From: Bill Bateman
To: Lawrence Burkhart NRR
Date: 10/16/01 11:20AM
Subject: Re: ORDER

I reworked some of the narrative based on a review of a GT response in which Jim Medoff used lots of words from the orders. He will be getting that input to you.

>>> Lawrence Burkhart 10/15/01 04:36PM >>>
Please find attached the latest with respect to the generic portion of order (which may be used for Davis-Besse, Surry 2, and/or D.C. Cook 2). I added an introductory paragraph to the safety issue section (p. 1). Added some discussion of why VHP leaks may not be able to be detected prior to gross failure (Allen please review) - p. 5.

A couple of questions/issues from Brian:

1. Need more re: justification of why 12/31/01 is the required date (Allen is working on this part)

2. The issue of undue risk is not fleshed out adequately. We state we are applying the RIS 2001-02 decision-making process 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. That leads to discussion of (1) compliance with regs and (2) no undue risk. The compliance with regs issue is discussed but not the undue risk portion.

Mark/Rich, Could you help out in this area?

CC: James Medoff

D-22

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UNITED STATES
NUCLEAR REGULATORY COMMISSION

In the Matter of)
) Docket No. _______
(LICENSEE)) License No. _______
(Facility Name)) EA _____

ORDER MODIFYING LICENSE
(EFFECTIVE DECEMBER 31, 2001)

I

Blank No. 1 (Name of licensee) (Licensee) is the holder of Facility Operating License No. Blank No. 2 issued by the Nuclear Regulatory Commission (NRC or Commission) pursuant to 10 CFR Part 50 on Blank No. 3 (Date-1). The license authorizes the operation of Blank No. 4 (XYZ facility) in accordance with conditions specified therein. The facility is located on the Licensee's site in Blank No. 5 (City, State).

II - Safety Issue

In February 2001, circumferential cracking reactor pressure vessel head penetration nozzles (including control rod drive mechanism penetration nozzles) has been observed at Oconee Nuclear Station, Unit 3, a previously unseen and unexpected cracking phenomenon. Circumferential cracking of vessel head penetration nozzles is significant because of the potential failure of a layer of defense in depth for plant safety. The cause of this cracking mechanism has been attributed to primary water stress corrosion cracking of the vessel head penetration nozzles which initiates from the outside diameter of the nozzles and could result in a small-break loss-of-coolant accident. In response to the occurrence of this new cracking phenomenon, the Commission issued NRC Bulletin, "Circumferential Cracking of Reactor Vessel Head Penetration Nozzles," on August 3, 2001, to address the generic safety implications of potential pressure boundary leakage from vessel head penetration nozzles and to discuss the staff's technical bases for the recommended graded-inspection program for U.S. vessel head penetration nozzles.

Reactor pressure vessel head penetrations, including control rod drive mechanism nozzles, are part of the reactor coolant pressure boundary, which is one of 3 principle barriers to the release of radioactive fission products to the environment. All control rod drive mechanism nozzles are currently fabricated from Inconel 600 (Alloy 600) and are joined to the upper vessel heads using interference fits and partial penetration J-groove welds fabricated from Alloy 182, which is an Inconel filler metal material with material properties similar to Alloy 600. Previous staff reviews of the preliminary safety assessments that were submitted to the NRC as topical reports by the pressurized water reactor owners groups (Westinghouse, Combustion Engineering, and Babcock and Wilcox Owners Groups), indicate that the methods for fabricating control rod drive mechanism nozzles have been basically the same for all U.S. pressurized water reactors.

Stress corrosion cracking can only occur if the following material properties and environmental conditions are present: (1) the material must be in a highly stressed environment, (2) a corrosive environment must be present, and (3) the material must be of a type that is susceptible to stress corrosion cracking (primary water stress corrosion cracking is an age-related form of stress corrosion cracking). Due to the high temperatures and pressures, as well as the borated coolant environment, vessel head penetrations, including control rod drive mechanism nozzles, are in a highly stressed and corrosive environment. In addition, all control rod drive mechanism nozzles are fabricated from Alloy 600, a material that is known to be susceptible to primary water stress corrosion cracking when exposed in highly stressed, high temperature, borated coolant environments. Reports of stress corrosion cracking in Alloy 600 pressurizer nozzles and instrumentation nozzles to the reactor coolant hot legs of pressurized water reactors confirm that Alloy 600 is a material that is susceptible to stress corrosion cracking.

Cracking of control rod drive mechanism nozzles and welds is a degradation of the primary reactor coolant system boundary. If undetected and left uncorrected, primary water stress corrosion cracking of a vessel head penetration nozzle has the potential to result in leakage of the reactor coolant from the pressure boundary and possibly a catastrophic failure of the nozzle. The latter event would result in a significant small-break loss-of-coolant-accident for the facility.

Crack initiation and growth models predict that cracks initiate and grow along planes that are perpendicular to the stress vectors acting on them. During power operations of the reactors, adjacent to the contour of the J-groove weld, the highest magnitude tensile stresses are the nozzle hoop stresses. Consequently, in the regions adjacent to the contour of the J-groove welds, control rod drive mechanism nozzle cracking is postulated to occur in axial-oriented planes. In contrast, in regions located at vertical positions just above the root of the J-groove welds, the axial stresses have the high tensile stress magnitudes (with the highest magnitude tensile stresses occurring at the outside diameter of the nozzles). Therefore, for the nozzle regions located just above the root of the J-groove welds, any cracking would be postulated to initiate from the outer surface of the nozzle along a circumferentially oriented plane. However, the following events would have to take place for initiation of a circumferential flaw to be possible at these locations: (1) initiation of an axial crack would need to occur, (2) the stress intensity factor for the crack tip would need to be of a magnitude high enough to grow the axial crack through-wall such that a sufficient leak path would exist to allow for leakage of the coolant into the annular region of the nozzle, and (3) the environment resulting from the leakage would have to be corrosive enough to initiate the circumferential flaw from the outside diameter surface of the nozzle.

A. History of Vessel Head Penetration Cracking in the Nuclear Power Industry and Generic Letter 97-01

Axial cracking in pressurized water reactor control rod drive mechanism nozzles has been previously identified, evaluated, and repaired. Numerous small-bore Alloy 600 nozzles and pressurizer heater sleeves have experienced leaks attributed to primary water stress corrosion cracking. Generally, these components are exposed to temperatures of 560°F or higher and to reactor coolant system inventory. However, circumferential cracks above the weld from the outside diameter to the inside diameter have not been previously identified in any of the vessel head penetration nozzles in the U.S.

In 1991, an action plan was implemented by the NRC staff to address primary water stress corrosion cracking of Alloy 600 vessel head penetrations at all U.S. pressurized water reactors. This action plan included a review of the safety assessments by the pressurized water reactor owners groups submitted for staff review on June 16, 1993, by the Nuclear Management and Resource Council (NUMARC, now the Nuclear Energy Institute [NEI]).

After reviewing the industry's safety assessments and examining the overseas inspection findings, the NRC staff concluded, in a safety evaluation dated November 19, 1993, that pressurized water reactor control rod drive mechanism nozzle and weld cracking was not an immediate safety concern. The bases for this conclusion were that if primary water stress corrosion cracking occurred, (1) the cracks would be predominately axial in orientation, (2) the axial cracks would result in detectable leakage before catastrophic failure, and (3) the leakage would be detected during visual examinations performed as part of surveillance walkdown inspections before significant damage to the reactor pressure vessel head would occur (based on the information gathered from the visual inspections of plants that have experienced vessel head penetration nozzle leakage, the staff believes that this leakage may be small enough so as not to be detectable prior to catastrophic failure). However, the NRC staff noted concerns about potential circumferential cracking (which would need to be addressed on a plant-specific basis), high residual stresses from initial manufacture and from tube straightening sometimes done after welding, and the need for enhanced leakage monitoring.

By letter dated March 5, 1996, NEI submitted a white paper entitled "Alloy 600 Reactor Pressure Vessel Head Penetration Primary Stress Corrosion Cracking," which reviewed the significance of primary water stress corrosion cracking in pressurized water reactor vessel head penetrations and described how the pressurized water reactor licensees were managing the issue. NEI assumed that the issue was primarily an economic issue rather than a safety issue, and described an economic decision tool to be used by pressurized water reactor licensees to evaluate the probability of a vessel head penetration developing a crack or a through-wall leak during a plant's lifetime. This information would then be used by a pressurized water reactor licensee to evaluate the need to conduct an inspection of the vessel head penetration nozzles at its plant.

To verify the conclusions in the industry's safety assessments, sampling inspections were performed at 3 pressurized water reactors in 1994. The results of these inspections were consistent with the February 1993 analyses by the owners groups, the staff's safety evaluation dated November 19, 1993, and the primary water stress corrosion cracking reported in the vessel head penetration nozzles of European reactors. On the basis of the results of the initial inspections of U.S. pressurized water reactors, the pressurized water reactor owners groups' analyses, and the European experience, the NRC staff determined that it was probable that control rod drive mechanism nozzles at other plants contained similar axial cracks, but that such cracking did not pose an immediate or near-term safety concern. Further, the NRC staff recognized that the scope and timing of inspections may vary for different plants, depending on their individual susceptibility to this form of degradation. However, in its safety evaluation of November 19, 1993, the staff identified that degradation of control rod drive mechanisms and other vessel head penetration nozzles was an important safety consideration in the long term because of the possibility of (1) exceeding the ASME Code safety margins if the cracks are sufficiently deep and continue to propagate during subsequent operating cycles and (2) eliminating a layer of defense in depth for plant safety.

On April 1, 1997, NRC issued Generic Letter 97-01, "Degradation of Control Rod Drive Mechanism Nozzle and Other Vessel Closure Head Penetrations," which requested addressees to inform the staff of their inspection activities related to vessel head penetrations. Based on the industry's Generic Letter 97-01 responses, which took credit for periodic inspections of the reactor pressure vessel head, the staff agreed that the conclusions in its November 19, 1993, safety evaluation remained valid.

B. Cracking at the Oconee Nuclear Station

On February 18, 2001, with Oconee Nuclear Station, Unit 3 in Operating Mode 5, Duke Energy Corporation, the licensee, 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. Nondestructive examinations of nine additional control rod drive mechanism nozzles from the same heat of material were conducted for "extent of condition" purposes. The licensee did not detect recordable indications in these nine additional control rod drive mechanism nozzles.

Upon commencement of required ASME Code Section XI repair activities of the affected control rod drive mechanism nozzles, Duke Energy Corporation implemented required dye-penetrant testing of the repair weld butter 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 dye penetrant test examinations. In addition, it was determined that several flaw indications were circumferential in orientation and grew into the nozzle just above the root of the J-groove weld. Further investigations and metallurgical examinations revealed 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 pin hole through-wall indications. These circumferential portions of the cracks followed the weld profile contour.

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. The recent identification of circumferential cracking in control rod drive mechanism nozzles at Oconee Nuclear Station, Units 2 and 3, along with axial cracking in the J-groove welds at these 2 units and at Oconee Nuclear Station, Unit 1 and Arkansas Nuclear One, Unit 1, has resulted in the staff reassessing its conclusion in Generic Letter 97-01 that cracking of vessel head penetration nozzles is not an immediate safety concern.

C. Topical Report MRP-44, Part 2

After the initial finding of significant circumferential cracking at Oconee Nuclear Station, Unit 3 , the NRC held a public meeting with the EPRI Materials Reliability Program (MRP) on April 12, 2001, to discuss control rod drive mechanism nozzle circumferential cracking issues. During the meeting, the industry representatives indicated that they were developing a generic safety assessment, recommendations for revisions of near-term inspections, and long-term inspection and flaw evaluation guidelines.

On May 18, 2001, the MRP submitted EPRI Report TP-1001491, Part 2, "PWR [pressurized water reactor] Materials Reliability Program Interim Alloy 600 Safety Assessments to U.S. Pressurized water reactor Plants (MRP-44), Part 2: Reactor Vessel Top Head Penetrations," (henceforth MRP-44, Part 2) to provide an interim safety assessment for primary water stress corrosion cracking of Alloy 600 vessel head penetration nozzles and Alloy 182 J-groove welds in pressurized water reactors. The approach taken in the MRP-44, Part 2, uses an assessment of the relative susceptibility of each pressurized water reactor to outside diameter-initiated or weld-initiated primary water stress corrosion cracking based on the operating time and temperature of the penetrations. Based upon this simplified model, provided in Appendix B of the MRP-44, Part 2, each pressurized water reactor plant was ranked by the MRP according to the operating time (in EFPY) required for the plant to reach an effective time-at-temperature condition equivalent to the worst case degraded condition for Oconee Nuclear Station, Unit 3 . To address the experience at Oconee Nuclear Station, the MRP recommended that plants ranked within 10 EFPY of Oconee Nuclear Station, Unit 3 and having fall 2001 outages should perform a visual inspection of the reactor pressure vessel top head capable of detecting small amounts of leakage similar to that observed at the Oconee Nuclear Station, Units 2 and 3 and Arkansas Nuclear One, Unit 1.

On June 7, 2001, the NRC held a public meeting at which the MRP provided initial responses to questions on the MRP-44, Part 2, report that the NRC staff had identified and transmitted to the MRP on May 25, 2001. The NRC staff provided additional questions on various aspects of the MRP-44, Part 2, report in a letter to the MRP, dated June 22, 2001. In this letter, the staff informed the MRP that the staff had two areas of contention with the industry's methodology provided in MRP-44, Part 2, the first issue being that the staff did not agree with the MRP's conclusion that nozzle leaks would be detectable on all vessel heads and the second issue being that the staff was concerned with the lack of consideration of an applicable crack growth rate for cracks initiated from the outer-diameter surfaces of the vessel head penetration nozzles or in their associated J-groove welds. With respect to the latter issue, the staff informed the MRP that it did not agree with the MRP conclusion in its responses of June 7, 2001, that the appropriate crack growth rate for outside diameter-initiated cracking of vessel head penetration nozzles is adequately represented by crack growth data for Alloy 600 steam generator tubes under primary water environments.

D. <u>NRC Bulletin 2001-01</u>

On August 3, 2001, the Commission issued NRC Bulletin 2001-01, "Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles," to address the generic safety implications of the pressure boundary leakage at the Oconee Nuclear Station, Unit 2 and 3 and Arkansas Nuclear One, Unit 1 power plants on the industry's vessel head penetration nozzles, and to discuss its technical bases for the recommended graded-inspection program for U.S. vessel head penetration nozzles. In the bulletin, the staff discussed the technical aspects of plant designs that could impede the ability of the VT-2 visual examination methods to detect leakage from the control rod drive mechanism nozzles of commercial U.S. pressurized water reactors.

The staff emphasized that the ability to detect reactor coolant leakage from the vessel head penetration nozzles could be limited if the visual examination methods for detecting the leakage were incapable of distinguishing between boric acid residue deposited as a result of vessel head penetration nozzle leaks and those previously deposited as a result from leakage from other sources. The staff also emphasized that it was critical for the industry to establish defensible crack growth rates for primary water stress corrosion cracking-type flaws in both vessel head penetration nozzle base metal and filler metal materials so that a determination could be made as to whether a partial through-wall flaw would be capable of growing beyond the critical flaw size during a scheduled operating cycle for a facility.

In the Bulletin, the staff stated that, as a result of its review of the susceptibility rankings given in Appendix B to MRP-44, Part 2, the population of control rod drive mechanism nozzles for U.S. pressurized water reactors could be categorized into the following populations. For the population of plants considered as having low susceptibility based upon a susceptibility ranking of more than 30 EFPY from the Oconee Nuclear Station, Unit 3 condition, the staff stated that the likelihood of primary water stress corrosion cracking degradation at these facilities was low, and that, therefore, enhanced examinations beyond those required by Section XI of the ASME Code were probably not necessary at the present time.

For the population of plants considered as having a moderate susceptibility to primary water stress corrosion cracking based upon a susceptibility ranking of more than 5 EFPY but less than 30 EFPY from the Oconee Nuclear Station, Unit 3 condition, the staff stated that an effective visual examination capable of detecting and discriminating small amounts of leakage or boric acid deposits from 100% of vessel head penetration nozzles, may be sufficient to provide reasonable assurance that primary water stress corrosion cracking degradation would be identified prior to posing an undue risk. The staff emphasized that this effective 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.

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 EFPY from the Oconee Nuclear Station, Unit 3 condition, 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 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% 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.

For the population of plants which have already identified the existence of primary water stress corrosion cracking in the control rod drive mechanism nozzles (for example, through the detection of boric acid deposits), the staff concluded there was a sufficient likelihood that the cracking of vessel head penetration nozzles will continue to occur as the facilities continue to operate, and that, therefore, a qualified volumetric examination of 100% of the vessel head penetration nozzles (with a demonstrated capability to reliably detect cracking on the outside diameter of the vessel head penetration nozzle) would be an appropriate method of providing evidence of the structural condition of their vessel head penetration nozzles.

Therefore, the staff requested that licensees addressed by the Bulletin submit the following information with respect to their nuclear power plants:

- the plant-specific susceptibility ranking for the plants (including all data used to determine each ranking) using the primary water stress corrosion cracking susceptibility model described in Appendix B to the MRP-44, Part 2, report;

- a description of the vessel head penetration nozzles in the plants, including the number, type, inside and outside diameter, materials of construction, and the minimum distance between vessel head penetration nozzles;

- a description of the reactor pressure vessel head insulation type and configuration;

- a description of the vessel head penetration nozzle and reactor pressure vessel head inspections (type, scope, qualification requirements, and acceptance criteria) that have been performed at the plant(s) over the past 4 years, and the findings, including a description of any limitations (insulation or other impediments) to accessibility of the bare metal of the reactor pressure vessel head for visual examinations; and

- a description of the configuration of the missile shield, the control rod drive mechanism housings and their support/restraint system, and all components, structures, and cabling from the top of the reactor pressure vessel head up to the missile shield, including the elevations of these items relative to the bottom of the missile shield.

In Bulletin 2001-01, the staff also requested that addressees discuss their plans, if any, to perform augmented examinations of their control rod drive mechanism nozzles consistent with the additional augmented examination recommendations provided in the Bulletin. The staff also requested, pursuant to 10 CFR 50.54(f), addressees to submit their responses within 30 days of issuance of the Bulletin.

By letter dated Blank No. 11 (Date-2) , the Licensee submitted its responses to NRC Bulletin 2001-01 for the Blank No. 12 (XYZ Facility). The Licensee's response to NRC Bulletin 2001-01 indicates that the Blank No. 13 - (provides the plant specific input from Al Hiser that summarizes the technical information in the response to NRC Bulletin 2001-01 for the XYZ facility). Based on the inadequacy of the ASME Section XI inspection methods to detect degrading control rod drive mechanism nozzle to reactor pressure vessel head welds, and the inability of the industry to establish a defensibly low initiating event frequency and core damage frequency for control rod drive mechanism nozzles failures, I lack assurance that the licensee's scheduled time for performing qualified visual examinations of the control rod drive mechanism nozzles of the Blank No. 14 (XYZ Facility) is sufficient to provide adequate protection of the health and safety of the public. As a result, there is a significantly increased probability of a control rod drive mechanism nozzle failure at this time, raising immediate concerns relative to the public health and safety.

The recent identification of significant circumferential cracking in 3 of the vessel head penetration nozzles at Oconee Nuclear Station, Unit 3 in March 2001, and later in a single vessel head penetration nozzle at Oconee Nuclear Station, Unit 2 in April 2001, raises concerns about a potentially risk-significant generic condition affecting all domestic pressurized water reactors. These identifications of circumferential cracking are 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. The manner in which the circumferential cracks were detected at Oconee Nuclear Station, Unit 3 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 (*i.e.*, 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 cracking reported at Oconee Nuclear Station, Units 2 and 3 reinforces the importance of examining 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. Presently, Paragraph IWA-5242 of Section XI, ASME Boiler and Pressure Vessel Code, does not require licensees to remove reactor pressure vessel head insulation materials before visually inspecting their reactor vessel heads and vessel head penetration nozzles.

The level of degradation (cracking) of vessel head penetration nozzles that has been found in other plants, if left undetected and uncorrected in a timely 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 small-break loss-of-coolant accident.

III

Under 10 CFR 2.202(a)(1), the Commission has the authority to modify, suspend, or revoke an operating license when the Commission finds violations of the Commission's requirements, 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 Blank No. 6 (Operating License No.) stated in Blank No. 7 (Section VI or VII) of this order is based on assuring that the adequate protection of the health and safety of the public will be maintained at the Blank No. 8 (XYZ Facility). As a result, pursuant to provisions in 10 CFR 50.109(a)(4)(ii), the staff is not required to, and hence did not, perform a backfit analysis for this order to modify Operating License Blank No. 9 (Operating License No.).

IV

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) to 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 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. However, pursuant to Paragraph IWA-5242 of Section XI of 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.

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., ≤ 1 gallon per year); 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. The location of thermal insulating materials, and physical obstructions may also 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. 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 gross 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 adequate assurance that the public health and safety are protected and significant risk implications may, therefore, exist. 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 decision-making 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 postulated occurrence of a vessel head penetration nozzle failure, and consequently a small-break 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 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 5 safety principles in the integrated decision-making 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 it (1) meets current regulations, (2) is consistent with "defense-in-depth philosophy", (3) maintains sufficient safety margin, (4) results is 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 pressure boundary leakage can go undetected before gross failure occurs.

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 the 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 type 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 (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. 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 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. Therefore, given the "special circumstance," the staff does not have reasonable assurance that adequate protection is achieved by plants that do not perform inspections that are sufficient to detect this type of degradation.

V

I find that issuance of an order to require licensees with most highly ranked (susceptible) vessel head penetration nozzles to perform inspections that are capable of detecting vessel head penetration nozzle degradation or leakage before the safety margins for the nozzles are lost and gross rupture (*i.e.*, a full 360° through-wall failure 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 [the licensee's commitment and,] the significance of concerns regarding the 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 on December 31, 2001.

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. <u>Blank No. 23 (XYZ Facility Name in CAPITAL LETTERS)</u> IS MODIFIED AS FOLLOWS:

1. Require a shutdown of the <u>Blank No. 24 (XYZ Facility Name)</u> to the cold shutdown Mode of Operation for the facility by <u>Blank No. 25 (Date-6).</u>
2. <u>Blank 26 - (Al Hiser to provide specific details of inspection methods the staff will require under modification of the license as consistent with the recommendations in NRC Bulletin 2001-01.)</u>
3. <u>Blank 27 - (Al Hiser to provide specific details of what the staff will require for</u>

removal of thermal insulation materials and cleanliness of the bare surfaces of the vessel head, as consistent with the recommendations in NRC Bulletin 2001-01.)

4. Blank 28 - (Al Hiser to provide specific details of what the staff will require for performing a qualified leak path and interference fit evaluation , as consistent with the recommendations in NRC Bulletin 2001-01.)

5. Blank 29 - (Al Hiser to provide specific details of the type of information the staff will require to be submitted by the licensee and the date for submitting this information to the Commission, as consistent with the recommendations in NRC Bulletin 2001-01.)

Require a shutdown of the Blank No. 24 (XYZ Facility Name) to the cold shutdown Mode of Operation for the facility by Blank No. 25 (Date-6) unless you provide information that is acceptable to the staff demonstrating that a qualified visual inspection was performed within the 18 months prior to December 31, 2001.

The Regional Administrator, Region X, 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.

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 Blank 31 - (Region No. , Blank 32 - (regional 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 IV 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 October 2001