From: Lawrence Burkhart
To: Allen Hiser; Andrea Lee; Bill Bateman; Jack Strosnider; Keith Wichman } NRR
Date: 11/2/01 2:55PM
Subject: DAVIS-BESSE ORDER

I honestly don't know what the status is regarding our intentions wrt the orders but I was directed by Brian to draft the order (as far as poosible) for Davis-Besse and give him a copy by 4:00 p.m. (which he can then give to the EDO's office for review over the weekend).

Please see the attached. I have highlighted (in CAPITAL LETTERS) where I think "beefing up" from EMCB may help. I have attempted "beefing up" in some places. I request that you (EMCB) review and make sure my comments are OK and/or if more is needed (if so, please provide).

CC: Anthony Mendiola; Douglas Pickett; F. Mark Reinhart; Farouk Eltawila; Jacob Zimmerman; John Zwolinski; Richard Barrett; Stephen Sands

D-27

UNITED STATES
NUCLEAR REGULATORY COMMISSION

In the Matter of)
) Docket No. 50-346
FirstEnergy Nuclear Operating Company) License No. NPF-3
Davis-Besse Nuclear Power Power Station, Unit No. 1) EA_____

ORDER MODIFYING LICENSE
(EFFECTIVE IMMEDIATLEY)

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 - Safety Issue

On February 18, 2001, with Oconee Nuclear Station, Unit 3, in Operating Mode 5, Duke Energy Corporation, the licensee for Oconee, 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.

Upon commencement of required American Society of Mechanical Engineers (ASME) Code Section XI repair activities of the affected control rod drive mechanism nozzles, Duke Energy Corporation 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 flawed areas, Duke Energy Corporation identified that the flaw indications (cracks) in each nozzle were significantly larger than originally detected by prior dye penetrant test examinations. In addition, it was determined that several flaw indications were circumferential in orientation and grew into the nozzle (e.g., from the outside diameter to the inside diameter) just above the root of the J-groove weld. Further investigations and metallurgical examinations revealed the cause to be primary water stress corrosion cracking and that these cracks had initiated from the outside diameter of the control rod drive mechanism penetration nozzles. The circumferential crack in the No. 56 control rod drive mechanism nozzle was through-wall and the circumferential crack was 165° in length. Also, the No. 50 nozzle had pinhole through-wall indications. These circumferential portions of the cracks followed the weld profile contour. Subsequent reexamination of inspection records has revealed a part-through-wall circumferential crack in the No. 23 nozzle, which was repaired along with the Nos. 50 and 56 nozzles.

Additional vessel head penetration nozzle cracking was discovered at Oconee Nuclear Station, Unit 1 (axial cracking) in November 2000; Arkansas Nuclear One, Unit 1 (axial cracking) in February 2001; and Oconee Nuclear Station , Unit 2 (circumferential cracking) in April 2001. (Subsequently in October 2001, circumferential cracking has been identified in one control rod drive mechanism nozzle at Crystal River, Unit 3, and deposits indicative of nozzle leakage have been identified near two nozzles at Three Mile Island, Unit 1.)

The identification of circumferential cracking in control rod drive mechanism nozzles at Oconee Nuclear Station, Units 2 and 3, is significant in that they represent 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 axial cracking in the J-groove welds at Oconee Nuclear Station, Units 1, 2, and 3, and Arkansas Nuclear One, Unit 1, have raised concerns about a potentially risk-significant generic condition affecting all 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. Consequently, it is reasonable to assume that this could result in a significant decrease in the assurance of adequate protection of the public health and safety.

The manner in which the circumferential cracks were detected is also significant in that they were detected only during the repair process. Although the normal inspection efforts and expanded 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, they were not capable of indicating the presence of the circumferential cracking that was occurring in the nozzles. The ASME Section XI inspection methods were and are inadequate in detecting degradation in control rod drive mechanism nozzle-to-reactor pressure vessel head welds. This reinforces the importance of performing a prompt and effective examination of the upper pressure vessel head area using nondestructive examination techniques that are capable of detecting recordable flaw indications 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 observed, the NRC issued Bulletin 2001-01, "Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles," on August 3, 2001. In the Bulletin, the staff stated that, as a result of its review of the susceptibility rankings established by the industry and documented in Appendix B to MRP-44, Part 2, the susceptibility of pressurized water reactors to cracking of the vessel head penetration nozzles could be categorized into four populations. 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 5 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 be characterized by the following aspects: (1) that, as a result of a plant-specific demonstration, 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) that 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 stated 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. It is the staff's judgement that performance of the recommended examinations of all vessel head penetration nozzles gives adequate assurance that a significantly-sized crack does not exist and, therefore, a reduced probability of a vessel head penetration nozzle failure. (DO WE NEED MORE THAN THE PREVIOUS SENTENCE - IS SO PROVIDE DISCUSSION OF WHY THE SCOPE OF INSPECTION REDUCES RISK OR PROVIDES A GREATER ASSURANCE OF THE ADEQUATE PROTECTION.)

This inspection would also need to be conducted in a prompt manner to preclude any potential undue risk to public health and safety. 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. December 31, 2001, was chosen based on the timing of the issuance of the Bulletin as well as on the logistics of lining up resources to perform the recommended inspections.

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.

By letter dated September 4, 2001, as supplemented by letter dated October 17, 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. The licensee also provided information regarding the basis for deferring the recommended inspections beyond December 31, 2001.

(DISCUSSION WHY INSPECTION MUST BE COMPLETED NOW - MY ATTEMPT FOLLOWS BUT NEEDS MORE)

Since issuance of the Bulletin, the staff has continued to assess the technical aspects of this cracking phenomenon and the resulting potential safety impacts. In addition, many plants have performed the recommended inspections and some have found cracking in the vessel head penetration nozzles. Using the limited amount of information gained from these inspections and the knowledge of cracking phenomena and risk implications, the staff has documented its findings and assessment of this issue in its "Safety Assessment Title," dated November xx, 2001 (Agencywide Documents Access and Management System Accession No. ML013xxxxxxx). In its assessment, the staff found that ....INSERT INFO RE: EXISTING CRACK SIZE/GROWTH RATE/RISK OF FAILURE. INCLUDE INFO RE: WHY CRACK PROPAGATES FROM NO DETECTABLE LEAKAGE TO FAILURE SO QUICKLY.

In addition the staff found that the recommended inspections should be completed immediately to assure adequate protection of the public health and safety. Delay in performance of these examinations presents an undue risk to public health and safety because it is reasonable to assume that a significantly-sized crack may exist and result in a nozzle failure and small-break loss-of-coolant accident.

Furthermore, the Nuclear Steam System Supply vendor for the Davis-Besse Nuclear Power Station, Unit No. 1 plant is Babcock and Wilcox. This may be significant with respect to the susceptibility to vessel head penetration nozzle cracking because all of the other plants that exhibit this commonality have observed some form of vessel head penetration nozzle cracking including circumferential cracking at Oconee Power Station, Units 2 and 3 and Crystal River, Unit 3.

Based on the information provided by the licensee and other relevant information available to the NRC, the staff finds that the licensee has not established a defensibly low initiating event frequency and core damage frequency for control rod drive mechanism nozzle failures, and, consequently, there is a significant probability of a control rod drive mechanism nozzle failure and related safety implications at this time. Therefore, the NRC staff finds that the proposed schedule of the recommended inspections for Davis-Besse Nuclear Power Station, Unit No. 1 does not provide assurance of adequate of public health and safety.

III

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 for detecting defects prior to a failure of a facility's vessel head penetration nozzle. 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 current data supplied by the industry to date, the staff cannot be assured that VT-2 visual examination methods used on the upper vessel heads in accordance with Inspection Category B-P of Table IWB-2500-1 to Section XI of the ASME Boiler and Pressure Vessel Code are capable of detecting leakage from a through-wall flaw in the nozzles or their adjacent J-groove welds. Additionally, leak rate calculations of the reactor coolant from leaking vessel head penetration nozzles at the Oconee Nuclear Station, Unit 3, demonstrate that the leaks may occur, and in all probability do occur, at very slow rates (i.e., less than 1 gallon per year), and leakage rates of this magnitude may not be high enough to allow for detectable indication of the leakage using typical instrumentation designed for the purpose of detecting reactor coolant pressure boundary leakage. 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. (Pursuant to Paragraph IWA-5242 of Section XI of the ASME Boiler and Pressure Vessel Code, the 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 is also a critical aspect, as it is important for visual examination methods to be capable of distinguishing between boric acid residues that result from vessel head penetration nozzle leakage and those residues that result from leaks in other reactor coolant system components.

Compliance with the inspection requirements of 10 CFR 50.55a is no longer considered adequate to detect and prevent potential cracking and failure of the vessel head penetration nozzles for pressurized water reactor-designed reactors. This situation constitutes a "special circumstance" in that compliance with the Commission's regulations does not provide assurance of adequate protection of the public health and safety and, therefore, undue risk exists. Regulatory Information 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" rebuts the presumption that compliance with the regulations provides adequate protection of public health and safety. Although developed for staff reviews of license amendment requests, the process in Regulatory Information Summary 2001-02 is appropriate for other regulatory decisionmaking purposes because it addresses the fundamental requirement for operation of a nuclear reactor: that there is reasonable assurance of adequate protection for the public health and safety.

Application of the Regulatory Information Summary 2001-02 process to this issue has three steps:

1. identification of a "special circumstance" involving a risk factor not addressed by regulations;
2. assessment of the factor with respect to the five safety principles of risk-informed

decisionmaking to establish whether its effect is sufficiently large to rebut the assumption that adequate protection is achieved by compliance with existing regulations; and

3. identification of an adequate basis for establishing reasonable assurance of adequate protection when the factor is considered.

A special circumstance is present because 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) are not adequate to assure the structural integrity of the vessel head penetration nozzles in that the specified examination method is not capable of detecting cracking in vessel head penetration nozzles. The Code requirements are inadequate to monitor for degradation in the vessel head penetration nozzles prior to leakage from the nozzles and possibly prior to a vessel head penetration nozzle failure, and consequently a loss-of-coolant accident scenario. This is contrary to the statement in the Preface to Section XI that states "The rules . . [of Section XI] . . require a mandatory program of examinations, testing and inspections to evidence adequate safety . . [of a nuclear power plant]." Thus, a "special circumstance" exists with respect to this issue, as the regulations specify compliance with ASME Code requirements that are not adequate to detect degradation in the nozzles and protect against a loss-of-coolant accident. This satisfies step one in the Regulatory Information Summary 2001-02 process.

Additionally, only one of the five safety principles in the integrated 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," is met. Applying this risk-informed process, a 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 the inspections being performed meet the requirements of 10 CFR 50.55a, the first principle is satisfied. However, 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 is contrary to the second principle regarding the "defense-in-depth" philosophy. Compliance with the ASME Code, Section XI, inservice inspection requirements fails to satisfy the third principle of maintaining the safety margins since it cannot be assured that pressure boundary leakage would be detectable 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 gross control rod drive mechanism nozzle failure. Conditional core damage probability values for the subject plants range from 2E-2 per reactor-year to 1.4E-3 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 would have to be demonstrated to be below 5E-4 to 7E-3 per reactor-year. For a plant that has a baseline core damage frequency of greater than 1E-4 per reactor-year, the initiating event frequency for the vessel head penetration nozzle failure would have to be demonstrated to be below 5E-5 per reactor-year. For the age of the plants in question and the lack of a qualified examination for detecting degradation in these nozzles, there does not appear to be an adequate basis to justify the necessarily low initiating event frequencies proposed by the industry for these types of failures.

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 Information Summary 2001-02 process involves identification of an adequate basis for establishing reasonable assurance of protection when the "special circumstance" is considered. The Commission has compiled a number of general design criteria for the design, fabrication, construction, testing, and performance of structures, systems and components important to safety in Appendix A to 10 CFR Part 50. The general design criteria provide the Commission's perspectives on the factors that are sufficient to achieve "adequate protection." Three general design criteria 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 leaktight integrity." Taken as a whole, these general design criteria 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.

Failure to inspect a portion of the reactor vessel in a manner that is sufficient to detect the extent of degradation caused by a mechanism known to be degrading other 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 plants, if left undetected and uncorrected, would result in a gross failure of the reactor coolant pressure boundary (loss-of-coolant accident). This creates the potential for the plant's core damage frequency to rise to values approaching the conditional core damage probability of a loss-of-coolant accident, approximately 5.3E-3 per reactor-year. This loss of confidence that the core damage frequency will not increase to an unacceptable level, plus the associated potential for loss of one of the "defense-in-depth" barriers constitutes an undue risk to the public health and safety. Therefore, I do not have reasonable assurance that adequate protection is achieved by plants that do not immediately perform the inspections that are sufficient to detect this type of degradation.

IV

I find that issuance of an Order to require licensees with the most highly ranked (susceptible) vessel head penetration nozzles to immediately perform inspections that are capable of detecting vessel head penetration nozzle degradation or leakage and before the safety margins for the nozzles are lost and rupture of the nozzle occurs is necessary to provide reasonable assurance of adequate protection of the health and safety of the public. Accordingly, pursuant to 10 CFR 50.109(a)(4)(ii), no backfit analysis is required for imposition of these inspection requirements. Pursuant to 10 CFR 2.202, I have determined, based on potentially hazardous condition, that a circumferential crack may exist undetected and uncorrected in the vessel head penetration nozzles of these facilities, 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 1 day after the date of issuance of this Order and proceed to the cold shutdown or refueling 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 to support 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 subpopulation of plants considered to have a high susceptibility to primary water stress corrosion cracking. This qualified visual examination should 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 any vessel head penetration nozzle exhibiting through-wall cracking will provide sufficient leakage to the reactor pressure vessel head surface (based on the as-built configuration of the vessel head penetrations). Secondly, 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, 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) may be appropriate to provide evidence of the structural integrity of the vessel head penetration nozzles. This examination shall be found acceptable by the staff prior to plant operation.

The Regional Administrator, Region II, or 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.

V

Under 10 CFR 2.202(a)(1), the Commission has the authority to modify, suspend, or revoke an operating license when the Commission finds, among other things, potentially hazardous conditions, or other facts deemed to warrant issuance of an Order. Under 10 CFR 2.202(a)(5), the Commission may make Orders immediately effective, without prior opportunity for hearing, in cases where the Commission determines that the public health, interest, or safety so requires, or where conduct causing the violation is willful.

The modification of Operating License NPF-3 stated in Section IV of this Order is based on assuring that the adequate protection of the health and safety of the public will be maintained at Davis-Besse Nuclear Power Station, Unit No. 1

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 Enforcement, 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 Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555, to the Assistant General Counsel for Materials Litigation and Enforcement at the same address, to the Regional Administrator, NRC Region II, Sam Nunn Atlanta Federal Center, 23 T85, 61 Forsyth Street, SW., Atlanta, GA 30303-8931, 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