B-1846

From: Douglas Pickett
To: Allen Hiser; Andrea Lee; Anthony Mendiola; Barry Marcus; Beth Wetzel; Bill Bateman; Brian Sheron; Bruce Mallett; Christine Lipa; Christopher Thomas; Chuck Paulk; David Lew; David Nelson (HQ-OE); Donald Naujock; Donna Chaney; Douglas Simpkins; Edward Andruszkiewicz; Edward Kendrick; Edwin Hackett, Ellis Merschoff, F. Mark Reinhart; Frank Akstulewicz; Gary Holahan; Gene Imbro; Geoffrey Grant; Ho Nieh; Hubert J. Miller; Ian Jung; Jack Strosnider; James Caldwell; James Wiggins; Jan Strasma; Jay Collins; Jeffrey Clark; Jerry Dozier, Jim Dyer; John Grobe; John Jacobson; John Zwolinski; Jose Calvo; Kenneth Karwoski; Kenneth O'Brien; Laura Collins; Laura Gerke; Lee Ellershaw; Linda Portner; Linda Smith; Luis Reyes; Maggalean Weston; Margie Kotzalas; Mark Lesser; Melvin Holmberg; Michael Cullingford; Mitzi Young; Nilesh Chokshi; Pat Gwynn; Ralph Caruso; Ram Subbaratnam; Richard Barrett; Robert Spence; Ronald Gardner; Roy Caniano; Satwant Bajwa; Spiros Droggitis; Stephen Sands; Steven Bloom; Steven Long; Steven Reynolds; Suzanne Black; Tad Marsh; Thomas Koshy, Timothy Steingass; Victor Dricks; Victor McCree; Walton Jensen; Wayne Sifre; William Beckner; William Cullen
Date: 3/27/02 1:28PM
Subject: Latest Davis-Besse Communications Plan

This email is intended to provide a general update to the Davis-Besse Communications Team. Also included is the latest version of the plan

- The licensee prepared a preliminary Probable Cause Summary Report dated March 22, 2002. A four page summary was released by the licensee which can be found on the NRC public web page.

- An information only telecon was held with the licensee and NRR staff on Monday, March 25, regarding proposed modifications to the reactor pressure vessel head. Modifications include plugging nozzles 2 and 11 and inserting a 13-inch diameter stainless steel plate where the former nozzle 3 existed The stainless steel plate will be approximately 3 inches thick whereas the original reactor vessel head contained approximately 6 inches of carbon steel Multiple relief requests are expected from the ASME Code The staff impressed upon the licensee that the public must be involved and anticipated a public meeting to discuss the proposed modifications.

- On March 27, the staff informed the licensee that their preliminary Probable Cause Summary Report is not considered complete and to expect a significant request for additional information from the staff The staff will run the questions through the AIT team prior to submitting it on the docket to the licensee.

- On March 27, the staff discussed the ACRS meetings of April 9 and April 11. There will be a subcommittee meeting on April 9 where we expect the licensee to make a presentation focusing on the root cause and information currently available. We stressed that this session should focus on root cause and that any discussion of proposed modifications would be more appropriate for the May ACRS meeting. The full ACRS committee meeting will be held on April 11. The licensee is expected to be present but is not expected to make a presentation before the full committee.

- The licensee had proposed a public meeting for April 4 to discuss their proposed modifications to the reactor pressue vessel head. On March 27, the staff informed the licensee that we could not support such a meeting prior to the licensee developing a more comprehensive root cause analyses. We suggested to the licensee that their next public meeting should address root cause

- Region III has reached agreement with the licensee that the public AIT meeting will be held at 9.00 a.m. at the Oak Harbor High School on Friday, April 5. Brian Sherson and Allen Hiser will represent NRR at the AIT exit. The end-of-cycle Reactor Oversight Program meeting will also be at Oak Harbon High School that same afternoon at 1:00 p.m.

- The staff is continuing work on a significant list of questions received from the public during the public meeting of March 20. When complete, they will be included on the NRC web page

OCIO-045

Communication Plan
Davis-Besse Nuclear Power Station
Significant Metal Loss Observed in Reactor Vessel Head

PURPOSE:

The purpose of this Communication Plan is to provide the outline for how the NRC will communicate with internal and external stakeholders during the staff's evaluation and response to the significant metal loss observed in the reactor vessel head at the Davis-Besse Nuclear Power Station.

GOALS:

This communication plan was developed to (1) establish a coordinated action plan to respond to the significant metal loss observed in the reactor vessel head at the Davis-Besse Nuclear Power Station (DBNPS), (2) solicit industry assistance in identifying the root cause and generic applicability, (3) generate appropriate generic correspondence to ensure that operating facilities have performed appropriate inspections or have developed acceptable justifications for continued operation of their facilities, and (4) provide guidance on how to respond to public questions. These actions will assist the NRC in addressing two performance goals: (1) maintain safety and protect the environment and (2) enhance public confidence. It is important to communicate the facts to the public and to assure the public that we are aware of the situation and are monitoring the actions of both the licensee and the industry. This document describes the methods and tools for communicating, both internally and externally, the agency's response to the identification of the degraded condition of the reactor vessel head at the DBNPS.

March 27, 2002

KEY MESSAGES:

- The NRC prohibits plant operation with known pressure boundary leakage of the reactor coolant system (RCS). The pressure containing components of the RCS and the portions of connecting systems out to and including the isolation valves define the reactor coolant pressure boundary (RCPB). The joints of the RCPB components are welded or bolted.

- Significant metal loss in the reactor vessel head has not previously been observed in domestic reactors. The mechanisms that have caused this condition are not fully understood. Had this condition not been identified, it could have resulted in significant RCS leakage emanating from the reactor vessel head. A loss-of-coolant accident would represent a serious challenge to the plant's safety systems.

- RCS leakage, outside of RCPB leakage, is an analyzed condition. There are pre-established procedures and safety-related emergency core cooling systems for handling a spectrum of RCS leakage.

- Operators are trained to respond to RCS leakage and are required to promptly shut down the reactor, classify the event, and make immediately-required, off-site notifications if the situation requires such actions.

- The DBNPS staff is analyzing conditions appropriately. The NRC resident inspectors and other NRC staff members have been monitoring the licensee's actions.

- The licensee has established a dedicated Root Cause Investigation Team, an Engineering Evaluation Team, a Field Activities Team, and an External Interface Team in response to this event.

DESCRIPTION OF CIRCUMSTANCES:

On February 16, 2002, the Davis-Besse facility began a refueling outage that would include inspection of the control rod drive mechanism (CRDM) nozzles in accordance with licensee commitments to NRC Bulletin 2001-01, "Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles," dated August 3, 2001. As described in Event Report 38732 supplemented on March 5, 2002, the licensee identified CRDM nozzles 1, 2 and 3 as exhibiting axial through-wall indications in the nozzles, indicative of pressure boundary leakage. The licensee decided to repair these three nozzles and two other nozzles with indications not related to pressure boundary leakage.

Subsequently on March 6, 2002, the licensee began repairs of 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 to a point above the indications in the nozzle material. During this machining process on nozzle 3, conditions were such that the machining process was terminated and the machining apparatus was removed from the nozzle. After the apparatus was removed, it was noted that CRDM nozzle 3 displaced towards the downhill portion of the reactor vessel head, until the CRDM flange contacted that of the adjacent CRDM nozzle.

The licensee then initiated efforts to characterize the reactor vessel head volume surrounding this CRDM nozzle to identify the cause of the CRDM nozzle displacement. These efforts included removal of the CRDM nozzle. In addition, existing boric acid deposits located on the top of the reactor vessel head were removed using hot water to dissolve the deposits. Upon completion of the boric acid removal efforts on March 7, 2002, the licensee identified a large cavity as a result of a significant, unexpected metal loss on the downhill side of CRDM nozzle 3. Follow-on characterization indicated that the cavity extended about 5 inches below nozzle 3 (towards the adjacent nozzle), with a depth of approximately 6 inches and a width of approximately 4 to 5 inches.

The licensee has begun to assess the extent of the damage and has formed a root cause team to investigate the cause of this phenomenon. Repair and replacement options are being considered.

On March 11, 2002, the licensee contacted the media along with local and state government officials to inform them of the identified conditions.

NRC Information Notice 2002-11, "Recent Experience with Degradation of Reactor Pressure Vessel Hed," issued on March 12, 2002.

Confirmatory Action Letter (CAL No. 3-02-001) issued on March 13, 2002.

NRC Bulletin 2002-01, "Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity," issued on March 18, 2002.

Licensee Intra-Company Memorandum from S. Loehlein to H. Bergendahl, dated March 22, 2002, provides preliminary Probable Cause Summary Report. The four page summary report was made publicly available on both the NRC external web page and on the Davis-Besse docket file.

On March 25, 2002, an information only telecon was held between the NRR staff and the licensee to discuss proposed modifications to the reactor pressure vessel head. The licensee proposes to plug current nozzles 2 and 11 and install a 13-inch diameter stainless steel plate over the current nozzle 3 location. This will require multiple relief requests from the ASME Code. In addition, the licensee was informed that the public must be involved in this review process and that the staff anticipate a public meeting to discuss the proposed modifications.

On March 25, 2002, the licensee was informed that the staff did not consider the preliminary Probable Cause Summary Report to be complete and that additional staff questions would be forthcoming. In addition, the staff informed the licensee that a public meeting to discuss the root cause would be appropriate and that such a meeting would need to be held in advance of submitting proposed modifications to the reactor pressure vessel head.

The licensee has been requested to address the ACRS subcommittee on April 9, 2002. The purpose is to discuss information available to date and the probable root cause. The staff informed the licensee that a discussion of proposed modifications would be more appropriate for the May ACRS meeting. The licensee was also requested to be in attendance, but not prepared to make a presentation, at the full ACRS committee meeting on April 11, 2002.

STAKEHOLDERS:

Interested external and internal stakeholders can include the following:

- External Stakeholders

  General public
  Media
  Public interest groups
  Nuclear industry organizations
  Licensees
  States
  Canada
  International Organizations (e.g., IAEA)

- Internal Stakeholders

  Office of Public Affairs
  Office of International Programs
  Office of Congressional Affairs
  Operations Center
  NRC employees in headquarters
  NRC employees in regions

Attachment 1: Internal NRC Action Plan for the Davis-Besse Loss of Metal in the Reactor Vessel Head

Attachment 2: Questions and Answers

Attachment 3: OPA's Questions and Answers

Attachment 4: Davis-Besse Communications Team Members

Attachment 5: Communication Action Plan Sequence

Attachment 6: NRC Press Release of March 12, 2002

Attachment 7: NRC Information Notice 2002-11

Attachment 8: Confirmatory Action Letter No. 3-02-001

Attachment 9: NRC Bulletin 2002-01

NRC INTERNAL ACTION PLAN FOR THE DAVIS-BESSE LOSS OF METAL IN THE REACTOR VESSEL HEAD

The staff is currently monitoring the licensee's activities relating to the identified condition. Daily telephone conference calls are being conducted at 2:00 p.m. EST between the licensee, NRR, and the NRC Region III office. The purpose of these calls is to provide the latest developments to the staff. Included in these calls are representatives of NRR's Division of Licensing Project Management, LPD3, the Division of Engineering, Materials & Chemical Engineering Branch, the Office of Research. Also included are NRC Region III's Division of Reactor Projects and the Division of Reactor Safety. Representatives from the AIT team will be present for these telephone conference calls.

Additional telephone conference calls are conducted as necessary for emergent issues.

Information Notice 2002-11, "Recent Experience with Degradation of Reactor Pressure Vessel Head, describing the Davis-Besse conditions, was issued on March 12, 2002. The staff is preparing further generic correspondence.

A separate web page has been developed for this issue that links from the Bulletin 2001-01 site. The web address is http://www.nrc.gov/reactors/operating/ops-experience/vessel-head-degradation.html.

NRC Region III has established an Augmented Inspection Team (AIT), comprised of metallurgical and engineering specialists, to monitor the utility's investigation and evaluation of the cavity and its determination of the conditions causing the damage. The entrance briefing for the AIT was held on March 12, 2002. The Region issued a press release concerning the AIT on March 12, 2002. Following completion of the inspection, the NRC will hold a public meeting in the plant vicinity to discuss the inspection findings.

Commissioner Assistants briefings will be held as requested. In addition, daily interaction with the EDO staff and Region III are maintained to provide updates. The initial briefing is scheduled for March 14, 2002.

A meeting with industry representatives at Headquarters in Rockville, Maryland, is scheduled for Tuesday, March 19, 2002, to better communicate issues and concerns. In addition, the staff also plans to conduct a separate public meeting at Headquarters on Wednesday, March 20, 2002, to brief NRC stakeholders on the overall issue.

ACRS subcommittee will be briefed on April 9, 2002.

Additional public meetings will be conducted prior to plant restart.

Attachment 1

QUESTIONS AND ANSWERS

Question 1: How was boric acid able to cause the substantial, undetected metal loss in the vessel head?

Response:

The root cause of the degraded condition at the Davis-Besse Nuclear Power Station (DBNPS) that resulted in a significant loss of metal in the reactor pressure vessel (RPV) head has not yet been determined. The characteristics surrounding the degraded area at Davis-Besse include a history of boron deposits on the RPV head for an extended period of time, a history of Control Rod Drive Mechanism (CRDM) flange leaks, and recently discovered through-wall leaks in the CRDM nozzle welds. At the present time, the relative roles of these characteristics in the degradation found at Davis-Besse is not fully understood and will be evaluated by the root cause evaluation.

The NRC has experience that corrosive effects of boric acid could have led to this condition. Boric acid is normally mixed with the reactor coolant system (RCS) water to provide reactivity control for the reactor. Any leakage from the RCS will result in the formation of boron crystals following the evaporation of water. Long-term RCS leakage from the CRDMs would lead to a buildup of boron crystals on the reactor pressure vessel (RPV) head. However, it is yet unclear how conditions conducive to boric acid corrosion were established on the vessel head. One theory, for example, is leakage from small axial cracks in the CRDM welds could lead to the wetting of the boron crystals by leaking primary coolant favoring the creation of a solution of boric acid.

The primary effect of boric acid leakage onto the ferritic steel RPV head is corrosion and general dissolution of the ferritic material. The rate of general corrosion (wastage) of ferritic steel from boric acid varies and depends on several conditions, including whether the boric acid remains in solution or is evaporated off. During shutdown conditions, the temperature of the RPV head is low and any moisture in the area condenses and mixes with the boron to form boric acid, leading to corrosive conditions on the RPV head. During power operations, the temperature of the RPV head is sufficiently high that any leaking primary coolant would be expected to flash to steam, leaving behind dry boric acid crystals. If the boric acid is dry (i.e., boric acid crystals), the corrosion rate is less severe.

Given the wide range of conditions around reactor primary coolant leakage sites and the wide variation in boric acid corrosion rates, the deleterious effects of boric acid on ferritic steel components indicate the importance of minimizing boric acid leakage, detecting and correcting leaks in a timely manner, and promptly cleaning any boric acid residue.

Licensees are not required to perform inspections of the reactor vessel head. Some facilities are not physically capable of inspecting the head due to encapsulated insulation that is custom fit directly on the reactor vessel. Boron deposits on the reactor vessel head at DBNPS were known but thought to be lacking the necessary moisture to create a corrosive environment.

Question 2: How serious an accident might have resulted from the unanticipated problem?

Response:

The exterior of the reactor vessel head is made of ferritic steel with a thickness of six inches or more. The underside of the reactor vessel head is clad with more corrosion-resistant stainless steel. At least one area of the RPV head at DBNPS experienced degradation that exposed the stainless steel clad of the RPV head to the atmosphere. If the yield strength of the remaining stainless steel clad had been exceeded, either due to continued corrosion of the vessel head ferritic steel, or due to an over-pressure transient, a Loss of Coolant Accident (LOCA) could have occurred. Such an accident would represent a serious challenge to the plant's safety systems.

All domestic reactors are designed to withstand a spectrum of LOCAs. The safety-related emergency core cooling systems are designed to mitigate a LOCA equivalent to a double ended rupture of the largest primary coolant pipe. If the area of the exposed cladding at DBNPS had failed, it is expected that consequences of this event would be enveloped by the large break LOCA that the plant is designed to withstand.

Question 3: Is this a generic issue? What other plants are candidates for potentially having the same problem?

Response:

While the root cause for the Reactor Pressure Vessel (RPV) head metal loss has not been identified, the issue could have generic implications. All pressurized water reactors use boric acid in the RCS and are susceptible to RCS leakage that may result in boron deposits on the reactor vessel head. The staff will issue generic communications to all pressurized water reactors in order obtain sufficient information to properly identify the safety implications.

Question 4: Will Davis-Besse be able (allowed) to repair the vessel head or must it replace it?

Response:

The licensee is currently evaluating options to either repair the reactor vessel head or replace it. Until the extent of the damage is fully understood, it is not clear whether repairs are feasible or practical.

Question 5: Will other plants be required to stand down promptly to perform the internal inspection required to determine if similar corrosion exists?

Response:

The NRC is aware of the generic implications and is currently working to formulate an appropriate response. The NRC issued an Information Notice on March 12, 2002, to inform licensees of the conditions at DBNPS and to alert licensees of the potential problem at their sites. NRC plans to generate generic communications that will request licensees to provide their bases for concluding that a condition similar to that found at Davis-Besse does not exist at their facility. It should be pointed out that the number of plants that have recently examined their vessel heads have concluded that boric acid corrosion/erosion had not taken place. Licensees need to inform us of the extent to which they have examined their vessel heads for this condition.

Question 6: If not, why not?

Response:

Not Applicable

Question 7: What is the availability of vessel heads? Could DBNPS jump the line on someone who's already ordered one but who doesn't have as imminent a need? Might there be a suitable vessel head from a plant that is being decommissioned?

Response:

Davis-Besse is evaluating options that include whether to repair or replace the RPV head. If they decide to choose the latter, NRC does not know what the availability or adaptability is of suitable vessel heads from other sources.

Question 8: Did the NRC take unnecessary chances with public safety in allowing DBNPS to delay by about two months its CRDM inspections and repairs which led to the current discovery?

Response:

NRC made a risk-informed decision to allow Davis-Besse to operate until February 16, 2002, before it shut down to inspect the CRDM nozzles based on the best information available to the staff at that time. The degradation of the reactor vessel head by boric acid corrosion/erosion was unforseen at the time of that decision. Had we been aware of it, we most likely would have shut the plant down immediately.

Question 9: **Is this discovery raising questions about the properties of boric acid and its potential corrosive interaction with critical internals?**

Response:

No. The reactor vessel internals are fabricated from stainless steel which is highly resistant to corrosion.

Question 10: **How substantial is the cladding's ability to withstand the corrosive effects of boric acid in this instance?**

Response:

The cladding material is constructed of stainless steel and is highly resistant to the corrosive effects of boric acid.

OPA's QUESTIONS AND ANSWERS

Question 1: How was boric acid able to cause the substantial, undetected metal loss in the vessel head?

Response:

The root cause has not yet been determined. Several possibilities are under consideration. The corrosive effects of boric acid could have led to this condition. Boric acid is normally mixed with the reactor coolant system (RCS) water to provide reactivity control for the reactor. Any leakage from the RCS through a cracked vessel head nozzle could have led to a buildup of boron crystals which played a role in corroding the steel of vessel head. The leaks at Davis Besse are located in an area that is difficult to inspect due to insulation that is custom fit directly onto the reactor pressure vessel head.

Question 2: How serious an accident might have resulted from the unanticipated problem?

Response:

Under a worst case scenario, if corrosion ate through the six-inch thick steel dome of the reactor pressure vessel head while the reactor was operating, this could cause a Loss of Coolant Accident. Parts of the vessel head could break loose under pressure (2,250 pounds per square inch) and damage surrounding control rods or other equipment. However, all reactors are designed to be able to deal with loss of coolant accidents. Each reactor is equipped with an emergency core cooling systems d to provide cooling water to the reactor core in the event of a large pipe break that would interrupt the normal flow of water. Automatic safety systems and highly trained reactor control room operators would take the necessary actions to ensure the reactor would be safely shut down.

Question 3: Is this a generic issue? What other plants are candidates for potentially having the same problem?

Response:

While the root cause has not been identified, the issue could have generic implications. All pressurized water reactors use boric acid in the reactor coolant system water and are thus susceptible to corrosion that might be caused by boric acid corrosion caused by leakage. The staff is issuing generic communications to all pressurized water reactors in order obtain sufficient information to properly identify the safety implications.

Question 4: Will Davis-Besse be able (allowed) to repair the vessel head or must it replace it?

Response:

The licensee is currently evaluating options to either repair the reactor vessel head or replace it. Until the extent of the damage is fully understood, it is not clear whether repairs are feasible or practical.

Question 5: Will other plants be required to stand down promptly to perform the internal inspection required to determine if similar corrosion exists?

Response:

The NRC issued an Information Notice on March 12, 2002, to inform licensees of the conditions at Davis Besse and to alert licensees of the potential problem at their sites. NRC plans to request additional information from its licensees, also. Many plants that have recently examined their vessel heads have concluded that boric acid corrosion/erosion had not taken place. Licensees need to inform us of any recent examinations/inspections.

Question 6: If not, why not?

Response:

Not Applicable

Question 7: What is the availability of vessel heads? Could DBNPS jump the line on someone who's already ordered one but who doesn't have as imminent a need? Might there be a suitable vessel head from a plant that is being decommissioned?

Response:

Davis-Besse is evaluating whether to repair or replace the vessel head. NRC does not know what the availability or adaptability is of suitable vessel heads from other sources.

Question 8: Did the NRC take unnecessary chances with public safety in allowing DBNPS to delay by about two months its CRDM inspections and repairs which led to the current discovery?

Response:

No. Using all of the best available information available at the time, NRC decided last December to allow Davis-Besse to operate until February 16 before shutting down to inspect its CRDM nozzles. That decision was made following a November 28 meeting with representatives from FirstEnergy, who provided the NRC technical staff with information on the condition of the reactor vessel head nozzles, the results of videotaped inspections during three prior refueling outages in 1996, 1998 and 2000, in which no nozzle defects were found, and a detailed analysis of the possible risks in permitting continued operation, and possible accident consequences. Inherent in NRC's decision to permit continued operation through mid-February, was the recognition that even if there was a failure of the vessel head or its nozzles, safety systems exist to cope with any resulting accident. To further reduce the likelihood of any vessel failure, the licensee said they would reduce reactor pressure vessel head temperature, to would reduce the growth of any cracks that did exist.

Question 9: Is this discovery raising questions about the properties of boric acid and its potential corrosive interaction with critical internals?

Response:

No. The reactor vessel internals are fabricated from stainless steel which is highly resistant to corrosion.

Question 10: How substantial is the cladding's ability to withstand the corrosive effects of boric acid in this instance?

Response:

The cladding material is constructed of stainless steel and is highly resistant to the corrosive effects of boric acid.

Davis-Besse Communications Team Members

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Attachment 4

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Tom Koshy - 415-1176
Jerry Dozier - 415-1014
Ian Jung - 415-1837

OSTP
Spiros Droggistis - 415-2367

COMMUNICATION ACTION PLAN
SEQUENCE

Date/Time	Action	Responsible Organization
Everyday 2:00 p.m.	Conduct RPV Status Calls with Davis-Besse **(ONGOING)**	DLPM DE Region III RES
03/08/2002	Issue Preliminary Notification of Event or Unusual Occurrence (PNO-III-02-006) **(COMPLETED)**	Region III (Holmberg/Jacobson)
03/12/2002	Issue NRC Information Notice (IN-02-011) on Davis-Besse Vessel Damage **(COMPLETED)**	RORP/DRIP
03/11/2002	Initiate Communication Plan for Davis-Besse Vessel Damage **(COMPLETED)**	DLPM (Mendiola)
03/12/2002	Issue AIT Press Release - "NRC to Conduct Augmented Inspection of Davis-Besse Vessel Damage" **(COMPLETED)**	Region III (Strasma)
03/13/2002 11:00 a.m.	Conduct information conference call with industry groups (MRP/NEI)	DE (Bloom/Wetzel)
03/14/2002 10:00 a.m.	Conduct Commissioner Technical Assistant Briefing	DLPM DE Region III RES
03/15/2002 COB	Complete NRC Bulletin for Davis-Besse Vessel Damage	DE (Karwoski)
03/18/2002 2:00 p.m.	Brief CRGR on Davis-Besse Vessel Damage Bulletin	DE (Karwoski)
03/19/2002 11:00 a.m.	Conduct public meeting between NRC and Industry Representatives at NRC Headquarters **(COMPLETED)**	DE (Bloom)
03/20/2002 1:00 - 5:00 p.m.	Conduct briefing for external stakeholders on Davis-Besse Vessel Damage at NRC Headquarters **(COMPLETED)**	DLPM DE Region III
03/13/2002	Create an NRC Website to support Davis-Besse Vessel Damage information. **(COMPLETED)**	DE
TBD	Provide pre-brief with local County Commissioners prior to public AIT exit meeting	Region III

Date/Time	Action	Responsible Organization
04/05/2002 9:00 a.m.	AIT public exit meeting at Oak Harbor High School in Oak Harbor, Ohio	Region III DE
04/05/2002 1:00 p.m.	End-of-Cycle Reactor Oversight Program public meeting at Oak Harbor High School in Oak Harbor, Ohio	Region III
04/09/2002	Conduct briefing for ACRS subcommittee on Davis-Besse Vessel Damage at NRC Headquarters	DLPM DE Region III
04/11/2002	Discuss Davis-Besse events before full ACRS Committee	Region III DE
TBD	Public Meeting to discuss root cause evaluation	Region III DE Licensee
TBD	Public Meeting to discuss proposed modifications to reactor pressure vessel head	Licensee DE

NRC NEWS

U.S. NUCLEAR REGULATORY COMMISSION
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Lisle IL 60532
Web Site: http://www.nrc.gov E-mail: opa3@nrc.gov

No. III-02-002 March 12, 2002
CONTACT: Jan Strasma (630) 829-9663

NRC TO CONDUCT AUGMENTED INSPECTION OF DAVIS-BESSE REACTOR VESSEL DAMAGE

The Nuclear Regulatory Commission has begun a Augmented Team Inspection into damage to a small area of the top of the reactor vessel, apparently caused by corrosion, at the Davis-Besse Nuclear Power Station. The plant, located at Oak Harbor, Ohio, is operated by FirstEnergy Corporation.

The plant has been shut down since February 15 for refueling and maintenance.

The cavity in the top of the reactor vessel was discovered during inspection and repair activities during the outage. It is about 4 inches by 5 inches and approximately 6 inches deep. The reactor vessel head, fabricated of carbon steel with a stainless steel liner, is about 6 1/2 inches thick.

During the outage, plant personnel inspected 69 control rod tubes which pass through the reactor vessel head. The NRC issued a bulletin last August requiring the detailed inspections at Davis Besse and other sites after cracking problems were found at several other nuclear plants.

Using ultrasonic techniques, FirstEnergy workers found cracks through the tube walls in two tubes, and lesser cracks in three additional tubes.

During repairs to one of the tubes with through-wall cracks, workers discovered the void adjacent to the tube.

The NRC's Augmented Inspection Team, comprised of metallurgical and engineering specialists, will monitor the utility's investigation and evaluation of the cavity and its determination of the conditions causing the damage. The inspection is being conducted to better understand the circumstances surrounding the corrosion and damage and to consider whether similar conditions might exist at other plants.

The preliminary cause of the damage appears to be corrosion as a result of boric acid deposits. Boric acid is a constituent of the water in the reactor cooling system and was apparently deposited on the reactor vessel through the leaking crack in the control rod tube or some other source.

The utility is developing its plans for repair of the reactor vessel head. The NRC will review the utility's plans.

Attachment 7

Following completion of the inspection, the NRC will hold a meeting in the plant vicinity to discuss the inspection findings. The meeting will be open to public observation.

The inspection report, issued about four weeks after the inspection, will be available on the agency's website and through its Electronic Reading Room at http://www.nrc.gov as an Agencywide Document Access and Management System (ADAMS) document. Help in using ADAMS is available through the NRC Public Document Room at 301/415-4737 or 800/397-4209.

The NRC has issued an Information Notice to operating nuclear plants to inform them of the corrosion damage at Davis-Besse. The notice will be available online in the Electronic Reading Room with the accession number of ML020700556.

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION
WASHINGTON, DC 20555-0001

March 12, 2002

NRC INFORMATION NOTICE 2002-11: RECENT EXPERIENCE WITH DEGRADATION OF REACTOR PRESSURE VESSEL HEAD

Addressees

All holders of operating licenses for pressurized-water reactors (PWRs), except those who have permanently ceased operations and have certified that fuel has been permanently removed from the reactor.

Purpose

The U.S. Nuclear Regulatory Commission (NRC) is issuing this information notice to inform addressees about findings from recent inspections and examinations of the reactor pressure vessel (RPV) head at Davis-Besse Nuclear Power Station. It is expected that recipients will review the information for applicability to their facilities and consider actions, as appropriate, to avoid similar problems. However, suggestions contained in this information notice are not NRC requirements; therefore, no specific action or written response is required.

Description of Circumstances

On February 16, 2002, the Davis-Besse facility began a refueling outage that included inspection of the vessel head penetration (VHP) nozzles, which focused on the inspection of control rod drive mechanism (CRDM) nozzles, in accordance with the licensee's commitments to NRC Bulletin 2001-01, "Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles," which was issued on August 3, 2001. These inspections identified axial indications in three CRDM nozzles, which had resulted in pressure boundary leakage. Specifically, these indications were identified in CRDM nozzles 1, 2, and 3, which are located near the center of the RPV head. These findings were reported to the NRC on February 27, 2002, and supplemented on March 5 and March 9, 2002. The licensee decided to repair these three nozzles, as well as two other nozzles that had indications but had not resulted in pressure boundary leakage.

The repair process for these nozzles included roll expanding the CRDM nozzle material into the surrounding RPV head material, followed by machining along the axis of the CRDM nozzle to an elevation above the indications in the nozzle material. On March 6, 2002, the machining process on CRDM nozzle 3 was prematurely terminated and the machining apparatus was removed from the nozzle. During the removal process, nozzle 3 was mechanically agitated and subsequently displaced in the downhill direction (i.e., tipped away from the top of the RPV head) until its flange contacted the flange of the adjacent CRDM nozzle.

ML020700556

To identify the cause of the CRDM nozzle displacement, the licensee began an investigation into the condition of the RPV head surrounding CRDM nozzle 3. This investigation included removing the CRDM nozzle from the RPV head, removing boric acid deposits from the top of the RPV head, and performing ultrasonic thickness measurements of the RPV head in the vicinity of CRDM nozzles 1, 2, and 3. Upon completing the boric acid removal on March 7, 2002, the licensee conducted a visual examination of the area, which identified a large cavity in the RPV head on the downhill side of CRDM nozzle 3. Followup characterization by ultrasonic testing indicated wastage of the low alloy steel RPV head material adjacent to the nozzle. The wastage area was found to extend approximately 5 inches downhill on the RPV head from the penetration for CRDM nozzle 3, with a width of approximately 4 to 5 inches at its widest part. The minimum remaining thickness of the RPV head in the wastage area was found to be approximately ⅜ inch. This thickness was attributed to the thickness of the stainless steel cladding on the inside surface of the RPV head, which is nominally ⅜ inch thick.

Background

The Davis-Besse Nuclear Power Station has an RPV head that is constructed from low alloy steel, fabricated in accordance with the American Society of Mechanical Engineers (ASME) specification SA-533, Grade B, Class1, and clad on the inside surface with stainless steel. Of those 69 VHP nozzles, 61 are used for CRDMs, 7 are spare (empty) nozzles, and 1 is used for the RPV head vent piping. Each of the 69 nozzles is approximately 4 inches in outside diameter, with a wall-thickness of approximately ⅝ inch. Each is constructed of Alloy 600 and is attached to the RPV head by a partial-penetration, J-groove weld using Alloy 82 and 182. The distance from the center of one nozzle to the center of the next is approximately 12 inches.

The vessel head is insulated with metal reflective insulation, which is located on a horizontal plane slightly above the RPV head (i.e., it is not in direct contact with the head). The minimum distance between the RPV head and the insulation is approximately 2 inches at the center (top) of the head. The CRDM nozzles pass from the RPV head through the insulation and terminate at flanges to which the CRDM housings are attached.

The limited gap between the insulation and the RPV head does not impede the performance of a visual inspection of the CRDM nozzles, as described in Bulletin 2001-01. This is because the top of the RPV head is surrounded by a service structure that has 18 openings (referred to as "weep holes") near the bottom of the structure, through which small cameras can be inserted to facilitate visual inspections of the RPV head.

During refueling outages in 1998 and 2000, the licensee performed visual inspections of the RPV head surface that was accessible through the service structure weep holes. The scope of these visual inspections covered the bare metal of the RPV head to identify the presence of boric acid deposits, which would be indicative of primary coolant leakage. These inspections also included checking for leakage from any of the CRDM flanges, located above the insulation, in response to Generic Letter 88-05, "Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components," which the NRC issued on March 17, 1988.

The visual inspections in 1998 showed an uneven layer of boric acid deposits scattered over the RPV head (including deposits near CRDM nozzle 3). The outside diameter of the CRDM nozzles had white streaks, which indicated to the licensee that the boric acid evident on the head flowed downward from leakage in the CRDM flanges.

During the refueling outage in 2000, the licensee also performed visual inspections of the CRDM flanges and nozzles. Above the RPV head insulation, those inspections revealed five CRDM flanges with evidence of leakage, including one flange that was the principal leakage point. Boric acid deposits on the vertical faces of three of these five flanges and the associated nozzles confirmed leakage from the flanges. Similarly, one of the other two leaking CRDM flanges had boric acid deposits between the flange and the insulation, which indicated leakage from the flange. All of these leaking flanges were repaired by replacing their gaskets. The faces of the flange that was the principal leakage point were also machined to ensure a better seal.

Visual inspections performed below the RPV head insulation during the 2000 refueling outage indicated some accumulation of boric acid deposits on the RPV head. These deposits were located beneath the leaking flanges, with clear evidence of downward flow from the flange area. No visible evidence of CRDM nozzle leakage (i.e., leakage from the gap between the nozzle and the RPV head) was detected. The licensee described that the RPV head area was cleaned with demineralized water to the greatest extent possible, while trying to maintain the dose as low as reasonably achievable (ALARA). Subsequent video inspection of the partially cleaned RPV head and nozzles was performed for future reference.

A subsequent review of the 1998 and 2000 inspection videotapes in 2001 confirmed that there was no evidence of leakage from the RPV head nozzles, although many areas of the RPV head were not accessible because of persistent boric acid deposits that the licensee did not clean because of ALARA issues (including the region around nozzle 3).

The inspections in 2002 did not reveal any visual evidence of flange leakage from above the RPV head. However, as discussed above, three CRDM nozzles had indications of cracking (identified by ultrasonic testing of the nozzles), which could result in leakage from the RPV to the top of the RPV head.

Discussion

The following documents describe reactor operating experience with boric acid corrosion of ferritic steel reactor coolant pressure boundary components in PWR plants:

- Information Notice 86-108, "Degradation of Reactor Coolant System Pressure Boundary Resulting from Boric Acid Corrosion," issued December 29, 1986
- Information Notice 86-108, Supplement 1, "Degradation of Reactor Coolant System Pressure Boundary Resulting from Boric Acid Corrosion," issued April 20, 1987
- Information Notice 86-108, Supplement 2, "Degradation of Reactor Coolant System Pressure Boundary Resulting from Boric Acid Corrosion," issued November 19, 1987
- Information Notice 86-108, Supplement 3, "Degradation of Reactor Coolant System Pressure Boundary Resulting from Boric Acid Corrosion," issued January 5, 1995
- Generic Letter 88-05, "Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants," issued March 17, 1988

Several instances of boric acid corrosion discussed in these generic communications are associated with corrosion of the RPV head. NRC Information Notice 86-108, Supplement 1, for example, described an instance in which boric acid had severely corroded three of the RPV flange bolts, the control rod drive shroud support, and an instrument tube seal clamp. Similarly, NRC Information Notice 86-108, Supplement 2, described an instance in which boric acid resulted in nine pits in the surface of the RPV head, ranging in depth from 0.9 to 1 cm [approximately 0.4 inch] and ranging in diameter from 2.5 to 7.5 cm [1 to 3 inches].

As discussed in Information Notice 86-108, Supplement 2, the primary effect of boric acid leakage onto the ferritic steel RPV head is wastage or general dissolution of the material. Pitting, stress corrosion cracking (SCC), intergranular attack, and other forms of corrosion are not generally of concern in concentrated boric acid solutions at elevated temperatures such as those that may occur on the surface of the RPV head. The rate of general corrosion (wastage) of ferritic steel from boric acid varies and depends on several conditions, including whether the boric acid is dry or in solution. If the boric acid is dry (i.e., boric acid crystals), the corrosion rate is less severe; however, boric acid crystals are not completely benign to carbon steel. During operation, the temperature of the RPV head is sufficiently high that any leaking primary coolant would be expected to flash to steam, leaving behind dry boric acid crystals.

Given the wide range of conditions around reactor primary coolant leakage sites and the wide variation in boric acid corrosion rates, the deleterious effects of boric acid on ferritic steel components indicate the importance of minimizing boric acid leakage, detecting and correcting leaks in a timely manner, and promptly cleaning any boric acid residue.

The investigation of the causative conditions surrounding the degradation of the RPV head at Davis-Besse is continuing. Boric acid or other contaminants could be contributing factors. As discussed above, factors contributing to the degradation might also include the environment of the head during both operating and shutdown conditions (e.g., wet/dry), the duration for which the RPV head is exposed to boric acid, and the source of the boric acid (e.g., leakage from the CRDM nozzle or from sources above the RPV head such as CRDM flanges).

<u>Related Generic Communications</u>

Bulletin 2001-01, "Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles," August 3, 2001.

Bulletin 82-02, "Degradation of Threaded Fasteners in the Reactor Coolant Pressure Boundary of PWR Plants," June 2, 1982.

Generic Letter 88-05, "Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants," March 17, 1988.

Generic Letter 97-01, "Degradation of Control Rod Drive Mechanism Nozzles and Other Vessel Closure Head Penetrations," April 1, 1997.

Information Notice 80-27, "Degradation of Reactor Coolant Pump Studs," June 11, 1980.

Information Notice 82-06, "Failure of Steam Generator Primary Side Manway Closure Studs," March 12, 1982.

Information Notice 86-108, "Degradation of Reactor Coolant System Pressure Boundary Resulting from Boric Acid Corrosion," December 29, 1986.

Information Notice 86-108, Supplement 1, "Degradation of Reactor Coolant System Pressure Boundary Resulting from Boric Acid Corrosion," April 20, 1987.

Information Notice 86-108, Supplement 2, "Degradation of Reactor Coolant System Pressure Boundary Resulting from Boric Acid Corrosion," November 19, 1987.

Information Notice 86-108, Supplement 3, "Degradation of Reactor Coolant System Pressure Boundary Resulting from Boric Acid Corrosion," January 5, 1995.

Information Notice 90-10, "Primary Water Stress Corrosion Cracking of INCONEL 600," February 23, 1990.

Information Notice 94-63, "Boric Acid Corrosion of Charging Pump Casing Caused by Cladding Cracks," August 30, 1994.

Information Notice 96-11, "Ingress of Demineralizer Resins Increases Potential for Stress Corrosion Cracking of Control Rod Drive Mechanism Penetrations," February 14, 1996.

Information Notice 2001-05, "Through-Wall Circumferential Cracking of Reactor Pressure Vessel Head Control Rod Drive Mechanism Penetration Nozzles at Oconee Nuclear Station, Unit 3," April 30, 2001.

This information notice does not require any specific action or written response. If you have any questions about the information in this notice, please contact one of the technical contacts listed below or the appropriate project manager in the NRC's Office of Nuclear Reactor Regulation (NRR).

/RA/
William D. Beckner, Program Director
Operating Reactor Improvements Program
Division of Regulatory Improvement Programs
Office of Nuclear Reactor Regulation

Technical contacts:

Allen Hiser, NRR
(301) 415-1034
E-mail: alh1@nrc.gov

Ken Karwoski, NRR
(301) 415-2752
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Jerry Dozier, NRR
(301) 415-1014
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Attachment: List of Recently Issued NRC Information Notices

March 13, 2002

CAL No. 3-02-001

Mr. Howard Bergendahl
Vice President - Nuclear, Davis-Besse
FirstEnergy Nuclear Operating Company
Davis-Besse Nuclear Power Station
5501 North State Route 2
Oak Harbor, OH 43449-9760

SUBJECT: CONFIRMATORY ACTION LETTER - DAVIS-BESSE NUCLEAR POWER STATION

Dear Mr. Bergendahl:

As a result of your identification of extensive degradation to the pressure boundary material of the reactor pressure vessel (RPV) head, the NRC dispatched an Augmented Inspection Team to your facility on March 12, 2002. A copy of the charter for the Augmented Inspection Team is enclosed for your information. The RPV head degradation was discovered on March 6, 2002, during repair activities that followed from the identification of cracks in several Control Rod Drive Mechanism (CRDM) penetration tubes. The initial penetration examinations which led to the crack identification were performed in response to Bulletin 2001-01, "Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles."

Following discussions between you and Mr. J. A. Grobe, Director, Division of Reactor Safety, Region III, you issued a letter to document commitments for activities to evaluate and resolve the RPV head degradation issue. In your letter to the NRC dated March 12, 2002, (Serial Number 1-1265), you identified several specific activities you intend to implement to resolve the reactor pressure vessel head material degradation issue. Those items are restated below as items (1) through (6) as clarified during additional telephone discussions with you on March 13, 2002. It is our understanding that you will take the following actions:

(1) Quarantine components or other material from the RPV head and CRDM nozzle penetrations that are deemed necessary to fully address the root cause of the occurrence of degradation of the leaking penetrations. Prior to implementation, plans for further inspection and data gathering to support determination of the root cause will be provided to the NRC for review and comment.

(2) Determine the root cause of the degradation around the RPV head penetrations, and promptly meet with the NRC to discuss this information after you have reasonable confidence in your determination.

(3) Evaluate and disposition the extent of condition throughout the reactor coolant system relative to the degradation mechanisms that occurred on the RPV head.

Attachment 8

(4) Obtain NRC review and approval of the repair or modification and testing plans for the RPV head, prior to implementation of those activities. Prior to restart of the reactor, obtain NRC review and approval of any modification and testing activity related to the reactor core or reactivity control systems.

(5) Prior to the restart of the unit, meet with the NRC to obtain restart approval. During that meeting, we expect you will discuss your root cause determination, extent of condition evaluations, and corrective actions completed and planned to repair the damage and prevent recurrence.

(6) Provide a plan and schedule to the NRC, within 15 days of the date of this letter, for completing and submitting to the NRC your ongoing assessment of the safety significance for the RPV head degradation.

Pursuant to Section 182 of the Atomic Energy Act, 42 U.S.C 2232, you are required to:

(1) Notify me immediately if your understanding differs from that set forth above;

(2) Notify me if for any reason you cannot compete the actions within the specified schedule and advise me in writing of your modified schedule in advance of the change; and

(3) Notify me in writing when you have completed the actions addressed in this Confirmatory Action Letter.

Issuance of this Confirmatory Action Letter does not preclude issuance of an order formalizing the above commitments or requiring other actions on the part of the licensee; nor does it preclude the NRC from taking enforcement action for violations of NRC requirements that may have prompted the issuance of this letter. In addition, failure to take the actions addressed in this Confirmatory Action Letter may result in enforcement action.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response will be made 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), to the extent possible, it should not include any personal privacy, proprietary, or safeguards information so that it can be made available to the public without redaction. ADAMS is accessible from the NRC Web site at http://www.nrc.gov/NRC/ADAMS/index.html (the Public Electronic Reading Room). If personal privacy or proprietary information is necessary to provide an acceptable response, then please provide a bracketed copy of your response that identifies the information that should be protected and a redacted copy of your response that deletes such information. If you request withholding of such material, you must specifically identify the portions of your response that you seek to have withheld and provide in detail the bases for your claim of withholding (e.g., explain why the disclosure of information will create an unwarranted invasion of personal privacy or provide the information required by 10 CFR 2.790(b) to support a request for withholding confidential commercial or financial information). If safeguards information is necessary to provide an acceptable response, please provide the level of protection described in 10 CFR 73.21.

Sincerely,

/RA/

J. E. Dyer
Regional Administrator

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

Enclosure: Augmented Inspection Team Charter - Davis-Besse Reactor Vessel Head Material Loss

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

UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION
WASHINGTON, DC 20555-0001

March 18, 2002

NRC BULLETIN 2002-01: REACTOR PRESSURE VESSEL HEAD DEGRADATION AND REACTOR COOLANT PRESSURE BOUNDARY INTEGRITY

Addressees

All holders of operating licenses for pressurized-water nuclear power reactors, except those who have permanently ceased operations and have certified that fuel has been permanently removed from the reactor pressure vessel, and all holders of operating licenses for boiling-water reactors for information.

Purpose

The U.S. Nuclear Regulatory Commission (NRC) is issuing this bulletin to require pressurized-water reactor (PWR) addressees to submit:

(1) information related to the integrity of the reactor coolant pressure boundary including the reactor pressure vessel head and the extent to which inspections have been undertaken to satisfy applicable regulatory requirements, and

(1) the basis for concluding that plants satisfy applicable regulatory requirements related to the structural integrity of the reactor coolant pressure boundary and future inspections will ensure continued compliance with applicable regulatory requirements, and

(3) a written response to the NRC in accordance with the provisions of Title 10, Section 50.54(f), of the *Code of Federal Regulations* (10 CFR 50.54(f)) if they are unable to provide the information or they can not meet the requested completion dates.

Background

On August 3, 2001, the NRC issued Bulletin 2001-01, "Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles" (ADAMS Accession Number ML012080284). That bulletin described instances of cracked and leaking Alloy 600 reactor pressure vessel head penetration nozzles, including control rod drive mechanism and thermocouple nozzles. In response to that bulletin, pressurized-water reactor licensees provided their plans for inspecting their reactor pressure vessel head penetrations and/or the outside surface of the reactor pressure vessel head to determine whether the nozzles were leaking. Some plants have completed these inspections.

In conducting these inspections at the Davis-Besse Nuclear Power Station in February and March 2002, the licensee identified three control rod drive mechanism nozzles with indications of axial cracking that resulted in reactor coolant pressure boundary leakage. One of these three control rod drive mechanism nozzles also had a circumferential indication which was not through-wall, and therefore, did not result in reactor coolant pressure boundary leakage. These were not unexpected findings, given the high susceptibility of the Davis-Besse plant to vessel head penetration nozzle cracking (as described in NRC Bulletin 2001-01). These axial indications were identified in control rod drive mechanism nozzles 1, 2, and 3, which are located near the center of the reactor pressure vessel head. Because of these indications, the licensee decided to repair control rod drive mechanism nozzles 1, 2, and 3, as well as two other nozzles that had indications but had not resulted in reactor coolant pressure boundary leakage.

The repair process for these nozzles included roll expanding the control rod drive mechanism nozzle material into the surrounding reactor pressure vessel head material, followed by machining along the axis of the control rod drive mechanism nozzle to an elevation above the indications in the nozzle material. On March 6, 2002, the machining process on control rod drive mechanism nozzle 3 was prematurely terminated and the machining apparatus was removed from the nozzle. During the removal process, control rod drive mechanism nozzle 3 was mechanically agitated and subsequently displaced, or tipped, in the downhill direction (away from its vertical position on top of the dome-shaped reactor pressure vessel head) until its flange contacted the flange of the adjacent control rod drive mechanism nozzle.

To identify the cause of the control rod drive mechanism nozzle displacement, the licensee began an investigation into the condition of the reactor pressure vessel head surrounding control rod drive mechanism nozzle 3. This investigation included removing the nozzle and boric acid deposits from the reactor pressure vessel head, and ultrasonically measuring the thickness of the reactor pressure vessel head in the vicinity of control rod drive mechanism nozzles 1, 2, and 3. Upon completing the boric acid removal on March 7, 2002, the licensee conducted a visual examination of the area, which identified a cavity in the reactor pressure vessel head on the downhill side of control rod drive mechanism nozzle 3 (i.e., the lowest portion of the nozzle extending out of the reactor pressure vessel head). Follow-up characterization by ultrasonic testing indicated thinning of the reactor pressure vessel head material adjacent to the nozzle. The thinned area was initially estimated to extend approximately 5 inches from the penetration for control rod drive mechanism nozzle 3; however, from more recent results, the thinned area extends approximately 7 inches from the nozzle at the stainless steel cladding, indicating the degradation was more severe at the bottom of the cavity than on the top. The width of the exposed area was approximately 4 to 5 inches at its widest part. The minimum remaining thickness of the reactor pressure vessel head in the thinned area was found to be approximately 3/8-inch. This thickness was attributed to the thickness of the stainless steel cladding on the inside surface of the reactor pressure vessel head, which is nominally 3/8-inch thick.

NRC Information Notice 2002-11, "Recent Experience with Degradation of Reactor Pressure Vessel Head," dated March 12, 2002, provides additional detail concerning the Davis-Besse inspection findings, the design and configuration of the Davis-Besse reactor pressure vessel head and service structure, and past inspections.

Since the NRC issued Information Notice 2002-11, additional information has become available concerning the condition of the reactor pressure vessel head at Davis-Besse. Specifically, the 3/8-inch stainless steel cladding near control rod drive mechanism nozzle 3 was found to be deflected upwards by about 1/8-inch over a 4-inch distance, indicating that the material had yielded. This is significant because the 3/8-inch cladding had essentially become the reactor coolant pressure boundary near the affected nozzle after the base material of the reactor pressure vessel head had degraded.

In addition, two areas of less severe thinning have been detected near control rod drive mechanism nozzle 2. At the time this bulletin was being prepared, it was not known whether these two areas were connected because one was detected on the outer surface of the reactor pressure vessel head and the other was detected at the inner surface. In addition, the dimensions of these areas were not known at the time this bulletin was being prepared. On the basis of preliminary information, the affected area appeared to be much smaller in size than the area located near control rod drive mechanism nozzle 3.

The investigation of the causative conditions surrounding the degradation of the reactor pressure vessel head at Davis-Besse is continuing. Boric acid or other contaminants could be contributing factors, as could steam jet cutting caused by leakage from the nozzle. Other factors contributing to the degradation might include the environment (e.g., wet/dry) surrounding the reactor pressure vessel head during both operating and shutdown conditions, the duration for which the reactor pressure vessel head was exposed to boric acid, and the source of the boric acid (e.g., leakage from cracks in the reactor pressure vessel head penetration nozzle or from sources above the reactor pressure vessel head such as control rod drive mechanism flanges).

Discussion

The reactor pressure vessel head is an integral part of the reactor coolant pressure boundary, and its integrity is important to the safe operation of the plant. The recent identification of thinning of the reactor pressure vessel head at Davis-Besse raises questions regarding licensees' practices for identifying and resolving degradation of the reactor coolant pressure boundary, including licensees' models for assessing corrosion that is caused by contaminants such as boric acid in the operating environment of the reactor pressure vessel head, or erosion that is caused by flow through a through-wall defect in a vessel head penetration nozzle.

As indicated above, the investigation of the causative conditions surrounding the degradation of the reactor pressure vessel head at Davis-Besse is continuing. An evaluation of the available information leads to several observations. First, the base metal of the reactor pressure vessel head degraded near leaking nozzles. Second, the reactor pressure vessel head has had boric acid deposits in the vicinity of the degraded areas for at least the past several years; that is, the deposits were not fully removed during the last several refueling outages. Third, some of the boric acid deposits on the top of the reactor pressure vessel head came from leaking control rod drive mechanism flanges, as discussed in NRC Information Notice 2002-11. Evaluations are on-going on whether similar degradation could occur (1) with just deposits and/or contaminants on the reactor pressure vessel head (i.e., without a leaking nozzle), (2) with just a leaking nozzle (i.e., without deposits and/or contaminants on the reactor pressure vessel head), or (3) whether both conditions are necessary to cause the observed degree of degradation. That is, the interaction between these two conditions and their respective influences in initiating the degradation of the reactor pressure vessel head is still being evaluated.

Although the root cause is still under investigation, preliminary assessments indicate that boric acid was a contributor. Corrosion of ferritic material, such as the base metal of the reactor pressure vessel head, is well documented in the list of related generic communications identified in this bulletin. In response to NRC Generic Letter 88-05, "Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants," dated March 17, 1988, licensees committed to implement a systematic program to monitor locations where boric acid leakage could occur, and to implement measures to prevent degradation of the reactor coolant pressure boundary by boric acid corrosion.

Historically, these programs have assumed that there is only a small potential for wastage of the reactor pressure vessel head attributable to leakage of primary coolant through the vessel head penetration nozzles. The supporting analyses assumed that coolant escaping from a penetration would flash to steam, leaving behind deposits of boric acid crystals. Typically, these crystals are assumed to accumulate on the reactor pressure vessel head; however, such deposits are assumed to cause minimal corrosion while the reactor is operating because the temperature of the reactor pressure vessel head is above 500 F during operation, and dry boric acid crystals are not very corrosive. Therefore, wastage is typically expected to occur only during outages when the boric acid could be in solution, such as when the temperature of the reactor pressure vessel head falls below 212 F. However, the findings at Davis-Besse bring into question the reliability of this model.

As indicated above, one of the contributing factors to the observed degradation could be the presence of boric acid deposits on the top of the reactor pressure vessel head. The procedures for determining whether these deposits could be present on the top of the reactor pressure vessel head are plant-specific because they are contingent on plant-specific design characteristics. For example, some plants have the reactor pressure vessel head insulation sufficiently offset from the head itself, in order to allow effective visual examination (as discussed in Bulletin 2001-01). Other plants have the insulation offset from the reactor pressure vessel head, but in a contour matching that of the head itself, in a design that requires special tooling and procedures to perform an effective visual examination. Still other plants have the reactor pressure vessel head insulation directly adjacent or attached to the head itself, in a design that potentially requires the removal of the insulation to permit an effective visual examination.

Plants for which limited data are available from direct visual inspection must use another method to determine whether boric acid deposits could be on the top of the reactor pressure vessel head. One method includes assessing whether boric acid (1) has leaked from locations above the reactor pressure vessel head, (2) has penetrated the insulation by flowing through the insulation or through gaps in the insulation, and (3) has precipitated onto the reactor pressure vessel head or has allowed precipitants to fall onto the reactor pressure vessel head.

One of the other factors suspected of contributing to the degradation observed at Davis-Besse is the presence of a leaking reactor pressure vessel head penetration nozzle. The integrity of reactor pressure vessel head penetration nozzles is discussed in NRC Bulletin 2001-01.

That bulletin discusses an industry model for assessing the susceptibility of plants to primary water stress corrosion cracking at the reactor pressure vessel head penetration nozzles. The industry's susceptibility ranking model has limitations, such as large uncertainties and the inability to predict when cracking will occur. Nonetheless, this model does provide a starting point for assessing the potential for cracking of reactor pressure vessel head penetration nozzles in pressurized water reactor plants.

Inspections performed to date at plants with high and moderate susceptibility have generally confirmed the ability of the model to predict a plant's relative susceptibilities; however, a plant with a ranking of 14.3 effective full-power years from the Oconee 3 condition (at the time when circumferential cracking was identified at Oconee 3 in March 2001) identified three nozzles with cracking; other plants with fewer effective full-power years from the Oconee 3 condition did not identify cracking.

Several plants have repaired nozzles with through-wall degradation (i.e., nozzles that leaked). Results from these inspections do not appear to indicate the presence of a degraded area in the reactor pressure vessel base metal. However, the extent to which the inspection techniques used would have detected such an area or the degree to which attention was placed on identifying this form of degradation, varies from plant to plant. Some inspection and repair methods may not have been capable of identifying the presence of a void in the carbon steel head adjacent to the cladding interface.

The NRC has developed Web pages to keep the public informed of generic activities related to Alloy 600 cracking and reactor pressure vessel head degradation:

> http://www.nrc.gov/reactors/operating/ops-experience/alloy600.html
> http://www.nrc.gov/reactors/operating/ops-experience/vessel-head-degradation.html

These Web pages provide links to information regarding the cracking identified to date, along with documentation of NRC interactions with industry (industry submittals, meeting notices, presentation materials, and meeting summaries). The NRC will continue to update these Web pages as new information becomes available.

Applicable Regulatory Requirements

Several provisions of the NRC regulations and plant operating licenses (Technical Specifications) pertain to reactor coolant pressure boundary integrity. The general design criteria (GDC) for nuclear power plants (Appendix A to 10 CFR Part 50), or, as appropriate, similar requirements in the licensing basis for a reactor facility, the requirements of 10 CFR 50.55a, and the quality assurance criteria of Appendix B to 10 CFR Part 50 provide the bases and requirements for NRC staff assessment of the potential for and consequences of degradation of the reactor coolant pressure boundary.

The applicable GDC include GDC 14 (Reactor Coolant Pressure Boundary), GDC 31 (Fracture Prevention of Reactor Coolant Pressure Boundary), and GDC 32 (Inspection of Reactor Coolant Pressure Boundary). GDC 14 specifies that the reactor coolant pressure boundary (RCPB) has an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture. GDC 31 specifies that the probability of rapidly propagating fracture of the RCPB be minimized. GDC 32 specifies that components which are part of the RCPB have the capability of being periodically inspected to assess their structural and leaktight integrity; inspection practices that do not permit reliable detection of degradation are not consistent with this GDC.

NRC regulations in 10 CFR 50.55a state that the American Society of Mechanical Engineers (ASME) Class 1 components (which includes the reactor coolant pressure boundary) must meet the requirements of Section XI of the ASME Boiler and Pressure Vessel Code. Various portions of the ASME Code address reactor coolant pressure boundary inspection. For example, Table IWA-2500-1 of Section XI of the ASME Code provides examination requirements for reactor pressure vessel head penetration nozzles and references IWB-3522 for acceptance standards. IWB-3522.1(c) and (d) specify that conditions requiring correction include the detection of leakage from insulated components and discoloration or accumulated residues on the surfaces of components, insulation, or floor areas which may reveal evidence of borated water leakage, with leakage defined as "the through-wall leakage that penetrates the pressure retaining membrane." Therefore, 10 CFR 50.55a, through its reference to the ASME Code, does not permit through-wall degradation of the reactor pressure vessel head penetration nozzles.

For through-wall leakage identified by visual examinations in accordance with the ASME Code, acceptance standards for the identified degradation are provided in IWB-3142. Specifically, supplemental examination (by surface or volumetric examination), corrective measures or repairs, analytical evaluation, and replacement provide methods for determining the acceptability of degraded components.

Criterion V (Instructions, Procedures, and Drawings) of Appendix B to 10 CFR Part 50 states that activities affecting quality shall be prescribed by documented instructions, procedures, or drawings, of a type appropriate to the circumstances and shall be accomplished in accordance with these instructions, procedures, or drawings. Criterion V further states that instructions, procedures, or drawings shall include appropriate quantitative or qualitative acceptance criteria for determining that important activities have been satisfactorily accomplished. Visual and volumetric examinations of the reactor coolant pressure boundary are activities that should be documented in accordance with these requirements.

Criterion IX (Control of Special Processes) of Appendix B to 10 CFR Part 50 states that special processes, including nondestructive testing, shall be controlled and accomplished by qualified personnel using qualified procedures in accordance with applicable codes, standards, specifications, criteria, and other special requirements. Within the context of providing assurance of the structural integrity of reactor coolant pressure boundary for the degradation observed at Davis-Besse, special requirements for visual examination and/or ultrasonic testing would generally require the use of qualified visual and ultrasonic testing methods. Such methods are ones that a plant-specific analysis has demonstrated would result in the reliable detection of degradation prior to a loss of specified reactor coolant pressure boundary margins of safety. The analysis would have to consider, for example, the as-built configuration of the system and the capability to reliably detect and accurately characterize flaws or degradation, and contributing factors such as the presence of insulation, preexisting deposits, and other factors that could interfere with the detection of degradation.

Criterion XVI (Corrective Action) of Appendix B to 10 CFR Part 50 states that measures shall be established to assure that conditions adverse to quality are promptly identified and corrected. For significant conditions adverse to quality, the measures taken shall include root cause determination and corrective action to preclude repetition of the adverse conditions. For degradation of the reactor coolant pressure boundary, the root cause determination is important for understanding the nature of the degradation present and the required actions to mitigate future degradation. These actions could include proactive inspections and repair of degraded portions of the reactor coolant pressure boundary.

Plant technical specifications pertain to this issue insofar as they do not allow operation with known reactor coolant system pressure boundary leakage.

Generic Letter 88-05, "Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants," pertains to this issue in that the staff concluded that in the absence of a program for addressing the corrosive effects of reactor coolant system leakage, compliance with General Design Criteria 14, 30, and 31 cannot be ensured.

<u>Required Information</u>

1. Within 15 days of the date of this bulletin, all PWR addressees are required to provide the following:

    A. a summary of the reactor pressure vessel head inspection and maintenance programs that have been implemented at your plant,

    B. an evaluation of the ability of your inspection and maintenance programs to identify degradation of the reactor pressure vessel head including, thinning, pitting, or other forms of degradation such as the degradation of the reactor pressure vessel head observed at Davis-Besse,

    C. a description of any conditions identified (chemical deposits, head degradation) through the inspection and maintenance programs described in 1.A that could have led to degradation and the corrective actions taken to address such conditions,

    D. your schedule, plans, and basis for future inspections of the reactor pressure vessel head and penetration nozzles. This should include the inspection method(s), scope, frequency, qualification requirements, and acceptance criteria, and

    E. your conclusion regarding whether there is reasonable assurance that regulatory requirements are currently being met (see the Applicable Regulatory Requirements, above). This discussion should also explain your basis for concluding that the inspections discussed in response to Item 1.D will provide reasonable assurance that these regulatory requirements will continue to be met. Include the following specific information in this discussion:

(1) If your evaluation does not support the conclusion that there is reasonable assurance that regulatory requirements are being met, discuss your plans for plant shutdown and inspection.

(2) If your evaluation supports the conclusion that there is reasonable assurance that regulatory requirements are being met, provide your basis for concluding that all regulatory requirements discussed in the Applicable Regulatory Requirements section will continue to be met until the inspections are performed.

2. Within 30 days after plant restart following the next inspection of the reactor pressure vessel head to identify any degradation, all PWR addressees are required to submit to the NRC the following information:

A. the inspection scope (if different than that provided in response to Item 1.D.) and results, including the location, size, and nature of any degradation detected,

B. the corrective actions taken and the root cause of the degradation.

3. Within 60 days of the date of this bulletin, all PWR addressees are required to submit to the NRC the following information related to the remainder of the reactor coolant pressure boundary:

A. the basis for concluding that your boric acid inspection program is providing reasonable assurance of compliance with the applicable regulatory requirements discussed in Generic Letter 88-05 and this bulletin. If a documented basis does not exist, provide your plans, if any, for a review of your programs.

The information required in Item 1.A, 1.B, and 1.C, should address:

- the material condition of the reactor pressure vessel head as determined through direct visual examinations dating back to the last time the entire reactor pressure vessel head was visually inspected to the bare metal. Include the date of the last 100 percent bare metal inspection, the results of that examination, and the extent and results of visual examinations conducted since the last 100 percent bare metal inspection. If no 100 percent bare metal inspection has ever been conducted, indicate so in your response.

- any leaks of boric acid or any other corrosive material onto the reactor pressure vessel head or insulation since the last 100 percent bare metal inspection (the results of which were provided in responding to 1.C). Include the extent to which boric acid deposits or other corrosive materials were removed from the reactor pressure vessel head, the length of time this material was left on the reactor pressure vessel head (and whether it is still on the reactor pressure vessel head), and the condition of the head following removal of the deposits. Also include a discussion of your program for preventing corrosion of the reactor pressure vessel head and the location of the leaks relative to any nozzle with through-wall cracks. If leakage was onto the insulation, discuss whether the leakage could have permeated the insulation or flowed through gaps in the insulation (e.g., around nozzles) such that deposits accumulated on the reactor pressure vessel head.

- the leakage integrity of the reactor pressure vessel head penetration nozzles. Include a summary of inspections performed (including scope and extent) to detect cracking and/or degradation of the vessel penetration weld or nozzle base metal, whether the inspection plan included any examination that could identify a potential cavity behind the reactor pressure vessel head nozzle, and if so, the potential for the inspection method used to accurately and reliably detect a cavity in the reactor pressure vessel head near the penetration nozzles (including the basis for this conclusion), particularly in cases where a leakage path has existed (i.e., even if the nozzle has been repaired). For repaired nozzles, the description should include the scope and results from the post-repair inspections.

Required Response

In accordance with 10 CFR 50.54(f), in order to determine whether any license should be modified, suspended, or revoked, each PWR addressee is required to respond as described below. This information is sought to verify licensee compliance with the current licensing basis for the facilities covered by this bulletin.

Within 7 days of the date of this bulletin, a PWR addressee is required to submit a written response if they are unable to provide the information or they can not meet the requested completion dates. The PWR addressee must address in their response any alternative course of action they propose to take, including the basis for the acceptability of the proposed alternative course of action.

The required written response should be addressed to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, 11555 Rockville Pike, Rockville, MD 20852, under oath or affirmation under the provisions of Section 182a of the Atomic Energy Act of 1954, as amended, and 10 CFR 50. 54(f). In addition, submit a copy of the response to the appropriate regional administrator.

Reasons for Information Request

Extensive degradation of the reactor coolant pressure boundary including leakage violates NRC regulations and plant technical specifications. Degradation of the reactor pressure vessel head or other portions of the reactor coolant pressure boundary can pose a significant safety risk if permitted to progress to the point that their integrity is in question and the risk of a loss of coolant accident increases. This information request is necessary to permit the assessment of plant-specific compliance with NRC regulations. This information will also be used by the NRC staff to determine the need for, and to guide the development of, additional regulatory actions to address degradation of the reactor pressure vessel head and/or other portions of the reactor coolant pressure boundary. Such regulatory actions could include regulatory requirements for augmented inspection programs under 10 CFR 50.55a(g)(6)(ii) or additional generic communication.

The NRC staff is interacting with the industry on the implications of the degradation observed at Davis-Besse. The NRC staff will continue to assess additional information it receives on this subject in determining the need for, and to guide the development of, additional regulatory actions to address degradation of the reactor pressure vessel head and/or other portions of the reactor coolant pressure boundary.

Related Generic Communications

- Information Notice 2002-11: "Recent Experience with Degradation of Reactor Pressure Vessel Head," March 12, 2002. [ADAMS Accession No. ML020700556]
- Bulletin 2001-01: "Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles," August 3, 2001. [ADAMS Accession No. ML012080284]
- Information Notice 2001-05, "Through-Wall Circumferential Cracking of Reactor Pressure Vessel Head Control Rod Drive Mechanism Penetration Nozzles at Oconee Nuclear Station, Unit 3," April 30, 2001. [ADAMS Accession No. ML011160588]
- Generic Letter 97-01, "Degradation of Control Rod Drive Mechanism Nozzle and Other Vessel Closure Head Penetrations," April 1, 1997.
- Information Notice 96-11, "Ingress of Demineralizer Resins Increases Potential for Stress Corrosion Cracking of Control Rod Drive Mechanism Penetrations," February 14, 1996.
- Information Notice 86-108, Supplement 3, "Degradation of Reactor Coolant System Pressure Boundary Resulting from Boric Acid Corrosion," January 5, 1995.
- NUREG/CR-6245, "Assessment of Pressurized Water Reactor Control Rod Drive Mechanism Nozzle Cracking," October 1994.
- Information Notice 94-63, "Boric Acid Corrosion of Charging Pump Casing Caused by Cladding Cracks," August 30, 1994.
- Information Notice 90-10, "Primary Water Stress Corrosion Cracking of INCONEL 600," February 23, 1990.
- Generic Letter 88-05, "Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants," March 17, 1988.
- Information Notice 86-108, Supplement 2, "Degradation of Reactor Coolant System Pressure Boundary Resulting from Boric Acid Corrosion," November 19, 1987.
- Information Notice 86-108, Supplement 1, "Degradation of Reactor Coolant System Pressure Boundary Resulting from Boric Acid Corrosion," April 20, 1987.
- Information Notice 86-108, "Degradation of Reactor Coolant System Pressure Boundary Resulting from Boric Acid Corrosion," December 29, 1986.

- Bulletin 82-02, "Degradation of Threaded Fasteners in the Reactor Coolant Pressure Boundary of PWR Plants," June 2, 1982.
- Information Notice 82-06, "Failure of Steam Generator Primary Side Manway Closure Studs," March 12, 1982.
- Information Notice 80-27, "Degradation of Reactor Coolant Pump Studs," June 11, 1980.

Backfit Discussion

Under the provisions of Section 182a of the Atomic Energy Act of 1954, as amended, and 10 CFR 50.54(f), this bulletin transmits an information request for the purpose of verifying compliance with existing applicable regulatory requirements (see the Applicable Regulatory Requirements section of this bulletin). Specifically, the required information will enable the NRC staff to determine whether current inspection and maintenance practices for the detection of degradation of the reactor coolant pressure boundary at reactor facilities (similar to that observed at Davis-Besse) provides reasonable assurance that reactor coolant pressure boundary integrity is being maintained. The required information will also enable the NRC staff to determine whether PWR addressee inspection and maintenance practices need to be augmented to ensure that the safety significance of this form of degradation remains low. No backfit is either intended or approved by the issuance of this bulletin, and the staff has not performed a backfit analysis.

Federal Register Notification

A notice of opportunity for public comment on this bulletin was not published in the *Federal Register* because the NRC staff is requesting information from power reactor licensees on an expedited basis for the purpose of assessing compliance with existing applicable regulatory requirements and the need for subsequent regulatory action. This bulletin was prompted by the discovery of degradation of the reactor pressure vessel head at Davis-Besse. Degradation of this extent has not been postulated or identified in PWRs. As the resolution of this matter progresses, the opportunity for public involvement will be provided.

Small Business Regulatory Enforcement Fairness Act

The NRC has determined that this action is not subject to the Small Business Regulatory enforcement Fairness Act of 1996.

Paperwork Reduction Act Statement

This bulletin contains an information collection that is subject to the Paperwork Reduction Act of 1995 (44 U.S.C. 3501 et seq.). This information collection was approved by the Office of Management and Budget, clearance number 3150-0012, which expires July 31, 2003. The burden to the public for this mandatory information collection is estimated to average 135 hours per response, including the time for reviewing instructions, searching existing data sources, gathering and maintaining the data needed, and completing and reviewing the information collection. Send comments regarding this burden estimate or any other aspect of this information collection, including suggestions for reducing the burden, to the Records Management Branch (T-6 E6), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by Internet electronic mail at INFOCOLLECTS@NRC.GOV; and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202, (3150-0012), Office of Management and Budget, Washington, DC 20503.

Public Protection Notification

If a means used to impose an information collection does not display a currently valid OMB control number, the NRC may not conduct or sponsor, and a person is not required to respond to, the information collection.

If you have any questions about this matter, please contact one of the persons listed below or the appropriate Office of Nuclear Reactor Regulation (NRR) project manager.

/RA/

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