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June 28, 2002

The Honorable Edward J. Markey
United States House of Representatives
Washington, D.C. 20515

Dear Congressman Markey:

I am responding on behalf of the Nuclear Regulatory Commission (NRC) to your letter of May 1, 2002, regarding the recent discovery of the cavity in the reactor pressure vessel (RPV) head at the Davis-Besse nuclear power station. You expressed concerns about the adequacy of licensee inspections at Davis-Besse and our oversight of the inspections. In your letter, you asked the Commission to provide responses to 22 questions to assist in carrying out your oversight and legislative responsibilities.

The NRC has taken regulatory actions following the initial discovery of events at the Davis-Besse reactor facility. Immediately after the discovery of the cavity, we dispatched an Augmented Inspection Team to gather information at the site to facilitate the understanding of the mechanisms that led to the RPV head degradation. We also formed a special oversight panel to coordinate the agency's efforts to assess the performance problems associated with the licensee's inadequate inspection of the RPV head. This panel, which will follow the framework of the NRC's Inspection Manual Chapter 0350 (IMC 0350), will evaluate the licensee's past and present performance and determine those activities that must be completed before a decision can be made on whether the reactor can be safely restarted. The panel will conduct public meetings to facilitate a clear understanding of the issues and the NRC's actions in response to the RPV head degradation.

We have also formed a lessons-learned task force to evaluate the procedures related to inspection activities, generic communications, and policies and practices within the Commission's regulatory structure. This self-assessment will allow NRC management to recommend areas of improvement applicable to the NRC and to the industry. The NRC conducted three public meetings on Wednesday, June 12, at 10a.m., 3 p.m., and 7 p.m. in Oak Harbor, Ohio to discuss the activities of the oversight panel and the task force with members of the public.

We have completed a preliminary evaluation of the technical aspects of the licensee's April 18, 2002, root cause analysis report on the RPV head degradation. While our evaluation is preliminary in nature, the information in the report appears credible and supports the general causes and timeline of the RPV head degradation. We are continuing to investigate the root cause and to review other information provided by the nuclear power industry so that we can formulate effective corrective actions to prevent recurrence.

Enclosed are responses to your questions. NRC documents and reference material are cited as necessary to support the responses. I appreciate your interest in this important matter. If you have any additional questions, please contact me.

Sincerely,

/RA/

Richard A. Meserve

Enclosure: Responses to Congressional Questions

Identical letter sent to:

The Honorable Marcy Kaptur
United States House of Representatives
Washington, D.C. 20515

The Honorable Edward J. Markey
United States House of Representatives
Washington, D.C. 20515

Responses to Congressional Questions

Question 1: Does the safety analysis for nuclear power plants consider cracks or holes specifically located in the reactor vessel head (in addition to any analysis of pressurized thermal shock)? In other words, have such analyses specifically examined the safety consequences of a hole in a reactor vessel head? If not, why not?

Response 1:

Cracks or holes specifically located in the reactor vessel head are considered in safety analyses in nuclear plants as the consequences of a break in a single control rod drive mechanism (CRDM) housing. The CRDM housing fully penetrates the reactor pressure vessel head. The break of a single CRDM housing which would produce a hole in the upper reactor vessel head is evaluated in each plant's final safety analysis report (FSAR). Analyses of this event include the additional conservative assumption that a control rod assembly is ejected from the core with an accompanying increase in power at that portion of the core. Pressurized water reactors (PWRs) operate at power with very few control assemblies inserted within the core, so these analyses are conservative. The offsite radiological consequences are evaluated and are required to be within the limits of the Commission's regulations. Only the early portions of control rod assembly ejection accidents are evaluated in the plant's FSAR, since the long-term recovery phase of this event, which involves the operation of the emergency core cooling systems, is bounded by the analyses of the larger breaks in the coolant piping that are also described in the plant's FSAR and are part of the plant design basis.

The staff believes that the consequences from a crack or hole in the reactor vessel head of a nuclear power plant would be bounded by the consequences of a break of equivalent size in the coolant piping. The consequences from breaks in the coolant piping are routinely analyzed in the plant's FSAR. For discussions of some recent evaluations of reactor vessel head breaks, please see the response to Question 3.

Question 2: If the stainless steel clad had failed, what assurance does the NRC have that the safety systems at Davis-Besse or any other plants would have mitigated the event? Please fully describe and document any back-up safety systems that would have prevented, slowed or minimized the uncontrolled meltdown that might have begun.

Response 2:

If the stainless steel clad in the upper head of the Davis-Besse reactor had failed, a hole approximately 8 inches in diameter with an equivalent cross-sectional area of about 100 square inches (in^2) would have been created in the reactor vessel head. Based on calculations performed by the staff and described in the response to Question 3, the consequences would be similar to a break of similar dimensions located in the coolant piping of the plant. The high pressure and low pressure safety injection systems are designed to mitigate the effects of breaks up to and including a double-ended rupture of the largest pipe in the reactor coolant system. The largest pipe in the Davis-Besse reactor coolant system is the hot leg, which measures 36 inches in diameter. In the case of a double-ended rupture, coolant discharges through both ends of the pipe. Therefore, the Davis-Besse safety injection systems can mitigate the consequences of a break with a cross-sectional area equal to approximately 2000 in^2, which is 20 times the size of the degraded area of the head. Even in the case of a double-ended rupture of the hot leg, the high and low pressure injection systems would keep the temperature of the fuel rod cladding and the cladding oxidation below the limits established in 10 CFR 50.46. The requirements for maintaining a coolable geometry and long-term cooling would also be met. The response to Question 3 cites recent evaluations of reactor vessel head breaks.

Question 3: What thermal hydraulic analysis has the NRC performed to determine whether or not a hole or crack in the reactor vessel head would have resulted in a core meltdown? Provide any analysis or supporting documentation.

Response 3:

The NRC staff performed a series of analyses for postulated breaks in the reactor vessel head as documented in an enclosure to this letter (Reference A). This work was begun following the discovery of a number of cracks and leaks in the CRDM nozzles of operating pressurized water reactors (PWRs). The study showed that the system would respond to a break in the reactor vessel head caused by the failure of up to 3 CRDMs in a similar manner as to a small break loss-of-coolant accident (LOCA), which is analyzed in the plant's Final Safety Analysis Report (FSAR). No new phenomena or unexpected results were identified.

Following the identification of the corrosion in the reactor vessel head at Davis-Besse, the staff performed additional analyses in which the break size was postulated to be approximately the size of the degraded area on the Davis-Besse reactor vessel head. To generate a conservative answer and to bound the consequences of a rupture of the degraded area, a complete failure of the control rods to enter the core was analyzed concurrent with the postulated break. This study is documented in an enclosure to this letter (Reference B). In these studies, the consequences of postulated breaks in the reactor vessel head were found to be similar to those of piping breaks analyzed in the plant's FSAR. No new phenomena or unexpected results were identified.

Plants designed by all three PWR reactor vendors were analyzed with similar assumptions. The consequences of a failure to insert any control rods were found to be minimal because the negative reactivity produced by core voiding (i.e., vaporization of the water in the core region) during the depressurization of the system following the break, and the later boric acid addition by the emergency core cooling system, were sufficient to shutdown the reactor.

More recently, the staff evaluated the consequences of a small leak or crack in the reactor vessel head. The results also showed that the reactor would shut down by core voiding and boric acid addition without core uncovery or overheating. A leak of this size is about the minimum size that would cause the reactor inventory to decrease and initiate a loss-of-coolant accident, since smaller leaks can be made up by a plant's normal charging system.

Question 4: If the clad had failed, what would the consequential damage to the control rod drive system have been?

Response 4:

Control rod drive mechanisms (CRDMs) are electromechanical devices which rely on electric power to engage the control rod and raise or lower it in response to signals from the reactor control panel. In response to an emergency, the electric power to the CRDMs is quickly interrupted causing the CRDMs to detach from the control rods, which drop into the reactor core. This is commonly called a "scram." The electric power to the CRDMs is turned off by opening electrical breakers (switches), which are located outside the reactor containment. The scram breakers open when they receive a signal from various safety sensors, which detect a variety of abnormal operating conditions, including loss-of-coolant accidents (LOCAs).

Damage to the CRDM electrical cables above the reactor vessel could cause the electrical power to be interrupted directly, in which case the control rods would drop into the core. Alternatively, an ejected housing could cause a short in the electrical power system, which would be detected and then open the scram breakers, which would cut off power to the control rods, dropping them into the core and shutting down the reactor. In both of these cases, pressure sensors in the reactor coolant system are designed to detect the occurrence of the LOCA and they would send their own separate signal to the scram breakers to open, allowing the control rods to drop into the core.

As discussed in the response to Questions 1-3, the staff has assessed the consequences of various size LOCAs in the top of the vessel. The analyses were performed assuming all control rods do not insert into the core to bound the consequences of an ejected rod mechanically damaging the other rods and preventing their insertion. The results show that the Davis-Besse plant and that the emergency core cooling system (ECCS) would operate within the design basis to mitigate the accident successfully.

A more detailed description of the operation of the CRDMs is enclosed with this letter (Reference C).

Question 5: Is it possible for the control rod drive systems to withstand the thermal and hydraulic forces generated by the breaking of the clad and successfully scram the reactor? Provide any supporting analysis to justify the answer.

Response 5:

The NRC did not analyze the mechanical damage that could be sustained by the control rods in the event of a rupture of the clad on the Davis-Besse head. Instead, a worse case scenario of all control rods failing to insert into the core was analyzed, and as discussed in response to Questions 1-3, the results show that the Davis-Besse plant would operate within the design basis to mitigate the accident successfully.

Question 6: If the control rod drive system were damaged as the result of a reactor vessel rupture in the reactor pressure vessel (RPV) head, would a reactor scram occur before or after the damage to the control rods? Provide any supporting analysis to justify the answer.

Response 6:

As discussed in response to Questions 1-3, the staff has assessed the consequences of a LOCA in the top of the vessel assuming that all control rods do not insert into the core. The analyzed scenarios bound the scenarios postulated in this question. The results show that even if all the control rods failed to insert into the core, the Davis-Besse plant would operate within the design basis to mitigate the accident successfully.

Question 7: If the damage occurred before the scram occurred, what would have been the consequences assuming the remaining safety systems worked?

a. Is it possible that the control rod insertion mechanism would have been disabled at or around the same time that the emergency core cooling systems were reflooding the reactor with water to replace that lost as a result of the rupture?
b. What would have been the consequences of such a chain of events?
c. Would the containment have failed?
d. Would there have been offsite releases?
e. What would have been the dose rates within the vicinity of Davis-Besse?
f. Would regulatory limits have been exceeded? If so, by how much?

Response 7:

The NRC staff believes it unlikely that a break in one area of the reactor vessel head would have prevented all of the control rod assemblies from entering the core. Nonetheless, as discussed in response to Questions 1-3, the staff has assessed the consequences of a LOCA in the top of the vessel assuming that all control rods do not insert into the core. The analyzed scenarios bound the scenarios postulated in this question. The results show that even if all the control rods failed to insert into the core, the Davis-Besse plant would operate within the design basis to mitigate the accident successfully.

Question 7a:

Is it possible that the control rod insertion mechanism would have been disabled at or around the same time that the emergency core cooling systems were reflooding the reactor with water to replace that lost as a result of the rupture?

Response 7a:

The NRC staff considers it unlikely that the control rod insertion mechanism would be disabled at or around the same time that the emergency cooling systems are operating. This is because the electrical signal which causes the control rods to fall by gravity into the core is activated at a higher pressure than the signal which causes the emergency core cooling systems to actuate. By the time reactor system pressure had fallen to the setpoint for activating the emergency core cooling systems, the control rods would already be in the core. In addition, and as indicated in the response to Question 4, damage to the CRDM electrical cables above the reactor vessel would cause the electrical power to be interrupted directly causing the control rods to drop into the core.

Question 7b:

What would have been the consequences of such a chain of events?

Response 7b:

As discussed in response to Questions 1-3, the staff has assessed the consequences of a LOCA in the top of the vessel assuming that all control rods do not insert into the core. The analyzed scenarios bound the scenarios postulated in this question. The results show that even if all the control rods failed to insert into the core, the Davis-Besse plant would operate within the design basis to mitigate the accident successfully.

Question 7c:

Would the containment have failed?

Response 7c:

Since the event is within the scope of existing analyses for bounding accidents for which the plant safety systems are designed, the containment response would also be within the scope of existing containment analyses. Containment systems would function to cool the containment atmosphere. Passive cooling from condensation of steam on the walls and other internal containment structures would contribute to maintaining the containment atmosphere pressure and temperature within acceptable values.

Therefore, the staff concludes that the containment would not fail.

Question 7d:

Would there have been offsite releases?

Response 7d:

If the clad lining the vessel had failed, the primary coolant in the reactor primary coolant system would have been discharged into the containment through the damaged penetration on the reactor vessel head. A small fraction of the volatile fission products normally present in the primary coolant would have been released into the containment atmosphere. Since the containment integrity would not be challenged by the event, the containment leak rate would be within the low levels allowed by NRC regulations, and the offsite releases would be well below the regulatory limits (10 CFR Part 100). The public health and safety would be ensured.

Question 7e:

What would have been the dose rates within the vicinity of Davis-Besse?

Response 7e:

The staff has estimated that the dose rates at the Davis-Besse site (exclusion area) boundary using a conservative meteorological model (i.e., poor fission product dispersion by low wind speed) would have been less than 0.01 rem to the whole body (less than 0.04 percent of the regulatory limit) and 0.1 rem to the thyroid (less than 0.04 percent of the regulatory limit) during the first 2 hours of the primary coolant discharge.

In evaluating the dose, the staff conservatively assumed that the reactor was operating at its peak fission product concentration experienced during current reactor fuel Cycle 13, all primary coolant in the reactor primary coolant system had discharged into the containment atmosphere through the damaged penetration on the reactor vessel head, and all the remaining emergency core coolant systems (ECCS) functioned as designed. The staff further assumed that radioactive iodine in the primary coolant would have been increased (spiked) due to the pressure and temperature transient while discharging the primary coolant into the containment atmosphere and that the containment would have been leaking at its maximum allowable leak rate of 0.5 percent of containment air volume per day.

Question 7f:

Would regulatory limits have been exceeded? If so, by how much?

Response 7f:

No. The dose rates at the Davis-Besse site boundary would have been well below the dose guidelines specified in 10 CFR Part 100. The regulation establishes the maximum allowable dose of 25 rem whole body and 300 rem to the thyroid from iodine exposure to an individual located at any point on the exclusion area boundary for two hours immediately following onset of fission product release from the core resulting from a major accident.

Question 8: Page 1 of the Probable Cause Summary Report of the Initial Investigative Team for Root Cause (hereinafter referred to as “March 22, 2002, Report”) states that “Deferral of the modification to the service structure for improved access when the modification was first considered resulted in the continued limited ability to prevent significant boric acid accumulations and allow for better visual determination of leakage sources.”

a. Why was modification of the service structure deferred?
b. Who made this decision?
c. Did the NRC staff approve such deferral, and if so, on what basis?
d. How many other licensees have deferred and/or never undertaken similar modifications to assure access to their service structures?

Question 8a:

Why was the modification of the service structure deferred?

Response 8a:

The decision to defer this modification was reviewed by the NRC Augmented Inspection Team (AIT). The inspection was completed on April 5, 2002, and is documented in Inspection Report (IR) 50-346/02-03. Section 5.5.1 of this report states, “This modification was canceled in 1992, because the current inspection techniques were considered adequate.” Then for budget reasons it was deferred to the next outage (Reference D, Section 5.5.1).

Question 8b:

Who made this decision?

Response 8b:

The initial decision to cancel the modification in 1992 was made by licensee management. The decision to fund the modification as proposed again in 1994 was made by the licensee’s Project Review Group in November of 1998. After funding approval, the licensee scheduled installation of the service structure modification for the 2002 refueling outage. Prior to the 2002 outage, based on budgetary concerns and the expected replacement of the head in 2004, the modification was deferred to the 2004 outage.

Question 8c:

Did the NRC staff approve such deferral, and if so, on what basis?

Response 8c:

The NRC is not part of the licensee’s approval process for plant modifications such as that proposed for the service structure. The licensee is authorized to make certain changes to their facility without prior NRC approval as discussed in 10 CFR Part 50.59 “Changes, Tests, and Experiments.” As permitted under 10 CFR 50.59, this particular modification likely would not require NRC approval.

Question 8d:

How many other licensees have deferred and/or never undertaken similar modifications to assure access to their service structures?

Response 8d:

The service structure is a feature of plants designed by Babcock & Wilcox (B&W). In response to the Davis-Besse situation, the NRC has contacted the B&W licensees and verified that all (except Davis-Besse) have made the necessary modifications to ensure access to their service structure.

Question 9: The March 22, 2002, Report states that "Boric acid that accumulated on the top of the RPV head over a period of years inhibited the station's ability to confirm visually that neither nozzle leakage nor vessel corrosion was occurring." Evidence available now shows that leakage from the nozzles began 2 to 4 operating cycles ago.

a. Why wouldn't the presence of boric acid on the top of the RPV head been an indication to the licensee or to NRC inspection personnel that there was a problem? Is it normal for there to be boric acid accumulations on the top of the RPV for years?
b. Is it the NRC's policy that if boric acid or anything else obscures the top of the RPV head, that the licensee is free to ignore it for years and thereby fail to confirm visually that neither nozzle leakage nor vessel corrosion was occurring?
c. Why was leakage from the nozzles not immediately detected at the time it was occurring?
d. Why was this leakage not successfully detected in routine inspections, as the NRC assured Congress it would be in 1996?
e. If the normal presence of insulation in the RPV has the effect of preventing inspections from successfully detecting cracks that could result in leaks, then what was the basis for the NRC's 1996 assurances to me that such cracks could be detected long before leaks occurred?
f. It now appears that both the Davis-Besse and Oconee nuclear reactors operated for many months (perhaps years) with through-wall cracks in the CRDM nozzles. Based on this experience, how can the NRC be sure that its reliance on "leak-before-break" and inspections is justified? Doesn't this experience strongly suggest that either the technical specification limits on unidentified leakage need to be tightened or that the vessel head penetrations need to be instrumented to allow leakage to be immediately detected?
g. The updated final safety analysis report (UFSAR) for Davis-Besse doesn't appear to allow for the presence of boric acid in the RPV head. Four modes of failure are described, including 1) ductile yielding; 2) brittle fracture; 3) fatigue; and 4) NDTT (Nil Ductility Temperature Transition, also known as "reactor embrittlement"). Nowhere is boric acid corrosion mentioned. In light of this, shouldn't the presence of boric acid <u>alone</u> in the RPV have immediately halted operation of the reactor and triggered a full investigation by the licensee and the NRC?
h. Wasn't Davis-Besse operating outside its design basis? If so, if a rupture had occurred, isn't it true that there would have been no basis for knowing whether the event could have been controlled?

Question 9a:

Why wouldn't the presence of boric acid on the top of the RPV head been an indication to the licensee or to NRC inspection personnel that there was a problem? Is it normal for there to be boric acid accumulations on the top of the RPV for years?

Response 9a:

It is not unusual for some PWR licensees to operate with what has been considered insignificant boron deposits (i.e., light dusting or very thin layer). The primary reason for operating with some amount of boron deposits is that in cleaning the RPV head, workers are exposed to significant radiation levels. However, in response to NRC Generic Letter 88-05, “Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants,” of March 17, 1988, pressurized water reactor (PWR) 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. Licensees can prevent significant boron accumulation on the RPV head if their systematic programs are properly implemented and monitored.

Prior to the discovery of the Davis-Besse degradation, boric acid corrosion on the reactor vessel head was not considered a risk-significant issue that warranted high-levels of NRC oversight. It was previously believed that boric acid corrosion on the reactor head was of low risk-significance because dry boric acid deposits are not corrosive. They are formed as the coolant leaking from the primary system is converted to steam because of the elevated temperatures of the reactor vessel head. Since the deposits are dry, they are not corrosive. A wetted environment, which apparently existed at Davis-Besse, is necessary for significant corrosion to occur. The discovery of the degradation at Davis-Besse has prompted the NRC and the industry to reconsider the boric acid corrosion management strategies. The NRC has established an independent lessons-learned task force to evaluate the procedures related to inspection activities, generic communications, and policies and practices within the Commission’s regulatory structure. This self-assessment will allow NRC management to ascertain the important lessons for the Commission and recommend areas of improvement applicable to the NRC and to the industry

In prior outages, before the discovery of reactor pressure vessel (RPV) head degradation at Davis-Besse, the licensee has indicated that it 1) misinterpreted information that may have signaled the onset of the degradation, 2) only partially cleaned the RPV head and left significant boron deposits, and 3) deferred a modification that would have made the RPV head more accessible. These examples as well as others described in the NRC Augmented Inspection Team report of May 3, 2002, and the licensee’s root cause report suggest that the licensee was not sensitive to the importance of removing significant boron deposits from the head. This may be indicative of broader licensee performance issues which will be examined by the IMC 0350 panel that was formed on April 29, 2002.

Question 9b:

Is it the NRC’s policy that if boric acid or anything else obscures the top of the RPV head, that the licensee is free to ignore it for years and thereby fail to confirm visually that neither nozzle leakage nor vessel corrosion was occurring?

Response 9b:

The NRC’s policy on boric acid management is based on Generic Letter 88-05. While there are no specific regulatory requirements that the bare head of the RPV be viewable, licensees are required to implement programs which monitor and take measures to prevent degradation of the reactor coolant pressure boundary by boric acid corrosion. Failure to comply with these requirements would be addressed through the NRC's Enforcement Policy and Reactor Oversight Program.

Question 9c:

Why was leakage from the nozzles not immediately detected at the time it was occurring?

Response 9c:

Leakage was not identified immediately at the time that it was occurring because the leakage rates from CRDM nozzle cracking were below the sensitivity of the on-line leakage detection systems at Davis-Besse and at other plants, which is roughly gallons per day. Also, the operators would not be in a position to observe the leakage during plant operation. However, there were other opportunities for the licensee to detect leakage and the resulting corrosion from the CRDM nozzles.

Davis-Besse personnel did perform visual inspections during the last several refueling outages. However, these inspections were not effective.

In addition, a photograph of the reactor vessel flange provided in the report of April 18, 2002, clearly indicates that some form of significant degradation was occurring. The licensee's response to this information was not effective.

Indirect indications of reactor vessel head degradation were also evident in fouling of the containment air coolers and the radiation element monitor filters. As described in the report of April 18, 2002, the licensee did not attribute these findings to the possibility of reactor vessel degradation. The NRC IMC 0350 Panel will be reviewing the licensee's performance in this area.

Question 9d:

Why was this leakage not successfully detected in routine inspections, as the NRC assured Congress it would be in 1996?

Response 9d:

The licensee chose to leave significant boron deposits on the RPV head. The staff believes that had the licensee properly implemented its boric acid corrosion program, it would have identified the leakage during routine inspections and prevented the significant corrosion.

As indicated by a photograph of the reactor vessel flange taken during the April 2000 refueling outage, which is provided in the report of April 18, 2002, the routine inspections conducted at Davis-Besse did provide an opportunity for the licensee to identify on-going reactor vessel head degradation. If this information had been appropriately handled by the licensee, it is likely that this condition would have been attributed to CRDM nozzle leakage. The report does not describe why the licensee failed to identify this indication of degradation. The NRC IMC 0350 Panel will be reviewing the licensee's performance in this area.

Question 9e:

If the normal presence of insulation on the RPV has the effect of preventing inspections from successfully detecting cracks that could result in leaks, then what was the basis for the NRC's 1996 assurances to me that such cracks could be detected long before leaks occurred?

Response 9e:

As referenced in your letter, the NRC made the following statement at the Subcommittee on Energy and Power's September 5, 1996 NRC oversight hearing:

> "the NRC staff had determined that VHP [Vessel Head Penetration] cracking does not pose an immediate safety concern because the cracks would result in detectable leakage before failure, and the leakage would be detected during visual examinations performed as part of surveillance walkdown inspections."

As described in Generic Letter 97-01, there was no expectation that cracks could be detected long before leaks occurred, but rather that cracking of nozzles did not pose an immediate safety concern. The bases for this assertion are (1) cracking was expected to predominantly occur in the axial orientation of the nozzle (not the circumferential orientation that could result in a nozzle ejection that would represent a more serious safety concern), (2) these axial cracks would result in detectable leakage before catastrophic failure, and (3) this leakage would be detectable during visual examinations performed as a part of surveillance walkdown inspections.   Analyses submitted by the industry owners groups and reviewed by the NRC staff provided the technical support for these bases. In part, these analyses indicated that a short axial crack (0.5 inches long) in combination with a tight annulus clearance of 0.0001 inch, would provide more than 50 pounds per year of boron deposit on the reactor vessel head. This amount of deposit, and far greater amounts for longer cracks, was found by the NRC staff to be detectable prior to catastrophic failure. This finding would hold whether the reactor vessel head to CRDM nozzle interface was directly observable, or even if the presence of insulation precluded direct observation of this interface.

From recent industry experiences with CRDM nozzle cracking, we now know that these analyses may be optimistic, in that deposits observed in the field may be much smaller than the analyses would indicate. This finding is possibly attributable to field observations of some cracking in the J-groove welds as opposed to cracking occurring solely in the CRDM nozzle base metal. As described in Bulletin 2001-01, these observations have led to a greater emphasis on enhanced visual observations of the reactor vessel head (as opposed to standard walkdown inspections) to detect small deposits of boron, and greater use of non-destructive evaluation to directly identify cracking in the CRDM nozzles.

The visual observations of the Davis-Besse head of increasing deposit amounts and deposits changing in character from loose, white and powdery to rock-hard, lava-like with significant colorations of red and orange, should have been identified by the licensee as having significance for follow-up actions. The licensee for Davis-Besse did not take appropriate actions in response to this available information.

Question 9f:

It now appears that both the Davis-Besse and Oconee nuclear reactors operated for many months (perhaps years) with through-wall cracks in the CRDM nozzles. Based on this experience, how can the NRC be sure that its reliance on “leak-before-break” and inspections is justified? Doesn’t this experience strongly suggest that either the technical specification limits on unidentified leakage need to be tightened or that the vessel head penetrations need to be instrumented to allow leakage to be immediately detected?

Response 9f:

The Davis-Besse head degradation has caused the staff to re-evaluate the need for more effective inspection techniques and has highlighted the need to reliably detect cracking of the CRDM nozzles before degradation of the RPV head can occur. The lessons learned task force is addressing this issue.

Question 9g:

The Updated Final Safety Analysis Report (UFSAR) for Davis-Besse doesn’t appear to allow for the presence of boric acid in the RPV head. Four modes of failure are described, including 1) ductile yielding; 2) brittle fracture; 3) fatigue; and 4) NDTT (Nil Ductility Temperature Transition, also known as “reactor embrittlement”). Nowhere is boric acid corrosion mentioned. In light of this, shouldn’t the presence of boric acid alone on the RPV have immediately halted operation of the reactor and triggered a full investigation by the licensee and the NRC?

Response 9g:

As discussed in response to part a. of this question, PWR 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 in response to NRC Generic Letter 88-05. The significant amount of boric acid deposits on the Davis-Besse reactor vessel head should have prompted the licensee to clean the head and to determine and repair the source of the leakage prior to restart during past inspections. The type of degradation discovered at Davis-Besse was unforseen, and had the NRC been aware of it, we would have shut the plant down immediately.

Question 9h:

Wasn't Davis-Besse operating outside its design basis? If so, if a rupture had occurred, isn't it true that there would have been no basis for knowing whether the event could have been controlled?

Response 9h:

Although the design basis for the reactor pressure vessel head at Davis-Besse was not maintained because of the degradation of the head, the design basis requirements for the ECCS continued to be satisfied because the ECCS systems would have retained their capability to provide the necessary core cooling to mitigate the loss of coolant accident that might have occurred at the Davis-Besse plant successfully. The NRC regulatory philosophy for maintaining safety incorporates the concept of defense-in-depth for the design basis of safety systems at nuclear power plants. Under this philosophy, systems that perform a safety function within the plant are designed to continue to function during scenarios where component failures occur, and they provide back-up capability when other systems fail. The emergency operating procedures that guide the operators in the use of plant systems to mitigate a LOCA would have continued to be applicable and effective in controlling the course of the event. As indicated in the response to Questions 1-3, the Davis-Besse plant would have operated within the design basis to mitigate the accident successfully.

Question 10: The March 22, 2002, Report states that “Historically, there have been problems with CRDM flange leakage both at Davis-Besse and the industry. This appears to have obscured the recognition that boric acid accumulation on the RPV might also be due to nozzle leakage.”

a. What is the nature and safety significance of the flange leakage problem?
b. Where did the leakage come from?
c. Please provide a list of all other reactors that have been affected.
d. What measures have been undertaken to address these problems at these reactors?
e. If, at Davis-Besse, CRDM flange leakage obscured recognition that boric acid accumulation on the RPV might also be due to nozzle leakage, couldn’t this have occurred elsewhere? What has been done to determine whether or not this has occurred?

Question 10a:

What is the nature and safety significance of the flange leakage problem?

Response 10a:

Flange leakage is not a significant safety concern. Leakage from the CRDM flanges that resulted in deposits on the reactor vessel head complicated the identification of the leakage that was coming from the CRDM nozzles. The CRDM nozzle leakage was safety significant in that it played a role in the root cause of the head degradation.

Question 10b:

Where did the leakage come from?

Response 10b:

CRDM flanges constitute a mechanical connection between two piping components as opposed to a welded connection. Borated water can leak through this mechanical joint connection, and possibly traverse down the nozzle to the reactor vessel head.

Question 10c:

Please provide a list of all other reactors that have been affected.

Response 10c:

The reactors that could be affected by this type of leakage (if present) include all operating Pressurized Water Reactors in the country, a total of 69 reactors of the Westinghouse, Babcock and Wilcox, and Combustion Engineering designs. A list of these reactors is enclosed with this response.

Question 10d:

What measures have been undertaken to address these problems at these reactors?

Response 10d:

Improved gaskets have been used to remedy the problems with leakage. In addition, the NRC's expectation is that leakage will be monitored by licensee's boric acid control inspection programs. The staff is also considering whether further generic regulatory response is appropriate.

Question 10e:

If, at Davis-Besse, CRDM flange leakage obscured recognition that boric acid accumulation on the RPV might also be due to nozzle leakage, couldn't this have occurred elsewhere? What has been done to determine whether or not this has occurred?

Response 10e:

On August 3, 2001, the NRC issued NRC Bulletin 2001-01 "Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles" to all holders of operating licenses for PWRs. The purpose of the bulletin was to request information related to the structural integrity of the reactor pressure vessel head penetration (VHP) nozzles at PWR facilities. Specifically, the NRC requested information on the extent of VHP nozzle leakage and cracking found to date, inspections and repairs undertaken to satisfy applicable regulatory requirements, and the basis for concluding that the plans for future inspections will ensure compliance with applicable regulatory requirements.

Bulletin 2002-01, "Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity," was issued on March 18, 2002, as a result of the Davis-Besse head degradation issue. The staff issued Bulletin 2002-01 to PWR licensees requesting that addressees provide information related to the structural 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. Bulletin 2002-01 also required licensees to provide the basis for concluding that plans for future inspections of the reactor coolant pressure boundary will satisfy applicable regulatory requirements at their respective PWR plants. The staff completed its first detailed review of licensee responses, and has not identified any plants with conditions similar to those that lead to the degradation at Davis-Besse. The staff has discussed questions with many licensees which have helped to clarify their responses, and the results of these discussions are documented on the NRC's public website.

Question 11: The March 22, 2002, Report states that "The potential for significant corrosion of the RPV head as a result of accumulating boric acid and local leakage was not recognized as a safety significant issue by the staff and management of the plant."

a. Isn't the RPV lined with stainless steel to protect it from significant corrosion?
b. If so, how could the potential for significant corrosion not be recognized as a safety significant issue?

Question 11a:

Isn't the RPV lined with stainless steel to protect it from significant corrosion?

Response 11a:

Yes, the inside of the reactor pressure vessel head is lined or "clad" with stainless steel. The purpose of the stainless steel lining or "cladding" is to limit the potential for corrosion due to boric acid in the primary coolant system water. The outside of the reactor pressure vessel head is not clad because the environment is typically hot and dry, and does not contain the moisture required to support boric acid corrosion.

Question 11b:

If so, how could the potential for significant corrosion not be recognized as a safety significant issue?

Response 11b:

NRC issued Generic Letter 88-05 identifying this concern to industry and licenses were required to implement programs to monitor for boric acid corrosion and prevent degradation of the reactor coolant pressure boundary. Failure to comply with these requirements would be addressed through the NRC's Enforcement Policy and Reactor Oversight Program. The NRC is investigating the cause of the licensee's failure to respond to this safety significant issue and their failure to prevent the degradation of the reactor pressure vessel head.

Question 12: The key events timeline set forth in the March 22, 2002, report notes that sometime between 1994-1996 “CRDM nozzle #3 crack propagates through wall of nozzle;” that in 1998 and 2000 the licensee “did not identify nozzle leakage on head, nor was boric acid accumulation successfully removed from nozzle #3;” and that in 1999 “noteworthy corrosion at nozzle #3 of the RPV head inititated, as evidenced by iron oxide in the containment atmosphere.” How and why were these apparent warning flags ignored by the licensee?

Response 12:

The boric acid corrosion control program at the site included both cleaning and inspection requirements, but was not effectively implemented to detect the leakage and prevent the significant corrosion of the reactor vessel head over a period of years. Similarly, on several occasions, maintenance and corrective action activities failed to detect and address the indications in the containment that the significant corrosion of the reactor vessel head was occurring. The NRC views these as missed opportunities to identify and correct the significant degradation to the reactor pressure vessel head.

The AIT report has an attached List of Documents Reviewed, which lists issues identified by the licensee and entered into its corrective action program for evaluation, disposition and resolution. Some of these documents addressed the filter plugging issues associated with the containment radiation monitors. As part of its corrective actions, the licensee did not make the association that the iron oxide plugging the radiation monitor filters originated from the reactor vessel head. An NRC investigation and inspections were initiated to determine whether FirstEnergy adequately followed their boric acid corrosion control program and corrective action program. Also, the NRC will be performing further inspections to ensure effective implementation of the licensee's programs.

Question 13: The March 22, 2002, report states “It should be noted that there is strong circumstantial evidence that the iron oxide that Davis-Besse began to collect in radiation monitor filters in 1999 was indicative of the RCS leak and corrosion in nozzle #3. As Operation Experience, this information would be potentially beneficial to other plants.” Has the NRC asked other plants to check for accumulation of iron oxide in their radiation monitor filters? If not, why not? If so, what have they found? Since iron oxide is rust, why didn’t the operators assume that they had a corrosion problem in 1999, take steps to identify its source, and then fix the problem?

Response 13:

The NRC issued Information Notice (IN) 2002-13, “Possible Indicators of Ongoing Reactor Pressure Vessel Head Degradation” dated April 4, 2002, to alert licensees to possible indicators of reactor coolant pressure boundary degradation including degradation of the reactor pressure vessel head material. Specifically, the IN discusses containment air cooler and radiation element filter clogging due to boron buildup. The NRC requested that recipients of the IN review the information for applicability to their facilities and consider taking appropriate actions. The NRC staff has conducted conference calls with licensees regarding inspections and the measures taken to ensure that they do not have reactor pressure vessel head degradation. Licensees who have addressed the IN 2002-13 issues during conference calls have stated that no boron buildup or iron oxide is present in containment or in containment filters, and that there has been no increase in the frequency of filter changes which would be indicative of reactor vessel head degradation.

In its root cause analysis report, the licensee attributes its inaction in 1999 to a misinterpretation of information. As discussed, the NRC’s IMC 0350 panel will examine licensee performance issues at Davis-Besse.

Question 14: According to the NRC reports, the air filters on the containment radiation monitors were replaced far more frequently than normal due to plugging from iron oxide (i.e., rust) and boric acid in the air. This buildup of material on the filters (i.e., plugging) was likely due to the corrosion that was occurring in the reactor vessel head.

a. What corrective action did the Davis-Besse licensee take, if any, in response to the abnormal condition?
b. Did anyone bring the problem to the attention of management?
c. Were any problem reports written? If so, provide copies.
d. What were the responses, if any, to these reports?
e. Is the absence of problem or corrective action reports a violation of 10 CFR Part 50 Appendix B, the NRC's quality assurance requirements?
f. Please provide any documentation related to any corrective action that Davis-Besse took in response to the plugging of the air filters in the containment radiation monitors. Did the resident inspector or regional personnel know of the problem with the plugging of the air filters?
g. If not, why not? Provide any NRC documentation related to the NRC knowledge of the plugging of the air filters.

Question 14a:

What corrective action did the Davis-Besse licensee take, if any, in response to the abnormal condition?

Response 14a:

The licensee generated condition reports that addressed the filter plugging issues associated with the containment radiation monitors. The corrective actions taken by the licensee are documented in these condition reports. Based on the licensee's Root Cause and Probable Cause Reports, the conditions were identified, but the collective significance was not recognized. As part of these corrective actions, the licensee states that licensee staff did not make the association that the iron oxide plugging the radiation monitor filters originated from the reactor vessel head. The licensee described that their focus for these evaluations was ensuring operability of the radiation elements to meet technical specification requirements. Copies of several condition reports and supporting documents are provided as an attachment. NRC investigations and inspections continue into the cause of the licensee's failure to identify the corrosion earlier.

Question 14b:

Did anyone bring the problem to the attention of management?

Response 14b:

Yes. The filter plugging issues were documented on condition reports. The licensee has a programmatic guideline which outlines its corrective action program and establishes the methods and requirements for identifying and documenting conditions, including those adverse to quality, their causes, and the actions necessary to correct and/or prevent recurrence. Issues identified by the corrective action program are brought to site management attention through the generation of condition reports.

The iron oxide found on the radiation element filters was noted by the licensee on several occasions, and entered into the corrective action program. Samples were sent to outside contractors for analysis, and independent investigators were brought in for resolution. Additionally, this was designated as a Plant Issue for review by senior management personnel on monthly intervals. Presentations were given to management on January 20, February 16 and March 17, 2000. These presentations described the issue, the actions to date and pending actions, as well as a proposed timeline.

Question 14c:

Were any problem reports written? If so, provide copies.

Response 14c:

Yes. Condition reports were written on the filter plugging issues. Several condition reports and supporting documents are provided as an attachment.

Question 14d:

What were the responses, if any, to these reports?

Response 14d:

The licensee's responses are documented in the condition reports and supporting documents provided as an attachment.

Question 14e:

Is the absence of problem or corrective action reports a violation of 10 CFR Part 50 Appendix B, the NRC's quality assurance requirements?

Response 14e:

The general requirements in 10 CFR Part 50 Appendix B Criterion XVI require that deficiencies and defective equipment are promptly identified and corrected. Licensees have corrective action programs that stipulate the reporting of the condition and resolution in condition reports or similar documents. Depending on the safety significance of the issue, not having a written condition report for an issue may be a violation.

At Davis-Besse, the licensee generated condition reports which addressed the filter plugging issues associated with the containment radiation monitors. As part of its corrective actions, the licensee did not make the association that the iron oxide plugging the radiation monitor filters originated from the reactor vessel head.

The NRC is conducting an investigation and additional inspections to review and characterize potential regulatory issues that were identified as a result of the recent Augmented Inspection Team inspection conducted at the Davis-Besse Station. Any violations identified as a result of this inspection and investigation will be dispositioned in accordance with the General Statement of Policy and Procedures for NRC Enforcement Actions, NUREG-1600 and the NRC's Reactor Oversight Process.

Question 14f:

Please provide any documentation related to any corrective action that Davis-Besse took in response to the plugging of the air filters in the containment radiation monitors. Did the resident inspector or regional personnel know of the problem with the plugging of the air filters?

Response 14f:

The corrective actions that Davis-Besse took are documented in the condition reports and supporting documents referenced in the previous section (a) through (e). These actions were principally focused on the operability of the radiation monitors.

Yes. The NRC Resident Inspectors documented problems with the reactor coolant system leak detection system air filters in Inspection Reports 50-346/99009 (DRP) and 50-346/99010 (DRP). Additionally, the inspectors reviewed a temporary modification associated with bypassing the charcoal filters on Radiation Elements RE 4597AB/BA as documented in Inspection Report 50-346/2001-013.

Response 14g:

If not, why not? Provide any NRC documentation related to the NRC knowledge of the plugging of the air filters.

Response 14g:

The three Inspection Reports named above are provided as an attachment.

Question 15: If a manufacturer of reactor vessels proposed to construct a vessel with a stainless steel plug of the same size using the same process (welding, heat treating, etc.) that Davis-Besse is likely to use, would the vessel be qualified for nuclear service (i.e., qualify for an N stamp)? If not, why should the proposed repair be acceptable?

Response 15:

FirstEnergy announced on May 23, 2002, that it has purchased the Midland Nuclear Plant unused reactor vessel head from Consumers Energy. The replacement head was manufactured by Babcock & Wilcox and is similar in specifications to the Davis-Besse head.

Question 16: In the responses to the NRC request for additional information related to this event, many of the licensees are relying on results from the current inspection process to justify the continued operation of their plants. Why is this justified given that those very inspection processes failed to discover the hole in Davis-Besse's vessel head?

Response 16:

In Bulletin 2002-01, "Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity," dated March 18, 2002, the NRC staff requested information from licensees regarding their inspection methods (past and planned future inspections), and the capability of these inspections to identify degradation similar to that at Davis-Besse. The inspections that licensees are relying upon have been performed with greater attention to the details of findings around the CRDM nozzles such as signs of degradation of the RPV head and indications of corrosion products.

Davis-Besse did not perform proper inspections and evaluations of the RPV head. The NRC has determined that other PWR licensees have performed proper inspections based on responses to Bulletin 2002-01.

Question 17: The nuclear industry is relying heavily on an Electric Power Research Institute (EPRI) report related to corrosion rates in their reactor vessel head inspection programs. Those programs are at least in part related to the continued operation of plants. The experience at Davis-Besse may not be consistent with the EPRI study. Will the NRC require an independent evaluation of the EPRI report in light of the Davis-Besse experience to justify its continued use as a basis for developing reactor vessel head inspection programs? Has the NRC performed an evaluation itself?

Response 17:

The original EPRI Boric Acid Corrosion Guidebook was issued in 1995 to incorporate experience that was available at that time. The industry continues to develop inspection techniques based on recent and evolving industry experience, and the EPRI Materials Reliability Program (MRP) is developing specific recommendations for inspection techniques in light of CRDM cracking and head degradation issues. The Boric Acid Guidebook would have recommended RPV head cleaning, which, if implemented by Davis-Besse, should have been successful in precluding RPV head degradation. Due to uncertainties associated with the boric acid corrosion mechanism and corrosion rates, the NRC is reexamining its long-term management of this issue, including the industry’s utilization of the EPRI report.

Question 18: Neither the NRC nor the nuclear industry has been able to pinpoint with any certainty the exact cause of the corrosion in the Davis-Besse head. It could have come from above the head from small leaks in the control drive housing flanges, from below through cracks in the penetration or both. Without knowing the origin of the leakage and its exact cause, how can any sort of effective corrective action program be developed to prevent occurrence elsewhere?

Response 18:

The licensee presented the probable cause for the degradation in the Davis-Besse reactor vessel head as boric acid corrosion from the leaking CRDM nozzle penetration attributable to primary water stress corrosion cracking (PWSCC). Some of the boron 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. The NRC staff is evaluating the Davis-Besse root cause analysis report and considering the corrosion mechanism and contributing factors.

Although the mechanistic processes that led to the Davis-Besse head degradation are not known with certainty, implementation of effective inspection and leakage detection/mitigation activities by plants is sufficient to provide assurance that similar conditions do not exist elsewhere. For the present, NRC staff review of licensee responses to Bulletins 2001-01 and 2002-01 indicates that direct visual inspections of the RPV surface and inspections to identify nozzle cracking and leakage provide sufficient assurance at this time that significant RPV head degradation has not occurred at other plants. We are currently following-up with some licensees to clarify their responses to Bulletin 2002-01.

Improved management of this issue will be achieved through the licensee's implementation of enhanced inspection programs to assure timely detection of relevant conditions indicative of CRDM nozzle degradation. In the near-term we expect to provide guidance on the attributes of inspection activities that will be effective in timely detection of CRDM nozzle cracking. This guidance will be based on conservative assumptions in recognition of the uncertainties in our understanding of the degradation phenomena.

Question 19: Both the nuclear industry and the NRC considered failures in the reactor vessel of the type that occurred at Davis-Besse to be not credible. As such, there is no analysis that demonstrates that the public health and safety is maintained. The Davis-Besse event shows that such failures are credible.

a. What changes to the regulations does the NRC anticipate in response to this event that was previously considered incredible?
b. What implications does this have for the NRC's decision to adopt what it has termed "risk-informed" regulation? Did the risk-informed approach to regulation successfully identify the Davis-Besse event as a risk for which appropriate regulations needed to be prepared?
c. If not, what does this say about the efficacy of the NRC's "risk-informed" regulation model?
d. Does the NRC intend to reconsider "risk-informed" regulation in light of the Davis-Besse experience?
e. If such events are to be reviewed, what criteria will the NRC use to judge whether or not new design-basis accidents should be backfitted to older plants or required for new designs? If the NRC does not intend to do such reviews, why not?

Question 19a:

What changes to the regulations does the NRC anticipate in response to this event that was previously considered incredible?

Response 19a:

The NRC has established an independent lessons-learned task force to evaluate the procedures related to inspection activities, generic communications, and policies and practices within the Commission's regulatory structure. This self-assessment will allow NRC management to ascertain the important lessons for the Commission and recommend areas of improvement applicable to the NRC and to the industry. The task force is expected to complete its review by early September 2002, and prepare a public written report containing findings, conclusions and recommendations for staff action.

Question 19b:

What implications does this have for the NRC's decision to adopt what it has termed "risk-informed" regulation? Did the risk-informed approach to regulation successfully identify the Davis-Besse event as a risk for which appropriate regulations needed to be prepared?

Response 19b:

Neither the risk-informed nor traditional regulatory approaches anticipated licensees repeatedly failing to implement effective required corrective action resulting in the reactor vessel head degradation at Davis-Besse. The NRC continues to support risk-informed regulation as a valuable approach. Our risk-informed approach builds upon our traditional processes by bringing risk insights into an integrated decision-making process that incorporates traditional principles, as well. As enumerated in Regulatory Guide 1.174, the integrated principles include meeting existing regulations, maintaining defense-in-depth, maintaining adequate safety margins, and monitoring the factors that are important to making the risk small, as well as considering the known uncertainties in the risk analysis. The applicability of the pertinent principles in the context of the Davis-Besse head degradation is discussed in response to Question 19c, below.

Question 19c:

If not, what does this say about the efficacy of the NRC's "risk-informed" regulation model?

Response 19c:

The NRC does not believe that the Davis-Besse event calls into question the efficacy of its risk-informed regulation model for the reasons discussed below.

It is important to recognize that the risk models that support our risk-informed regulatory approaches do not attempt to identify all individual causes for loss-of-coolant accidents. Neither do the safety analysis models used in our traditional licensing process. Both risk-informed regulation and regulation based on traditional engineering approaches rely on the same basic understanding of phenomena and the engineering analyses of their effects on systems and structures. Neither approach is better or worse than the other in addressing unknown phenomena. Both regulatory processes incorporate the engineering principles for providing extra margins in design analyses and defense-in-depth in designs. These practices can help cope with the effects of unknown phenomena.

Ample design margins allow for substantial degradation without failure. This engineering practice has evolved because degradation mechanisms are known to be hard to predict, both as to the types of degradation that will be important and the rates at which they will occur. Margin is helpful because it usually allows for degradation to be found and corrected before failure occurs.

Defense-in-depth is valuable in case an unknown phenomenon does initiate an accident or cause failure of a safety system. When defense of the public health and safety is provided in-depth, there will still be remaining means of defense, even if one means is defeated by an unanticipated occurrence. In the case of the cavity at Davis-Besse, even if the clad material under the cavity had ruptured, the emergency core cooling system would have been adequate to prevent core damage. With defense-in-depth, the probability that all of the defenses will fail at the same time, due to known or unknown phenomena, is kept very small.

Question 19d:

Does the NRC intend to reconsider “risk-informed” regulation in light of the Davis-Besse experience?

Response 19d:

The staff will review the lessons learned from this experience to determine whether any changes should be made to any of our regulations, processes or procedures, including those that are risk-informed.

Question 19e:

If such events are to be reviewed, what criteria will the NRC use to judge whether or not new design basis accidents should be backfitted to older plants or required for new designs? If the NRC does not intend to do such reviews, why not?

Response 19e:

The degradation in the reactor vessel head experienced at Davis-Besse would result in a loss-of-coolant accident that is within the bounds of such accidents that are part of the current plant licensing design-basis as has been discussed previously. Such an accident would have been within the mitigation capability of the emergency core cooling systems.

Question 20: There is some indication that the Europeans (especially the French) have taken much more aggressive corrective action than the US in response to cracking around CRDM nozzles.

a. What caused the Europeans to adopt this more aggressive approach?
b. Has the NRC office monitored the actions of Europeans?
c. By whom and what office?
d. Did that office know of the actions taken by the French?
e. Did that office inform the Commissioners?
f. To what extent did the NRC take any action knowing what the Europeans did?
g. Provide any papers, correspondence or other documents related to the European response to this problem.
h. Is there any technical basis that the Europeans should have taken more aggressive action?

Question 20a:

What caused the Europeans to adopt this more aggressive approach?

Response 20a:

Other countries repair cracks in their vessel head penetrations, just as the U.S. licensees have done. As stated in NRC's Generic Letter 97-01, "Degradation of Control Rod Drive Mechanism Nozzle and Other Vessel Closure Head Penetrations," "European and Japanese utilities have . . . repaired the nozzles or replaced the heads, as appropriate." The French program to replace the reactor vessel heads, which began in the mid-1990s, is only about half-completed at this point, and is projected to be completed in 2007. The affected regulatory authorities, including the NRC, have strengthened the inspection requirements for vessel heads and CRDM nozzles in particular. For example, and as described in the slides from the meeting between NRC and Duke Power on September 7, 2001, the CRDM cracks discovered at the three Oconee plants have been repaired and reinspected, 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-and-a-half years. The meeting slides are posted on the NRC's Alloy 600 web site, www.nrc.gov/reactors/operating/ops-experience/alloy600.html, under "Public Meetings."

Question 20b:

Has the NRC office monitored the actions of Europeans?

Response 20b:

Yes, as discussed in the response to part a. of this question, the NRC has monitored the actions of European licensees with regard to CRDM cracking issues. In addition, the staff had planned international trips to exchange information on CRDM cracking with its counterparts. These trips were canceled after the events of September 11, 2001. The staff is considering reinstatement of these trips in the future.

Question 20c:

By whom and what office?

Response 20c:

Technical staff in the Office of Nuclear Reactor Regulation, working with the Office of International Programs, and the Office of Nuclear Regulatory Research monitor the activities of our foreign counterparts concerning this issue and other issues related to nuclear power plants.

Question 20d:

Did that office know of the actions taken by the French?

Response 20d:

Yes, all three offices knew of the repairs of CRDMs and reactor vessel head replacements that were and continue to be performed by the French.

Question 20e:

Did that office inform the Commissioners?

Response 20e:

Yes, the NRC staff identified PWSCC as an emerging technical issue for domestic reactors and raised it to the Commission's attention in 1989. The staff issued SECY-97-063, "Proposed NRC Generic Letter: "Degradation of Control Rod Drive Mechanism and Other Vessel Closure Head Penetrations," to the Commission on March 18, 1997. As mentioned in the response to part a. of this question, in April 1997, the NRC issued Generic Letter 97-01 which included a discussion of foreign experience with CRDM cracking beginning in the early 1990's. Specifically, Generic Letter 97-01 included the following statement with regard to the French experience, "In France, Electricite de France (EdF) is planning on replacing all vessel heads as a preventative measure." Therefore, the Commission has been aware of PWSCC issues for both domestic and foreign reactors.

Question 20f:

To what extent did the NRC take any action knowing what the Europeans did?

Response 20f:

As discussed in response to part a. of this question, the NRC issued Generic Letter 97-01 in April 1997, and referenced European and Japanese experience which helped to strengthen the inspection requirements for vessel heads and CRDM nozzles in particular.

Question 20g:

Provide any papers, correspondence, or other documents related to the European response to this problem.

Response 20g:

The NRC obtained information from the French with regard to nozzle cracking in the early to mid-1990s. This information was identified as sensitive when it was provided to the NRC staff. The NRC staff will be requesting permission from the French to release the specific information to the public. However, a summary of previous foreign experience, including the French experience, is provided in Generic Letter 97-01.

Question 20h:

Is there any technical basis that the Europeans should have taken more aggressive action?

Response 20h:

There is no technical basis that caused the Europeans to take more aggressive action. The decision to replace reactor vessel heads was made by the French utility to mitigate future CRDM cracking.

Question 21: We understand that some NRC staff members wanted to ask FirstEnergy additional questions last fall about its request to delay CRDM nozzle inspections, but that NRC senior management overruled this request. Is this true? Please provide a copy of all correspondence between NRC staff and between NRC staff and the Commission relating to the delay in the CRDM nozzle inspections.

Response 21:

In the Fall of 2001, following review of licensee responses to Bulletin 2001-01, the NRC staff became concerned about the lack of inspection information regarding the condition of CRDMs at Davis-Besse. The staff considered ordering the plant to shut down in advance of their scheduled April 2002 refueling outage. To the best of staff's knowledge, Davis-Besse was in full compliance with NRC regulations and the provisions of their operating license. Thus, the burden of proof was on the NRC staff to establish that adequate protection of the public health and safety required a shutdown to perform additional inspections. The December 31, 2001, date that was referenced in Bulletin 2001-01, was not a deadline and did not constitute a legal requirement.

NRC management determined that the most systematic path for justifying a shutdown Order was to make a risk-informed safety case based on recently developed regulatory guidance made public in RIS-2000-07, "Use of Risk-Informed Decisionmaking in License Amendment Reviews," dated March 28, 2000. The staff drafted a proposed Order and, based on information available early in the review, recommended issuing it to Davis-Besse. Attached is a copy of slides from a November 14, 2001, briefing to the Technical Assistants of the Commissioners, in which the staff informed the Commission of this decision. Additionally, a copy of all correspondence between NRC staff relating to the delay in the CRDM nozzle inspections at Davis-Besse is enclosed.

Subsequently, several public meetings were held at which the licensee for Davis-Besse supplied additional inspection and analysis information relevant to the decision. Based on the information supplied at that time, NRC management concluded that the original justification for the proposed Order could not be sustained. Central to this decision was the conclusion that CRDM cracking at Davis-Besse was very unlikely to initiate a loss of coolant accident. An additional factor was the realization that even if there was a failure of the RPV head due to a CRDM failure, the resulting loss of coolant accident (LOCA) was within the design basis envelope of the plant. The degradation that was discovered on the Davis-Besse head was unforseen when the NRC allowed Davis-Besse to operate until February 16, 2002. Had we been aware of the degradation, the Agency would have taken the appropriate regulatory actions to shut down the reactor for the required inspection.

Based on the information available, the staff reached a consensus and recommended to management that the licensee be allowed to operate the plant until its proposed date of February 16, 2002, which was earlier than its originally scheduled outage at the end of March.

Question 22: We have seen press reports indicating that FirstEnergy ordered a new reactor vessel head for Davis-Besse in December, months before it reported the hole in the existing RPV to the NRC. What does this suggest regarding what the licensee knew about problems with the reactor? Did the licensee tell the NRC staff that it was planning to replace the RPV head at the time that it was requesting a delay in the CRDM nozzle inspections? If not, should it have done so?

Response 22:

FirstEnergy had determined it would be less expensive to replace the head than to repeatedly inspect for and repair defective nozzles. Many other nuclear power plants have followed the same course of action by ordering new reactor vessel heads. The licensee placed the order for the new head with Framatome in October 2001, and expected delivery of the head in time for a planned outage in early 2004. During a November 2001 meeting to discuss CRDM nozzle inspections, the licensee informed the NRC that it was planning to replace the RPV head. We are not aware of a relationship between FirstEnergy's order of a new reactor vessel head and its request to delay the vessel head inspection six weeks beyond December 31, 2001.

List of Attached Reference Material

Question 3:

A. Walton Jensen, NRC, Memorandum to Stephen Dinsmore, NRC, “CRDM Thimble Rupture Study,” August 1, 2001.

B. Walton Jensen, NRC, Memorandum to Gary Holahan, NRC, “Sensitivity Study of PWR Reactor Vessel Breaks,” May 10, 2002.

Question 4:

C. Detailed description of the operation of the Control Rod Drive Mechanisms.

Question 8:

D. NRC Inspection Report 50-346/02-03.

Question 9:

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

F. NRC Bulletin 2001-01, “Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles,” August 3, 2001.

G. NRC Bulletin 2002-01, “Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity,” March 18, 2002.

Question 10:

H. A list of operating pressurized-water reactors in the United States.

Question 13:

I. NRC Information Notice (IN) 2002-13, “Possible Indicators of Ongoing Reactor Pressure Vessel Head Degradation,” April 4, 2002.

Question 14:

J. CR 1999-1947, An Area For Improvement From the Institute of Nuclear Power Operation Evaluation Identified That Equipment Problems Have Complicated Plant Transients and Have Contributed to Plant Events.

K. CR 1999-1300, Several Filters From the Containment Radiation Monitors Were Sent For Analysis. This CR Documents the Results.

L. PCAQR 98-0942, Cleanliness Inspection on Transfer Tubes.

M. CR 01-1110, RE4597BA Filter Change Occurring More Frequently.

N. CR 01-2795, RE4597BA Alarm.

O. CR 01-1822, Increasing Frequency of RE4597BA Filter Changeout.

P. Plant Issues Indexes for January, February, and March of 2000. These are included because the “high containment impurities” issue was dropped from the list after March 2000.

Q. NRC Inspection Report 50-346/99010(DRP).

R. NRC Inspection Report 50-346/99009 (DRP).

S. NRC Inspection Report 30-346/2001013 (DRP).

Question 18:

T. NRC Information Notice 2002-11, “Recent Experience with Degradation of Reactor Pressure Vessel Head,” March 12, 2002.

Question 20:

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

Question 21:

V. RIS-2000-07, “Use of Risk-Informed Decisionmaking in License Amendment Reviews,” March 28, 2000.

W. Copy of slides from a November 14, 2001, staff briefing to the Commissioners Technical Assistants.

X. Requested correspondence between NRC staff relating to the delay in the CRDM nozzle inspections at Davis-Besse.

Reference C: Control Rod Drive Mechanism Operation

The control rod drive mechanism (CRDM) is an electromechnical device. During normal operation the drive mechanism is used to raise, lower, or maintain control rod position within the reactor in response to electrical signals from the control rod drive motor control system. The control system provides a sequentially programed direct current (dc) input to the four-pole, reluctance-type drive motor to produce a rotating magnetic field for the rotor assembly. The rotor assembly is split so that when power is applied to the stator, the rotor assembly arms pivot to mechanically engage the roller nuts with the lead screw threads. When electric power is applied to the electric motor, it causes the operating mechanism to engage the lead screw of the control rod. The rotation of the operating mechanism causes the leadscrew motion. The electric motor drive is designed to trip whenever electric power is removed from it. This disengages the operating mechanism from the leadscew causing the control rods to fall under gravity into the reactor core. This is known as a reactor trip.

The control rod indication system is an integral part of the control rod drive housing and provides absolute position indication by the use of reed switches. In addition to providing indication, these switches also provide control, limit, and alarm function capability for the control system in the control room.

The CRDM consists of a motor tube which acts as the pressure boundary and houses the leadscrew, the leadscrew rotor assembly, and a snubber assembly. The top end of the motor tube is sealed by a closure and vent assembly. The motor stator is mounted externally and surrounds the motor tube. The rotational motion of the rotor assembly is translated to the non-rotating leadscrew coupled to the control rod. The leadscrew is driven by separating roller nut assemblies attached to segment arms which are rotated magnetically by the motor stator outside the motor tube. Current flow through the stator windings establishes a magnetic field which causes the separating roller nut assembly arms to close and engage the leadscrew. When current is removed from the stator, the loss of the magnetic field allows mechanical springs to force the segment arms apart disengaging the roller nut halves from the leadscrew.

The control rod drive mechanism is designed to “trip” whenever power to the stator is interrupted, due to a transient, such as a small-break loss-of-coolant accident. When the drive mechanisms are required to respond to a trip signal, the action of the control rod drive system and the drive mechanism results in a positive, nonreversible initiation of the trip function. The trip command has priority over all other commands. The CRDM system is required to trip the CRDM whenever it receives an automatic trip command signal from the reactor protection system (RPS) or a manual trip command signal from the operator. During a power loss, the rotor assembly segment arms pivot, releasing the mechanical contact between the roller nut and the leadscrew. The lead screw and control rod are then pulled into the reactor core to the full-in positions by gravity. During the free fall condition, coolant is allowed to pass from the reactor head area into the motor tube housing, through the ball check valves in the lead screw. This prevents the formation of a low-pressure area that could affect rod drop times. The hydraulic snubber assembly, within the motor tube housing, decelerates the moving control rod assembly (CRA) to a low speed just before it reaches the CRA full-in position. The final deceleration energy is absorbed by the belleville spring assembly. The CRDM system is designed to provide safe shutdown and to provide for positive and safe reactor shutdown from all operating and transient load conditions without damage to the reactor.