Union of Concerned Scientists

Citizens and Scientists for Environmental Solutions

June 12, 2002

Mr. John A. Grobe, Director
Division of Reactor Safety
United States Nuclear Regulatory Commission
801 Warrenville Road
Lisle, IL 60532-4351

SUBJECT: QUESTIONS FOR DAVIS-BESSE MANUAL CHAPTER 0350 PANEL

Dear Mr. Grobe

The Union of Concerned Scientists (UCS) is pleased that the Nuclear Regulatory Commission (NRC) established a Manual Chapter 0350 Panel for Davis-Besse and named you as its Chair. The 0350 Panel you chaired for the restart of D C Cook worked very well in our view. It enabled us to monitor the status of numerous issues as they were added to, revised on, and worked off the restart schedule. The 0350 Panel process also provided us with a vehicle for inputting our questions and concerns into a formal process and track their resolution.

We have every expectation that the 0350 Panel process will work as well, if not better, for Davis-Besse. There is already evidence of it working better in the commendable efforts of the NRC staff in providing telephone access to Davis-Besse public meetings. When the NRC grants the 2.206 petition that UCS and fourteen other organizations submitted regarding Davis-Besse, the 0350 panel can easily incorporate those tasks into its activities.

This letter conveys a number of concerns and related questions we feel the 0350 Panel should address prior to restart. It is not our expectation that the NRC explicitly answer each of our questions. If the answer is contained within a publicly available document such as an inspection report, all we ask is that for the NRC to point us to that document.

Because we anticipate additional or follow-up questions may arise and answers will not arrive all at once, we adopted the practice of numbering our questions. We would appreciate the NRC citing our question numbers when answering the questions and pointing to documents containing answers.

We attempted to provide sufficient background information as to explain the underlying concerns and place the questions in context. If the 0350 Panel wants clarifying information about any concern or question, I am prepared to discuss it during a 0350 Panel public meeting or other forum.

Sincerely,

David A. Lochbaum

David Lochbaum
Nuclear Safety Engineer
Washington Office

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Apparent Failure to Comply with Federal Regulations for Updating the FSAR

On October 9, 1996, James M. Taylor, Executive Director for Operations at the Nuclear Regulatory Commission (NRC), notified senior management for every nuclear power plant of the NRC's expectations regarding conformance with design and licensing bases requirements. Mr. Taylor stressed the importance of the Final Safety Analysis Report (FSAR):

> The FSAR is required to be included in, and is one portion of, an application for an operating license (OL) for a production or utilization facility. 10 CFR 50.34(b) describes the information which must be included in an FSAR. **The FSAR is the principal document upon which the Commission bases a decision to issue an OL and is, as such, part of the licensing basis of a facility. It is also a basic document used by NRC inspectors to determine whether the facility has been constructed and is operating within the license conditions.** [emphasis added]

NRC Chairman Shirley Ann Jackson explained the FSAR's vital role during a speech at the 17th Annual Institute of Nuclear Power Operations Conference for utility chief executive officers on November 7, 1996:

> The NRC uses the FSAR when evaluating license amendment requests and other issues at particular facilities and will use the FSAR in reviewing applications for license renewals. **The accuracy of the FSAR, and the design basis generally, has a direct impact on the accuracy of recurring reviews and safety analyses performed by the NRC staff.** NRC inspectors continue to use the FSAR as a baseline when conducting inspections. [emphasis added]

Thus, the NRC relies on an accurate FSAR when determining that nuclear power plants comply with federal safety regulations. Leonard Bickwit, Jr., NRC General Counsel, stated in a letter dated August 14, 1980, to the NRC Commissioners, "**...compliance with the Commission's regulations is essential to a determination of adequate protection of the public health and safety under the Atomic Energy Act** [of 1954]" [emphasis added]. It logically follows that an accurate FSAR is essential for adequate protection of public health and safety and that an inaccurate, incomplete FSAR jeopardizes adequate protection.

Title 10 of the Code of Federal Regulations (CFR) Section 50.34(b) specifies the NRC's regulations covering information contained in the FSAR:

> A description and analysis of the structures, systems, and components of the facility, with emphasis upon performance requirements, the bases, with technical justification therefor, upon which such requirements have been established, and the evaluations required to show that safety functions will be accomplished. The description shall be sufficient to permit understanding of the system designs and their relationship to safety evaluations.

As reported in the *Federal Register* on May 8, 1980, the NRC issued a final rule effective July 22, 1980, requiring periodic updating of FSARs at all nuclear power plants:

> The Nuclear Regulatory Commission is amending its regulations [10 CFR 50.71(e)] to require each person licensed to operate a nuclear power reactor to submit periodically to the Commission revised pages for its Final Safety Analysis Report (FSAR). **These revised pages will indicate changes which have been made to reflect information and analyses submitted to the Commission or prepared as a result of Commission requirement. The amendment is**

> **being made to provide an updated reference document to be used in recurring safety analyses performed by the licensee, the Commission, and other interested parties.** [emphasis added]

The current requirements specified in 10 CFR Section 50.71(e):

> Each person licensed to operate a nuclear power reactor pursuant to the provisions of Section 50.21 or Section 50.22 of this part shall update periodically, as provided in paragraphs (e)(3) and (4) of this section, the final safety analysis report (FSAR) originally submitted as part of the application for the operating license, to assure that the information included in the FSAR contains the latest material developed. This submittal shall contain all the changes necessary to reflect information and analyses submitted to the Commission by the licensee or prepared by the licensee pursuant to Commission requirement since the submission of the original FSAR or, as appropriate, the last update FSAR. **The updated FSAR shall be revised to include the effects of:** all changes made in the facility or procedures as described in the FSAR; all safety evaluations performed by the licensee either in support of requested license amendments or in support of conclusions that changes did not involve an unreviewed safety question; and **all analyses of new safety issues performed by or on behalf of the licensee at Commission request. The updated information shall be appropriately located within the FSAR.** [emphasis added]

The regulation's purpose seems obvious. The Final Safety Analysis Report describes design features, procedures, and safety analyses that ensure plant operation conforms with federal nuclear safety standards. The regulation requires that any changes to design features (i.e., plant modifications), revisions to procedures, and any new or revised safety analyses be reflected in the FSAR by periodic updates. Periodic updating is essential because the FSAR is a "living" document used in the day-to-day operation and maintenance of the nuclear power plant. Complete and accurate FSAR information ensures that safety decisions are based on all relevant and applicable information.

The FSAR is an important source and reference document that is used daily at nuclear power plants. It is a source document for benchmarking control room simulators, writing emergency procedures, developing lesson plans, conducting probabilistic risk assessments, and performing numerous other activities. It is a reference document for determinations if proposed changes to the plant and its procedures can be made without adversely affecting safety margins and for safety inspections conducted by the NRC.

An incomplete and inaccurate FSAR corrupts all of these efforts. It makes it more likely that safety margins will be inadvertently eroded by individuals basing decisions on incomplete or inaccurate information. Some might claim that updating the FSAR to include safety analyses required by the NRC is not necessary because the relevant information has been shared with the NRC and is available. However, as pointed out by James M. Taylor, NRC Executive Director for Operations, in memo SECY-95-300 dated December 20, 1995:

> **"Those commitments not contained in the FSAR are not controlled by a defined regulatory process such as 10 CFR 50.59. Therefore, licensees have the ability to change docketed commitments not contained in the FSAR without informing the Commission.**" [emphasis added]

It is theoretically possible that everyone using the FSAR as a source or reference document also reviews correspondence to and from the NRC to extract applicable information from safety analyses performed at the NRC's request. It is extremely improbable that this intensive effort is made every time.

A review of the Davis-Besse UFSAR by UCS did not reveal any discussion of the analyses of safety issues performed in response to NRC requests such as Bulletin 2001-01, "Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles:" Generic Letter 97-01, "Degradation of Control Rod Drive Mechanism Nozzle and Other Vessel Closure Head Penetrations;" Generic Letter 88-05, "Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants;" and Bulletin 82-02, "Degradation of Threaded Fasteners in the Reactor Coolant Pressure Boundary of PWR Plants." Assuming for the moment that Davis-Besse performed the requested analyses, it appears that they did not comply with 10 CFR 50.71 paragraph (e) by incorporating information from said analyses into the UFSAR. That omission, in our view, contributed to the repeated failures of plant workers to fully appreciate the numerous warning signs of reactor vessel head damage.

Two specific NRC requests are provided as case studies:

<u>NRC Bulletin 88-04</u>
On May 5, 1988, the NRC issued Bulletin 88-04, "Potential Safety-Related Pump Loss" to all holders of operating licenses for nuclear power plants. The safety issue described in Bulletin 88-04 has significant nuclear safety implications because it involved emergency core cooling system (ECCS) performance and core cooling capability (in other words, the system that would have had to function properly had the damaged reactor head at Davis-Besse given way).

The required actions specified in Bulletin 88-04 included:

> **Evaluate the adequacy of the minimum flow bypass lines for safety-related centrifugal pumps with respect to damage resulting from operation and testing in the minimum flow mode.** [emphasis added]

UCS did not find a summary of this evaluation in the Davis-Besse UFSAR.

<u>NRC Bulletin 88-11</u>
On December 20, 1988, the NRC issued Bulletin 88-11, "Pressurizer Surge Line Thermal Stratification" to all holders of operating licenses for pressurized water reactors.

The required actions specified in Bulletin 88-11 included:

> Within four months of receipt of this Bulletin, licensees of plants in operation over 10 years (i.e., low power license prior to January 1, 1979) are requested to demonstrate that the pressurizer surge line meets the applicable design codes and other FSAR and regulatory commitments for the licensed life of the plant, considering the phenomenon of thermal stratification and thermal striping in the fatigue and stress evaluations. **This may be accomplished by performing a plant specific or generic bounding analysis. If the latter option is selected, licensees should demonstrate applicability of the referenced generic bounding analysis.** [emphasis added]

According to the NRC in Bulletin 88-11, pressurizer surge line stratification is a safety concern because:

> Unexpected piping movements are highly undesirable because of potential high piping stress that may exceed design limits for fatigue and stresses. The problem can be more acute when the piping expansion is restricted, such as through contact with pipe whip restraints. **Plastic deformation can result, which can lead to high local stresses, low cycle fatigue and functional impairment of the line**. Analysis performed by the Trojan licensee indicated that thermal stratification occurs in the pressurizer surge line during heatup, cooldown, and steady-state operations of the plant. [emphasis added]

UCS did not find a summary of this evaluation in the Davis-Besse UFSAR.

The Federal regulations of 10 CFR 50.71(e) require plant owners to incorporate the results of evaluations and analyses performed at the request of the NRC into their Final Safety Analysis Reports. It appears that FirstEnergy has not complied with this regulation, which has been in effect since 1980.

UCS-01a **Will the Davis-Besse FSAR be in compliance with 10 CFR 71(e) prior to restart?**
UCS-01b **Since NRC Chairman Jackson asserted that "The accuracy of the FSAR, and the design basis generally, has a direct impact on the accuracy of recurring reviews and safety analyses performed by the NRC staff," what steps will the NRC staff take to ensure that its past decisions based on the incomplete, inaccurate Davis-Besse FSAR were proper?**

Potential Inability of Plant to Cope with Accident

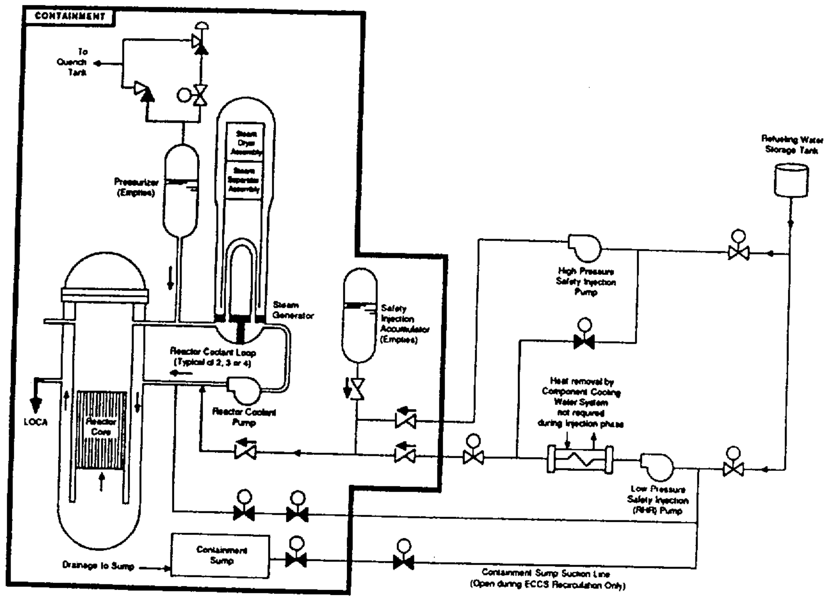
The reactor vessel head damage at Davis-Besse caused the 3/16-inch stainless steel cladding to become the reactor coolant pressure boundary. For the as-found cavity, the stainless steel cladding reportedly retained sufficient strength to withstand the reactor pressures from normal operation and postulated design basis accidents. Had the reactor continued to operate and the cavity continued to enlarge, the stainless steel cladding would have been breached at some point. The pressurized water within the reactor vessel would have rapidly left via the opening, creating the need to replenish the lost inventory.

Although not specifically designed for a broken reactor vessel, Davis-Besse was designed with several systems installed to mitigate a loss-of-coolant accident (LOCA). If these systems function properly, the reactor core remains adequately cooled to prevent significant fuel damage.

A draft report prepared by the Los Alamos National Laboratory and issued by the NRC in July 2001 entitled "GSI-191: Parametric Evaluations for Pressurized Water Reactor Recirculation Sump Performance," outlined this potential problem:

> In the event of a loss-of-coolant accident (LOCA) within the containment of a pressurized water reactor (PWR), piping thermal insulation and other materials in the vicinity of the break will be dislodged by break-jet impingement. **A fraction of this fragmented and dislodged insulation, and other materials such as paint chips, paint particulates, and concrete dust, will be transported to the containment floor** by the steam/water flows induced by the break and by the containment sprays. Some of this debris will eventually be transported to and accumulated on the recirculation sump suction-screens. Debris accumulation on the sump screen may challenge the sump's capability to provide adequate, long-term cooling water to the ECCS and to the containment spray (CS) pumps for extended periods of time. [emphasis added]

The schematic illustrates the systems involved following a loss-of-coolant accident. The LOCA is represented by the opening on the left side of the reactor vessel allowing cooling water to pour out into the containment building where it drains to the containment sump. The high pressure safety injection pumps and the low pressure safety injection pumps (one each shown on the right portion of the schematic) initially supply water from the refueling water storage tank to the reactor vessel to compensate for the lost inventory. Thirty minutes to an hour into the accident, the refueling water storage tank empties. Operators close valve (shown as the white bow-ties on the schematic) and open other valves (the black bow-ties) to allow the pumps to take water from the containment sump and deliver it to the reactor vessel.



PWR Coolant Injection and Heat Transport Paths During a Large LOCA-ECCS Injection Phase

The NRC's Advisory Committee on Reactor Safeguards reviewed this draft report and related documents in September 2001. In a letter dated September 14, 2001, ACRS Chairman George E. Apostolakis informed NRC Executive Director of Operations William Travers:

> We agree with the staff that potential issues associated with the performance of pressurized water reactor containment sumps have been identified. The NRC staff should expeditiously resolve GSI-191.

The need for expeditious resolution of this safety problem is illustrated by the following table from the executive summary of the Los Alamos draft report:

Sump Failure Potential	SLOCA	MLOCA	LLOCA
Very Likely	23	32	57
Likely	10	8	4
Possible	10	3	0
Unlikely	26	26	8
Total	69	69	69

SLOCA, MLOCA, and LLOCA stand for small-break loss-of-coolant accident, medium-break loss-of-coolant accident, and large-break loss-of-coolant accident, respectively. The numbers represent the operating fleet of 69 pressurized water reactors. According to Los Alamos, 57 of the 69 operating PWRs (82 percent) will "very likely" experience containment sump failure in event of a large-break loss-of-coolant accident. Only 8 reactors (12 percent) are "unlikely" to avoid sump failure. The reactor vessel head damage at Davis-Besse approximated the area of a medium-break loss-of-coolant accident. Thirty-two reactors (56 percent) are "very likely" to experience sump failure following a medium-break loss-of-coolant accident.

Sump failure is a very serious problem. It means the low pressure safety injection pumps are no longer able to take water from the containment sump and recycle it to the reactor vessel. Operators might be able to re-fill the refueling water storage tank or align the pumps to other external sources of water, but that doesn't completely save the day. Continued reliance on make-up water from outside the containment means that the flood level of water inside the containment rises and rises. Important safety equipment, like motors, could be submerged and disabled.

Numerous near-misses have already pointed to this vulnerability. By letter dated March 1, 2001, NRC Executive Director for Operations William Travers informed the NRC Chairman and Commissioners on the status of the accident sequence precursor program (SECY-2001-0034). Dr. Travers reported:

> A review of final and preliminary precursor results for the period 1994-2000 shows that several precursors involved event initiators or conditions not typically modeled in PRAs or IPEs. **These events make up approximately 20 percent of the precursors for this period. Almost half of these events involved conditions that could render safety-related equipment inoperable during a postulated high-energy line break**. [emphasis added]

Sump failure would indeed be a condition "that could render safety-related equipment inoperable during a postulated high-energy line break." This vulnerability is even more chilling given how close Davis-Besse came to needing the containment sump to function properly.

The draft report by Los Alamos masked the names of the 69 PWRs in the study, so it is not publicly known whether Davis-Besse is one of the more vulnerable or one of the least vulnerable plants to the sump failure problem. It is publicly known that Davis-Besse operated for years with one or more cracked CRDM nozzles that leaked hundreds of thousands of gallons of borated water into containment. When the water evaporated, it left behind lots of boric acid crystals all over the place inside containment. That powder is not unlike the "paint chips, paint particulates, and concrete dust" identified by Los Alamos as prime suspects for blocking the sump screens.

UCS-02a **Will the plant-specific evaluation of GSI-191 vulnerability recommended by the NRC staff in its presentation to the ACRS in September 2001 be completed for Davis-Besse prior to restart?**

UCS-02b **If not, would the NRC staff not be guilty of the exact same, or at least eerily similar, tolerance of degraded conditions that caused the current problem at Davis-Besse?**

UCS-02c **Will all of the boric acid deposited inside the reactor containment at Davis-Besse be found and removed prior to restart?**

UCS-02d **If not, what assurance exists that this debris won't be transported to the containment sump and contribute to sump failure?**

Apparent Lack of Proper Controls Over Maintenance Activity

Davis-Besse work order 00-001846-000 (Enclosure 1) from April/May 2000 resulted in the reactor vessel head being power washed to remove boric acid accumulation. According to the work order, "Some boron deposits are hardened and soaking time may be required" and "The cleaning media will be pressurized de-mineralized water heated to approximately 175°F."

In numerous public meetings conducted earlier this year, NRC staffers and FirstEnergy representatives have repeatedly stated that the reactor vessel head damage was surprising even though large amounts of boric acid had been evident over many years. The oft-cited explanation was that the high temperatures in the area surrounding the reactor vessel head when the plant was operating caused leaked borated water to quickly evaporate, leaving behind boric acid crystals in a benign form.

Even if this hypothesis were correct, it would seem to be undermined by adding water to the boric acid crystals and then letting this potent liquid acid "stew" awhile on the unprotected reactor vessel head. Thus, it would seem that FirstEnergy' actions (and inactions) suggest that the company did not view boric acid to be potentially damaging to the carbon steel reactor vessel head whether in the form of dry crystals or when dissolved in water. Davis-Besse UFSAR Section 5.2.3.2, page 5.2-15 states

> All materials exposed to the reactor coolant exhibit corrosion resistance for the expected service condition. ... Sensitized stainless steel weld overlay (cladding) is permitted.

The reactor coolant is borated water formed by adding boric acid crystals to deminerlized water. When demineralized water is added to boric acid crystals lying on the reactor vessel head, the result is essentially reactor coolant. It is well known and now re-established that the carbon steel reactor vessel head is not resistant to corrosion. It is also well known that the outer surface of the reactor vessel head is not covered with stainless steel, as is the inner surface. Hence, it is questionable if the maintenance activity of adding demineralized water to boric acid crystals on the unprotected reactor carbon steel reactor vessel head conforms with UFSAR Section 5.2.3.2. A safety evaluation performed under 10 CFR 50.59 for the proposed power washing activity should have evaluated whether it was a good idea.

UCS-03a **Did FirstEnergy perform a safety evaluation (or screening) in accordance with 10 CFR 50.59 for the power washing activity described in work order 00-001846-000?**

UCS-03b **If not, how can the company and the NRC assure adequate protection against potential damage to safety equipment and components caused by maintenance activities?**

<u>Opportunity to Eliminate Causal Factor</u>

It is our understanding that Davis-Besse is one of a very small handful of nuclear power plants using mechanical flanges for the control rod drive mechanisms (CRDMs). The overwhelming majority of nuclear power plants use CRDM seal welds. While both methods have experienced leakage, the mechanical flanges have experienced more frequent problems.

Leakage of borated water from the mechanical flanges at Davis-Besse over many years contributed to the severity of the reactor head damage. Whether the boric acid from that leakage actually corroded the reactor vessel head or not, its accumulation on the head impeded inspections.

FirstEnergy is replacing the reactor vessel closure head with a similar head intended for the Midland nuclear plant. It would seem an opportunity to also modify the CRDM nozzles to replace the mechanical flanges with seal welds. It would seem to have both economic and safety benefits to reduce the likelihood of future CRDM leakage.

UCS-04 **Will the CRDM mechanical flanges be replaced with seal welds prior to restart?**

Musical Chairs

Several management changes have already occurred at Davis-Besse and more are likely to occur prior to restart.

The NRC recently banned Ms. Gail C. Van Cleave from working in the nuclear industry because she had used her dead mother's social security number in an application for a job as a clerk in the warehouse at the D C Cook nuclear plant in Michigan. Although the NRC's investigation did not identify a single instance where Ms. Van Cleave's performance as a warehouse clerk actually or potentially affected the safety of the D C Cook nuclear plant in any adverse way, the agency considered this single working mother to lack the trustworthiness and reliability needed to work in the nuclear industry.

UCS-05a	**How many of the top twenty managers/supervisors in place at Davis-Besse on February 16, 2002 (when the plant shut down for its current outage) have left FirstEnergy? [NOTE: For privacy reasons, please do not provide the names of anyone who has departed.]**
UCS-05b	**If the behavior of Ms. Van Cleave was such a "clear and present danger" to public health and safety that the NRC could simply not trust her to work in the nuclear industry, is the behavior of FirstEnergy management at Davis-Besse from 1996 through 2002 better than or worse than that of Ms. Van Cleave?**
UCS-05c	**If Ms. Van Cleave's behavior is better, will the NRC ban any FirstEnergy managers and supervisors from working in the nuclear industry?**
UCS-05d	**If Ms. Van Cleave's behavior is worse, please return to Question UCS-05b and try again.**
UCS-05e	**The NRC banned Ms. Van Cleave from working at D C Cook and every other nuclear power plant in the United States because of her indiscretion at D C Cook. Does the NRC care that managers and supervisors fired or otherwise removed from Davis-Besse because of that troubled plant's problems find work at other US nuclear power plants?**

Investigations

UCS is aware that the US Congress dispatched an investigator to Davis-Besse and that the NRC Office of Investigations is pursuing a multi-faceted investigation into possible wrong-doing at Davis-Besse.

UCS-06a	**Will the results of these investigations be publicly available prior to restart?**
UCS-06b	**If not, how can people living around the plant have any assurance that workers and managers at Davis-Besse did not violate federal safety regulations and place them at undue risk?**

NRC Acceptance of "Interim" Head Transplant

During public meetings between FirstEnergy and the NRC staff on June 4th and FirstEnergy and the ACRS on June 6th, FirstEnergy stated its intention of operating Davis-Besse with a reactor vessel closure head transplanted from the cancelled Midland nuclear plant until 2010 or 2012, when a permanent reactor vessel closure head will be installed. FirstEnergy indicated that the permanent reactor vessel closure head installation would not occur before 2010 or 2012 so as to coordinate that effort with the steam generator replacements, when another hole must be cut in the reactor containment.

The NRC staff is in the process of reviewing the head transplant and must concur with FirstEnergy's plans as a condition for restart.

UCS-07 **Will the NRC's acceptance of the interim head be conditional on a firm commitment from FirstEnergy to install the permanent reactor vessel closure head no latter than the outage when the steam generators are replaced?**

Not so ironic Déjà vu

The NRC was harshly criticized following the March 1979 accident at Three Mile Island because the agency had information about a similar precursor event occurring at Davis-Besse in 1978, but failed to share that warning with the owners of potentially vulnerable plants.

In a letter dated January 28, 1972 (Enclosure 2), the NRC notified the owner of Three Mile Island about an event at a foreign reactor. Specifically, the NRC noted:

> As you know, an event occurred at a foreign pressurized water power reactor in which **an unusual corrosion mechanism occurred when prolonged leakage of borated reactor coolant onto the reactor vessel head was undetected. Subsequent tests have indicated that this corrosion potential might exist under certain conditions when borated fluid has prolonged contact with stainless steel.** [emphasis added]

UCS was unable to find any correspondence from NRC (then called the Atomic Energy Commission) to Davis-Besse's owner in 1972 regarding this potential problem. There were several letters in the old records showing that NRC warned other plant owners in 1972 about it. We suspect that the reason NRC did not warn Davis-Besse's owner in 1972 was that the plant was still under construction. It did not receive an operating license from the NRC until April 1977. The NRC's warning went out to owners of operating plants, not to owners of plants soon to be operating.

It's ironic that NRC had information about safety problems at Three Mile Island that it did not share with Davis-Besse and later had information about safety problems at Davis-Besse that it did not share with Three Mile Island.

The difference, of course, is that the NRC did share plenty of information about boric acid problems with Davis-Besse on numerous occasions after the plant received its operating license in 1977. FirstEnergy failed to properly heed that information, making it highly unlikely that they would have heeded the 1972 warning had they received it.

UCS-08 **What was the similar problem that occurred at a foreign reactor to prompt the agency to warn some plant owners in 1972?**

Enclosure 1

FENOC DAVIS-BESSE PLANT

T1*RC

Work Order 00-001846-000
Subsystem: SUB052-01

Asset: T1*RC REACTOR VESSEL 1-1
Problem Locn: CTMT9_213*_565

Action: ROUTINE MAINTENANCE
WO Type:
Clearance: N

Work Class:
Nat.Acct: 4537 4537 15 NR13A DBRX 00000 00
Printed: 25-APR-00 15:51 D2L

Clearance number:
Tech Spec: N
Test Requirements: N
Lead Craft: RADIATION TEST

Quality Class: Q
Environmental Qualification: N
ASME Component: ASMEXI
Repair Tag Number: U0896
Train:

Permission to Commence Work

SS/SM Authorization: N/A DATE

SUPERVISOR [illegible] DATE [illegible]

Requested by: ANDREW SIEMASZKO **Phone:** 7341
Planner: DENNIS A LISKA **Phone:** 8338

Problem Description:
LARGE BORON ACCUMULATION WAS NOTED ON THE TOP OF THE RX HEAD AND ON TOP OF THE INSULATION. BORIC ACID CORROSION MAY OCCUR

NO TAGS HUNG (IN CONTAINMENT)
SS/SM APPROVED BY: GARY MELSSEN
FAILURE DATE: 04-21-00
JDA04-21-00

Work Description:
CLEAN BORON ACCUMULATION FROM TOP OF REACTOR HEAD AND ON TOP OF INSULATION.
SEE ANDREW SIEMASZKO (PLANT ENGINEERING), EXT 7341 FOR ADDITIONAL DETAILS

Work Order Review

Plant Engineering	Andrew Siemaszko	DATE	4/25/00
SRO	[illegible]	DATE	4/25/00
ALARA	[illegible]	DATE	4/25/00
QC Mechanical	[illegible]	DATE	4/25/00
Lead Shop Review	[illegible]	DATE	5/3/00

Special Instructions:
Ensure water does not enter [illegible] [illegible] 4/25/00

Page 1 of 1

04/17/2002 DBNPS

A-5

APR-17-2002 09:33 4198980691 98% P.02

T1-RC

Work Order 00-001846-000
Subsystem: SVB062-01

Permits

RWP
Transient Combustible

Steps

	Craft	Crew Size	Crew Name	Hrs
1	RADIATION TEST	4		10

CLEAN BORON ACCUMULATION FROM TOP OF REACTOR HEAD AND ON TOP OF INSULATION.

SEE ANDREW SIEMASZKO (PLANT ENGINEERING), EXT 7341 FOR ADDITIONAL DETAILS.

1) RAISE LEAD BLANKETS AS REQUIRED TO PROVIDE ACCESS TO WEEP HOLES. ALL BLANKETS WILL HAVE TO BE RAISED TO PROVIDE ACCESS 360 DEGREES AROUND HEAD AT WEEP HOLE LEVEL.

2) INSTALL PROTECTIVE COVERING ON REACTOR HEAD BOLT HOLES. THIS IS REQUIRED TO PREVENT WATER RUN OFF FROM DRAINING THROUGH BOLT HOLES.

3) COVER WEEP HOLES AND PROVIDE DRAIN

4) POWER WASH REACTOR VESSEL HEAD

5) REMOVE PLASTIC AND PROTECTIVE COVERS.

6) RESTORE LEAD BLANKETS AS DIRECTED BY RP

SIGNATURE: CAK DATE: 4/25/00

	Craft	Crew Size	Crew Name	Hrs
2	MECHANICAL	4		4

REMOVE AND REPLACE LEXAN COVERS ON REACTOR VESSEL HEAD TO FACILITATE CLEANING.

SIGNATURE: N/A. [illegible] DATE: 4/25/00

	Craft	Crew Size	Crew Name	Hrs
3	MAINTENANCE SERVICES	1		6

IF NECESSARY MANUFACTURE REPLACEMENT LEXAN COVERS

SIGNATURE: N/A. [illegible] DATE: 4/25/00

Closeout

Lead Shop / MDT Removed	CAK	Date:	5/1/00 [illegible]
SS/SM Authorization	N/A [illegible]	Date:	
QC Mechanical	N/A [illegible]	Date:	
Planner Review	[illegible]	Date:	[illegible]

Completion Date: 4/ [illegible] /00 Completed By [illegible]

2

DESCRIPTION OF WORK PERFORMED

Work performed without deviations.

[illegible signature]
4/25/00

CONTINUED ON BACK SIDE

Reactor Vessel Head cleaning.

Large deposits of boron have accumulated on the top of the insulation and on the Reactor Vessel Head. Nuclear Regulatory Commission (NRC) issued Generic Letter 97-01 to holders of operating licenses for pressurized water reactors (PWR's). The letter requires to maintain the program for ensuring the timely inspection of the control rod drive mechanism (CRDM) and other vessel closure head penetrations. The program is required due to degradation of the CRDM nozzle caused by Primary Water Stress Corrosion Cracking process. In order to perform required inspections the nozzles as well as the penetrations must be free of boron deposits. Once the head is free from the boron new boron deposits may be easily noted and remedial actions taken.

Background and technical information.
Beginning in 1986, Alloy 6000 CRDM nozzle leaks have been reported

Overview of the cleaning effort.
There are two areas requiring cleaning. The area above the insulation and the area below the insulation on the top of the reactor vessel head. The area above the insulation is accessible through the ventilation duct openings located approximately seven feet above the head flange. Scaffolding (movable platform) will be utilized to gain access to the ventilation duct openings after Lexan covers will be removed. The area below the insulation on the top of the reactor vessel head will be accessible via the weep holes (other name is mouse holes). The cleaning media will be pressurized de-mineralized water heated to approximately 175 °F. Water will be sprayed on the boron deposits through the ventilation duct openings and through the weep holes. One weep hole will be used to drain the liquid out of the head to the plastic drums. The remaining weep holes will be blocked with a plastic tape. The plastic drums will be located outside of the head stand area at the base of the water shield tanks. Two inch diameter corrugated plastic hose will provide means of transporting the liquid from the weep hole to the plastic drums. Accumulated liquid will be disposed off as directed by Health Physics and or Decontamination Department personnel. The estimated volume of water used will be between 100 and 600 gallons. Some boron deposits are hardened and soaking time may be required.

Major challenges of the cleaning effort will be associated with the spill protection. Recently installed inner and outer Reactor Vessel Head gaskets can not become soaked with the boric acid solution. To protect the gaskets number of protective measures will be taken.

- All but one weep hole will be blocked with the plastic cover. In the event the water is escaping from the covered weep hole the cleaning effort will be stopped and spill contained.
- All stud holes will be covered with the plastic covers and secured with the black tape. Should the liquid escape from the weep hole it will float toward the edge of the head and drip down on the floor surface. It is not likely that the liquid would continue its flow under the flange for approximately 30 inches to reach the gaskets
- The spray and drain process will be coordinated such that when the sill is noted the spraying operation is stopped immediately. Only small amount of water will be used at a time.

Another challenge of the cleaning effort will be associated with the protection of the CRDM motors. To prevent water damage to the motors the only area where water will be permitted and sprayed is located between the flange plain and the top of the insulation. The spray operator will be briefed about the need to control the spray and not to create any splashing. The operator will be briefed not to spray any water on the motor assemblies. Motor assemblies are sealed and are not easily impregnable with water.

ALARA considerations include time/distance principle. The cleaning effort will mainly consist of preparation work. The cleaning effort is scheduled to last approximately 4 hours. With majority of time devoted to the head area. The dose is significantly lower at the weep hole area in comparison with the ventilation duct openings area. Equipment operator will minimize stay time in the "shine" area while spraying. If feasible a mirror will be utilized to inspect the results of spray at the ventilation duct openings area. After initial cleaning a video inspection will be performed by the Framatome Technologies. Should additional cleaning be required the process will be repeated until most boric acid deposits are removed or as directed by HP.

<u>Work Order instructions.</u>

The following items are required for support of head cleaning effort.

Scaffolding- the scaffold is needed on the North side of the head. The scaffold is needed for wrapping the head with the plastic to block all weep holes. In addition to scaffolding a movable platform will be constructed to enable access to the Lexan covers.

Uncover the weep holes- this can be accomplished by partially rising the bottom portion the lead blankets presently installed on the head. All blankets will need to be raised since plastic tape will be strapped all around the head.

Cover the Reactor Head bolt holes- this can be accomplished by rising the plywood decking and covering the holes with plastic or wrap. Cover each hole separately by cutting square piece of plastic and tape it to the flange with the black tape. Reinstall the plywood flooring.

Remove all Lexan covers- Lexan covers are bolted to the ventilation duct openings. The Lexan material is fragile. Special care should be taken during removal and re-installation not to chip any corners and not to overtorque the bolts. This will result in cracks, and covers will have to be replaced. As a precaution, more Lexan sheet material should be ordered in the event that replacement covers are needed. Verify Lexan sheets are available in stores. Materials required to perform the work are: plastic, tarpaulin, black tape, and stainless steel hooks for rising the lead shielding.

NRC FORM 386
(9-1995)

U.S. NUCLEAR REGULATORY COMMISSION

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ENCLOSURE 2

Docket No. 50-289

JAN 28 1972

Metropolitan Edison Company
ATTN: Mr. John G. Miller
Vice President
P. O. Box 542
Reading, Pennsylvania

Gentlemen:

As you know, an event occurred at a foreign pressurized water power reactor in which an unusual corrosion mechanism occurred when prolonged leakage of borated reactor coolant onto the reactor vessel head was undetected. Subsequent tests have indicated that this corrosion potential might exist under certain conditions when borated fluid has prolonged contact with carbon steel.

To preclude additional experiences of this type, an appropriate program of inservice inspection should be implemented to detect such effects at an early stage. The ASME Code Committee for Inservice Inspection is considering revision of the ASME Code for Inservice Inspection of Nuclear Reactors. However, as an interim measure, we believe that the inspection program described in the enclosure should be incorporated into your inservice inspection program.

Please advise us within 30 days concerning your adoption of the provisions of the enclosure.

Sincerely,

Original Signed By
R. C. DeYoung

R. C. DeYoung, Assistant Director
for Pressurized Water Reactors
Division of Reactor Licensing

Enclosure:
PWR Inservice Inspection Program

1486 329

cc:
See page 2

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DATE ▶	1/27/72	1/27/72	1/ /72			

Recommended PWR Inservice Inspection Program for Detection of Effects of Reactor Coolant Leakage

A. Inspection Requirements

(1) Prior to reactor startup following each refueling outage, all pressure-retaining components of the reactor coolant pressure boundary shall be visually examined for evidence of reactor coolan' leakage while the system is under a test pressure not less than the nominal system operating pressure at rated po.er.

This examination (which need not require removal of insulation) shall be performed by inspecting (a) the exposed surfaces and joints of in ulations, and (b) the floor areas (or equipment) directiy underneath these components.

At locations where reactor coolant leakage is normally expected and collected (e.g., valve stems, etc), the examination shall verify that the leakage collection system is operative and [illegible].

(2) During the conduct of the examinations of (1) above, particular attention shall be given to the insulated areas of components constructed of ferritic steels to detect evidence of boric acid residues resulting from reactor coolant leakage which might have accumulated during the service period preceding the refueling outage.

(3) The visual examinations of (1) and (2) above shall be conducted in conformance with the procedures of Article IS-211 of Section XI of the ASME Boiler and Pressure Vessel Code.

B. Corrective Measures

(1) The source of any reactor coolant leakage detected by the examinations of A(1) above shall be located by the removal of insulation where necessary and the following corrective measures applied:

(a) Normally expected leakage from component parts (e.g., valve stems) shall be minimized by appropriate repairs and maintenance procedures. Where such leakage may reach the surface of ferritic components of the reactor coolant pressure boundary, the leakage shall be suitably channeled for collection and disposal.

(b) Leakage from through-wall flaws in the pressure-retaining membrane of a component shall be eliminated, either by corrective repairs or by component replacement. Such repairs shall conform with the requirements of Article IS-400 of Section XI of the ASME Boiler and Pressure Vessel Code.

(2) In the event boric acid residues are detected by the examinations of A(2) above, insulation from ferritic steel components shall be removed to the extent necessary for examination of the component

surfaces wetted by reactor coolant leakage to detect evidence of corrosion.

The following corrective measures shall be applied:

(a) An evaluation of the effect of any corroded area upon the structural integrity of the component shall be performed in accordance with the provisions of Article IS-311 of Section XI Code.

(b) Repairs of corroded areas, if necessary, shall be performed in accordance with the procedures of Article IS-400 of Section XI Code.