From: John Zwolinski
To: Lawrence Burkhart — NRR
Date: 11/7/01 3:48PM
Subject: Re: LATEST DAVIS-BESSE ORDER

Get Davis Besse right.....Z

>>> Lawrence Burkhart 11/07/01 03:47PM >>>
Mitzi just told me that she has more comments. I will go to talk with her about these comments and make the changes. Also, the way the memo is written now, I attach both the Davis-Besse and D.C. Cook 2 order to the memo. Perhaps we should write 2 memos - one for Davis Besse and one for D.C. Cook 2, otherwise we need to wait until we get D.C. Cook 2 order finalized before we can put the memo package into concurrence.

>>> John Zwolinski 11/07/01 03:43PM >>>
Is there a copy going into concurrence....Z

>>> Lawrence Burkhart 11/07/01 12:22PM >>>
Attached is the latest order. Mitzi and I have incorporated Larry's and Allen's comments of this morning.

Mark is working on "tightening up" the RIS 2001-02 discussion in Section IV.

Larry.

D-39

1 November 7, 2001 (12:10PM)

UNITED STATES
NUCLEAR REGULATORY COMMISSION

In the Matter of)
) Docket No. 50-346

Davis-Besse Nuclear Power Station, Unit No. 1) EA_____

ORDER MODIFYING LICENSE
(EFFECTIVE IMMEDIATELY)

I

FirstEnergy Nuclear Operating Company (the licensee) is the holder of Facility Operating License No. NPF-3 issued by the Nuclear Regulatory Commission (NRC or Commission) pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50 on April 22, 1977. The license authorizes the operation of Davis-Besse Nuclear Power Station, Unit No. 1, in accordance with conditions specified therein. The facility is located on the licensee's site in Oak Harbor, Ohio.

II

On February 18, 2001, Duke Energy Corporation (Duke), the licensee for Oconee Nuclear Station Unit 3, performed a VT-2 visual examination of the outer surface of the unit's reactor pressure vessel head to inspect for indications of borated water leakage. This reactor pressure vessel head inspection was performed as part of a normal surveillance during a planned maintenance outage. The VT-2 visual examination revealed the presence of small amounts of boric acid residue in the vicinity of nine of the 69 control rod drive mechanisms. Subsequent nondestructive examinations identified 47 recordable crack indications in these nine degraded control rod drive mechanism nozzles. The licensee initially characterized these flaws as either axial or below-the-weld circumferential indications, and initiated repairs of the degraded areas.

As part of American Society of Mechanical Engineers (ASME) Code Section XI repair activities of the affected control rod drive mechanism nozzles, Duke implemented required dye-penetrant testing and detected the presence of additional indications in two of the nine degraded penetration nozzles. While implementing the excavations and repairs of these two nozzles, Duke found that each nozzle had a circumferential crack that extended approximately 165° around the nozzle above the weld, i.e., at a location that is part of the reactor pressure vessel pressure boundary. Further investigations and metallurgical examinations revealed the cause to be primary water stress corrosion cracking initiated from the outside diameter of the control rod drive mechanism penetration nozzles. These circumferential portions of the cracks followed the weld profile contour. Reexamination of ultrasonic inspection records revealed a part-through-wall circumferential crack in a third nozzle, which was repaired along with the other two nozzles. This ultrasonic inspection reexamination also determined that the two longer cracks were evident but one was indicated to be substantially shorter (59°) than the destructive examination determined it to be.

Other vessel head penetration nozzle cracking and leakage observed at approximately the same time as the circumferential cracking at Oconee Nuclear Station, Unit 3 include:

- Oconee Nuclear Station, Unit 1, axial cracking in November 2000,
- Arkansas Nuclear One, Unit 1, axial cracking in February 2001,
- Oconee Nuclear Station, Unit 2, circumferential cracking in April 2001,

The identification of circumferential cracking in control rod drive mechanism nozzles at Oconee Nuclear Station, Units 2 and 3, is significant in that it represents the first reported occurrences of circumferential cracking in the control rod drive mechanism nozzles of U. S. pressurized water reactors. These occurrences of circumferential cracking along with the recently identified cracking in the nozzles and J-groove welds at the above-listed plants raised concerns about a potentially risk-significant condition affecting some domestic pressurized water reactors. The level of cracking of vessel head penetration nozzles that has been found and that may exist undetected at other facilities, if left undetected and uncorrected in a prompt manner, could result in a gross failure of the reactor coolant pressure boundary in the form of a vessel head penetration nozzle failure, and consequently a loss-of-coolant accident. Such a failure would result in a significant decrease in the assurance of adequate protection of the public health and safety.

It is also significant that the full extent of the circumferential cracks was determined at Oconee Nuclear Station, Units 2 and 3, only during the repair process and through a destructive examination process. Although the normal and expanded ASME Code required inspection efforts to monitor for additional signs of degradation (e.g., bare metal examinations) did reveal the evidence of leakage from the vessel head penetration nozzles, these efforts were not capable of indicating the extent of the circumferential cracking that was occurring in the nozzles. Additionally, calculations of the reactor coolant leakage rate from the vessel head penetration nozzles at the Oconee, Unit 3, indicate that the leakage occurs at very low rates (i.e., less than 1 gallon per year), and leakage rates of this magnitude are not high enough to allow for detection using typical instrumentation designed for the purpose of detecting reactor coolant pressure boundary leakage.

The current method for managing primary water stress corrosion cracking in the vessel head penetration nozzles of U.S. pressurized water reactors is dependent on the implementation of inspection methods intended to provide early detection of degradation of the reactor coolant pressure boundary. Section (g)(4) of 10 CFR 50.55a requires, in part, that ASME Code Class 1, 2, and 3 components must meet the inservice inspection requirements of Section XI of the ASME Boiler and Pressure Vessel Code throughout the service life of a boiling or pressurized water reactor. Pursuant to Inspection Category B-P of Table IWB-2500-1 to Section XI of the ASME Boiler and Pressure Vessel Code, licensees are required to perform VT-2 visual examinations of their vessel head penetration nozzles and reactor vessel heads once every refueling outage for the system leak tests, and once an inspection interval for the hydrostatic pressure test.

Based on the experience with this cracking phenomenon, the VT-2 visual examination methods used on the vessel head nozzles in accordance with Inspection Category B-P of Table IWB-2500-1 to Section XI of the ASME Boiler and Pressure Vessel Code do not provide reasonable assurance that leakage from a through-wall flaw in a nozzle will be detected. The VT-2 visual examination methods specified by the ASME Code are not directed at detecting the very small amounts of boric acid deposits, e.g., on the order of a few grams, that have been associated with vessel head penetration nozzle leaks in operating plants. In addition, the location of thermal insulating materials and physical obstructions may limit the capability of VT-2 visual examination methods to identify minute amounts of boric acid deposits on the outer surface of the vessel head. Paragraph IWA-5242 of Section XI of the ASME Boiler and Pressure Vessel Code does not require licensees to remove thermal insulation materials when performing ASME VT-2 visual examinations of their reactor vessel heads. Cleanliness of reactor vessel heads during the examinations which is critical for visual examination methods to be capable of distinguishing between boric acid residues that result from vessel head penetration nozzle leaks and those residues that result from leaks in other reactor coolant system components is not addressed by the ASME Code. Finally, the ASME Code, as referenced in 10 CFR 50.55a, does not require surface or volumetric examinations to detect cracking in vessel head penetration nozzles.

The characteristics of this type of leakage coupled with the inadequate aspects of the existing ASME Code inspection requirements reinforces the importance of performing a prompt examination of the upper pressure vessel head area using examination techniques that are capable of detecting cracking in the vessel head penetration nozzles and their associated J-groove welds and heat-affected zones.

To address the generic safety implications of the pressure boundary leakage resulting from the extent of vessel head penetration nozzle cracking and leakage observed at Oconee Nuclear Station, Units 1, 2, and 3, and Arkansas Nuclear One, Unit 1, the NRC issued Bulletin 2001-01, "Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles," on August 3, 2001. In the Bulletin, the susceptibility of pressurized water reactors to cracking of the vessel head penetration nozzles was categorized into four populations based on the susceptibility rankings established by the industry and documented in Appendix B to MRP-44, Part 2, entitled "PWR Materials Reliability Program, Interim Alloy 600 Safety Assessments for US PWR Plants (MRP-44): Part 2: Reactor Vessel Top Head Penetrations," and dated May 2001. For the population of plants considered as having a high susceptibility to primary water stress corrosion cracking based upon a susceptibility ranking of less than five effective full-power years from the Oconee Nuclear Station, Unit 3, condition (which includes Davis-Besse Nuclear Power Station, Unit No. 1), the staff stated that the possibility for leaks to occur from a vessel head penetration nozzle at one of these facilities would dictate the need to use a qualified visual examination that would be capable of reliably detecting and accurately characterizing leakage from through-wall cracks in all of the vessel head penetration nozzles. The staff concluded that the qualified visual examination methods should have the following characteristics: (1) a plant-specific demonstration that any vessel head penetration nozzle exhibiting through-wall cracking would be capable of providing a sufficient leakage path to the reactor pressure vessel head surface (based on the as-built configuration of the vessel head penetrations); and (2) the effectiveness of the qualified visual examination should not be compromised by the presence of insulation, existing deposits on the reactor pressure vessel head, or other factors that could interfere with the detection of leakage. Absent the use of a qualified visual examination, the staff noted in the Bulletin that a qualified volumetric examination of 100 percent of the vessel head penetration nozzles (with a demonstrated capability to reliably detect cracking on the outside diameter of a vessel head penetration nozzle) would be appropriate to provide evidence of the structural integrity of the vessel head penetration nozzles. Performance of the recommended examinations of all vessel head penetration nozzles is expected to provide reasonable assurance that a crack of significant size does not exist.

To assess the prevalence and severity of vessel head penetration cracking and determine plant-specific compliance with NRC regulations, the staff requested that addressees of the Bulletin submit information regarding the scope, timing, and results of completed inspections and the scope and schedule of future inspections of their vessel head penetration nozzles. The bulletin requested that licensees not planning to perform inspections prior to December 31, 2001, provide the technical basis for their planned inspection schedules. At the time of issuance of the Bulletin, the staff considered that performance of the recommended inspections by December 31, 2001, was a timely action given the very limited experience and observations regarding this cracking phenomenon and the associated safety implications. December 31, 2001, was chosen based on the need to acquire additional information in a timely manner and to allow licensees time to plan and perform the recommended inspections, and to make any needed repairs.

Since issuance of the Bulletin, additional facilities have identified vessel head penetration nozzle cracking including:

- North Anna, Unit 1, nine nozzles identified with cracking in September 2001,
- Crystal River, Unit 3, circumferential cracking in October 2001,
- Three Mile Island, Unit 1, seven nozzles identified with cracking (five nozzles to be repaired) in October 2001,
- Surry Power Station, Unit 1, ten nozzles identified with cracking (five nozzles to be repaired) in October 2001 (total number and orientation of cracks still under investigation), and
- North Anna, Unit 2, five nozzles identified for additional examination in October 2001.

The results of these inspections have not revealed conditions of incipient failure. However, considering the uncertainties and variability in plant susceptibilities, the inspections have identified conditions supporting the need to perform inspections in the near term to verify the absence of conditions worse than those found to date. In light of these results, the staff believes that operation, of facilities considered to be highly susceptible to this cracking phenomenon, beyond December 31, 2001, is unacceptable unless the recommended inspections to identify this potentially hazardous condition are completed and found acceptable by the staff.

III

By letter dated September 4, 2001, as supplemented by letters dated October 17, October 30, and November 1, 2001, the licensee submitted its responses to Bulletin 2001-01 for Davis-Besse Nuclear Power Station, Unit No. 1, that documented the "high susceptibility" ranking of Davis-Besse Nuclear Power Station, Unit No. 1. The licensee also described its intention to perform the recommended inspection, including a qualified visual examination of all of the vessel head penetration nozzles, in April 2002, with the licensee subsequently indicating a plan to shutdown by the end of March 2002. The licensee's bulletin responses also provided information regarding licensee's basis for deferring the recommended inspections beyond December 31, 2001.

As a part of its basis for delaying the recommended inspection beyond December 31, 2001, the licensee cited a history of reactor vessel head visual examinations at the Davis-Besse Nuclear Power Station, Unit No. 1 plant using a remote camera in Spring 2000, Spring 1998, and Spring 1996. Davis-Besse Nuclear Power Station, Unit No. 1 has a total of 69 CRDMs. In 1996, 94 percent of the nozzles (i.e., 65) were visually examined (four were not examined). In 1998, 72 percent of the nozzles (i.e., 50) were visually examined (19 were not examined), and in 2000, 65 percent of the nozzles (i.e., 45) were visually examined (24 were not examined). As a consequence, 24 nozzles have not been inspected since 1998, 19 nozzles have not been inspected since 1996, and 4 nozzles have never been inspected. In its Bulletin response, and supplemental information provided by letter dated October 30, 2001, the licensee stated that the nozzles that were not examined in the recent examinations were obscured by boric acid leakage from other sources, such as control rod drive mechanism motor tube flanges. In addition, the licensee stated that for the four nozzles not examined in 1996, it could not demonstrate the presence of a gap between the nozzles and the reactor pressure vessel head, which is not consistent with one of the characteristics of a qualified visual examination identified in the Bulletin.

As documented in the foregoing, the licensee has not, at any time, performed a qualified visual examination of 100 percent of the nozzles at Davis-Besse Nuclear Power Station, Unit

No. 1. In addition, based on information provided by the licensee, the visual inspections that were performed did not utilize lights and inspection angles that would ensure the likelihood of detection of the very small amount of boric acid deposits associated with vessel head penetration leakage. Consequently, these inspections provide little or no insight in evaluating the condition of the vessel held penetration nozzles weld in 1996, 1998, or 2000.

As stated previously, Davis-Besse Nuclear Power Station, Unit No. 1, is ranked in a population of thirteen plants that have high susceptibility or have previously identified leakage or cracking in their vessel head penetration nozzles. As indicated from the recent operating experience described previously, nine out of ten of the plants in the same population as Davis-Besse Nuclear Power Station, Unit No. 1 that have performed recent inspections have found evidence of cracking in the vessel head penetration nozzles. The tenth plant identified no leakage or cracking after inspection, and the remaining two plants have near-term plans to inspect their nozzles [can it be distinguished from D-B???].

The nuclear steam system supply vendor for the Davis-Besse Nuclear Power Station, Unit No. 1, plant is Babcock & Wilcox. For the population of seven plants designed by Babcock & Wilcox, six have performed recent examination of their nozzles. All six of the plants have identified leaking and cracked nozzles. In addition, three out of the six units have identified circumferential cracking. Davis-Besse Nuclear Power Station, Unit No. 1, is the only Babcock & Wilcox plant that has not performed a recent visual examination of 100 percent of their nozzles.

Since Davis-Besse Nuclear Power Station, Unit No. 1, is in the population of highly susceptible plants that have found vessel head penetration nozzle cracking, and in some cases the cracking has been significant, i.e., circumferential, it is reasonable to expect that Davis-Besse Nuclear Power Station, Unit No. 1, could have significant cracking, which would violate the reactor coolant pressure boundary integrity.

Furthermore, Technical Specification, Section 3/4.4.6, which is part of the Davis-Besse Nuclear Power Station, Unit No. 1 operating license, addresses violations of reactor coolant pressure boundary integrity by requiring the plant to shutdown. The technical specification bases specifically state that "PRESSURE BOUNDARY LEAKAGE of any magnitude is unacceptable since it may be indicative of an impending gross failure of the pressure boundary." Given that all of the other Babcock & Wilcox-designed plants have identified reactor coolant pressure boundary leakage, it is highly probable that Davis-Besse Nuclear Power Station, Unit No. 1, is also currently experiencing this leakage and is operating in violation of its technical specifications.

Based on the information provided by the licensee and the extent of vessel head penetration cracking and leakage found at multiple facilities, I find that the licensee has not provided an adequate basis to operate beyond December 31, 2001, without performing inspections to verify the integrity of the reactor coolant pressure boundary at Davis-Besse Nuclear Power Station, Unit No. 1. Performance of the recommended inspections prior to operation beyond December 31, 2001, is timely and necessary given Davis-Bessie's high-susceptibility ranking and the extent of cracking and leakage found at other similarly-designed facilities. Consequently, I find that a potentially hazardous condition exists and warrants the issuance of an Order that modifies the operating license for Davis-Besse Nuclear Power Station, Unit No. 1, to require that (1) the facility be shut down by December 31, 2001, and proceed to the cold shutdown or lower Mode, (2) the licensee perform inspections to demonstrate to the NRC that there is reasonable assurance that the vessel head penetration nozzles are free of defects prior to subsequent plant operation.

IV

In addition to cracking phenomena observed at other facilities, the risk implications associated with vessel head penetration nozzle cracking and leakage warrant issuance of this order. Regulatory Issue Summary 2001-02, "Guidance on Risk-Informed Decisionmaking in License Amendment Reviews," dated January 18, 2001, provides a process for the staff to consider whether a "special circumstance" exists which may rebut the presumption that compliance with the regulations provides adequate protection of public health and safety. Although developed as a tool for staff reviews of license amendment requests, the process in Regulatory Issue Summary 2001-02 is appropriate for other regulatory decisionmaking purposes because it addresses the fundamental requirement for operation of a nuclear reactor: there is reasonable assurance of adequate protection for the public health and safety.

A special circumstance is present because compliance with 10 CFR 50.55a inservice inspection requirements for inspection of vessel heads (i.e., pursuant to Category B-P to Table IWB-2500-1 of Section XI, ASME Boiler and Pressure Vessel Code) is not adequate to detect degradation in the nozzles and protect against a loss-of-coolant accident and assure the structural integrity of the vessel head penetration nozzles. Failure of the regulations to require adequate monitoring for degradation in the vessel head penetration nozzles which could lead to a vessel head penetration nozzle failure, and consequently a loss-of-coolant accident, constitutes a risk factor not addressed by the regulations. Given that ASME Code requirements are not adequate to detect degradation in the nozzles, the licensee's reactor vessel head inspections, described in Section III, above, did not ameliorate the above deficiencies in the ASME Code inspection requirements. Thus, consistent with the Regulatory Issue Summary 2001-02 process, a special circumstance exists for the Davis-Besse Nuclear Power Station, Unit No. 1.

Applying the risk-informed decisionmaking process described in Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," a special circumstance is acceptable if (1) it meets current regulations, (2) it is consistent with "defense-in-depth philosophy," (3) it maintains sufficient safety margin, (4) it results in only a small increase in core damage frequency, and (5) the basis for the risk estimate is monitored using performance measurement strategies. Given that inspections that have been performed at Davis-Besse Nuclear Power Station, Unit No. 1, have met the requirements of 10 CFR 50.55a, the first principle is satisfied. However, as noted above, compliance with the regulations may not be adequate to prevent the failure of the reactor coolant pressure boundary, one of the three barriers to release of radioactive materials from the reactor core, and thus the second principle regarding the "defense-in-depth" philosophy is not satisfied. Compliance with the ASME Code, Section XI, inservice inspection requirements fails to satisfy the third principle of maintaining safety margins since it cannot be assured that pressure boundary leakage would be detected prior to a gross failure of a vessel head penetration nozzle.

The fourth principle is not met because the core damage frequency could eventually approach the relatively high numerical value of the conditional core damage probability for the loss-of-coolant accident that would result from a control rod drive mechanism nozzle failure. Based on the licensee's submittal dated November 1, 2001, the conditional core damage probability value is 2.7E-3 for a control rod drive mechanism nozzle failure that produces a medium break loss-of-coolant accident. Based on the individual plant examination (IPE) submittal, dated February 26, 1993, Davis-Besse Nuclear Power Station, Unit No. 1, has a baseline core damage frequency of 6.6E-5 per reactor year.

To fall below the Regulatory Guide 1.174 guidelines of a core damage frequency increase (i.e., change in core damage frequency) of less than 1E-5 per reactor-year for a plant that has a baseline core damage frequency of less than 1E-4 per reactor-year, the initiating event frequency for a vessel head penetration nozzle failure at Davis-Besse Nuclear Power Station, Unit No. 1, would have to be demonstrated to be below 3.7E-3 per reactor year.

Based on the inadequacy of ASME code required inspections and the extent of vessel head penetration nozzle cracking and leakage found at multiple facilities, it cannot be concluded without conducting inspections capable of identifying vessel head penetration nozzle cracking that the initiating event frequency, i.e., vessel head penetration nozzle failure, will remain sufficiently low to satisfy Regulatory Guide 1.174 guidelines for Davis-Besse Nuclear Power Station, Unit No. 1.

Finally, the fifth principle is not satisfied because the basis for any licensee analysis that shows risk levels below Regulatory Guide 1.174 numerical guidelines must be based on assumptions that cannot be verified without performing inspections that are capable of detecting the form of degradation being modeled.

In summary, this "special circumstance" does not satisfy four of the five safety principles, and therefore, the assumption that compliance with the regulations is sufficient to provide reasonable assurance of adequate protection of public health and safety is not valid.

The final step for application of the Regulatory Issue Summary 2001-02 process involves identification of an adequate basis for establishing reasonable assurance of adequate protection when the "special circumstance" is considered. The Commission has established General Design Criteria (GDC) for the design, fabrication, construction, testing, and performance of structures, systems and components important to safety in Appendix A to 10 CFR Part 50, that identify features necessary for adequate protection. Three GDC are relevant to this issue. Criterion 14 states that "[t]he reactor coolant pressure boundary shall be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage or rapidly propagating failure, and of gross rupture." Criterion 30 states that "[m]eans shall be provided for detecting and, to the extent practical, identifying the location of the source of reactor coolant leakage." Criterion 32 states, in part, that "components of the reactor coolant pressure boundary shall be designed to permit . . . periodic inspection and testing of important areas and features to assess their structural integrity and leak-tight integrity." Taken as a whole, these GDC emphasize that the Commission considers that it is extremely important from a safety standpoint to maintain the reactor coolant pressure boundary in a leaktight and structurally sound condition, with an extremely low probability of gross failure. These GDC are consistent with the requirements of Technical Specification, Section 3/4.4.6, that does not allow continued operation with any pressure boundary leakage, and the intent of the inservice inspection requirements of 10 CFR 50.55a(g)(4).

Failure of the licensee for Davis-Besse Nuclear Power Station, Unit No. 1, to conduct inspections of the reactor vessel head penetration nozzles in a manner that is sufficient to detect the extent of degradation caused by a mechanism known to be degrading other similar plants in that portion of the vessel and prior to a significant reduction in safety margin is inconsistent with these general design criteria. The level of degradation that has been found in other similar plants, if left undetected and uncorrected, could result in a gross failure of the reactor coolant pressure boundary (loss-of-coolant accident).

In summary, compliance with the ASME Code requirements specified in 10 CFR 50.55a(g)(4) is not considered adequate to detect cracking and prevent failure of the vessel head penetration nozzles for pressurized water reactors, and the licensee has not conducted additional inspections that would ameliorate this situation. This situation constitutes a special circumstance, the potential consequence of which is the loss of the reactor coolant pressure boundary, one of the "defense-in-depth" barriers, and the potential for the plant's core damage frequency to rise to a value approaching the conditional core damage probability of a loss-of-coolant accident, constituting an undue risk to public health and safety. Therefore, I do not have reasonable assurance that adequate protection will be maintained without performance of timely inspections that are sufficient to detect this type of degradation.

V

Based on the above, I have determined, based on a potentially hazardous condition, that the integrity of the reactor coolant pressure boundary may not be maintained at the Davis-Besse Nuclear Power Station, Unit No. 1. Accordingly, pursuant to 10 CFR 2.202, I find it necessary to require the licensee to shutdown the facility by December 31, 2001, and to demonstrate, by inspection, that the vessel head penetration nozzles are free of defects. These inspections shall be capable of detecting vessel head penetration nozzle degradation or leakage to determine the extent of condition and identify nozzles requiring repair to assure that reactor coolant pressure boundary integrity is maintained and to provide reasonable assurance of adequate protection of the health and safety of the public. Also, based on the above, I further find that the assurance of the public health and safety requires that this Order be effective immediately.

Accordingly, pursuant to Sections 103, 161b, 161i, 161o, 182 and 187 of the Atomic Energy Act of 1954, as amended, and the Commission's regulations in 10 CFR 2.202 and 10 CFR Part 50, IT IS HEREBY ORDERED, EFFECTIVE IMMEDIATELY, THAT LICENSE NO. NPF-3 IS MODIFIED AS FOLLOWS:

1. The plant shall be shutdown no later than December 31, 2001, and proceed promptly to

the cold shutdown or lower Mode of operation.

2. A demonstration to the NRC that there is reasonable assurance that the vessel head penetration nozzles at Davis-Besse Nuclear Power Station, Unit No. 1, are free of defects that exceed the requirements of the ASME Code is required prior to subsequent power operation. This demonstration shall include the performance of a qualified visual examination of 100 percent of the vessel head penetration nozzles as recommended in NRC Bulletin 2001-01 for the population of plants considered to have a high susceptibility to primary water stress corrosion cracking. This qualified visual examination must be able to reliably detect and accurately characterize leakage from cracking in vessel head penetration nozzles considering two characteristics. One characteristic is a plant-specific demonstration that a flow path exists between the nozzles and the reactor pressure vessel head to the exterior of the reactor pressure vessel head such that any vessel head penetration nozzle exhibiting through-wall cracking will provide sufficient leakage to the reactor pressure vessel head surface. The second characteristic that must be considered is to ensure the effectiveness of the qualified visual examination is not compromised by the presence of insulation, existing deposits on the reactor pressure vessel head, or other factors that could interfere with the detection of leakage. Absent the use of a qualified visual examination, a qualified surface or volumetric examination of 100 percent of the vessel head penetration nozzles (with a demonstrated capability to reliably detect cracking on the outside diameter of a vessel head penetration nozzle) is required to provide evidence of the structural integrity of the vessel head penetration nozzles.

3. Operation in Modes higher than the cold shutdown Mode is not authorized until the staff determines that the examination scope and results are acceptable.

The Director of the Office of Nuclear Reactor Regulation, may relax or rescind, in writing, any of the above conditions upon a showing by the licensee of good cause.

VI

In accordance with 10 CFR 2.202, the licensee must, and any other person adversely affected by this Order may, submit an answer to this Order, and may request a hearing on this Order, within 20 days of the date of this Order. Where good cause is shown, consideration will be given to extending the time to request a hearing. A request for extension of time must be made in writing to the Director, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Washington, DC 20555, and include a statement of good cause for the extension. The answer may consent to this Order. Unless the answer consents to this Order, the answer shall, in writing and under oath or affirmation, specifically admit or deny each allegation or charge made in this Order and set forth the matters of fact and law on which the licensee or other person adversely affected relies and the reasons as to why the Order should not have been issued. Any answer or request for a hearing shall be submitted to the Secretary, U.S. Nuclear Regulatory Commission, ATTN: Rulemakings and Adjudications Staff, Washington, DC 20555. Copies also shall be sent to the Director, Office of Nuclear Reactor Regulation, U. S. Nuclear Regulatory Commission, Washington, DC 20555, to the Director, Office of Enforcement, and the Assistant General Counsel for Materials Litigation and Enforcement at the same address, and to the licensee if the answer or hearing request is by a person other than the licensee. If a person other than the licensee requests a hearing, that person shall set forth with particularity the manner in which his interest is adversely affected by this Order and shall address the criteria set forth in 10 CFR 2.714(d).

If a hearing is requested by the licensee or a person whose interest is adversely affected, the Commission will issue an Order designating the time and place of any hearing. If a hearing is held, the issue to be considered at such hearing shall be whether this Order should be sustained.

Pursuant to 10 CFR 2.202(c)(2)(i), the licensee may, in addition to demanding a hearing at the time the answer is filed or sooner, move the presiding officer to set aside the immediate effectiveness of the Order on the ground that the Order, including the need for immediate effectiveness, is not based on adequate evidence but on mere suspicion, unfounded allegations, or error.

In the absence of any request for hearing, or written approval of an extension of time in which to request a hearing, the provisions specified in Section IV above shall be final 20 days from the date of this Order without further order or proceedings. If an extension of time for requesting a hearing has been approved, the provisions specified in Section V shall be final when the extension expires if a hearing request has not been received. AN ANSWER OR A REQUEST FOR HEARING SHALL NOT STAY THE IMMEDIATE EFFECTIVENESS OF THIS ORDER.

FOR THE NUCLEAR REGULATORY COMMISSION

Samuel J. Collins, Director
Office of Nuclear Reactor Regulation

Dated this day of November 2001