety significance because the cavity in the reactor vessel head represented a loss of the design basis barrier integrity. The significance of this finding will be determined by the SDP for the issue, which was begun following the AIT activities.

d. Enforcement

10 CFR Part 50, Appendix B, Criterion V, states, in part, 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. The licensee failed to properly characterize CRs 2000-0782 and 2000-1037 as significant conditions adverse to quality, in accordance with the guidance contained in the licensee's corrective action program procedure. This issue is encompassed within the licensee root cause analysis, prescribed by CR 2002-01128. There is no current safety concern because the plant is presently shut down, cooled down, and defueled. Because the safety significance of the apparent violation has yet to be determined, the noncompliance will be classified as an unresolved item. This will be identified as URI 50-346/02-08-9, "Failure to Follow Corrective Action Program Procedure."

b.4 Completeness and Accuracy of Information

a. Introduction

In addition to the information in the AIT report subsequent questions were raised regarding completeness and accuracy of documents either required by the USNRC to be maintained by the licensee or submitted to the USNRC. Consequently, as licensee documents associated with this issue were reviewed for regulatory compliance, they were concurrently reviewed for completeness and accuracy in all material respects as required by 10 CFR 50.9, “Completeness and Accuracy of Information.”

b. Description

The inspectors’ review of the extensive documentation associated with this issue revealed a series of examples of information provided to the Commission or required by the Commission’s regulations to be maintained by the licensee that were not complete and accurate.

1. The cancellation of Modification 90-012 was based on a statement in a Document Void Request, approved September 23, 1993: “Current inspection techniques using high-powered cameras preclude the need for inspection ports, additionally, cleaning of the reactor vessel head during last three outages was completed successfully without requiring access ports.” This statement is inaccurate because boric acid deposits were left on the head at the end of both the seventh and eighth refueling outages, the two outages preceding this statement.

2. PCAQR 98-0649, dated April 18, 1998, made the statement, “Accumulation of boric acid on the reactor vessel caused by leaking CRDMs has not resulted in any boric acid corrosion. This was identified through inspections following reactor vessel head cleaning in past outages.” This statement is inaccurate because areas of the reactor head were not cleaned of boric acid deposits nor was the base metal under all the deposits inspected.

3. PCAQR 98-0649 also contained the following statement, “Additionally, B&W documentation discussing CRDM nozzle cracking further stated that boric acid deposits on the head caused by leaking CRDM flanges would not result in head corrosion.” The PCAQR did not state which B&W document was being referenced. The inspectors reviewed the following documents:

    a. 51-1219275-01, CRDM Leakage Detection Evaluation, December 13, 1993

    b. 51-1229638-00, Boric Acid Corrosion Data Summary and Evaluation, April 15, 1994

    c. BAW-10190P, Safety Evaluation for B&W-Design Reactor Vessel Head Control Rod Drive Mechanism Nozzle Cracking

d. BAW-10190P, Addendum 1, External Circumference Crack Growth Analysis for B&W-Design Reactor Vessel Head CRDM Nozzles

e. BAW-10190P, Addendum 2, Safety Evaluation for Control Rod Drive Mechanism Nozzle J-Groove Weld

f. B&W Materials Committee Report 51-1201160-00, Alloy 600 SCC Susceptibility: Scoping Study of Components at Crystal River 3

g. B&W Report 51-1218440-00, Alloy PWSCC Time-to-Failure Models

h. B&W Report 51-1219143-00, CRDM Nozzle Characterization

i. BAW-2301, B&WOG Integrated Response to USNRC Generic Letter 97-01, Degradation of Control Rod Drive Mechanism Nozzle and Other Vessel Closure Head Penetrations

The inspectors did not find that statement in any of those nine documents, nor did the licensee identify the source document for that statement.

4. PCAQR 98-0767, dated April 25, 1998, Section 4A, Item F, stated, “The boric acid deposits were removed from the head.” This is incorrect information; it has been acknowledged that the head was not completely cleaned at the end of eleventh RFO.

5. Condition report CR 2000-1037, dated April 17, 2000, page 6 of 7, under Remedial Actions stated, “Accumulated boron deposited between the reactor head and the thermal insulation was removed during the cleaning process performed under W.O. 00-001846. No boric acid induced damage to the head surface was noted during the subsequent inspection.” This statement was inaccurate in that the accumulated boric acid was only removed from some areas of the head, and the subsequent inspection of the head surface for boric acid induced damage was only for that portion of the head where the boric acid deposit had been removed.

6. Work Order 00-001846-000, “Clean Boron Accumulation from Top of Reactor Head and Top of Insulation,” dated April 25, 2000, was prepared and issued to clean the reactor head as directed by the boric acid corrosion control procedure. The Work Order log stated, “work performed without deviation.” This was inaccurate since CRs clearly indicated that boric acid deposits were left on the head after the cleaning.

7. QA Audit report AR-00-OUTAG-01, dated July 7, 2000, stated, in part, “Boric Acid Corrosion Control Checklists and Condition Reports were initiated by inspectors when prudent to document and evaluate boric acid accumulation and leaks. Boric acid leakage was adequately classified and corrected when appropriate. Engineering displayed noteworthy persistence in ensuring boric acid accumulation from the reactor head was thoroughly cleaned.” This audit report contains inaccurate information: (1) the reactor head was not thoroughly

cleaned during the outage; (2) a boric acid corrosion control checklist was not prepared for the boric acid left on the head after the cleaning attempt; and (3) the boric acid accumulation and leaks were not identified, properly classified, nor corrected.

8. Davis-Besse letter, Serial 2731, September 4, 2001, Response to Bulletin 2001-01, contained the following four inaccuracies:

    a. The response to Item 1.c on page 2 of 19 contained the statement that "the minimum gap being at the dome center of the RPV head where it is approximately 2 inches, and does not impede a qualified visual inspection." This is contradicted by statements in several PCAQRs, most notably 94-0295, which prompted the reintroduction of the service structure modification, and 96-0551 which clearly stated that inspection capability at the top of the head was limited. This limitation was caused by the restricted access to the area through the service structure "weep holes", the curvature of the reactor pressure vessel head, and by the limited space to manipulate a camera due to the insulation that creates the two inch gap.

    b. Item 1.d of the Bulletin directed inclusion of a description of any limitations (insulation or other impediments) to accessibility of the bare metal of the RPV head for visual examinations. The response was incomplete in that it did not mention that accessibility to the bare metal of the reactor head was impeded by the significant accumulations of boric acid deposits in both the eleventh and twelfth RFOs.

    c. Item 1.d of the bulletin directed a discussion of the findings of vessel head inspections. The response to this on page 3 for the twelfth RFO was that inspection of the RPV head/nozzles indicated some accumulation of boric acid deposits. This was a mischaracterization of the accumulations as evidenced by the pictures and the video examination of conditions on the head at the beginning and ending of the outage.

    d. Additionally on page 3, the response stated, "The boric acid deposits were located beneath the leaking flanges with clear evidence of downward flow. No visible evidence of nozzle leakage was detected." This was inaccurate, in that the boric acid deposits were not all located under leaking flanges and there was no clear evidence of downward flow for all nozzles. The deposits were not limited to the area beneath the flanges as implied by that statement and, in fact, the build-up was so significant that all of the nozzles could not be inspected. There was no basis for stating that no visible evidence of nozzle leakage was detected.

9. Davis-Besse letter, Serial 2735, October 17, 2001, "Supplemental Response to Bulletin 2001-01," stated, "In May 1996, during a refueling outage, the RPV head was inspected. No leakage was identified, and these results have been recently verified by a re-review of the video tapes obtained from that inspection. The

RPV head was mechanically cleaned at the end of the outage. Subsequent inspections of the RPV head in the next two refueling outages (1998 and 2000), also did not identify any leakage in the CRDM nozzle-to-head areas that could be inspected. Video tapes taken during these inspections have also been re-reviewed." These statements were inaccurate in that they implied that the head was completely cleaned and inspected. The RPV head could not be completely inspected, as evidenced by PCAQR 96-0551. The RPV head was not cleaned as evidenced by PCAQR prepared at the start of the 1998 outage which stated that there were old boric acid deposits on the head.

c. Analysis

This issue represented a licensee performance deficiency. Completeness and accuracy of information are essential to the ability of the USNRC to establish a regulatory position on issues which can affect the health and safety of the general public. Completeness and accuracy in the documents listed above may have provided an earlier alert to the licensee staff and the USNRC about the problems with CRDM nozzle leakage or may have caused the USNRC to establish a different regulatory position concerning the urgency of inspections for the RPV head. Consequently, this was considered a finding of more than minor safety significance because the cavity in the reactor vessel head represented a loss of the design basis barrier integrity.

d. Enforcement

10 CFR 50.9 requires that information provided to the Commission by a licensee or information required by statute or by the Commissions regulations, order, or license conditions maintained by the licensee shall be complete and accurate in all material respects. The examples listed contain incomplete or inaccurate information material to the USNRC. The significance of these examples requires additional review as specified in NUREG-1600, "General Statement of Policy and Procedures for USNRC Enforcement." Because the safety significance of the apparent violation has yet to be determined, the issue will be classified as an unresolved item and will be identified as URI 50-346/02-08-10, "Completeness and Accuracy of Information."

4OA6 Management Meetings

Exit Meeting Summary

The inspectors presented the inspection results to Mr. L. Myers and other members of licensee management and staff at the conclusion of the inspection on August 9, 2002. The licensee acknowledged the information presented. Proprietary information reviewed and retained by the inspectors was identified.

KEY POINTS OF CONTACT

DAVIS-BESSE

D. Baker, LCM(A) Manager
R. Fast, Plant Manager
J. Grabnar, Design Basis Engineering Manager
D. Gudger, Learning Organization Manager
D. Haskins, Human Resources Manager
S. Loehlein, Principal Nuclear Consultant
L. Myers, Site Vice President
L. Pearce, Vice President, Oversight
J. Powers, Engineering Director
P. Roberts, Maintenance Manager
M. Roder, Operations Manager
J. Rogers, Plant Engineering Manager
R. Slyker, Licensing Staff Engineer
H. Stevens, Quality Assurance Manager
G. Wolf, Licensing Staff Engineer

NUCLEAR REGULATORY COMMISSION

J. Grobe, Chairman, Davis-Besse Oversight Panel
C. Lipa, Chief, Reactor Projects Branch 4
S. Thomas, Senior Resident Inspector

LIST OF ACRONYMS USED

AIT	Augmented Inspection Team
ASME	American Society of Mechanical Engineers
B&W	Babcock and Wilcox
CAC	Containment Air Cooler
CR	Condition Report
CRDM	Control Rod Drive Mechanism
EPRI	Electric Power Research Institute
GL	Generic Letter
gpm	Gallon Per Minute
LOCA	Loss of Coolant Accident
PCAQR	Potential Conditions Adverse to Quality Report
PDR	Public Document Room
RCS	Reactor Coolant System
RE	Radiation Element
RFO	Refueling Outage
RPV	Reactor Pressure Vessel
SDP	Significance Determination Process
URI	Unresolved Item
USNRC	U. S. Nuclear Regulatory Commission

ITEMS OPENED

50-346/2002-08-01	URI	Reactor Operation with Pressure Boundary Leakage
50-346/2002-08-02	URI	Reactor Vessel Head Boric Acid Deposits
50-346/2002-08-03	URI	Containment Air Cooler Boric Acid Deposits
50-346/2002-08-04	URI	Radiation Element Filters
50-346/2002-08-05	URI	Service Structure Modification Delay
50-346/2002-08-06	URI	Reactor Coolant System Unidentified Leakage Trend
50-346/2002-08-07	URI	Inadequate Boric Acid Corrosion Control Program Procedure
50-346/2002-08-08	URI	Failure to Follow Boric Acid Corrosion Control Program Procedure
50-346/2002-08-09	URI	Failure to Follow Corrective Action Program Procedure
50-346/2002-08-10	URI	Completeness and Accuracy of Information

LIST OF DOCUMENTS REVIEWED

The following is a list of licensee documents reviewed during the inspection, including documents prepared by others for the licensee. Inclusion on this list does not imply that USNRC inspectors reviewed the documents in their entirety, but, rather that selected sections or portions of the documents were evaluated as part of the overall inspection effort. Inclusion on this list does not imply USNRC acceptance of the document, unless specifically stated in the inspection report.

Procedures

NG-EN-00324	Boric Acid Corrosion Control, Revisions 1, 2, and 3
NG-NA-00305	Operating Experience Program
NG-NA-00702	Corrective Action Program, Revision 3
DB-PF-00204	ASME Section XI Pressure Testing, Revision 3
DB-OP-01200	Reactor Coolant System Leakage Management, Revision 3

Potential Conditions Adverse to Quality Reports (PCAQR)

1991-0353	Boron on Reactor Vessel Head
1992-0072	CAC Cleaning
1993-0132	Reactor Coolant Leakage from CRD Flange
1994-0295	Improper Closure of the Nozzle Leakage Inspection Commitment
1994-0912	Documents Results of CRDM leakage Video Inspection
1996-0551	Boric Acid on RX Vessel Head
1996-0650	VT-2 Inspection Revealed Evidence of Leakage and Boric Acid Residue
1998-0649	Reactor Vessel Head Boron Deposits
1998-0767	Reactor Vessel Head Inspection Results
1998-0824	CAC Boric Acid Accumulation
1998-1164	Water in RE4597 Sample Lines
1998-1895	CTMT Normal Sump Leakage in Excess of 1 gpm
1998-1965	Water and Boron Accumulation on Filter Cartridges
1998-1980	Potential CAC Fouling
1998-2071	Accumulation of Boric Acid on CTMT Service Water Piping

Condition Reports (CR)

1992-0139	Boron Found on Containment Air Sample Filter
1993-0187	Boric Acid Accumulation on SW Piping
1999-0372	Received Computer PT-RE4597AA/AB High
1999-0510	Low Flow Alarm Observed on RE4597BA While Out of Service for Maintenance
1999-0745	Small Clumps of Boric Acid Present on Wall Opposite of DH108
1999-0861	RE4597AA Sample Lines Were Found to be Full of Water
1999-0928	Increased Frequency of Particulate and Charcoal Filters for RE 4597BA Being Changed
1999-1300	Analysis of CTMT Radiation Monitor Filters
1999-1614	Due Date of LER Commitment Missed: Boric Acid Control Program Procedure Change

2000-0781	Leakage from CRD Structure Blocked Visual Exam of Reactor Vessel Head Studs
2000-0782	Inspection of Reactor Flange Indicated Boric Acid Leakage From Weep Holes
2000-0994	RV Head CRDM Nozzle at Location F-10 has Large Pit in Outer Gasket Groove
2000-0995	RV Head CRDM Nozzle Flange at Location D-10 has Extensive Pitting Across the Outer Gasket Groove. Inner Gasket Also Has Pitting
2000-1037	Inspection of Reactor Head Indicated Accumulation of Boron in Area of the CRD Nozzle Penetration
2000-1547	CAC Plennum Pressure Drop Following 12 RFO
2000-4138	Frequency for Cleaning Boron From CAC Fins Increased to Interval of Approximately 8 Weeks
2001-0039	CAC Plenum Pressure Experienced Step Drop
2001-0890	Unidentified RCS Leak Rate Varies Daily by as Much as 100 percent of the Value
2001-1110	Chemistry is Changing Filters on RE4597BA More Frequently
2001-1822	Frequency of Filter Changes for RE4597BA is Increasing
2001-1857	RCS Unidentified Leakage at .125 to .145 gpm
2001-2769	RE2387 Identified Spiked Above ALERT and High Setpoints
2001-2795	RE4597BA Alarmed on Saturation
2001-2862	Calculated Unidentified Leakage for Reactor Coolant System has Indicated Increasing Trend
2001-3025	Increase in RCS Unidentified Leakage
2001-3411	Received Equipment Fail Alarm for Detector Saturation on RE4597BA
2002-0685	Loose Boron 1-2" deep 75% Around Circumference of Flange
2002-0846	More Boron Than Expected Found on Top of Head

<u>Modifications</u>

MOD 90-0012	Modification Reactor Closure Head Access Ports
MOD 94-0025	Install Service Structure Inspection Openings

<u>USNRC Generic Communications for Control of Boric Acid Corrosion</u>

IN 86-108	Degradation of Reactor Coolant System Pressure Boundary Resulting from Boric Acid Corrosion, December 29, 1986
IN 86-108	Supplement 1, April 20,1987
IN 86-108	Supplement 2, November 19, 1987
IN 86-108	Supplement 3, January 5, 1995
GL 88-05	Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants, March,17, 1988
IN 90-10	Primary Water Stress Corrosion Cracking (PWSCC) of Inconel 600, February 23,1990
IN 94-63	Boric Acid Corrosion of Charging Pump Casing Caused by Cladding Cracks, August 30, 1994
IN 96-11	Ingress of Demineralizer Resins Increases Potential for Stress Corrosion Cracking of Control Rod Drive Mechanism Penetrations, February 14,1996
GL 97-01	Degradation of CRDM/CEDM Nozzle and other Vessel Closure Head Penetrations, April 1, 1997

IN 2001-05	Through-wall Circumferential Cracks of Reactor Pressure Vessel Head Control Rod Drive Mechanism Penetration Nozzles at Oconee Nuclear Station, Unit 3, April 30, 2001
Bulletin 2001-01	Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles, dated August 3, 2001

<u>Other Documents</u>

RAS02-00132	Probable Cause Summary Report for CR2002-0891, March 22, 2002
BAW-10190P	Safety Evaluation For B&W Design Reactor Vessel Head Control Rod Drive Mechanism Nozzle Cracking, May 1993
BAW-10190P, Addendum 1	B&W Owners Group Proprietary, External Circumferential Crack Growth Analysis for B&W-Design Reactor Vessel Head Control Rod Drive Mechanism Nozzle Cracking, December 1993
BAW-10190P, Addendum 2,	B&W Owners Group Proprietary, Safety Evaluation for Control Rod Drive Mechanism Nozzle J-Groove Weld
BAW-2301	B&W Owners Group Proprietary, B&WOG Integrated Response to Generic Letter 97-01, July 1997
Framatome 51-5015818-00	Davis-Besse CRDM Nozzle Heat Information, 2002
51-125825-00	CRDM Nozzle Heat Treatment
51-1218440-00	Alloy PWSCC Time-to-Failure Models
51-1219143-00	CRDM Nozzle Characterization
51-1219275-01	CRDM Leakage Detection Evaluation
51-1229638-00	Boric Acid Corrosion Data Summary and Evaluation
51-1201160-00	Alloy 600 SCC Susceptibility: Scoping Study of Components at Crystal River 3
Memorandum	Control Rod Drive Nozzle Cracking, May 8, 1996
Root Cause Plan	Dated March 18, 2002.
Intra-Company Memorandum	Probable Cause Summary Report for CR2002-0891, March 22, 2002
Meeting Minutes	DBPRC Meeting Minutes for MOD 94-0025.
WO 00-001846-00	Work Order - Clean Reactor Head
AR-00-OUTAG-01	QA Audit Report Refueling Outage 12
Serial 2472	Davis-Besse Letter: Response to Generic Letter 97-01, July 28, 1997
Serial 2731	Davis-Besse Letter: Response to Bulletin 2001-01, September 4, 2001
Serial 2735	Davis-Besse Letter: Supplemental Response to Bulletin 2001-01, October 17, 2001
ASME	ASME Boiler and Pressure Vessel Code, Section XI, 1986 Edition and 1995 Edition with 1996 Addenda

Photographic Records

Video Tape	Davis-Besse Reactor Head Inspection Under Insulation Alloy 600, 12 RFO
Video Tape	Davis-Besse 12 RFO Final Head Inspection
Video Tape	Davis-Besse Reactor Head Cleaning 11 RFO
Video Tape	Davis-Besse Weep Hole Cleaning Nozzle 67, 10 RFO
Video Tape	Davis-Besse Weep Hole Video Inspection 10 RFO
Video Tape	13 RFO Reactor Head Nozzle Remote Visual Inspection
Video Tape	Root Cause Video of Nozzle #3 and Adjacent Nozzles, March 13, 2002 to March 14, 2002
Video Tape	PT of Nozzle #46 J-groove Weld, March 24, 2002