May 3, 2002

Mr. Howard Bergendahl
Vice President - Nuclear
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 - DEGRADATION OF THE
REACTOR PRESSURE VESSEL HEAD - REPORT NO. 50-346/02-03(DRS)

Dear Mr. Bergendahl:

Your staff provided information to the NRC between March 6 and 10, 2002, concerning the identification of a large cavity in the reactor vessel head adjacent to a control rod drive nozzle. On March 13, 2002, the NRC issued a Confirmatory Action Letter outlining specific actions your staff are expected to take in response to this event. One of those actions is obtaining NRC approval prior to restart of the Davis-Besse plant.

On March 12, 2002, the NRC dispatched an Augmented Inspection Team (AIT) to the Davis-Besse site in accordance with NRC Management Directive 8.3, “NRC 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 enclosed report provides the AIT findings which were summarized for you and your staff during a public exit meeting on April 5, 2002.

The cavity in the reactor vessel head was discovered during maintenance activities for problems found during inspections conducted pursuant to NRC Bulletin 2001-01, “Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles.” The degraded area covers approximately 30 square inches where the thick low-alloy structural steel was corroded away, leaving only the thin stainless steel cladding layer as a pressure boundary for the reactor coolant system. This represents a loss of the reactor vessel’s pressure retaining design function, since the cladding was not considered as pressure boundary material in the structural design of the reactor pressure vessel. While the cladding did provide a pressure retaining capability during reactor operations, the identified degradation represents an unacceptable reduction in the margin of safety of one of the three principal fission product barriers at the Davis-Besse Nuclear Power Station.

The AIT concluded that the cavity was caused by boric acid corrosion from leaks through the control rod drive nozzles in the reactor vessel. These leaks were caused by primary water stress corrosion cracking of the nozzle material leading to a through-wall crack and corrosion of low alloy steel that went undetected for an extended period of time. The boric acid corrosion control program at the site included both cleaning and inspection requirements, but was not effectively implemented to detect the leakage and prevent the significant corrosion of the reactor vessel head over a period of years. Similarly on several occasions, maintenance and corrective action activities failed to detect and address the indications in the containment that the significant corrosion of the reactor vessel head was occurring. The NRC views these as missed opportunities to identify and correct this significant degradation to the reactor pressure vessel head.

The AIT did not address the verification of compliance with NRC rules and regulations, provide recommendations for enforcement actions, or assess the risk significance of this issue. A followup special inspection effort will be scheduled in the near future to pursue these aspects of the regulatory process.

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

Sincerely,

/RA by J. L. Caldwell for/

J. E. Dyer
Regional Administrator

Enclosure: NRC Augmented Inspection Report
No. 50-346/02-03(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

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U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket No:	50-346
License No:	NPF-3
Report No:	50-346/02-03
Licensee:	FirstEnergy Nuclear Operating Company
Facility:	Davis-Besse Nuclear Power Station
Location:	5501 North State Route 2 Oak Harbor, OH 43449
Dates:	March 12 - April 5, 2002
Team Leader:	R. Gardner, Engineering Branch Chief, DRS
Inspectors:	J. Davis, Sr. Materials Engineer, RES M. Holmberg, Sr. Reactor Inspector, DRS J. Gavula, Sr. Reactor Inspector, DRS D. Simpkins, Resident Inspector, Davis-Besse, DRP
Approved by:	John A. Grobe, Director Division of Reactor Safety

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SUMMARY OF FINDINGS

IR 05000346-02-03, on 03/12-04/05/2002, FirstEnergy Nuclear Operating Company, Davis-Besse Nuclear Power Station. Augmented Inspection Team.

This report covers a 3-week inspection by an NRC Augmented Inspection Team for the substantial loss of material from the reactor pressure vessel head.

- On March 5 and 6, 2002, workers at Davis-Besse were repairing control rod drive penetration Nozzle 3, following the identification of cracks detected through inspections performed pursuant to NRC Bulletin 2002-01, "Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles." The workers discovered a large cavity, a significant loss of metal adjacent to the control rod drive nozzle in the reactor vessel head, that apparently resulted from boric acid corrosion of the reactor vessel head due to leakage from the cracks in Nozzle 3.

- The cracks in the control rod drive nozzles were apparently due to primary water stress corrosion cracking of the Alloy 600 nozzle material. This type of cracking in this type of material has been identified at other facilities. However, the cracks at Davis-Besse appear to have initiated earlier than expected due to fabrication issues and plant operating conditions.

- The Davis-Besse staff, through their boric acid corrosion control program, did not clean and inspect the reactor vessel head sufficiently to identify the leakage due to nozzle cracking, nor the degradation of pressure boundary material.

- The apparent rate of boric acid corrosion was consistent with certain industry data. However, the corrosion rate used by the Babcock and Wilcox Owners Group, in their past assessment of potential head degradation associated with nozzle cracking, was significantly less than the apparent corrosion rate at Davis-Besse.

- The Davis-Besse staff missed several opportunities to identify the boric acid corrosion of the reactor vessel head at an earlier time. These opportunities involved the failure to identify the source of corrosion products that had accumulated on the containment air cooler fins, deposited on the containment radiation element filters, and noted as emanating from the inspection ports on the reactor vessel head service structure.

Report Details

1.0 BACKGROUND AND EVENT OVERVIEW

On March 6, 2002, Davis-Besse personnel notified the NRC of degradation to the reactor vessel head material adjacent to a control rod drive nozzle. The NRC issued a Confirmatory Action Letter on March 13, 2002. An Augmented Inspection Team (AIT) was chartered in Attachment A to determine the facts and circumstances related to the degradation of the reactor vessel head pressure boundary material, and to identify any precursor indications of this condition. The AIT developed a sequence of events, interviewed plant personnel, collected and analyzed factual information and evidence relevant to the reactor vessel head material loss, and conducted visual inspections of the reactor vessel head. The inspection was conducted in accordance with the AIT Charter, NRC Inspection Procedure 93800, “Augmented Inspection Team,” and NRC Management Directive 8.3, “NRC Incident Investigation Program.” In accordance with NRC procedures, the AIT charter did not include the verification of compliance with NRC rules and regulations, the recommendation of enforcement actions, nor the determination of risk significance for this issue. A public exit was conducted on April 5, 2002, using the presentation material in Attachment B.

1.1 Description of Reactor Vessel Head and Penetration Nozzles

Davis-Besse Nuclear Power Station is a two-loop pressurized water reactor designed by Babcock and Wilcox (B&W). The Davis-Besse reactor vessel has a torispherical shaped closure head constructed from low alloy steel (American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), SA-533, Grade B, Class 1), with approximately an 87-inch inside crown radius, 6.63 inches thick. The inside surface of the vessel head is clad with Type 308 and 308L stainless steel using a 6-wire submerged arc welding process. The cladding is provided for corrosion resistance and is not credited as pressure boundary material.

There are 69 vessel head penetration nozzles arranged in a rectangular pattern, with a center-to-center distance of approximately 12 inches, and are numbered sequentially starting at the center and progressing concentrically outward. The nozzles are fabricated from Alloy 600 tubes, with an outside diameter of approximately 4.00 inches and a wall thickness of 0.65 inches. The nozzles vary in length, depending on the location on the vessel head, from approximately 30 inches in the center to approximately 50 inches on the periphery. This includes a flange at the top for connecting to the control rod drive mechanism (CRDM) housings. Refer to Slide 5 in Attachment B for a diagram of the CRDM configuration. The nozzles extend through 4.00 inch bores in the vessel head, and are welded to the head with a J-groove weld at the inner surface of the head using Alloy 82 and 182 weld material. Refer to Slide 7 in Attachment B for a diagram of the CRDM nozzle.

The service structure is an enclosure attached to the reactor vessel head, approximately 18 feet high and 10 feet in diameter. This structure stabilizes and houses the CRDMs and contains a horizontal layer of metallic reflective insulation approximately 2 inches above the top of the vessel head. The CRDM nozzles welded to the vessel head pass through the insulation layer and attach to the CRDM housings with bolted flanges. These flanges are located about 9 inches above the horizontal insulation layer.

1.2 Sequence of Events: Discovery of Reactor Vessel Head Degradation

On February 16, 2002, the Davis-Besse facility began its 13th refueling outage (13 RFO), which included inspections of the CRDM nozzles in accordance with NRC Bulletin 2001-01, "Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles." On February 27, 2002, the licensee notified the NRC that CRDM Nozzles 1, 2 and 3 exhibited axial through-wall indications. The licensee decided to repair these three nozzles plus two other nozzles which had crack indications that did not appear to be through-wall.

On March 5, 2002, the licensee began repair work on CRDM Nozzle 3. The repair process included roll expansion of the CRDM nozzle material into the surrounding reactor vessel head material, followed by machining along the axis of the CRDM nozzle from the bottom to a point above the cracks in the nozzle material. After machining up past the J-groove weld, the machine unexpectedly rotated 15 degrees. The machining process was stopped and the machining tool was removed. Subsequent investigation identified that CRDM Nozzle 3 had tilted and was resting against an adjacent nozzle flange, which indicated a loss of some vessel head material.

On March 6, 2002, the licensee began an investigation to identify the cause of the movement by removing the CRDM nozzle. At the same time, activities were underway to remove boric acid residue from the top of the reactor vessel head using high pressure hot water to dissolve the deposits. After removing the boric acid deposits, the licensee identified a large cavity in the head material on the downhill side of CRDM Nozzle 3. In addition, during this same time period, the licensee identified a smaller cavity in the reactor vessel head after machining away the lower portion of Nozzle 2 during repair activities.

2.0 CHARACTERIZATION OF NOZZLE CRACKING AND REACTOR VESSEL HEAD WASTAGE AREAS

2.1 CRDM Nozzle Cracking

In response to NRC Bulletin 2001-01, “Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles,” the licensee ultrasonically examined all 69 CRDM nozzles during the current outage (13 RFO). These examinations were conducted inside the penetration tube from below the vessel head, and data was recorded from at least 1 inch above the J-groove weld down to the lower end of the nozzle. For these examinations, the ultrasonic transducers used were mounted in a blade probe head and setup for time-of-flight-diffraction. The transducer orientation was such that it provided maximum sensitivity for circumferentially oriented cracks near the outside diameter of the tube. Six nozzles were initially identified with crack-like indications using this technique.

For the six nozzles with crack indications a supplemental ultrasonic examination was conducted using a rotating head probe from above the vessel head. This probe head contained several types and angles of transducers designed to maximize the response to cracks oriented in both the circumferential and axial directions. This rotating probe confirmed cracks in five of the six nozzles identified by the blade probe. The cracks in these five nozzles initiated from the outside diameter of the nozzle near the J-groove weld. In three of the nozzles, through-wall axial cracks were identified that traversed the J-groove weld area of the nozzle. In addition, one circumferentially (circ.) oriented crack was identified in Nozzle 2 just above the J-groove weld, that was about 50 percent through-wall in depth. The number and dimensions of nozzle cracks are identified below:

Nozzle Number	Cracks and Orientation	Through-Wall Cracks	Through-Wall Crack Length (inches)	Crack Length Above J-weld (inches)
1	9 Axial	2	1.77 and 3.49	0.0, 0.5
2	8 Axial 1 Circ.	5 None	3.86, 2.71, 2.59, 3.95, 3.04 Not Applicable	0.8, 0.5, 0.5, 1.0, 0.5 Not Applicable
3	4 Axial	2	4.08, 3.84	1.3, 0.8
5	1 Axial	None	Not Applicable	Not Applicable
47	1 Axial	None	Not Applicable	Not Applicable

Although cracking was not identified at Nozzle 46, ultrasonic examinations revealed evidence of possible leakage and minor wastage in the annulus between the nozzle and the vessel head. Because a crack entirely within the J-groove weld could provide a leakage path and would not be detected with ultrasonic techniques, the licensee performed a dye penetrant examination of the J-groove weld. Four rounded indications were found, one 0.13 inches in diameter and three 0.06 inches in diameter. At the conclusion of this inspection, the licensee had not yet confirmed whether these indications were indicative of J-groove weld cracking.

2.2 Reactor Vessel Head Wastage Areas

The cavity adjacent to Nozzle 3 extended downhill toward Nozzle 11 for approximately 5 to 7 inches and was 4 to 5 inches wide. Within this area the 6.63 inch thick low alloy steel head was corroded away leaving only the stainless steel cladding layer on the inside of the reactor vessel head. The remaining cladding layer, ranging in thickness from 0.24 to 0.38 inches, had deflected upward into the cavity approximately 0.12 inches. This cladding layer is designed as a corrosion resistant layer and no credit is taken for the structural or pressure retaining capability of this layer. Therefore, the cavity at Nozzle 3 represented a loss of the design basis structural/pressure retaining boundary for the vessel head.

The cavity sides contained uneven ridges tapering downward, such that the cavity was larger at the outer surface of the head. Additionally, an undercut shelf existed at the downhill end of the cavity near Nozzle 11. An ultrasonic examination was conducted from the inner surface of the head to determine the extent of the cavity near Nozzle 3. This examination found that the cavity potentially had a “debonding” area between the stainless steel cladding layer and the vessel head material which extended for several inches around the cavity. The licensee intended to conduct additional examinations to further quantify the extent of this debonding. Refer to Slides 8 and 9 of Attachment B for a diagram and picture of this cavity.

In addition to the cavity adjacent to Nozzle 3, a comparatively small cavity was identified behind Nozzle 2. This cavity was approximately 1.75 inches wide and 0.25 inches deep. The licensee determined that the cavity extended from the top of the weld to the top of the vessel behind Nozzle 2 (approximately 4.2 inches). Refer to Slide 10 of Attachment B for a diagram of this area. The licensee removed Nozzle 2 to provide a more detailed characterization of this cavity after the AIT inspection.

3.0 PROBABLE CAUSE OF NOZZLE CRACKING AND HEAD WASTAGE

3.1 Probable Cause for Nozzle Cracking

For the five penetration nozzles with indications characterized as cracks (Section 2.1), four of these nozzles (Nos. 1, 2, 3, 5) were made from material heat No. M3935 manufactured by B&W Tubular Products. This same heat of tube material was found to have cracks in 14 of 68 penetrations used at Oconee Unit 3. This cracking was confirmed to be primary water stress corrosion cracking (PWSCC) based on analysis of cracked nozzles removed from Oconee Units 2 and 3 (these units also have a vessel head designed and constructed by B&W). Therefore, based on the observed susceptible heat of nozzle material under a similar environment, the AIT concluded that the Davis-Besse nozzle cracking was likely caused by PWSCC.

3.1.1 Factors Affecting Primary Water Stress Corrosion Cracking of Nozzles

Cracking of Inconel Alloy 600 penetration nozzle materials near the J-groove weld has been observed at several pressurized water reactors. The area of the J-groove weld on the nozzle is susceptible to PWSCC as discussed in NRC Generic Letter (GL) 97-01, “Degradation of Control Rod Drive Mechanism Nozzle and Other Vessel Closure Head Penetrations,” and in NRC Information Notice 2001-05, “Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles at Oconee Nuclear Station, Unit 3.” The susceptibility of a nozzle to cracking has been reviewed and documented in NUREG/CR-6245, “Assessment of Pressurized Water Reactor Control Rod Drive Mechanism Nozzle Cracking.” The susceptibility of a nozzle to PWSCC may be dependant on material, operating temperature, time, environment and residual stress. Because the operating environment of domestic pressurized water reactors is similar, the susceptibility of a particular nozzle to cracking may be dependant upon time, temperature, material microstructure and residual tensile stress. Thus, a particular heat of Alloy 600 used to fabricate a penetration nozzle may be more likely to experience cracking as each of these variables is increased (e.g., longer service time, higher operating temperatures, or a higher residual tensile stress). For the J-groove weld connecting the nozzle to the vessel head, a high residual tensile hoop stress is developed in the nozzle because of weld shrinkage. The magnitude of this residual tensile stress can range up to the yield strength of the material.

Crack initiation for PWSCC is strongly dependant on temperature (NUREG/CR-6245). The 605ºF operating temperature at Davis-Besse is higher than the other B&W plants (typically 602ºF). This higher operating temperature may have shortened the required operating time required to initiate cracking in the nozzles at Davis-Besse relative to other B&W designed plants.

Once a crack is formed (at a given temperature and environment) in a nozzle, the speed of crack propagation may be influenced by the tensile hoop stress induced from plant operating pressure and residual tensile hoop stresses induced by welding. As an axial crack in the nozzle progresses in length above the J-groove weld, welding induced residual tensile stress decreases rapidly, leaving only the operating pressure hoop stresses to extend the crack length. This results in slower crack growth as a crack increases in length above the J-weld. Therefore, the cracks identified in Section 2.1 which extend for the greatest distance above the J-groove weld are potentially the oldest cracks.

3.1.2 <u>CRDM Nozzle Materials and Contributing Factors</u>

Of the 69 Alloy 600 nozzles at Davis-Besse, 60 were manufactured by B&W Tubular Products and 9 were fabricated by Huntington Alloys. The nozzles are attached to the vessel head with an Alloy 82/182 "butter" and Alloy 82/182 J-groove weld. The specific method of fabricating the nozzle tubes was not recorded, but it would include rotary piercing or extruding over a mandrel followed by a mill anneal. The mill annealing heat treatment temperature should be in the range of 1850ºF to 1950ºF to put carbon into solution so that the carbides will precipitate at the grain boundaries during cooling. This heat treatment also redistributes chromium in the region of the grain boundaries. However, based on review of production records, the nozzles for all B&W plants were mill annealed in the temperature range of 1600ºF to 1700ºF. This lower temperature can increase susceptibility to primary water stress corrosion cracking.

As stated above, four of the Davis-Besse nozzles (Nos. 1, 2, 3, 5) exhibiting cracks were fabricated from material heat No. M3935 manufactured by B&W Tubular Products. This nozzle material heat had the highest yield strength (48,500 pounds per square inch) of the four material heats used to fabricate Davis-Besse head penetrations. It appears that this heat of Alloy 600 is more susceptible to primary water stress corrosion cracking than other heats of Alloy 600 used for B&W penetration tubes. However, the Owners Groups for B&W, Westinghouse, and Combustion Engineering have not been able to establish a definitive correlation between the yield strength and susceptibility to primary water stress corrosion cracking. Penetration tube 47 was also manufactured by B&W Tubular Products (heat number C2649-1) and contained a small crack below the J-Groove weld. This heat of material had the second highest yield strength (44,900 pounds per square inch). An additional factor affecting the material's yield stress was the straightening process used during manufacturing. This process will work harden the outside diameter of the nozzle resulting in the outside diameter yield stress being substantially above inside diameter yield stress.

3.2 Probable Cause for Vessel Head Wastage Cavities

Corrosion experiments (discussed in Section 3.2.1.2) simulating a cracked nozzle have confirmed that corrosion rates in excess of 2 inches per year are possible in low alloy steel. Nozzle 3 contained two through-wall axial cracks, which traversed the J-groove weld. The longest of these two cracks extended for approximately 1.3 inches above the J-groove weld. This crack would likely be the oldest crack in this nozzle as discussed in Section 3.1.1. The crack was on the downhill side of Nozzle 3 in direct alignment with the long dimension of the cavity. Therefore, the AIT concluded that the cavity observed on Nozzle 3 was associated with boric acid corrosion from crack induced leakage at this nozzle. Further, the AIT concluded, based on corrosion products observed on the head and in the containment air coolers and radiation element filters, that the corrosion process had been in progress for at least 4 years.

For Nozzle 2, the crack with the longest dimension above the J-weld was also located in the same area as the observed area of metal loss behind this nozzle. Again, the AIT considered that the metal loss was caused by boric acid corrosion from crack induced leakage at this nozzle.

3.2.1 Boric Acid Corrosion Mechanism

Pressurized water reactors use boric acid in the reactor coolant as one means of controlling the nuclear reaction rate. The levels of boric acid in the reactor coolant can range up to 2000 parts per million, which is generally not corrosive to materials used in the reactor plant. However, if boric acid is allowed to reach a concentrated solution it can become very corrosive to carbon steel components. The NRC issued GL 88-05, “Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR [Pressurized Water Reactor] Plants,” in March of 1988. The Generic Letter was in response to several industry incidents where concentrated boric acid solution, formed by evaporation of water from leaking reactor coolant, corroded reactor coolant pressure boundary components. The Generic Letter requested that licensees implement a program consisting of systematic measures to ensure that the reactor coolant pressure boundary would have an extremely low probability of abnormal leakage, rapidly propagating failure, or gross rupture.

3.2.1.1 Boric Acid Corrosion Processes

Compounds of boron can develop from the precipitation of boric acid from solution. Boric acid (H_3BO_3) and boric oxide (B_2O_3) can exist in a solid or molten state. The solid form of boric acid produced during evaporation depends on the rate of evaporation with faster evaporation creating smaller particles. When a boric acid solution comes in contact with boric acid crystals, larger crystals tend to form. It is also possible to form a salt tree when previously precipitated solids form a porous structure that can wick more solution to the vapor phase interface.

Boric acid solution that leaks onto the vessel head will cause the water to flash to steam, leaving behind white, popcorn-like boric acid crystals. This form of boric acid crystals is relatively easy to remove after the reactor is cooled down to ambient temperature. Dry, white, powdery boric acid crystals on the reactor vessel head have been found to be relatively benign while the reactor head is at operating temperatures. Although some darkening of the boric acid crystals may occur with age, brown or rust colored boric acid is a strong indication that corrosion has occurred and a problem potentially exists.

Above 302ºF, boric acid begins to dehydrate to form boric oxide:

$$2\ H_3BO_3 \Rightarrow B_2O_3 + 3H_2O$$

The final condition of the mixture of boric acid and boric oxide is site specific, depending on the relative quantities of each component and the amount of flow of boric acid, the porosity created by steam escaping, and the presence of impurities such as iron oxide. Boric oxide begins to soften at 617ºF and becomes highly viscous at 842ºF.

As boric acid that is not converted to the oxide is heated above 365ºF, it may become a viscous fluid (A. S. Myerson, Handbook of Industrial Crystallization, Butterworth-Heinemann, Boston, 1993), conforming to the surrounding geometry under the influence of gravity. Molten boric acid can contain between 8 and 14 percent water and can be highly corrosive under some conditions (U. Gurbuz Beker and N. Bulutcu, “A New Process to Produce Granular Boric Oxide by High Temperature Dehydration of Boric Acid in a Fluidized Bed,” *Transactions of the Institute of Chemical Engineers,* 74A, 133, 1996). Discussions with the NRC staff and staff members at the Brookhaven National Laboratory indicate that the boric acid/boric oxide mixture can vitrify if concentrated sufficiently and held at a high enough temperature.

3.2.1.2 Industry Accepted Boric Acid Corrosion Rates

In GL 88-05 corrosion rates were identified for pressure boundary materials of up to 0.019 inches per year (in/yr) at 500ºF. For lower temperatures, corrosion rates up to 4.8 in/yr were identified. However, these corrosion rates were established for configurations which were not representative of the CRDM nozzle to head annulus gap configuration.

A Babcock & Wilcox (B&W) owners group report, BAW-10190P, “Safety Evaluation For B&W Design Reactor Vessel Head Control Rod Drive Mechanism Nozzle Cracking,” was completed in May of 1993. In this report, a Combustion Engineering pressurizer heater sleeve mockup was used as the basis for establishing a 1.07 cubic inches per year corrosion rate as the applicable rate for the vessel head due to cracks in CRDM nozzles. The test results used by B&W were documented in EPRI report TR-102748S, “Boric Acid Corrosion Guidebook.” This B&W analysis concluded that with this corrosion rate, a plant would remain within ASME Code structural requirements for a minimum of 6 years. The AIT identified a test note in the EPRI report which stated that the maximum volume loss of 1.07 cubic inches per year may not be conservative for all cases since the volume loss is likely to increase as the corrosion depth and wetted surface area increase.

In November of 2001, an EPRI test was documented in Revision 1 to the Boric Acid Corrosion Guidebook. This test was performed utilizing a configuration, temperature, materials and leak rates which more closely matched the CRDM nozzle to vessel configuration. This test identified a corrosion rate of up to 2.37 in/yr. This test also indicated that the maximum corrosion occurred at the location where the boric acid entered the annulus gap. The contour of the degradation observed at Nozzle 2 and Nozzle 3 appeared to support this test result.

3.2.2 Licensee Preliminary Identified Cause

The preliminary conclusions of the licensee's root cause team were documented in a memorandum to the Davis-Besse Site Vice President, dated March 22, 2002 (Attachment C). In this memorandum, the root cause team concluded: "The factors that caused corrosion of the reactor pressure vessel (RPV) head in the regions of nozzles #2 and #3 are the CRDM nozzle leakage associated with through-wall cracking, followed by boric acid corrosion of the RPV low-alloy steel." The root cause team concluded that the cracking initiated in Nozzle 3 in 1990 (+/- 3 years) and the crack had propagated through-wall between 1994 and 1996. The average rate of RPV head corrosion was identified as 2 inches per year along the line from Nozzle 3 to Nozzle 11.

In this memorandum, the root cause team also stated that: "The estimated corrosion rates are compatible with test results reported in Electric Power Research Institute's (EPRI) Boric Acid Corrosion Guidebook. They are also consistent with the video, photographic and supporting plant data, that show that significant corrosion was occurring by the 1998 to 1999 time-frame." In addition, the root cause team identified a number of causal factors such as boric acid accumulation on the top of the RPV head and flange leakage.

The AIT concluded that the licensee's root cause team had reviewed the applicable historical data and established an appropriate time-line that supported the root cause. Although the AIT agreed with the preliminary root cause conclusions, there were several crucial questions left unanswered. The licensee's root cause efforts were continuing at the conclusion of the NRC's inspection. After the conclusion of the AIT, the licensee provided their final root cause analysis report to the NRC, on April 18, 2002, and provided responses to the NRC's questions associated with the preliminary root cause report on April 30, 2002. These documents are currently under review.

4.0 HISTORY OF VESSEL HEAD INSPECTIONS AND MATERIAL CONDITION

4.1 Background CRDM Flange Leakage

Historically, CRDM flange leakage had been observed at several B&W designed plants. At Davis-Besse, CRDM flange leakage typically resulted in deposits of boric acid on the service structure above the reflective insulation. However, flange leakage in liquid form also ran down the nozzles through the clearance gaps in the insulation and became boric acid deposits on the vessel head. The access for removing the boric acid deposits and inspecting the vessel head for corrosion is through (18) 5-inch by 7-inch rectangular openings or "weep holes." These openings are at the bottom of the service structure where it is attached to the vessel head. This location combined with the curvature of the vessel head made it difficult to inspect and clean the top center portion of the vessel head. Visual inspections of the vessel head have typically been accomplished using small video cameras inserted through the weep holes. Refer to Slide 5 in Attachment B for a diagram of the vessel head.

The CRDM flanges and flange bolts are made of stainless steel, corrosion resistant materials. Although the split nut-rings, located on the underside of the lower flange face, are made of a low alloy steel and are susceptible to corrosion, they have been coated with a corrosion resistant product. The nut-rings have not been found with boric acid corrosion at Davis-Besse. Because of these corrosion resistant materials, leakage from CRDM flanges typically does not result in corrosion, and any boric acid deposits from flange leakage are normally white or light in color. Conversely, as documented in the Davis-Besse Boric Acid Corrosion Control Procedure, boric acid deposits with red or rust color indicate that corrosion has occurred.

The licensee systematically resolved CRDM flange leakage by replacing the flange gaskets with a new design. Starting in 6 RFO (1990), gaskets were replaced on flanges which had developed leaks during the previous operating cycle, such that by 10 RFO (1996), the last nine old-design gaskets were replaced even though these flanges were not leaking.

4.2 History of Flange Leakage and Reactor Head Inspections

Inspections of the reactor head associated with identifying boric acid deposits were recorded after the licensee established a Boric Acid Control Program in 1988 in response to NRC GL 88-05. The following inspection results were documented in the licensee's corrective action system through PCAQRs [potential conditions adverse to quality reports] or CRs [condition reports] and/or recorded on video-tapes:

- In April of 1990 (6 RFO) 22 leaking CRDM flanges were identified and repaired (PCAQR 90-0120).

- In September of 1991 (7 RFO) 15 out of 21 leaking CRDM flanges were repaired. Boric acid was observed on the reactor vessel head that ran along the curvature of the head and stopped on the vessel closure bolts (PCAQR 91-0353). The source of these deposits was identified as flange leakage. Cleaning was performed with a wire brush and vacuum. No surface irregularities were noted following cleaning; however, the extent of deposits if any that remained after cleaning was not documented.

- In March of 1993 (8 RFO) 14 leaking CRDM flanges were identified and 11 were repaired (PCAQR 93-0132). The boric acid from flange leakage was removed to the extent possible by washdown of the head (PCAQR 96-551). The AIT viewed a videotape of the head inspection conducted during this outage and prior to the head washdown. Discrete patches of brown and white boric acid deposits were observed which were more numerous near the center of the head.

- In October of 1994 (9 RFO) eight CRDM flanges were leaking. All eight were repaired including three leaking flanges from the previous outage (PCAQR 94-0912). No record of a reactor vessel head inspection could be found.

- In April of 1996 (10 RFO) the remaining nine CRDM flanges (non-leaking) not previously repaired were modified with an enhanced gasket design. The head was inspected and video-taped using a remote camera mounted to a hand-held pole inserted through the weep holes. Several patches of boric acid accumulation were identified including a brown stained deposit at Nozzle 67 (PCAQR 96-551). The licensee documented that boron deposits could be indicative of flange leakage or nozzle leakage. A vacuum was used to remove boric acid deposits, but was not fully effective at removing the deposits of boric acid near the center of the head. The corrosion on the head from remaining boric acid was evaluated and considered minimal based on B&W Document 51-1229638, which identified minimal boric acid corrosion of carbon steel head material at temperatures corresponding to the normal head operating temperature. The licensee concluded that 50 to 60 percent of the head had been examined during this inspection. The limited head examination appeared to be due to access restrictions caused by the weep hole access limitations and the curvature of the head. The AIT observed the videotaped inspection and noted that the boric acid deposits were generally white in color and appeared to be the consistency of loose powder and discrete lumps.

- In May of 1998 (11 RFO) one leaking CRDM flange was identified and not repaired (PCAQR 98-0649). The head was inspected and video-taped using a remote camera mounted to a hand-held pole inserted through the weep holes. This inspection identified areas near the center of the head covered with an uneven layer of boric acid (PCAQR 98-767). The licensee documented that the boric acid deposits were removed “as best as we can.” The boric acid color was rust brown, which the licensee attributed to “old deposits” of boric acid. The previous root cause investigation and source documents from PCAQR 96-551 were referenced as the basis for leaving boric acid deposits on the head. The licensee concluded that due to the minimal operating time below 550°F, there was no impact on vessel head integrity. Based on review of this video-taped inspection, the AIT identified consolidated boric acid deposits near the center region including Nozzle 2 and 3 locations. On the head at an elevation below Nozzles 3 and 11, the AIT noted that the boric acid appeared highly adherent and rust brown in color.

- In April of 2000 (12 RFO) five leaking CRDM flanges were identified and repaired (CR 2000-0782). The head was inspected and video-taped using a remote camera mounted to a hand-held pole inserted through the weep holes. “Lava-like” brown/red deposits of boric acid over 1-inch thick were observed on much of the vessel head (CR 2000-1037). The corrective action for this condition was to repeat cleaning of the head until “most of the boric acid deposits are removed.” Licensee logs recorded that crowbars were needed to remove the “solid rock hard deposits of boron on the head.” In addition, pressurized heated water was used to remove the boric acid deposits. The extent of remaining boric acid deposits or evaluation of the effects on the head was not documented in the corrective action system after this cleaning. The system engineer also reported a large amount of boric acid deposits were observed above the mirror insulation due to flange leakage. The AIT viewed the video-taped examination made with a remote camera after the cleaning. This videotape showed a thick layer of “lava-like” brown/red boric acid that remained around the nozzles in the center of the head.

- In February of 2002 (13 RFO) no CRDM flange leakage was identified. The head was inspected and video-taped using a remote camera mounted to a hand-held pole inserted through the weep holes. The licensee documented that “more boron than expected was found on the top of the head” (CR 02-00685). Because the head was covered with boric acid and debris deposits, indications of nozzle crack induced leakage could not be positively identified at any nozzle location. The AIT reviewed pictures and tapes of this head inspection, which showed a thick lava-like brown/red deposit of boric acid covering the center of the head. Specifically, for 12 nozzles near the center of the head, the boric acid layer was several inches thick and precluded access for the remote camera inspection. The licensee subsequently removed the boric acid deposits from the head using hot pressurized water and identified the large head cavity at Nozzle 3.

The AIT noted the following important aspects in the above history of inspections and material condition of the RPV head:

(1) No flange leakage was found during 10 RFO (1996), and very limited flange leakage was noted during 11 RFO (1998). However, boric acid accumulation on the reactor vessel head increased from 9 RFO (1994) to 10 RFO (1996) and from 10 RFO (1996) to 11RFO (1998). Although the boric acid accumulation did not come from flange leakage, the licensee apparently did not deduce that it then must have come from pressure boundary leakage, such as nozzle cracking.

(2) Although five flanges were documented as leaking during 12 RFO (2000), according to CR-2000-0782, only four of the flanges showed positive evidence of gasket leakage. The fifth flange did not show the typical signs of flange leakage, but boric acid deposits had built up under the flange to the extent that the flange could not be fully inspected. This flange was for Nozzle 3, and the licensee concluded that the boric acid buildup was due to the flange leaking. The licensee apparently did not consider that the boric acid buildup could be due to nozzle leakage from below.

(3) Pictures of the reactor vessel, attached to CR-2000-0782, showed rust colored boric acid deposits emanating from the inspection openings on the reactor vessel head service structure. Although the licensee’s boric acid corrosion control procedure specifically stated that corrosion will most likely be exhibited by rust stained boric acid, the source of these corrosion products was not addressed in the condition report.

5.0 OPPORTUNITIES FOR EARLY DETECTION OF HEAD DEGRADATION

The AIT evaluated plant indications that could have provided an early opportunity to detect the corrosion occurring in the vessel head. The AIT identified the following indicators which could have provided early detection of the head corrosion.

5.1 <u>Boric Acid Corrosion Control Program</u>

Leakage from the reactor coolant system (RCS) with the reactor at power will flash to steam and leave behind boric acid crystals. Averaged over the course of a fuel cycle, there is approximately 0.03 pounds of boric acid per gallon of primary coolant. Assuming a leak rate of 0.001 gallons per minute, approximately 15 pounds of boric acid crystals would be produced in the vicinity of the vessel head by a postulated crack in a CRDM nozzle over one year. This leak rate would be significantly less than the minimum detection capability of the plant leakage detection systems. Therefore, inspection of the reactor head for boric acid deposits is potentially the most sensitive method available for detecting small leaks caused by cracked nozzles. However, there are limitations to this method. First, depending on location, a leak may not be accessible with the reactor at power. Consequently, certain leaks can only be identified when the reactor is shut down, which may only occur during refueling outages every two years. Second, this method depends on removing all existing boric acid accumulation, so any new leak can be detected without being masked by previous accumulations. This is critical because very small leaks may not be identifiable if the preexisting accumulation is not removed.

As previously discussed in Section 4.2, the licensee had preformed visual inspections of the reactor vessel head in 7 RFO (1991), and 8 RFO (1993) in accordance with GL 88-05 guidance. Davis-Besse’s implementing procedure for GL 88-05 was NG-EN-00324, “Boric Acid Corrosion Control.” Although recurring CRDM flange leakage was documented during 9 RFO (1994), licensee personnel were unable to identify any records documenting the visual inspections of the head during that outage.

In addition, boric acid deposits have historically been left on the head from flange leakage as discussed in Section 4. A leaking flange typically results in boric acid deposits which travel down past the head insulation resulting in a deposit/buildup of boric acid on the head. In accordance with the boric acid control program, these deposits should have been removed and the head inspected and any corrosion evaluated.

During 10 RFO (April 1996), a licensee engineer initiated PCAQR 96-0551, “Boric Acid on Reactor Vessel Head,” to document that the steps required by Procedure NG-EN-00324, “Boric Acid Corrosion Control,” had not been followed during the previous outage and that the procedure could not be fully implemented due to limited access to the reactor vessel head. The evaluation presented in this PCAQR acknowledged the need to clean the vessel head, such that nozzle leakage could be detected in the future. Also, the initial assessment in this PCAQR stated that the failure to clean the boric acid deposits made it difficult to determine if the deposits occurred because of leaking flanges or because of a crack in the CRDM nozzle. Licensee managers approved the PCAQR's initial assessment subject to the following comment:

> "Nozzle cracking is of course a significant issue. However, at present, the probability of occurrence is relatively low. We should remove boron from the reactor pressure vessel head as best we can and so as to minimize dose. This will allow us to monitor any leakage, should a nozzle crack initiate."

The corrective action for this PCAQR became a Request for Modification 94-0025 (see Section 5.5.1 for additional discussion on the delay of this modification).

Because of access limitations (see Sections 4.1 and 4.2), the RPV head was not completely cleaned and some portions were not thoroughly inspected, as specified by the licensee's Boric Acid Corrosion Control Program. The bases for not cleaning or inspecting the CRDM nozzles near the center of the RPV head was documented in PCAQR's or provided by licensee staff during interviews with the AIT. Specifically, the following information was utilized by the licensee to justify leaving boric acid deposits on the RPV head as identified during inspections in 10 RFO, 11 RFO and 12 RFO:

1) B&W Owners Group stress analyses had predicted that peripheral nozzles were more likely to crack than nozzles near the center of the vessel head.
2) Dried boric acid was not corrosive to the vessel and moderate amounts of boric acid from CRDM flange leakage had historically been found and cleaned up in the past, with no vessel corrosion.
3) Very limited boric acid corrosion occurs in the temperature range existing at the vessel head.
4) EPRI's "Boric Acid Corrosion Guidebook" indicated that, under specific circumstances, a layer of boric acid potentially protects a surface from ongoing corrosion by keeping water away from the surface.
5) CRDM nozzle cracking was an age related phenomenon, and the Davis-Besse staff believed they should not see any cracking because it was several years younger than Oconee where significant problems had not yet occurred. This was codified by the B&W Owners Group in July 1997 through a probabilistic susceptibility ranking that was developed in response to the NRC's GL 97-01.

The identification of nozzle cracks at Oconee Units 1 & 3, prompted the NRC to issue Bulletin 2001-01, which requested licensees to provide information, including a description of their previous inspections of the reactor vessel head. The Davis-Besse responses of September 4 and October 17, 2001, described their previous inspection and noted that, since 1996, four of the nozzles in the center of the vessel head were obscured with boric acid deposits and could not be viewed. In addition, the licensee's responses described their analytical efforts to verify that gaps would exist between the CRDM nozzles and the reactor vessel head, permitting through-wall leakage from a crack in a nozzle to be observed via boric acid deposits.

The licensee's analyses concluded that, except for Nozzles 1, 2, 3, and 4 (center nozzles), gaps would exist during normal operating conditions through which leakage could occur and boric acid deposits would be evident. In their supplemental response to the NRC Bulletin, dated October 30, 2001, the licensee stated that based on the above analytical results, the Davis-Besse staff would not expect to see boric acid residue around Nozzles 1, 2, 3, or 4 if a crack were present. This was based on the manufactured interference fit between the nozzles and the vessel head. The notable aspect of this conclusion was that the analytically predicted interferences ranged from 0.000025 to 0.000004 inches. Because the fabrication tolerances were more than an order of magnitude greater than the analytical results, the AIT considered the licensee's conclusion, relative to not expecting boric acid residue if a crack were present in these nozzles, to be unrealistic.

During interviews with the AIT, licensee personnel acknowledged that the reactor vessel head was treated less rigorously than other components in the plant, within the context of the GL 88-05 program. Although the boric acid corrosion control program was appropriately entered when boric acid was identified on the reactor vessel head, the resolution of the issue was not treated the same. Using the longstanding rationale discussed above, the licensee used a philosophy that boric acid had been on the reactor head for many years and no problems had ever been found.

5.2 Reactor Coolant System Leakage Detection

Because leakage from the through-wall cracks in Nozzle 3 would result in reactor coolant leakage into the containment atmosphere, the leakage detection systems in containment were reviewed to determine whether this system could have provided an early indicator of head corrosion. The observed leakage rate from a cracked nozzle would be expected to be very small based on a leakage rate (0.003 gallons per minute (gpm)) attributed to CRDM nozzle cracks observed at a foreign reactor plant (Bugey).

Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems," details requirements for leakage monitoring equipment such as the containment atmosphere particulate and gaseous radioactivity monitoring systems and containment sump level/flow monitoring system. The licensee has implemented a leak detection program in accordance with Regulatory Guide 1.45 as described in the Updated Safety Analysis Report, Section 5.2.4.

Reactor coolant system (RCS) leakage is grouped into two categories: identified and unidentified. Identified leakage is that which is captured and metered through closed systems, such as a collecting tank (e.g., pump seals and valve packing leaks); leakage into containment atmosphere from sources that are both specifically located and known not to interfere with the operation of leakage detection systems or not to be pressure boundary leakage; leakage through the steam generators to the secondary system; and reactor coolant pump seal returns. Unidentified leakage is everything which is not identified leakage.

Unidentified RCS leakage was normally less than 0.1 gpm (monthly average), until October of 1998, when a decision was made to remove the rupture disks downstream of the pressurizer relief valves for design concerns (PCAQR 98-1980). Specifically, a drain line, designed to collect relief valve leakage in the quench tank, was bypassed in this modification. This allowed leakage past the relief valves to be vented directly into the containment atmosphere, which 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 design concern 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. Subsequent investigations and containment entries were not successful in identifying definitive sources of this leakage. The licensee concluded, based upon the history of CRDM flange leakage and that unidentified leakage values observed at Davis-Besse were near industry averages, the leakage was most likely from the CRDM flanges.

Because of historical variations in unidentified leakage compared to the relatively small amount of leakage associated with CRDM cracks, the AIT concluded that, by itself, unidentified leakage trends were not a reasonable method of detecting nozzle cracking. However, when considered together with other indications of corrosion products as discussed in Section 4.2, above, and in Sections 5.3 and 5.4 below, the AIT concluded that this was a missed opportunity to detect the corrosion occurring on the reactor vessel head.

5.3 Containment Air Coolers

Reactor coolant leakage through the cracks in Nozzle 3 would travel as steam and liquid in the annulus behind the nozzle and leave boric acid deposits on the top of the head. In addition, this steam leakage would cause boric acid and corrosion products from the head cavity to be divided into fine particles which would be dispersed into the air space above the head. These fine particles would then be captured by the service structure ventilation system intake and be distributed throughout the containment. A key area which could collect these airborne particles of boric acid and corrosion products is at the containment air coolers (CAC).

The vessel head service structure ventilation pulls a suction from the CRDM flange area through the fans located on the 603 feet elevation, exhausting through ductwork to the top of the East D-ring. This provided a potential pathway for any corrosion fines and boric acid particulate dispersion originating from the vessel head. In November of 2001, radiological surveys showed a contamination plume effect originating from the service structure ventilation exhaust over the East D-ring. However, an isotopic analysis was not performed of the plume to fully characterize the source of the contamination. Additionally, two containment recirculation fans provide a mixing of the containment atmosphere, further dispersing the fines and particulates.

The CAC system consists of three separate tube/fin coolers (which are cooled by the service water (SW) system) located inside containment, and connected to a common supply plenum. Downstream of this plenum is a ductwork distribution system, designed to distribute air over and around all heat producing equipment, such as the reactor vessel, D-rings (housing the steam generators, pressurizer and reactor coolant pumps) and incore instrument tank. The external surfaces of the cooler tube banks are readily visible from the outside of the coolers, and have a remote indication of plenum pressure (used to determine cooling fin fouling) in the control room.

If a leak occurs from the RCS during normal operations, an aerosol mist is produced from the water flashing and evaporating as it exits the leak, increasing containment ambient humidity. Since the inlet water temperature of SW to the CACs is normally between 40°F and 75°F, substantially cooler than containment air temperatures, the CACs condense this ambient humidity to water, which is ultimately collected in the normal containment sump. In the process of removing the humidity, the CACs also collect particulate boric acid (which would be released with the RCS leakage as fine particles) on the cooling fins, in the discharge plenum and the associated ductwork. This fouling will decrease the plenum pressure, as read remotely in the control room, during periods of high boric acid accumulation.

In 1992, the licensee had experienced a CAC fouling from a leak in the reactor head vent line flange to the primary side of the steam generator. As a result, the licensee cleaned the boric acid, evident by the uniformly white coating on all three coolers. After repairs to the flange, no further boric-acid precipitated cleanings were required for several years.

In October of 1998, the removal of the rupture disks downstream of the pressurizer relief valves substantially contributed to the RCS unidentified leakage. In November 1998, PCAQR 98-1980 identified that the CAC fouling had increased correspondingly to increased leakage from the pressurizer reliefs. The CACs were cleaned 17 times from November 1998 to May 1999. During a mid-cycle outage in May 1999, the design concern was resolved, the rupture disks reinstalled, and the drain line reconnected. However, two additional CAC cleanings were conducted, one in June 1999 and one in July 1999. The post-job critique observed the boric acid to be “rust color on and in the boron being cleaned away” from CAC No. 1. Subsequent interviews indicated this was presumed to be the result of restoring from the mid-cycle outage, and the residual humidity in containment from outage-related repairs. After being cleaned in July 1999, the CACs did not need any further cleaning for approximately 10 months. Although the licensee installed high efficiency particulate air filters (inside containment) during August and September 1999, this did not appear to factor into the need for CAC cleaning.

After 12 RFO (May of 2000), CAC deposits were again forming, as evidenced by the decrease in plenum pressure. Eight CAC cleanings were conducted between June 2000 and May 2001, with no further cleanings required through the end of cycle. However, for 13 RFO (February 2002), the licensee reported (15) 5-gallon buckets of boric acid were removed from the ductwork and plenum. Significant boric acid was found elsewhere within containment, including on SW piping, stairwells and other areas of low ventilation.

After the 1999 mid-cycle outage, the licensee had attributed the excessive boric acid accumulation and CAC cleanings to leakage from CRDM flanges. In 12 RFO (May 2000), several leaking flanges were repaired, the results of which could not be verified throughout the cycle. However, 13 RFO (February 2002) inspections indicated the repairs had been successful, and no flange leakage was detected. Furthermore, earlier experience with leaking flanges (pre-1992, and 1992-1998) did not result in the need to clean the CACs. Therefore, CRDM flange leakage would not have reasonably been the major contributor to the increased boric acid loading on the CACs during this time frame. The licensee had also attributed the discoloration of the boric acid to migration of the surface corrosion on the CACs into the boric acid and the aging of the boric acid itself.

The AIT considered the sudden change to rust colored boric acid deposits in June of 1999, to indicate corrosion product accumulation from the formation of the head cavity near Nozzle 3. The failure of the licensee to identify the source of these deposits represented a missed opportunity to identify the corrosion cavity in the head at that time.

5.4 Radiation Elements

As discussed in Section 5.2, steam leakage through the cracks in Nozzle 3 would result in fine particles of boric acid and corrosion products. These particles would then be captured by the service structure ventilation system intake and distributed throughout the containment. An area where these fine particles of boric acid and corrosion products would be collected and observed is in the radiation element (RE) system filters.

There are two identical radiation element air sampling systems, drawing from two sample locations within containment. Air samples are drawn from within containment, passed through a particulate filter, an iodine sample cartridge and a noble gas detector before being exhausted back into containment. Both systems normally draw a sample from near the top of the “D-ring” structures, but can also draw from near the polar crane, and near the personnel airlock on the 603 feet elevation.

Boric acid accumulation on the RE filters can clog the filters and decrease flow to below acceptable levels, necessitating a filter change. Licensee records correlate past RCS leakage increases with RE filter changes, such as in 1992 when the reactor head vent flange leakage caused this to occur. In March of 1999, RE filter clogging from boric acid deposits was attributed to the pressurizer relief valve rupture disk maintenance which occurred in 1998. Filter changes normally occurred based on a monthly schedule rather than low flow rates. Beginning in May of 1999, the schedule of filter change out went from a monthly interval to an irregular 1 to 3 week interval, occasionally dropping to a 1 to 2 day interval by November 1999. In response to the increased frequency of filter changeouts, the licensee installed two large high efficiency particulate air filter units inside containment to capture a large portion of the corrosion fines. Additionally, the RE sample points were changed to the alternate locations. This action appeared to improve the service life of the filters, but did not eliminate the filter loading conditions completely.

In May of 1999, the RE filters began accumulating a yellowish-brown material. This material was sent to an external laboratory for analysis. The results of this analysis were received in November 1999, and positively identified the presence of ferric oxide. Specifically, this analysis 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.”

Accumulation of boric acid on the RE filters was readily recognized as a symptom of RCS leakage. During 12 RFO, CRDM flange D10 was attributed as the source of the RCS leakage, since the flange required machining to correct the leakage. However, the presence of ferric oxide fines was not explained, nor were multiple containment entries successful in determining a source. Additionally, past CRDM flange leakage had not significantly contributed to the CAC fouling, nor the RCS leakage indications. Therefore, the AIT believed that the corrosion deposits first identified in the RE filters beginning in May of 1999, indicated that corrosion was occurring due to the formation of the head cavity near Nozzle 3. The failure of the licensee to identify the source of these corrosion products represented a missed opportunity to identify the corrosion cavity in the head at that time.

5.5 Causal Factors Influencing Head Degradation Detection

Several decisions made by Davis-Besse personnel at various times directly influenced or potentially affected their ability to detect the head degradation associated with the CRDM nozzle leakage. These are discussed below.

5.5.1 Decision to Delay Modification to Service Structure

In March of 1990, modification 90-0012 was initiated to install multiple access ports in the service structure to permit inspection and cleaning of the vessel head. This modification was canceled in 1992, because the current inspection techniques were considered adequate.

In March of 1994, a licensee engineer initiated PCAQR 94-0295 to question why there was no commitment requiring a visual inspection of the reactor vessel head every refueling outage, as referenced in the NRC 1993 Safety Evaluation for the Alloy 600 CRDM nozzle cracking issue. The PCAQR's response from the Nuclear Assurance Director indicated that the commitment for the visual inspection did not appear to have been a licensee commitment to the NRC. Regulatory Affairs and Design Engineering personnel indicated that, although an enhanced visual was not a commitment to the NRC, they recommended the visual inspection be done. However, the plant engineering staff's comment in the PCAQR stated that there was a low risk of a crack in CRDM nozzles since none had been identified in the United States, and that the available inspection methods were not highly reliable. On that basis, the plant engineer felt it was not necessary to perform the inspections.

In May of 1994, the licensee engineer who wrote the above PCAQR initiated a Request for Modification (RFM 94-0025) to install openings in the CRDM service structure to allow thorough inspection and cleaning of the reactor vessel head. The modification request noted that, out of all of the B&W plants, only Davis-Besse and Arkansas Nuclear One, Unit 1, had not installed the access openings in the service structure. The modification request cited the following reasons for the modification:

1) there was no access to the reactor vessel head or CRDM nozzles without the modification, and there was an ongoing industry concern for Alloy 600 nozzle cracking;

2) inspection of the reactor vessel head for boric acid corrosion was difficult and not always adequate, because the video inspections did not encompass a 100 percent inspection of the head;
3) cleaning boric acid residue from the vessel head did not encompass 100 percent, because the size and geometry of the weep holes only permitted cleaning of the lower one-third of the head with scrapers and wire brushes.

The modification was approved by the plant in July of 1994, but remained unfunded by the Project Review Committee/Project Review Group until November of 1998, when it was scheduled for implementation in 13 RFO (2002). The modification was subsequently deferred until 14 RFO by the Project Review Group, as part of an effort to meet the 2001/2002 expenditure targets by reducing the number of projects implemented. In discussing the reasons for not implementing this modification, the rationale identified in Section 5.1 were also applied. The AIT considered the delay in implementing the modification as contributing to the failure to detect head degradation.

5.5.2 <u>Decision to Delay Repair of CRDM Flange on Nozzle 31 in 11RFO</u>

During 8 RFO (1993), CRDM flange leakage was noted on several CRDM flanges including the flange for Nozzle 31. The corrective actions included polishing the flange surface and replacing the gasket with a new design. The PCAQR issued to document this condition (93-0132) contained a recommendation that the flange surface be inspected during each subsequent maintenance outage and be machined if further leakage occurs. During 11 RFO (1998), the CRDM flange for Nozzle 31 was found to be leaking, and as indicated in PCAQR 98-0649, the amount of leakage was not considered significant compared to flange leakage from previous outages. Consequently, no corrective actions were taken, even though the vendor (Framatome) reiterated their recommendation from 1993 to machine the flange. The PCAQR did contain a recommendation to reexamine the flange for Nozzle 31 during 12 RFO and to replace the gasket if the flange was leaking.

During 12 RFO (2000), significant flange leakage was noted and five leaking flanges were identified during the video inspections of the CRDM flanges, including Nozzle 31's. The majority of the boric acid accumulation was attributed to Nozzle 31's flange due to steam cutting of the flange face. Condition Report 2000-1037 was written to describe the boric acid accumulation on the RPV head and on top of the insulation. The boric acid accumulation was attributed to leaking CRD flanges. The AIT considered the delay in repairing Nozzle 31's flange as a contributing cause of this event, because the extensive amount of flange leakage contributed to the boric acid deposits on the head which masked evidence of the nozzle leakage occurring at this time.

6.0 CONCLUSIONS

The AIT presented the inspection results to Mr. Saunders and other members of the licensee management at the conclusion of the inspection on April 5, 2002. The licensee acknowledged the conclusions presented as discussed in Attachment B and summarized below.

The AIT concluded that the probable cause of the cavity at Nozzle 3 was boric acid corrosion of the head associated with reactor coolant leakage from a through-wall crack in this nozzle. Further, the AIT concluded based on corrosion products observed on the head, and in the CAC and RE filters that the corrosion process had been in progress for at least 4 years.

The AIT concluded that the probable cause of the cracking observed in the five penetration nozzles was PWSCC. This was based on similar cracking identified at two other B&W plants that performed destructive analysis of cracked nozzles fabricated from the same heat of material to confirm PWSCC.

The AIT evaluated the indications which existed that could have provided an early opportunity to detect evidence of the formation of the corrosion cavity in the head at Nozzle 3. The AIT identified several opportunities which were available to the licensee to potentially identify this corrosion cavity at an earlier point in time. Specifically, these missed opportunities were associated with the failure to identify the source of the corrosion products deposited in the CAC and RE filters in early 1999 and the failure to remove boric acid or evaluate the source of corrosion products which accumulated on the vessel head.

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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
in/yr	Inches Per Year
NRC	Nuclear Regulatory Commission
PCAQR	Potential Conditions Adverse to Quality Report
PDR	Public Document Room
PWSCC	Primary Water Stress Corrosion Cracking
RCS	Reactor Coolant System
RE	Radiation Element
RFO	Refueling Outage
RPV	Reactor Pressure Vessel
SW	Service Water

LIST OF DOCUMENTS REVIEWED

Calculation

SIA Calc W-ENTP-11Q-306	Finite Element Gap Analysis of CRDM Penetrations (Davis-Besse), October 8, 2001.

Condition Reports (CR)

1992-0139	Boron Found on Containment Air Sample Filter
1993-0187	Boric Acid Accumulation on SW Piping
1998-0020	Multiple Problems Identified with RC-2
1998-0330	Industry Event (Prairie Island) Crack in the Motor Tube of the Control Rod Drives
1998-1963	Design Over-Stress of the Pressurizer Nozzles for Safety Valve
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
1999-0998	Awareness of Approaching the Tech Spec Limit for Maximum Ctmt Air Temp
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
2000-0903	Two of 40 CRDM Hold Down Bolts Had Indications Found During VT-1 Inspection
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
2000-1210	During Installation of Control Rod Drive Assembly at Location D-10, on the Reactor Head, it was Discovered that Top of Motor Tube for this Drive was out of Line with Surrounding Motor Tubes
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
2001-0487	Certain Areas Inside CTMT in Year 2000 Seeing Higher Temperatures
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
2001-2012	NRC Issuance of IEB 01-01 Circumferential Cracking of RX Pressure Vessel Head Penetration Nozzles

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
2001-2936	Monthly Functional Test for RE4597BA/BB Count Not Performed
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
2002-0891	UT Performed on #3 CRDM Nozzle Revealed Indication of Through-Wall Axial Flaws
2002-0932	Completion of UT on All 69 CRDM Nozzles Revealed Additional CRDM Cracks Beyond #3 Nozzle
2002-1053	While Machining Reactor Vessel Head Nozzle #3 the Nozzle Machining Tool Moved Approximately 15 Degrees
2002-1128	Evaluation of Bottom up Ultrasonic Test Data in Area of RX Pressure Vessel Head Nozzle #3 Shows Significant Degradation of RX Vessel Head Pressure Boundary
2002-1159	During Video Tape Review, Indication Found on Newly Machined Face on Mid-Span of CRDM Nozzle. Appears to be Through-wall in Immediate Vicinity of Base Metal Indications.

Drawings

M-503-127-3	Closure Head Assembly, Revision 3
M-503-212-1	Closure Head Subassembly Drawing, Revision 1
M-503-213-2	Closure Head Subassembly Drawing, Revision 2
03-1221681-03	Framatome Drawing of RV Nozzle/Nur Ring Modification

Modifications

MOD 90-0012	Modification Reactor Closure Head Access Ports
MOD 94-0025	Install Service Structure Inspection Openings
TM 1998-0036	Temporary Modification: Preliminary Evaluation of Pressurizer Nozzles for Relief Valves Demonstrates that an Overstress Condition May Exist in the Nozzle Flange

NRC Generic Communications for Control of Boric Acid Corrosion

IN 80-27	Degradation of Reactor Coolant Pump Studs, dated June 11, 1980
IEB 82-02	Degradation of Threaded Fasteners in the Reactor Coolant Pressure Boundary of PWR Plants, dated June 2, 1982
IN 82-06	Failure of Steam Generator Primary Side Manway Closure Studs, dated March 12, 1982
IN 86-108	Degradation of Reactor Coolant System Pressure Boundary Resulting from Boric Acid Corrosion, dated December 29, 1986
IN 86-108	Supplement 1, dated April 20,1987
IN 86-108	Supplement 2, dated November 19, 1987
IN 86-108	Supplement 3, dated January 5, 1995

GL 88-05	Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants, dated March,17, 1988
IN 90-10	Primary Water Stress Corrosion Cracking (PWSCC) of Inconel 600, dated February 23,1990
IN 94-63	Boric Acid Corrosion of Charging Pump Casing Caused by Cladding Cracks, dated August 30, 1994
IN 96-11	Ingress of Demineralizer Resins Increases Potential for Stress Corrosion Cracking of Control Rod Drive Mechanism Penetrations, dated February 14,1996
GL 97-01	Degradation of CRDM/CEDM Nozzle and other Vessel Closure Head Penetrations, dated 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, dated April 30, 2001
Bulletin 2001-01	Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles, dated August 3, 2001

Other Documents

RAS02-00132	Probable Cause Summary Report for CR2002-0891, dated March 22, 2002
NPE-96-00260	Control Rod Drive Nozzle Cracking, dated May 8, 1996
Books	RCS System Performance Books Volumes 1 though 11
BAW-10190P	Safety Evaluation For B&W Design Reactor Vessel Head Control Rod Drive Mechanism Nozzle Cracking, dated May of 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, dated December 1993
BAW-2301	B&W Owners Group Proprietary, B&WOG Integrated Response to Generic Letter 97-01, dated July 1997
Exam Report	Reactor Vessel Head ID Clad Thickness Measurements in Region of Wastage Between Nozzles 3 and 11, dated March 18, 2002
Exam Report DB-5	CRDM Nozzle 46 J-Groove Weld, dated March 24, 2002
Examination Report	Davis Besse 13 RFO CRDM Nozzle Examination Report, dated March 11, 2002
Framatome 51-5015818-00	Davis-Besse CRDM Nozzle Heat Information, 2002
EPRI Report TR-102748s	Boric Acid Corrosion Guidebook, Revision 0, dated April 1995
EPRI Report 1000975	Boric Acid Corrosion Guidebook, Revision 1, dated November 2001
Report 2779	Oconee Unit 3 CRDM Nozzle Crack and Material Characterization - Oconee Unit 1 Thermocouple Tube material Characterization - Metallurgical Analysis Report
Dominion Engineering Report	Volume and Weight of Material Lost at Nozzle 3
51-125825-00	CRDM Nozzle Heat Treatment

Material Test Report	DBNPS Reactor Vessel Head Certified Material Test Report
Intra-Company Memorandum	Control Rod Drive Nozzle Cracking, dated May 8, 1996
Root Cause Plan	Dated March 18, 2002.
Intra-Company Memorandum	Probable Cause Summary Report for CR2002-0891, dated March 22, 2002
Meeting Minutes	DBPRC Meeting Minutes for MOD 94-0025.
Standing Order 87-015	RCS Leakage Management and Attached Policy Reactor Coolant System Leakage Management
0620-00143210	Lukens Steel Company, Test Certificate, Chemical Analysis and Physical Properties

<u>Photographic Records</u>

Picture	Vessel Head on Stand
Picture	Head and Service Structure Looking NE
Picture	Scaffolding Around Service Structure
Picture	View From Newly Cut Service Structure Manway Opening Looking into Drives
Picture	Looking Through Manway Cut in Service Structure
Picture	View of CRD Flanges Above Insulation Showing Some Removed and Some Installed
Picture	Control Rod Drive Flanges Above Insulation
Pictures	Shielded Work Platform on Top of Service Structure
Picture	Pictures of Nozzle 2 and 3
Pictures	Area Surrounding Nozzle 3 Penetration
Picture	Nozzle 16 Quad C
Pictures	Nozzle 2
Pictures	Nozzle 3 Remnant
Pictures	From Bare Head Video Exam Conducted in 13 RFO.
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

<u>Potential Conditions Adverse to Quality Reports (PCAQR)</u>

1988-0494	Condition Not Satisfactorily Resolved per PCAQ
1990-0221	CRDM Flanges #F02 and F-4 Erosion and Irregularities.
1991-0353	Boron on Reactor Vessel Head
1992-0072	CAC Cleaning
1993-0098	Boric Acid Corrosion on OTSGA Head Vent Flange
1993-0132	Reactor Coolant Leakage from CRD Flange

1994-0912	Documents Results of CRDM leakage Video Inspection
1994-0974	CRDM Flange Indication
1994-0975	CRDM Flange Indication
1994-1338	10 CFR Part 21 RX Adaptor Tubes
1996-0551	Boric Acid on RX Vessel Head
1996-0650	VT-2 Inspection Revealed Evidence of Leakage and Boric Acid Residue
1996-1018	IN 96-032 RV Augmented ISI
1998-0020	Inadequate Testing
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
1998-1885	Found Two Carbon Steel Nuts on RC2
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

Procedures

NG-EN-00324	Boric Acid Corrosion Control, Revisions 1, 2, and 3
PP-1102.10	Surveillance Test Procedure: Plant Shutdown and Cooldown, Revision 16
DB-OP-06903	Operations Procedure: Plant Shutdown and Cooldown
DB-PF-00204	ASME Section XI Pressure Testing, Revision 3
DB-OP-01200	Reactor Coolant System Leakage Management, Revision 3

ATTACHMENT A TO NRC AUGMENTED INSPECTION REPORT NO. 50-346/02-03(DRS)

USNRC Memorandum from J. E. Dyer to R. N. Gardner, dated March 12, 2002: Augmented Inspection Team Charter - Davis-Besse Reactor Vessel Head Material Loss

Documented in ADAMS (Accession Number ML020730194)

ATTACHMENT B TO NRC AUGMENTED INSPECTION REPORT NO. 50-346/02-03(DRS)

NRC Briefing Slides for the Public AIT Exit Meeting Conducted on April 5, 2002

Documented in ADAMS (Accession Number ML021070811).

ATTACHMENT C TO NRC AUGMENTED INSPECTION REPORT NO. 50-346/02-03(DRS)

FirstEnergy Intra-Company Memorandum from S. A. Loehlein to H. W. Bergendahl, dated March 22, 2002

Documented in ADAMS (Accession Number ML020860035)

March 12, 2002

MEMORANDUM TO: Ronald N. Gardner, Chief
Electrical Engineering Branch
Division of Reactor Safety

FROM: J. E. Dyer ***/RA/***
Regional Administrator

SUBJECT: AUGMENTED INSPECTION TEAM CHARTER - DAVIS BESSE REACTOR VESSEL HEAD MATERIAL LOSS

In response to preliminary information provided by the licensee on March 10, 2002, regarding the significant loss of pressure boundary material from the reactor vessel head, an augmented inspection team (AIT) is being sent to the Davis-Besse Plant. You are hereby designated as the AIT leader.

A. Basis

On March 6, 2002, during repair activities to control rod drive mechanism (CRDM) nozzles, the licensee identified an area of wastage in the reactor pressure vessel head surrounding the No. 3 CRDM nozzle. The licensee initially identified five CRDM nozzles that required repairs due to cracking in the J-groove welds found during the nozzle examinations required by Bulletin 2001-01. Wastage area in the head was discovered when the licensee removed the No. 3 CRDM nozzle, after the penetration tube unexpectedly moved during repair activities.

Because this was a significant unplanned degraded condition having potential generic safety implications, an AIT was initiated in accordance with NRC Management Directive 8.3, "NRC Incident Investigation Program." The purpose of the AIT is to better understand the facts and circumstances related to the degradation of the reactor vessel head pressure boundary material. It is also to identify any precursor indications of this condition so that appropriate followup actions can be taken. All followup actions associated with the extent of condition, repairs/replacements, or corrective actions related to plant restart will be covered through other inspection activities.

CONTACT: John A. Grobe, Director, DRS
(630) 829-9700

B. Scope

Specifically, the augmented inspection team is expected to collect, analyze, and document factual information and evidence sufficiently to address the following:

1. The plant history of reactor coolant system operational leakage indications, including trends in unidentified leakage, containment air cooler fouling, containment radiation monitor readings, etc.

2. The plant history of reactor vessel head material condition issues, including control rod drive flange leakage or other sources of corrosive substances.

3. The plant history of reactor vessel head inspection, including visual inspections, ultrasonic testing, prior video-records of head examinations, reactor vessel head cleaning activities, and licensee action in response to generic correspondence for leakage and degradation of the reactor coolant system.

4. Characterization of all reactor vessel head wastage areas, including the best available geometric details of cavity volumes, surface conditions, surface contaminants, etc.

5. The probable cause(s) for the vessel head wastage.

C. Guidance

This memorandum designates you as the AIT leader. Your duties will be as described in Inspection Procedure 93800, "Augmented Inspection Team." The team composition has been discussed with you directly. During performance of the augmented inspection, designated team members are separated from their normal duties and report directly to you. The team is to emphasize fact-finding in its review of the circumstances surrounding the event, and it is not the responsibility of the team to examine the regulatory process, to determine whether NRC requirements were violated, to address licensee actions related to plant restart, or to address the applicability of generic safety concerns to other facilities. Safety concerns identified that are not directly related to the event should be reported to the Region III office for appropriate action.

The team will report to the site, conduct an entrance meeting, and begin inspection on Tuesday, March 12, 2002. Tentatively, the inspection should be completed by close of business March 22, 2002, with a report documenting the results of the inspection, including findings and conclusions, issued within 30 days of the public exit meeting. While the team is on site, you will provide daily status briefings to Region III management.

This Charter may be modified should the team develop significant new information that warrants review.

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