May 22, 2002

LICENSEE: FirstEnergy Nuclear Operating Company

FACILITY: Davis-Besse Nuclear Power Station, Unit 1

SUBJECT: DAVIS-BESSE NUCLEAR POWER STATION, UNIT 1 - MEETING SUMMARY OF MARCH 20, 2002, TO DISCUSS THE OBSERVED DEGRADATION OF THE REACTOR PRESSURE VESSEL HEAD

On March 20, 2002, Nuclear Regulatory Commission (NRC) management and staff conducted a public meeting in Rockville, Maryland, to discuss our understanding of the events and current conditions concerning the observed degradation of the reactor pressure vessel (RPV) head at the Davis-Besse Nuclear Power Station with interested members of the public. A telephone conference bridge was established for members of the public who were unable to attend and approximately 75 individuals participated on the bridge. Following a presentation by the staff, questions were taken. The NRC participants are included as Enclosure 1. The meeting handouts are included as Enclosure 2. Questions received from the public along with our responses are included as Enclosure 3.

On February 16, 2002, the Davis-Besse Nuclear Power Station in Oak Harbor, Ohio, began a refueling outage that included inspecting the nozzles entering the head of the RPV, the specially designed container that houses the reactor core and the control rods that regulate the power output of the reactor. The licensee's inspections focused on the nozzles associated with the mechanism that drives the control rods, known as the control rod drive mechanism (CRDM). Both the inspections and their focus were consistent with the licensee's commitments in response to NRC Bulletin 2001-01, "Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles," which the agency issued on August 3, 2001.

When conducting its inspections, the licensee found that three CRDM nozzles had indications of axial cracking, which had resulted in leakage of the reactor's pressure boundary. Specifically, the licensee found these indications in CRDM nozzles 1, 2, and 3, which are located near the center of the RPV head. The licensee reported these findings to the NRC on February 27, 2002, and provided supplemental information on March 5 and March 9, 2002. The licensee also decided to repair the three leaking nozzles, as well as two other nozzles that had indications of leakage, but had not resulted in pressure boundary leakage.

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

To identify the cause of the displacement, the licensee investigated 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 ultrasonically measuring the thickness 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 and was 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 0.24-inch. This thickness was attributed to the thickness of the stainless steel cladding on the inside surface of the RPV head, which has a nominal design thickness of 0.1875 inch with a maximum design thickness of 0.375 inch and a minimum design thickness of 0.125 inch. Local measurements at the area of concern identified an average thickness of 0.297 inch.

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. Other factors contributing to the degradation might include the environment of the RPV 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).

/RA/

Douglas V. Pickett, Project Manager, Section 2
Project Directorate III
Division of Licensing Projects Management
Office of Nuclear Reactor Regulation

Docket No. 50-346

Enclosures: 1. NRC Participants
2. Meeting Handout
3. Questions and Answers

cc w/encls: See next page

To identify the cause of the displacement, the licensee investigated 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 ultrasonically measuring the thickness 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 and was 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 0.24-inch. This thickness was attributed to the thickness of the stainless steel cladding on the inside surface of the RPV head, which has a nominal design thickness of 0.1875 inch with a maximum design thickness of 0.375 inch and a minimum design thickness of 0.125 inch. Local measurements at the area of concern identified an average thickness of 0.297 inch.

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. Other factors contributing to the degradation might include the environment of the RPV 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).

/RA/
Douglas V. Pickett, Project Manager, Section 2
Project Directorate III
Division of Licensing Projects Management
Office of Nuclear Reactor Regulation

Docket No. 50-346

Enclosures: 1. NRC Participants
2. Meeting Handout
3. Questions and Answers

cc w/encls: See next page

ADAMS ACCESSION NO. ML021230181 **Meeting Handout: ML020990215**
Package: ML021230496

OFFICE	PM:PDIII-2		LA:PDIII-2		SC:EMCB		SC:PDIII-2	
NAME	DPickett		THarris		SCoffin		AMendiola	
DATE	05/13/02		05/10/02		05/21/02		05/22/02	

OFFICIAL RECORD COPY

Davis-Besse Nuclear Power Station, Unit 1

cc:
Mary E. O'Reilly
FirstEnergy Corporation
76 South Main St.
Akron, OH 44308

Manager - Regulatory Affairs
First Energy Nuclear Operating Company
Davis-Besse Nuclear Power Station
Oak Harbor, OH   43449-9760

Director
Ohio Department of Commerce
Division of Industrial Compliance
Bureau of Operations & Maintenance
6606 Tussing Road
P.O. Box 4009
Reynoldsburg, OH   43068-9009

Regional Administrator
U.S. Nuclear Regulatory Commission
801 Warrenville Road
Lisle, IL   60523-4351

Michael A. Schoppman
Framatome ANP
1911 N. Ft. Myer Drive
Rosslyn, VA   22209

Resident Inspector
U.S. Nuclear Regulatory Commission
5503 North State Route 2
Oak Harbor, OH 43449-9760

Plant Manager, Randel J. Fast
FirstEnergy Nuclear Operating Company
Davis-Besse Nuclear Power Station
5501 North State - Route 2
Oak Harbor, OH   43449-9760

Dennis Clum
Radiological Assistance Section Supervisor
Bureau of Radiation Protection
Ohio Department of Health
P.O. Box 118
Columbus, OH   43266-0118

Carol O'Claire, Chief, Radiological Branch
Ohio Emergency Management Agency
2855 West Dublin Granville Road
Columbus, OH 43235-2206

Zack A. Clayton
DERR
Ohio Environmental Protection Agency
P.O. Box 1049
Columbus, OH 43266-0149

State of Ohio
Public Utilities Commission
180 East Broad Street
Columbus, OH   43266-0573

Attorney General
Department of Attorney
30 East Broad Street
Columbus, OH   43216

President, Board of County
Commissioners of Ottawa County
Port Clinton, OH   43252

President, Board of County
Commissioners of Lucas County
One Government Center, Suite 800
Toledo, Ohio 43604-6506

DISTRIBUTION FOR MEETING SUMMARY OF MARCH 20, 2002, WITH EXTERNAL STAKEHOLDERS

Hard Copy w/encls
PD III-2 Reading
PUBLIC
GGrant, RIII

Hard Copy w/Encl 1
OGC
ACRS

E-mail w/Encl 1
SCollins
BSheron
JZwolinski
TMarsh
CCasto
WBateman
KWichman
AHiser
SBajwa
SCoffin
SBloom
BWetzel
JChung
ALee
SLong
AMendiola
SSands
DPickett
THarris
HNieh
SRosenberg

LIST OF MEETING PARTICIPANTS

MARCH 20, 2002

NRC
Brian Sheron
Jack Strosnider
Bill Bateman
Tony Mendiola
Ken Karwoski
Allen Hiser

ENCLOSURE 1

QUESTIONS FROM MARCH 20, 2002, PUBLIC MEETING

Question 1: **The damage to the reactor vessel was apparently discovered when repairs to one of the nozzles went awry. Had this difficulty not been encountered, is it credible that Davis-Besse would have restarted without detection of the damage?**

Response:

No. The repair process was terminated during machining of the nozzle when the nozzle began to rotate. If the nozzle had not rotated and the machining process had been completed successfully, the cavity would have been identified by other aspects of the repair process. For example, the entire thickness of the cracked length of the nozzle is machined out during the repair (along with a small part of the reactor pressure vessel head). This machining process would have exposed the degraded area, which would have been identified by inspections performed during the repair process. Another opportunity for identifying the degraded area would have occurred during the welding part of the repair. If for some reason the degradation was not identified during these steps, additional post-repair inspections would have provided another opportunity to detect the degradation. Lastly, cleaning and visual inspection of the surface of the reactor vessel head would have provided another opportunity for detecting the degradation.

Question 2: **What is the maximum pressure that the reactor coolant is expected to experience during design bases transients and accidents?**

Response:

The limiting transient and accident pressure in terms of peak reactor coolant system pressure are estimated to be:

Transient: 2590 pounds per square inch absolute (psia) for a Loss of Normal Feedwater (Reference updated safety analysis report (USAR) Table 15.2.8-2)

Accident: 2628 psia for Control Rod Withdrawal from Subcritical Condition (Startup Accident) (Reference USAR Table 15.2.1-2)

Question 3: **Would the 3/8" stainless steel liner have withstood the peak pressure from question (2)?**

Response:

As indicated in the public meeting on March 19, 2002, the nominal design thickness of the clad (stainless steel liner) is 3/16-inch (0.188-inch). The minimum designed clad thickness is 1/8-inch (0.125-inch) and the maximum designed clad thickness is 3/8-inch (0.375-inch). This is different than previously reported by the licensee and reported in Nuclear Regulatory Commission (NRC) Information Notice 2002-11, and NRC Bulletin 2002-01.

Actual measurements of the cladding thickness are presently being evaluated. Preliminary results indicate the average clad thickness in the degraded area was 0.297-inch.

The licensee has performed analyses to estimate the pressure that this region is capable of withstanding. Structural integrity analyses performed by independent contractors for the licensee demonstrate that the structural integrity of the as-found reactor pressure vessel (RPV) head, though degraded, would have functioned to maintain the facility within its design basis during anticipated operational occurrences and postulated accidents. The NRC staff is continuing its review of the licensee's calculations. In addition, the staff is performing its own, independent confirmatory calculations. Based on the NRC efforts to-date, it is expected that the cladding would have been able to withstand the peak pressures as identified in question 2 above.

Question 4: The schematic and images provided on the Vessel Head Penetration webpage suggest that the condition of the outer surface of the reactor vessel head is difficult to monitor. How can the outer surface of the reactor vessel head be checked for damage?

Response:

Some plants have the reactor pressure vessel head insulation sufficiently offset from the head itself which permits 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.

Even with the physical challenges and challenges from occupational radiation exposure, the condition of the vessel head can be assessed. First, if boric acid on the outer surface of the head is a contributing factor and the head cannot be inspected (because of physical limitations), reviewing the plant history for leakage of boric acid in the region above the head can give an indication of whether boric acid has leaked onto the head. If leakage has occurred and insulation cannot be removed readily for a direct inspection, an assessment of whether the boric acid could have leaked through the insulation (i.e., because it is not watertight or because it has tears in it) or through gaps in the insulation (such as around penetrations) can provide insights into whether boric acid could be on the head itself. If this assessment cannot rule out the presence of boric acid on the top of the head, additional actions may be necessary. These actions include, but are not necessarily limited to, removing a portion of the insulation to permit a direct visual inspection of the head or performing ultrasonic thickness measurements of the head from the bottom of the vessel head (i.e., at the cladding) to ensure there is no wastage of the head.

Secondly, if leaking penetrations are a contributing factor, licensees can inspect from the bottom of the vessel head to identify cracks/defects in the nozzle area to ensure that the area is free from through-wall cracking. This is discussed further in Bulletin 2001-01.

Question 5: Is the configuration shown for Davis-Besse typical for all pressurized-water reactors (PWRs) or just for Babcock & Wilcox Co. (B&W) PWRs?

Response:

The configuration illustrated for the insulation is typical of B&W PWRs. Plants from other NSSS vendors (Combustion Engineering and Westinghouse) have a variety of insulation configurations, with some of the configurations readily accessible for visual examination of the outer surface of the vessel head and others inaccessible without destructive removal of the insulation.

Question 6: NRC TI2515/145 listed Davis-Besse in Bin 2, or the most susceptible for problems except for those plants that had already identified problems. With benefit of hindsight, does the NRC staff believe its decision to allow Davis-Besse to defer the inspection by 12/31/2001 to be proper? If so, why?

Response:

Based on the information available to the staff at the time, along with our understanding of active degradation mechanisms, we believed our decision was proper. The NRC's decision to require Davis-Besse to shutdown and perform an inspection earlier than their planned end of March 2002 refueling outage had a technical basis from deterministic and probabilistic analyses which assessed the likelihood and potential consequences of through-wall circumferential cracking of control rod drive mechanism (CRDM) nozzles. The inspection results at Davis-Besse confirmed the staff's technical assessment related to through-wall circumferential cracking of CRDM nozzles in that Davis-Besse identified only one part through-wall circumferential crack that did not provide a safety challenge for the plant. However, this inspection also revealed a separate degradation mode of vessel head wastage that may be related to through-wall cracking of CRDM nozzles. Had we been aware of this degradation mechanism, it is likely we would have required the plant to shut down and inspect earlier.

Question 7: A recurring theme out of the discoveries at Oconee, Crystal River, and now Davis-Besse, is "We haven't seen this before." Given the frequency of such surprises, aren't we really just experimenting with commercial power reactors? Research (not intended to be the NRC Office of Research) appears to be, at best, one step behind the problems instead of one or more steps ahead.

Response:

NRC and industry-sponsored research on many forms of environmental degradation has been successful in producing the technical bases for management (inspection, mitigation, assessment and repairs) of a wide variety of issues (boiling-water reactor (BWR) pipe cracking, cracking in BWR internals, fatigue, erosion corrosion, etc.). However, we clearly are not capable of foreseeing all potential events, which is why we operate under the philosophy of defense-in-depth.

The recent experiences (2001/2002) with environmental degradation (cracking, wastage, etc.) at operating plants have occurred through mechanisms that have been experienced before. It is rather the character and extent of the degradation that has been evolving with age. As the underlying mechanisms for environmental attack are age-related, this is to be expected. In the case of the Oconee reactor vessel head penetration (VHP) nozzle cracking experience, the severity of the stress corrosion cracking attack was not anticipated and does involve technical elements that are the subject of research. The NRC Office of Nuclear Regulatory Research has maintained a significant research effort in environmentally assisted cracking since the early 1980s. This effort has been successful in producing both data and models of types of environmental degradation that have been used in regulatory evaluations and in confirming industry-proposed methodologies for inspection, mitigation and repairs. Due to the recent occurrences of VHP degradation noted above, in addition to the V.C. Summer pipe cracking event (2001), the Office of Research has increased resources and re-focused efforts in this technical area, with a particular emphasis on developing an enhanced understanding of the fundamental mechanisms of these forms of environmental degradation. Other essential elements of this effort include: (1) development/evaluation of probabilistic models addressing initiation and progression of degradation, stress state, and the overall assessment for structural integrity; and (2) evaluation of reliability and effectiveness of inspection methods. These efforts will enable the NRC to be more pro-active in anticipating future occurrences.

With regard to the degradation recently observed on the Davis-Besse reactor vessel head, it is premature to comment definitively until the root cause evaluation is complete. However, the developing root cause is currently focused on boric acid degradation. This form of attack is well-known and can cause aggressive dissolution of carbon steel under certain conditions. This form of attack was also the subject of NRC Generic Letter 88-05 and, in response, licensees instituted boric acid corrosion control programs to deal with the issue. Therefore, it is not clear at this point if there are "research" issues associated with the Davis-Besse degradation. However, if the root cause evaluation were to indicate that the corrosion attack on the Davis-Besse head was dependent on the adjacent leaking penetration and was able to proceed at operating temperatures, then research into this mechanism is indicated.

Question 8: 10 CFR 50.71(e) requires periodic updating of the USAR. Davis-Besse USAR Sections 5.2 describe the integrity of the reactor pressure vessel and the associated failure modes. The failure modes do not appear to describe the failure modes associated with CRDM nozzle cracking or flange leakage. NRC Information Notice 2002-11 describes flange leakage at Davis-Besse occurring in the 1998 and 2000 time frame. Why hasn't the NRC required Davis-Besse to update their USAR?

Response:

All licensees, including Davis-Besse, are required by 10 CFR 50.71(e) to maintain the USAR in accordance with the design basis of their facility. Reactor coolant system leakage during normal plant operation is expected and limited by plant technical specifications (TSs). At the Davis-Besse facility, TS 3/4.4.6.2, “Reactor Coolant System - Operational Leakage,” identifies those limits. Specifically, TS 3/4.4.6.2 prohibits Pressure Boundary Leakage and limits unidentified reactor coolant system leakage.

TS 1.16 of the Definitions defines Pressure Boundary Leakage as follows:

> Pressure Boundary Leakage shall be leakage (except steam generator tube leakage) through a non-isolable fault in a Reactor Coolant System component body, pipe wall or vessel wall.

The Bases section for TS 3/4.4.6.2 states:

> Pressure Boundary Leakage of any magnitude is unacceptable since it may be indicative of an impending gross failure of the pressure boundary. Therefore, the presence of any Pressure Boundary Leakage requires the unit to be promptly placed in Cold Shutdown.

The staff considers leakage through cracks in the CRDM nozzles to be Pressure Boundary Leakage and prohibited by plant TSs. Therefore, the Davis-Besse licensee is being required to repair cracks in the CRDM nozzles prior to plant restart.

Leakage through the CRDM flanges is considered to represent "Unidentified Leakage" and not Pressure Boundary Leakage. Davis-Besse TS 3/4.4.6.2 limits unidentified leakage to less than one gallon per minute (1 gpm). Unidentified reactor coolant system leakage is that leakage generally associated with threaded connections, valve packing, and flange connections. This type of leakage is typically small and not "indicative of an impending gross failure of the pressure boundary" as defined in the above quoted TS Bases. Therefore, the staff considers flange leakage to be included in the design basis and not prohibited by the plant TSs.

Question 9: Generic Safety Issue (GSI) 191 focused on blockage of the containment sump screens during the recirculation phase of emergency core cooling system (ECCS). Has Davis-Besse adequately addressed GSI 191 with respect to insulation surrounding the reactor vessel pressure head?

Response:

The potential for ECCS recirculation sump screen clogging has been confirmed to be a generic concern for pressurized-water reactors in a parametric research study performed for the NRC in support of GSI-191. However, the level of detail in this generalized parametric study was not sufficient to allow conclusive findings to be made regarding the susceptibilities of specific plants. Therefore, although the particular type of insulation (i.e., reflective metallic) used on Davis-Besse's reactor vessel head is generally shown in the parametric study to be relatively resistant to clogging sump screens, it is currently premature, due to a number of important plant-specific factors, to speculate as to whether a rupture in Davis-Besse's reactor vessel head would have resulted in sump screen clogging. As part of the NRC's action plan for resolving GSI-191, the NRC is evaluating the need for plant-specific assessments to be performed to conclusively identify the susceptibility to recirculation sump screen clogging for each PWR. The NRC's current GSI-191 action plan is publicly available in the Director's Quarterly Status Report, dated February 7, 2002 (Accession number ML020150515).

Question 10: It would appear that the recent events at Davis-Besse meet the threshold to enter Manual Chapter (MC) 0350. Why hasn't the NRC initiated a 0350 panel?

Response:

The purposes of the MC 0350 process include: establishing criteria for oversight of licensee performance for licensees that are in a shutdown condition as a result of significant performance problems or a significant event, establishing a record of major regulatory and licensee actions leading to NRC approval for restart, and assurance that following restart the plant is operated in a manner that provides adequate protection of public health and safety.

There are several considerations for deciding when to enter the MC 0350 Process, including: 1) whether there have been significant performance problems or a significant plant event, 2) the plant is in a shutdown condition and addressing performance problems, and 3) the NRC has a regulatory hold in effect, such as a Confirmatory Action Letter.

By letter dated April 29, 2002, the NRC informed FirstEnergy that an oversight panel was being formed in accordance with NRC Inspection Manual Chapter 0350, "Oversight of Operating Reactor Facilities in a Shutdown Condition with Performance Problems."

Question 11: The NRC is relying on information supplied by the industry's Nuclear Energy Institute (NEI). Is NEI and information supplied to the public by NRC held under the same standards of 10 CFR 50.9, "Completeness and accuracy of information," that licensees are held to?

Response:

NEI is an industry advocate and works with the industry and NRC on resolution of specific technical issues. While the NRC does obtain information supplied by the industry's NEI, the information supplied is not subject to 10 CFR 50.9 since the NEI is not a licensed entity and does not come under NRC purview. In the present situation, the information provided by NEI was used to provide an initial indication of conditions at each plant. The staff plant-specific reviews focused on the more thorough information provided by the licensees in response to Bulletin 2002-01.

Question 12: Does the accumulation of boric acid on the reactor pressure vessel head satisfy the Davis-Besse USAR?

Response:

Generic Letter (GL) 88-05, "Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants," was issued to assess the safe operation of PWRs when reactor coolant leaks below TS limits develop and the coolant containing dissolved boric acid comes into contact with and degrades low alloy carbon steel components. The principal concern is whether such operation continues to meet the requirements of General Design Criteria 14, 30, and 31 of Appendix A to Title 10 of the Code of Federal Regulations (CFR) Part 50, when the concentrated boric acid solution or boric acid crystals, formed by evaporation of water from the leaking reactor coolant, corrode the reactor coolant pressure boundary. The Davis-Besse licensee responded to GL 88-05 in letters dated May 27, 1988, and June 26, 1989. By letter dated February 8, 1990, the staff concluded that the Davis-Besse licensee had adequately implemented a program for managing small primary coolant leakage to prevent boric acid corrosion of carbon steel components in accordance with GL 88-05.

As indicated in Bulletin 2002-01, all PWR licensees, including Davis-Besse, have been requested to submit their basis for concluding that their boric acid inspection program is providing reasonable assurance of compliance with the applicable regulatory requirements discussed in GL 88-05. As described in question 8 above, the staff considers flange leakage to be included in the design basis. While not specifically addressed in the Davis-Besse USAR, flange leakage will result in the accumulation of some amount of boric acid crystals on the RPV head. However, the estimated 900 pounds of boric acid crystals found on the RPV head is clearly outside of the staff's expectations and the Davis-Besse USAR.

The NRC will conduct further inspections, following the efforts of the Augmented Inspection Team, to determine whether any NRC requirements were violated. The NRC Enforcement Policy will be applied to any findings developed during these further inspections, as appropriate.

Question 13: The American Society of Mechanical Engineers (ASME) Code does not appear to address modifications to the reactor pressure vessel head currently being contemplated by the Davis-Besse licensee. Describe the process that the staff intends to use in reviewing any proposed modifications to the reactor pressure vessel head. In addition, what will be the time constraints for this review?

Response:

As stated in the staff's Confirmatory Action Letter dated March 13, 2002 (CAL No. 3-02-001), the licensee must obtain NRC review and approval of any repair or modification and testing plans for the reactor pressure vessel head prior to implementation of those activities. We will review any proposed repair or modification and testing plan in conjunction with ASME Code requirements and in accordance with provisions of 10 CFR 50.55a(a)(3). The Code addresses specific areas including weld processes and limitations, materials, material stresses, and qualifications of welders. The staff will monitor and inspect any repairs or modifications that are made to the reactor pressure vessel head.

The staff does not have any time constraints associated with this review process. The staff will take the necessary time and will not be inhibited by any restart schedule.

Question 14: How does the NRC plan to address this problem with other reactor facilities? Were their inspections sufficient to identify this type of problem? Why are other plants currently safe to operate?

Response:

Bulletin 2002-01, "Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity," was issued on March 18, 2002, to all PWRs. Within 15 days of the date of the bulletin, licensees were requested to provide:

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

b. an evaluation of the ability of their 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. their 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. their 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 the 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.

The staff has received the 15-day responses to Bulletin 2002-01 from all 69 pressurized water reactors except for Davis-Besse, who indicated they will provide a response after completing their root cause evaluation. The staff has completed its review of the 15-day responses and has not identified any plants with conditions similar to those that lead to the degradation at Davis-Besse.

Additional information on plant specific bulletin responses can be found on the NRC web page at www.nrc.gov under the heading "Key Topics."

Question 15: What is the NRC doing with regard to addressing this issue on a world-wide basis?

Response:

The staff is actively sharing information with other countries. Information is being provided on the NRC external web site. In addition, we are aware that other countries are monitoring this situation.

Question 16: Specific subsets of PWRs may have similar problems to Davis-Besse. What plants are most susceptible?

Response:

We do not have a list of "most susceptible plants." As described in Bulletin 2002-01, we are continuing to evaluate the conditions surrounding the degradation at Davis-Besse. The staff has not identified the root cause of the degradation problems at Davis-Besse.

Boric acid or other contaminants could be contributing factors, as could steam jet cutting caused by leakage from the nozzle. Reactor facilities have variations in insulation surrounding the reactor pressure vessel head and inspection practices. Each licensee needs to examine their facility in order to determine their susceptibility to this form of degradation. The information requested in the bulletin is intended to provide this information.

Question 17: When will FirstEnergy provide their planned course of action and identify the root cause?

Response:

As described in Confirmatory Action Letter (CAL) No. 3-02-001, the licensee must obtain staff review and approval for repair or modification and testing plans for the reactor pressure vessel head, prior to implementation of those activities. The licensee is currently considering actions that would include either repairing the existing RPV head or replacing the existing RPV head.

By letter dated April 25, 2002, the licensee submitted their proposed repair plans for the RPV head. The staff is currently reviewing the licensee's proposal.

By letter dated April 18, 2002, the licensee submitted their Root Cause Analysis Report. The staff is currently reviewing the licensee's report.

Question 18: Are the areas of degradation at other facilities being found in similar locations (e.g., center of head, periphery of head)?

Response:

Areas of degradation are not limited to specific areas of the reactor pressure vessel head. Minor, localized degradation has been observed at both the center and periphery of the head.

As indicated in the response to question 14 above, no plants have been identified with degradation similar to that found at Davis-Besse.

Question 19: What issues will be considered in an ASME Code repair?

Response:

The ASME Code addresses areas such as metals to be used, welding procedures, qualifications of welders, stress and fatigue considerations, in-process repair examinations, post-inspection repair examinations, and post-modification testing.

Question 20: A safety analysis has been discussed assuming a nominal clad thickness of 0.3 inches. Shouldn't the safety analysis assume the worst as-found condition of 1/8 inch?

Response:

The worst as-found condition was 0.24 inches in a single location. The average clad thickness was 0.297 inches. The licensee has performed calculations for both the minimum clad thickness (i.e., 0.24 inch) and the average clad thickness (i.e., 0.297 inch). These analyses demonstrated that the structural integrity of the as-found reactor pressure vessel (RPV) head, though degraded, would have functioned to maintain the reactor vessel within its design basis during anticipated operational occurrences and postulated accidents. As indicated in the response to question 3 above, the staff is continuing its review of the licensee's analyses. In addition, the staff is performing its own, independent confirmatory analyses.

Question 21: We are hearing problem descriptions using the words corrosion, erosion, wastage, and degradation. Are there differences in these descriptions or are they being used synonymously?

Response:

In the context of public meetings, these terms are used synonymously. Wastage is a specific corrosion mechanism. Erosion is a process by which the material is worn away gradually as a result of a fluid flowing across its surface. Degradation is a more general term that could refer to a corrosion related mechanism or some other type of process that results in "degrading" the material.

Question 22: Page 3 of Bulletin 2002-01 describes a deflection of the stainless steel clad of the RPV. Please describe what this means.

Response:

As described in Bulletin 2002-01, the stainless steel cladding near CRDM nozzle 3, was found to be deflected upwards by about 1/8-inch for a 4-inch distance, indicating that the cladding had yielded. This is significant because the cladding had essentially become the reactor coolant pressure boundary near the affected nozzle after the base metal of the RPV head had degraded. The stainless steel cladding has a nominal design thickness of 0.1875 inch with a maximum design thickness of 0.375 inch and a minimum design thickness of 0.125 inch. Local measurements at the area of concern identifies an average thickness of 0.297 inch.

Stainless steel has physical characteristics known as load limits. When exposed to moderate loading, stainless steel will return to its original shape. This translates into loading within the elastic load limit. When stainless steel is exposed to significant loads and does not return to its original shape, the steel has exceeded its elastic limit and entered plastic behavior. The fact that the stainless steel clad was deformed indicated that it experienced significant loads and high stress.

Question 23: Page 4 of Bulletin 2002-01 discusses how the insulation at Davis-Besse was offset to permit visual inspections. Since it appears to be easier to perform visual inspections at Davis-Besse, does this cause concern for inspections at other facilities? Can leakage make its way through insulation to contact the RPV head at other facilities?

Response:

As described in Bulletin 2002-01, plant-specific design characteristics limit the ability of some licensees to perform visual inspections of the RPV head. While performing visual inspections was somewhat easy at Davis-Besse, other facilities will incur additional costs and dose to plant personnel in performing visual examinations. All licensees will be expected to examine their facilities to determine whether leakage can come into contact with the RPV head.

Question 24: Will plants that are currently shut down be required to respond to the Bulletin prior to restarting?

Response:

All facilities were subject to the 15-day reporting requirement of the bulletin. As discussed in the response to question 14 above, all licensees (with the exception of Davis-Besse) have responded to the Bulletin and no facilities were found to have degradation similar to that at

Davis-Besse. We did not place any holds on facilities that were shut down at the time. They were alerted to the concerns of the bulletin.

Question 25: Regarding those facilities that are currently shut down, does the staff intend to keep those facilities down pending review of the bulletin response?

Response:

As discussed above, all facilities have responded to the Bulletin and none were found to have degradation similar to that found at Davis-Besse. No facilities were kept shut down as a result of staff review of the Bulletin.

Question 26: Greenpeace has been following this issue for 10 years. Why has it taken the NRC so long to react to this? Where is the paper trail? Considering NRC's decision to permit Indian Point Unit 2 to defer their steam generator tube inspections and the resultant tube rupture accident, is the public being adequately protected by the NRC?

Response:

The NRC has been aggressive in dealing with the issue of primary water stress corrosion cracking of vessel head penetrations nozzles, dating to the initial findings of degradation in France in the early 1990s. The NRC issued GL 97-01, "Degradation of Control Rod Drive Mechanism Nozzle and Other Vessel Closure Head Penetrations," on April 1, 1997, in order to request addressees to describe their program for ensuring the timely inspection of PWR CRDM and other vessel head penetration nozzles. This action was the culmination of industry data gathered from both domestic and foreign reactors dating back to 1986. With the findings of cracking in U.S. PWRs, the NRC has taken appropriate steps through the issuance of Information Notices and ultimately Bulletin 2001-01, "Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles," on August 3, 2001.

Recognizing the significance of the findings at Davis-Besse, the NRC has taken unprecedented steps to inform and involve both the industry and the public. The public meeting held with industry representatives on March 19, 2002, and this public meeting (on March 20, 2002) with external NRC stakeholders represent a new initiative for the agency. In addition to these meetings, the NRC has issued Information Notice 2002-11, "Recent Experience with Degradation of Reactor Pressure Vessel Head," and Bulletin 2002-01, "Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity." These events are significant and it is important to learn from them. However, there have been no significant adverse consequences and the public health and safety is being maintained.

Question 27: In the 1975 time frame, cracking was identified in 4-inch bypass lines of BWRs. The NRC was sufficiently concerned that it ordered the shutdown of all BWRs to perform inspections. Why isn't the NRC taking similar actions today?

Response:

It is difficult to make comparisons of the decision-making processes between 1975 and today. Clearly, both licensees and the NRC currently have sufficiently more operating experience, better inspection techniques, increased monitoring capabilities, and vastly improved communication capabilities than in 1975. We believe that the actions taken to date are sufficient to ensure the public health and safety and that a large-scale, industry shutdown is not justified.

Question 28: Can the plant be repaired and return to operation by July 1, 2002?

Response:

The staff is unable to provide any date for plant restart. The licensee's letter of April 25, 2002, provided their proposed repair plan for the RPV head. It should also be noted that the licensee is also considering replacing the RPV head in lieu of repairs. Once the licensee determines their course of action, the NRC staff will evaluate their proposal and determine if it meets NRC regulations and the applicable provisions of the ASME Boiler and Pressure Vessel code before repair or replacement can be affected.

Question 29: Following repairs or modifications, what tests must be performed and will they be performed prior to plant restart?

Response:

Some form of testing will be required dependent upon the repairs or modifications that are ultimately made. The exact form of testing cannot be ascertained at this time. It is anticipated that post-modification testing will be required prior to plant restart. In addition, core physics testing may be required during plant startup.

Question 30: The bulletin describes how the stainless steel liner of the RPV was deflected upwards. Regarding the ability of the plant to restart with the current liner, will the licensee's and NRC's analysis be made public?

Response:

The Oak Ridge National Laboratory is currently performing analyses for the NRC regarding deflection of the stainless steel clad. This analyses will be made publicly available. The licensee's analyses will be discussed at a public meeting prior to plant restart.

Question 31: Who identified that boric acid was accumulating on the RPV head? Leakage of boric acid appears to be persistent from a number of sources and over a number of years. What is the NRC doing about this more persistent problem?

Response:

The licensee identified that boric acid was accumulating on the RPV head.

The NRC addressed the overall issue of boric acid control in GL 88-05. The purpose of that GL was to verify that all PWRs implemented programs for monitoring small primary coolant leakage to prevent boric acid corrosion of carbon steel components. Bulletin 2002-01 requires licensees to provide information regarding inspections of the RPV head. In addition, within 60 days of the date of the bulletin (March 18, 2002), licensees are required to provide information related to the remainder of the reactor coolant pressure boundary that includes their basis for concluding that the boric acid inspection program is providing reasonable assurance of compliance with the applicable regulatory requirements discussed in GL 88-05 and Bulletin 2002-01. If a documented basis does not exist, licensees are required to provide their plans, if any, for a review of their programs.

Question 32: Considering that the CRDM sleeve moved during the repair process, was anything other than friction holding the sleeve in place during plant operation?

Response:

The nozzle did not rotate until after the J-groove weld had been machined out. Since there was no adjacent reactor vessel head material to provide support to the nozzle, it rotated. The nozzle was held in place during plant operation by the J-groove weld material.

Question 33: What is a LOCA and what does the staff mean when describing the consequences of a reactor pressure vessel head rupture as a “medium loss-of-coolant (LOCA)?”

Response:

LOCA is the acronym for "loss-of-coolant accident." Specifically, the term means that the pressure boundary around the reactor coolant system (RCS) has failed in a manner that allows the reactor coolant to escape the system faster than the normal coolant make-up system can replenish the loss. The emergency core cooling system (ECCS) is designed to cope with LOCAs. It is designed to be able keep the reactor core covered with water by replenishing reactor coolant lost through holes in the pressure boundary that are up to and including the size created by rupture of the largest pipe in the RCS. The ECCS has a variety of high-pressure and low pressure pumps plus tanks of water that use pressurized gas to inject water to the RCS. The ECCS is designed to be highly reliable, with two sets of equipment and back-up electrical power. Different size holes in the RCS require different sets of the ECCS equipment to ensure that the reactor core is adequately cooled. Small pipe breaks, less than about 2-inches in diameter, are too small to cool and depressurize the RCS. These are called "small LOCAs." For small LOCAs, the steam generators are still needed to cool the RCS and one of the high pressure pumps with a modest flow rate is needed to replenish the coolant loss. For the Davis-Besse plant, breaks between about 2-inches and 9-inches in diameter are large enough to cool and eventually to depressurize the RCS, but the ECCS must replenish the reactor coolant while the depressurization slowly proceeds. These are called "medium LOCAs." For medium LOCAs, both a high pressure pump and a low pressure pump plus the gas-pressurized water tanks are required to replenish coolant. Breaks greater than 9-inches in diameter will depressurize the RCS so fast that high pressure pumps are unnecessary. These are called "large LOCAs." A low pressure pump and the gas-pressurized tanks are needed to replenish the reactor coolant fast enough for large LOCAs. A design requirement for the ECCS is that it be capable of successfully protecting the reactor core during the double-ended break of the largest pipe in the RCS. At Davis-Besse, the largest pipe is 36-inches in diameter. The void in the carbon steel portion of the Davis-Besse RPV head leaves an unsupported section of the stainless steel clad that is about 20 square inches in area. That is, the same area as a pipe about 5 inches in diameter. So, if the clad ruptured under the void found in the carbon steel at Davis-Besse, it would create a medium LOCA.

Question 34: Are the cracks in the J-groove welds surrounding the CRDM nozzles attributed to radiation and embrittlement? What actions are we taking?

Response:

The cracks in the J-groove welds are not attributed to radiation-induced embrittlement. The cracks are induced by primary water stress corrosion cracking. Radiation-induced embrittlement occurs in the beltline region of the reactor vessel where a high neutron flux exists. Embrittlement does not occur on the RPV head.

Embrittlement is limited by regulations. Capsules are periodically removed from the RPV and tested at laboratories to verify that limits of embrittlement are not exceeded.

Question 35: Could you please walk me through your thought process as to how you will determine if the licensee should replace the reactor head cap or just weld the corroded area?

Response:

The licensee will determine whether the RPV head is either repaired or replaced. The licensee's letter of April 25, 2002, provided their proposed repair plan for the RPV head. It should also be noted that the licensee is also considering replacing the RPV head in lieu of repairs. Once the licensee determines their course of action, the NRC staff will evaluate their proposal and determine if it meets NRC regulations and the applicable provisions of the ASME Boiler and Pressure Vessel code before repair or replacement can be affected.

Question 36: Can you please share with me your understanding of the potential cost and timing involved in a replacement scenario and how that weighs into your decision-making process.

Response:

The licensee is evaluating all options available to them at this time, however, the process is ongoing and preliminary, and no one course of action has been decided to date. The NRC's primary function is to ensure the safe operation of the facility and that public health and safety are maintained. Replacement scenario costs do not weigh in our decision making process.

Question 37: What are the pros and cons from a safety standpoint relative to a replacement solution versus a repair solution for the reactor head cap?

Response:

It is the responsibility of the licensee to choose either to repair the existing head, or obtain a replacement head. In either case, the licensee will submit the appropriate information to the NRC for review and approval after determining if it meets NRC regulations and the applicable provisions of the ASME Boiler and Pressure Vessel code.

Question 38: Could the degradation of the reactor head found in this year's shutdown been caused by boric acid deposits from previous years?

Response:

The licensee's Root Cause Analysis Report was submitted by letter dated April 18, 2002. The licensee has determined that the cause of the degradation was boric acid corrosion resulting from leakage through a crack in a RPV penetration nozzle attributable to primary water stress corrosion cracking. The licensee further determined that the degradation at Davis-Besse had likely occurred over a period of years, but was not recognized until its discovery in March 2002. The staff is continuing its review of the licensee's Root Cause Analysis Report.

Question 39: If the stainless steel inner lining of the head is not corroded by boric acid, then why is the entire head not made of a thicker piece of stainless steel instead of the carbon steel?

Response:

There are many trade-offs in selecting materials for particular applications including inspectability, strength, corrosion susceptibility, and cost. Carbon steel offers certain advantages over stainless steel including strength (i.e., a carbon steel vessel head will be thinner than a corresponding stainless steel), better inspectability, and cost savings. Although carbon steel is susceptible to boric acid corrosion, stainless steel is also susceptible to other forms of corrosion some of which present challenges to conventional inspection techniques.

Question 40: Could Davis-Besse be facing any NRC fines for any operations violations based on what the investigations may or may not uncover?

Response:

The NRC's Enforcement Policy (NUREG-1600, dated May 1, 2000) does provide for civil penalties, under certain limited conditions, as outlined in Section VI.C. The NRC's initial actions in response to this event included dispatching an Augmented Inspection Team (AIT) to the site on March 12, 2002. The AIT's charter emphasis was to collect, analyze, and document factual information and evidence. The AIT did not examine the regulatory process to determine whether NRC requirements were violated or assess the licensee's performance associated with this event. The AIT report was issued on May 3, 2002, and is publicly available at the NRC web site.

The NRC will conduct further inspections, following the AIT's efforts, to determine whether any NRC requirements were violated. The NRC Enforcement Policy will be applied to any findings developed during these further inspections, as appropriate.

Question 41: Is there any indication that past CRDM nozzle cracking was a result of operator error?

Response:

There is no indication that the past CRDM cracking was a result of operator error. The fact of the cracking is fundamentally due to the age of the plant, coupled with the details of the materials properties, the design, and the methods of assembly and joining (the machining of the head and the CRDM housing, the nature of the interference fit, and the welding process) of the CRDM nozzle in the reactor head.

Question 42: Is there any indication that the degradation on the Davis-Besse RPV is a result of operator error?

Response:

There is no indication that the degradation on the Davis-Besse RPV is a result of operator error. It was noted in several of the most recent inspections that the top of the head was covered with a thick layer of boric acid that was trapped under the insulation support structure, preventing direct observation of the penetrations near the top, or center, of the head.

Question 43: If the licensee has to change the RPV head, are there spare heads available in the US for purchase?

Response:

RPV heads exist with both formerly operating facilities that have since been decommissioned and with facilities that were never completed. However, the licensee must make the final determination whether any of the existing RPV heads are appropriate for replacing the RPV head at Davis-Besse.

Question 44: Could Davis-Besse go back on line without making repairs to the indentation in the stainless steel lining of the reactor vessel?

Response:

The stainless steel lining is not part of the reactor coolant pressure boundary. The cladding serves as a protective liner for the carbon steel to limit the potential for corrosion. As a result, the vessel head could be repaired without making repairs to the stainless steel lining; however, this is unlikely since any repair will most likely require removal of the affected portion of the cladding in order to gain access to the degraded area.

Question 45: What type of insulation configuration do the Beaver Valley reactor vessel heads have? Are these heads easy to inspect?

Response:

The Beaver Valley reactor vessels have reflective stepped insulation. This type of insulation rests on top of the RPV head, and as such, requires lifting/removal of the insulation in order to gain visual access. The Beaver Valley reactor vessel heads can be inspected.

Question 46: The possibility of Davis-Besse reorganizing the placement of the control rods in the reactor, and performing additional tests and physics analyses that would accompany such a change was frequently mentioned as a possible repair option. To the layman, such as myself, this seems like a very large undertaking. Has such a redesign of an operating reactor occurred, and if so, how long did it take to complete? I understand that the NRC does not wish to speculate on possible repair plans, but could you venture a guess as to the time frame required for such a change to the physics of a reactor. Is it several months, several years, etc?

Response:

The reorganization of the placement of the control rods in the reactor and performing the necessary analyses to support operation with the revised control rod design is not that unusual. The staff has recently completed a review of a license amendment to the Waterford plant to change the control rod length and placement as part of a standard license amendment for an upcoming reload. The supporting physics and transient calculations are part of the normal analyses performed to support operation following any reload of a reactor. As such, the staff would not characterize the analyses of and control rod modification as a significant undertaking. The staff review process for the Waterford amendment took several weeks.

The extent of the analyses and the time to complete all the necessary reviews will not be known until the licensee actually submits their plans in writing.

Question 47: The caller from Greenpeace asked a question that I do not believe was addressed specifically in the course of the response to his barrage of questions, so I'll re-work it a bit. In my own research into the Davis-Besse situation, I've spoken with various people in the plant design and construction field who say that the problem with DB is actually a sign of the age of the US nuclear fleet. A similar problem (circumferential cracking) was discovered overseas about a decade ago (the Greenpeace point) and was dealt with by Sweden, France and Japan through the replacement of the entire reactor vessel or the entire reactor vessel head with designs using new materials less prone to such cracking. My question involves the decision to focus on fixing these cracks in US plants when they arise instead of replacing as was done in other countries. Did the NRC investigate the merits of repair vs. replacement, and if so, is there a report or other such document I can consult for explanation.

Response:

In fact, other countries, at first, "fixed" cracks in their VHPs just as the US licensees are planning to do. As stated in NRC's GL 97-01, "European and Japanese utilities have . . .

repaired the nozzles or replaced the heads, as appropriate." In fact, the French program to replace the heads, began in the mid-1990's, is only about half-completed at this point, and is projected to continue into 2007. All the affected regulatory authorities, including the USNRC, have strengthened the inspection requirements for vessel heads and CRDM nozzles in particular. For example, and as described in the Sept. 7, 2001, slides from the meeting of Duke Power with the NRC (viewable at NRC's Alloy 600 website URL - under "Public Meetings"), the CRDM cracks discovered at the three Oconee plants have been repaired and re-inspected, with NRC approval. The plants are in operation, pending receipt of replacement vessel heads, which will commence about a year from now, and continue, one vessel at a time, for about one year.

Question 48: Does this corrosion problem apply to Three Mile Island and Peach Bottom?

Response:

Three Mile Island (TMI), Unit 1, is a PWR and is subject to the corrosion problems experienced at Davis-Besse. The TMI licensee has received Bulletin 2002-01 and is expected to provide the information requested in the bulletin. Peach Bottom Units 2 & 3 are BWRs which do not include boron in their RCS. Therefore, the Peach Bottom facilities are not subject to the corrosion problem identified at Davis-Besse.