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# White Paper - Policy for Issuance of Orders to Utilities Licensed to Operate Nuclear Power Generation Facilities

SUBJECT: BASES FOR ISSUING ORDERS TO REQUIRE SPECIFIC LICENSEES TO MODIFY THEIR OPERATING LICENSES AND PERFORM AUGMENTED VOLUMETRIC OR SURFACE EXAMINATIONS OF THEIR PENETRATION NOZZLES TO THE UPPER REACTOR VESSEL HEADS

## PURPOSE

To inform the NRC Executive and Leadership Teams of the regulatory bases for issuing an order to modify the operating license for a specific pressurized water reactor (PWR) and require augmented volumetric or surface examinations of the plant's vessel head penetration (VHP) nozzles if the license's response to NRC Bulletin 2001-01 is insufficient to ensure the structural integrity of the facility's reactor coolant pressure boundary during power operation.

## BACKGROUND

Recently identified circumferential cracking in control rod drive mechanism (CRDM) nozzles at Oconee Nuclear Station Units 2 and 3, along with axial cracking in additional CRDM nozzles at these and two other facilities (Oconee Nuclear Station Unit 1 and Arkansas Nuclear One Unit 1), has raised concerns regarding the potential safety implications of the active degradation mechanism–primary water stress corrosion cracking (PWSCC)–and compliance with applicable regulatory provisions. Although the staff addressed cracking of VHP nozzles in Generic Letter (GL) 97-01, "Degradation of Control Rod Drive Mechanism Nozzle and Other Vessel Closure Head Penetrations," dated April 1, 1997, the recent identification of circumferential cracking calls into question the conclusions in GL 97-01 and the adequacy of industry actions for detecting and managing cracking in VHP nozzles.

## APPLICABLE REQUIREMENTS

The VHP nozzles to the upper reactor vessel head of a pressurized water reactor are part of the reactor coolant pressure boundary (RCPB). The Commission has established a number of requirements for maintaining the structural integrity of the RCPB and for ensuring that the RCPB will have sufficient safety margins against failure during normal operating and transient operating conditions. These requirements include the RCPB leakage requirements in the Technical Specification Limiting Conditions for Operation for the reactor coolant system, and appropriate inspection requirements found in Section XI of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, as invoked by reference in 10 CFR 50.55a (Codes and Standards). In addition, the NRC has established in 10 CFR Part 50, Appendix B, quality assurance requirements for the purpose of ensuring that activities important to safety will be conducted and implemented in a manner that will assure safe operation of their facilties and protection of the health and safety of the public.

### Code Requirements

Section (g)(4) to 10 CFR 50.55a, "Codes and Standards," requires that utilities holding licenses to operate nuclear power generation facilities to comply with the inservice inspection

D-15

requirements of Section XI to the ASME Boiler and Pressure Vessel Code. The current version of 10 CFR 50.55a allows licensees to use the edition of record up through the 1995 edition of Section XI. The Section III of ASME Boiler and Pressure Vessel Code treats VHP nozzles as appurtenances to the reactor vessel. Unless a specific relief or a specific exemption has been granted, the rule requires that the following ISI requirements for the reactor vessel be met:

- Inspection Item B15.10 of Inspection Category B-P to Section XI Table IWB-2500-1 requires licensees to perform a system leak test using VT-2 type visual examination methods of the reactor pressure vessel once every refueling outage, and a system hydrostatic test using VT-2 visual examination methods once an ISI inspection interval.

- VT-2 examinations are to be performed in accordance with the requirements of IWA-5240. IWA-5241 provides the examination criteria for performing the VT-2 examinations on non-insulated ASME Code Class components. In this case, IWA-5241 requires that the VT-2 examinations be conducted on the accessible external surfaces of the pressure retaining component (for non-insulated components that are obstructed from direct VT-2 examination, IWA-5241 requires that the only visual examination of the surrounding area, including the adjacent floor areas of equipment surfaces located underneath the component need be performed.). IWA-5242 provides the examination criteria for conducting VT-2 examinations of insulated ASME Code Class components. In this case, IWA-5242 requires that the VT-2 examinations be performed at least on the accessible and exposed surfaces and joints of the insulation. For vertical insulation surfaces, the surfaces need only be examined at the lowest vertical elevation where leakage would be expected; for horizontal surfaces, the rule requires that the surfaces be examined at the insulation joints. IWA-5242 allows licensees to perform the VT-2 examinations without the need for removal of the insulation materials. IWA-5242 also requires that discoloration or residues on the insulation surfaces be given particular attention as it may provide evidence of leakage from the primary system.

- IWB-3140 requires that flaw indications detected as a result of VT-2 examinations be evaluated for acceptability, either by performing supplemental examinations, implementing corrective repair/replacement measures, or performing analytical evaluations. If supplemental examinations are conducted for the purpose of further characterizing the degradation, IWB-3200 allows the supplemental examinations to be conducted using ASME volumetric or surface examination methods.

<u>Technical Specification Limiting Conditions of Operation for the Reactor Coolant Pressure Boundary</u>

The current improved technical specifications (ITS) for the reactor coolant system (RCS) do not allow for any identified leakage from the reactor coolant pressure boundary (i.e., 0 gpm of leakage from the RCPB), and allow for only 1 gpm of leakage from unidentified RCS sources. The ITS action statements on RCS leakage require licensees to bring within required limits with in 4 hours of detection or else bring the reactor to hot shutdown within the following 6 hours and cold shutdown within the following 36 hours. Plant specific versions of PWR technical specifications have corresponding limiting condition of operation and action statement requirements that are equivalent in intent to those in the ITS.

Appendix B to 10 CFR Part 50, Quality Assurance Requirements for Nuclear Power Plants and Fuel Reprocessing Plants

Appendix B to 10 CFR Part 50 provides the staff's quality assurance requirements for nuclear power generation and fuel production factilities. There are eighteen quality assurance requirements (Criteria) specified in Appendix B to Part 50. Of these criteria, Criteria II, V, IX, and XVI are of particular interest to the safety function of maintaining the structural integrity of ASME Code Class 1 vessel head penetration nozzles (VHP nozzles).

- Criterion II, "Quality Assurance Program," requires licensees to establish sufficient quality assurance programs at their facilities. With respect to these programs, the criterion, in part, requires that these programs identify structures, systems and components (SCC) falling within the scope of the programs, and provide control over activities affecting the quality of these SSC, consistent with their importance to safety. Criterion II, also in part, requires that activities affecting quality must be performed under suitably controlled conditions. The conditions include use of appropriate equipment, suitable environmental conditions for accomplishing the activity, such as adequate cleanliness, and assurance that all prerequisites for performing the activity have been satisfied. The criterion requires that the programs take into account the need for special controls, processes, test equipment, tools, and skills to attain the required quality, as well as the need to verify the level of quality by inspection or test.

- Criterion V, "Instruction, Procedures, and Drawings," requires that activities affecting quality shall be prescribed by documented instruction, procedures, or drawings, of a type appropriate to the circumstances and shall be accomplished in accordance with these instructions, procedures or drawings. These instructions procedures or drawings shall include appropriate qualitative or quantitative acceptance criteria for determining that important activities have been satisfactorily accomplished.

- Criterion IX, "Special Processes," requires that measures be established to assure that special processes such as welding, non-destructive testing or examination, and heat treating are controlled and accomplished by qualified personnel using qualified procedures in accordance with established codes, standards, specifications, criteria, and other special requirements.

- Criterion XVI, "Corrective Action," requires that measures be established to ensure that conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and nonconformances are promptly identified and corrected. For cases of conditions that are determined to be significantly adverse to quality, the criterion requires the measures include a determination of the cause of the adverse condition, and corrective measures to preclude repetition.

REGULATORY BASES AND POLICY FOR ISSUING ORDERS

10 CFR 2.202 provides the Commission's regulatory basis for issuing orders to holders of NRC operating licenses. 10 CFR 2.202(a)(i) implies that orders are given only when the Commission determines that a given licensee has failed to respond and resolve the regulatory and safety issues associated with a documented violation of the Commission's requirements, or when there exist other potentially hazardous conditions at a facility that would warrant issuance of the order.

According to 10 CFR 2.202, orders are made effective immediately, without prior opportunity for hearing, only it is determined that the public health, interest, or safety so requires, or when the order is responding to a violation involving willfulness. Otherwise, any licensee that receives an order from the Commission is afforded an opportunity for a hearing. Pursuant to 10 CFR 2.204, for cases in which the Commission believes a basis could reasonably exist for not imposing the order as proposed, the Commission may issue a Demand for Information to the licensee for the purpose of determining whether an order should be issued pursuant to 10 CFR 2.202. Any Demand for Information is required by the rule to allege the violations with which the licensee is charged or the potentially hazardous conditions or other facts deemed sufficient for issuing the demand. Any licensee issued a Demand for Information is required to submit a written response to the demand under Oath and Affirmation within 20 days of its issuance date, or other time as may be specified.

The Commission's policy for issuing orders is given in Section VI.D of the NRC's Enforcement Policy. The NRC's Enforcement Manual provides the guidance for implementing the Commission's Enforcement Policy. These documents are available through the NRC's Office of Enforcement Web Site, which is accessible at the following world-wide-web address: www.nrc.gov/OE According to the NRC's Enforcement Policy, an order is a written NRC directive to modify, suspend, or revoke a license; to cease and desist from a NRC-prohibited practice or activity; or to take such other action as may be proper (see 10 CFR 2.202). Orders may also be issued in lieu of, or in addition to, civil penalties, as appropriate for Severity Level I, II, or III violations. Unless a separate response is warranted under 10 CFR 2.201, a Notice of Violation need not be issued where an order is based on violations described in the order. The violations described in an order need not be categorized by severity level.

The Atomic Energy Act of 1954 and NRC Management Directive 9.17 allow the Commission to delegate the authority to take enforcement or other appropriate actions that may be authorized under 10 CFR Part 2, Subpart B, to the Executive Director for Operations (EDO) or his designee; the scope of this authority includes the authority to issue orders of behalf of the Commission, as promulgated in 10 CFR 2.202.

TOPICAL REPORT MRP-44, PART 2, AND ISSUANCE OF NRC BULLETIN 2001-01

After the initial finding of significant circumferential cracking at ONS3, the NRC held a public meeting with the EPRI Materials Reliability Program (MRP) on April 12, 2001, to discuss CRDM 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 the MRP-44, Part 2, report to provide an interim safety assessment for PWSCC of Alloy 600 VHP nozzles and Alloy 182 J-groove welds in PWR-designed plants. The approach taken in the MRP-44, Part 2, report uses an assessment of the relative susceptibility of each PWR to OD-initiated or weld-initiated PWSCC 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, report, each PWR 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 ONS3 as relative to the time the above-weld circumferential cracks were identified in early 2001. To address the experience at ONS, the MRP recommended that plants ranked within 10 EFPY of ONS3 and having fall 2001 outages should perform a visual inspection of the RPV top head capable of detecting small amounts of leakage similar to that observed at the Oconee units and ANO1.

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 Topical Report MRP-44, Part 2. With respect to the first area of contention, the staff informed the MRP that it did not agree with the MRP's conclusion that nozzle leaks would be detectable on all vessel heads. With respect to the second area of contention, the staff informed the MRP that it was concerned with the lack of consideration of an applicable crack growth rate for cracks initiated from the outer-diameter (OD) surfaces of the VHP nozzles or in their associated J-groove welds. With respect to this area of contention, the staff informed the MRP that it did not agree with the MRP conclusion in the responses of June 7, 2001, that the appropriate crack growth rate for OD-initiated cracking of VHP nozzles is adequately represented by crack growth data for Alloy 600 steam generator tubes under primary water environments.

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 and Arkansas Nuclear One-Unit 1 power plants on the industry's VHP nozzles, and to discuss its technical bases for the recommended graded-inspection program for U.S. VHP nozzles. In the bulletin, the staff emphasized that the issue of whether VHP nozzle leaks would be detectable at any particular PWR is a difficult issue to address. The staff therefore stated that the question as to whether a given licensee's visual examination methods would capable of detecting leakage from a given CRDM nozzle would require a licensee to complete the following assessments:

- a determination of the plant-specific, as-built nozzle geometries, including those of the measured in-shop nozzle dimensions for both VHP nozzles and vessel head penetrations to allow for a determination of the interference fit population for a particular RPV head
- a sufficiently detailed modeling of the RPV head and expected through-wall crack characteristics, such as surface roughness and crack tightness, to allow for a determination as to whether any cracking in the nozzles or associated J-groove welds is sufficient to provide for a sufficient leakage path to outer surface of the upper RPV head.
- a determination as to whether leakage paths to the outer surface of the upper vessel heads will result in sufficient amounts of boric acid residue deposits that will be detectable during visual examinations of the reactor vessel head.

- a determination of the materials, structures or components, such as thermal insulation materials, or supports that have the ability to make it difficult to examine the vessel heads or mask any signs of leakage that would normally be detected if the surfaces were clean and unobstructed.

In addition, the staff emphasized that the ability to detect reactor coolant leakage from the VHP 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 VHP nozzle leaks and those previously deposited as a result from leakage from non-safety-significant sources. The staff also emphasized that it was critical for the industry to establish defensible crack growth rates for PWSCC-type flaws in both VHP 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 CRDM nozzles for U.S. PWRs could be categorized into the following populations:

- those plants which have demonstrated the existence of PWSCC in their VHP nozzles (through the detection of boric acid deposits) and for which cracking can be expected to recur and affect additional VHPs;
- those plants which can be considered as having a high susceptibility to PWSCC based upon a susceptibility ranking of less than 5 EFPY from the ONS3 condition;
- those plants which can be considered as having a moderate susceptibility to PWSCC based upon a susceptibility ranking of more than 5 EFPY but less than 30 EFPY from the ONS3 condition; and
- the balance of plants which can be considered as having low susceptibility based upon a susceptibility ranking of more than 30 EFPY from the ONS3 condition.

As a result of its independent review of the MRP-44, Part 2 report, the staff informed the addressees that it would be appropriate for licensees to be cognizant of extenuating circumstances at their respective plant(s) that would suggest a need for more aggressive inspection practices to provide an appropriate level of confidence in VHP nozzle integrity.

For the population of plants in the low susceptibility population, the staff stated that the likelihood of PWSCC 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 in the moderate susceptibility population, the staff stated that an effective visual examination capable of detecting and discriminating small amounts of leakage or boric acid deposits from 100% of VHP nozzles, may be sufficient to provide reasonable confidence that PWSCC 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 RPV head, or other factors that could interfere with the detection of leakage.

For the population of plants in the high susceptibility population, the staff stated that the possibility for leaks to occur from a VHP 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 VHP nozzles. With respect to an examination of this sort, 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 VHP nozzle exhibiting through-wall cracking would be capable of provide sufficient leakage path to the RPV head surface (based on the as-built configuration of the VHPs), and (2) that the effectiveness of the qualified visual examination should not be compromised by the presence of insulation, existing deposits on the RPV 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 VHP nozzles (with a demonstrated capability to reliably detect cracking on the OD of a VHP nozzle) may be appropriate to provide evidence of the structural integrity of the VHP nozzles.

For the population of plants which have already identified the existence of PWSCC in the CRDM nozzles (for example, through the detection of boric acid deposits), the staff concluded there was a sufficient likelihood that the cracking of VHP nozzles will continue to occur as the facilities continue to operate, and that, therefore, a qualified volumetric examination of 100% of the VHP nozzles (with a demonstrated capability to reliably detect cracking on the OD of the VHP nozzle) would probably be the most appropriate method of providing evidence of the structural condition of their VHP nozzles.

In the bulletin the staff requested that addressees provide relevant information relative to the designs of the CRDM nozzles for their facilities, and their plans, if any, to perform augmented examinations of their CRDM nozzles consistence with the recommended augmented examinations recommendations provided in the bulletin.

## DISCUSSION AND TECHNICAL BASES FOR ISSUING ORDERS

The staff proposes to use the following technical bases and adequate protection arguments, as supported with appropriate compliance arguments in the following section of this paper, as its bases for issuing orders to specific licensees if their responses to NRC Bulletin 2001-01 cannot assure: (1) that structural integrity of their CRDM nozzles or other VHP nozzles will be maintained against a catastrophic failure of nozzle, and therefore a loss-of-coolant accident for the facility during power operations of the facility, and (2) that the probability for developing core damage as a result a postulated CRDM failure will be within acceptable values of risk as established by the NRC.

PWSCC is a form of stress corrosion cracking (SCC), which is an age-related cracking phenomenon. SCC can only occur if the following material property 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 suceptible to SCC. All CRDM nozzles are fabricated from Inconel 600 (Alloy 600), a material that is known to be susceptible to PWSCC when exposed in highly stressed, high temperature, borated coolant environments. These nozzles 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 evaluation submittals from the Westinghouse, Combustion Engineering, and Babcock and

Wilcox Owners Groups (i.e., as given in Topical Reports, indicate that the methods for fabricating CRDM nozzles has been basically the same for all U.S. PWRs.

Crack initiation and growth theories model that cracks initiate and grow along planes that are perpendicular to the stress vectors acting on them. During power operations of the reactors, for regions of the nozzles adjacent to the contour of the J-groove weld, the highest tensile stresses of magnitude are represented by the nozzle hoop stresses. Therefore, in the regions of the nozzles adjacent to the contour of the J-groove welds, cracking is postulated to occur in axial-oriented planes. In contrast, in nozzle 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 tensile stress magnitudes occurring at the O.D. of the nozzles). Thus, 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 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 O.D surface of the nozzle. The Duke Power Company's modeling of the circumferential flaws reported in the four ONS CRDM nozzles is consistent with these stress and leakage analyses and assumptions.

Through-wall, circumferential cracking of a CRDM nozzle is significant in that it has the potential to lead to a catastrophic failure of the nozzle and a small-break loss-of-coolant accident for the nuclear facility. Since there are currently no methods for preventing or mitigating the initiation and growth of PWSCC in Alloy 600 VHP nozzles, the current method for managing the effect is dependent on the implementation of inspection methods for detecting the effect prior to catastrophic failure of a facility's VHP nozzle. Section (g)(4) to 10 CFR 50.55a requires, in part, that ASME Code Class 1, 2, and 3 component 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 type visual examinations of their VHP nozzles and reactor vessel heads once every refueling outage for the system leak tests, and once an inspection interval for the hydrostatic pressure test. Subarticle IWA-3100 requires licensees to preform evaluations of flaw indications that have been detected as a result of required inservice inspections of ASME Code Class 1 components. These examinations are only capable of monitoring for leakage of the reactor coolant past the pressure boundary at best, and are not sufficient for detecting unacceptable partial throughwall flaws that reduce the effective nozzle thickness to magnitude less than the minimum allowable wall thickness for the nozzle.

The reports of circumferential cracking in the four ONS CRDM nozzles (i.e., one circumferential flaw reported in Unit 2 and a total three circumferential flaws reported in Unit 3) is signficant in that it represents the first reported occurrence of circumferential cracking in the CRDM nozzles of U.S. PWRs. The manner in which the circumferential cracks were detected is significant in that they were only detected after the licensee for ONS had performed non-destructive dye-penetrant examinations of weld-butters that were deposited as part of the licensee's repair activities for the degraded nozzles or in the case of one circumferential flaw, after the licensee had submitted its NDE results for the ONS CRDM nozzles to a contractor for an independent review of the examination results. The circumferential cracking was not detected as part of the licensee's normal ISI or expanded inspection efforts to monitor for additional signs of degradation in other ONS CRDM nozzles.

The current ASME Section XI inservice inspection requirements for VHP nozzles are only sufficient to reveal a problem with the structural integrity of the reactor coolant pressure boundary after the Technical Specifications have been violated, and are therefore insufficient to demonstrate whether leakage has occurred from the pressure boundary (i.e., from the VHP nozzles) during power operations of the reactors (i.e., during normal and transient operating conditions). Based on current data supplied by the industry to date, the staff cannot be assured that VT-2 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 throughwall flaw in the nozzles or their adjacent J-groove welds. Specifically, significant uncertainties exist as to whether power operation will result in the opening of a sufficient gap in the interference fits and provide for a sufficient path of leakage that would be capable of being detected on the outer surface of the vessel head. Leak-rate calculations of the reactor coolant from leaking VHP nozzles at the Oconee Unit 3 station demonstrate that the leaks may occur, and in all probability do occur, at very slow rates (i.e., ≤ 1 gpy); leakage rates of this magnitude may not high enough to allow for detectable indication of the leakage in the control room during power or transient operating conditions. The location of thermal insulating materials, and physical obstructions may also limit the capability of VT-2 examination method equipment to resolve minute amounts of boric acid deposits on the outer surface of the vessel head that would be capable of being detected by the human. 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 VHP nozzle leakage and those residues that result from leaks in other reactor coolant system components.

Currently, the Babcock and Wilcox designed vessel heads are the only PWR vessel heads that can be effectively examined by VT-2 methods without extensive efforts to remove thermal insulation materials from the vessel heads. Based on information supplied by the industry, extensive efforts are needed to remove the insulation materials from CE and Westinghouse designed vessel heads if visual examinations are to be effective methods of detecting leakage from the CRDM nozzles. As stated in IWA-5242, Section XI of ASME Code does not currently require licensees to remove thermal insulation materials when performing ASME VT-2 examinations of the reactor vessel head. There is therefore no guarantee that licensees owning CE or Westinghouse designed will remove the thermal insulation materials from their heads when conducting their VT-2 examinations. Based on these uncertainties, and the fact that the Duke Power did not detect the circumferential cracking in the CRDM nozzles until after it had initiated its repair activities for the nozzles creates a significant uncertainty as to whether the current Section XI ISI methodology for conducting visual examinations and dispositioning recordable flaw indications is capable of detecting the presence of significant O.D. initiated, circumferential cracks in U.S. CRDM nozzles.

*Reactor Systems Branch and Probabilistic Risk Assessment Branch to provide system and risk arguments supporting the adequate protection basis for issuing order*

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. Among them, GDC Criterion 14 states that the RCPB 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. Although the GDC do not apply to all facilities, the Commission has taken the position in GDC that it is important from a safety standpoint to protect the RCPB from the occurrences of leaks, rapidly propagating failure, and gross rupture. Based on the uncertainties in the ability of current NDE inspection methods to detect and size recordable PWSCC-type flaws in U.S. CRDM nozzles, and the ability of the industry to establish defensibly low initiating event frequence and core damage frequency for CRDM nozzles failures, the staff cannot come to the conclusion that the current licensing bases for inspecting the CRDM nozzles of U.S. PWRs are sufficient to provide adequate protection of the health and safety of the public.

COMPLIANCE BASES IN SUPPORT OF AN ADEQUATE PROTECTION BASIS FOR ISSUING ORDERS

Criterion XVI of 10 CFR Part 50, Appendix B, requires licensees to establish measures for the purpose of assuring "that conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and non-conformances are promptly identified and corrected. The staff is taking the position that the potential for circumferential CRDM nozzle cracks to exist and remain undetected in the VHP nozzles of U.S. PWRs is representative of a significant condition adverse to safety, as defined in Criterion XVI of 10 CFR Part 50, Appendix B. In the case of significant conditions adverse to quality, the rule requires that these measures shall be sufficient to assure that the cause of the condition is determined and corrective action taken to preclude repetition. In this case, the rule states that the identification of the significant condition adverse to quality, the cause of the condition, and the corrective action taken shall be documented and reported to appropriate levels of management".

How the NRC applies Criterion XVI to a given licensee's program for examining the VHP nozzles of a given PWR facility may depend on both on how a licensee opts to inspect for postulated degradation in the nozzles and their associated J-groove welds, and whether previous inspections of the nozzles, J-groove welds, or vessel heads have indicated a significantly degraded condition, as indicated by either existence of a recordable flaw indication of unacceptable size or by the presence of recordable through-wall reactor coolant pressure boundary leakage.

The technical specification limiting conditions of operation for reactor coolant systems do not allow for any identified leakage from the reactor coolant pressure boundaries of U.S. PWRs. For licensees that have performed visual VT-2 examinations of their plants' vessel heads during previous refueling outages for their units and have detected prior evidence of leakage on the outer surfaces of the heads, should subsequent visual examinations reveal the presence of additional pressure boundary leaks from the CRDM nozzles, it may be applicable for the NRC to apply provisions of Criterion XVI as a basis for concluding that the additional round of visual examinations were not sufficient to preclude detection of a known degradation mechanism (i.e., PWSCC) prior to falling out of compliance with the RCPB leakage requirements in technical specifications, and for citing the licensee against the technical specifications.

The compliance issue is not as cut and dry for the case of licensees that detect CRDM nozzle leakage for the first time as a result of their VT-2 examinations. However, the Office of Nuclear

Regulatory Regulation (NRR) and the Office of the General Coucil (OGC) have provided some enforcement guidelines in the internal memorandum of January 24, 1997, from Mr. Roy P. Zimmerman, Associate Director for Projects, NRR, to Mr. James Lieberman, Director of the Office of Enforcement (OE). In this memorandum, the staff states that it has been a long-standing policy that the allowed outage time (AOT) for a non-compliance with a limiting condition for operation should commence with the time upon which the inoperability or degraded condition was discovered; however, staff also states there has also been a long-standing recognition of the need to also consider potential enforcement based on the total duration that the condition may have existed (i.e., from the time of occurrence), where it can be readily determined, and the extent to which the licensee should have identified the condition earlier.

In the memorandum of January 24, 1997, the staff provides the following enforcement guidance as to whether a discovered non-compliance with a given LCO should result in citation of a violation against the technical specifications:

- If the time between the occurrence of the condition and the discovery of the condition is greater than the AOT for that condition (i.e., $t_D - t_O > AOT$), then the licensee should be cited for a failure to satisfy the LCO. Depending on the total time of the occurrence and other factors revealed by the root cause evaluation, the severity level could be increased to Severity Level III; however, any citation should be written to acknowledge that the licensee otherwise satisfied the technical specification required actions from the time of discovery.

- If the time between the occurrence of the condition and the discovery of the condition is less than the AOT for that condtion (i.e., $t_D - t_O < AOT$), and upon discovery the required actions are completed within the AOT or the shtudown track is satisfied, there would be no violation of the LCO condition; however, root cause determination would have to be examined to see if there are specific root cause issues that could warrant taking appropriate enforcement action.

- If the time between the occurrence of the condition and the completion of the required actions is less than the AOT, there would be no basis for warranting citing of a violation against the technical specifications.

The staff has therefore taken the position that, when making determinations as to cite a licensee for a failure to comply with a given LCO in technical specifications, appropriate consideration should be given to other citations, such as root causes that may focus on the actions taken by a licensee to detect and correct a non-conforming condition. Depending on the regulatory and safety significance, the LCO, the root causes or both should be combined into one escalated issue and used to determine whether enforcement is warranted. As stated in the applicable regulatory requirements section of this paper, the current improved technical specifications (ITS) for the reactor coolant pressure boundary (RCPB) do not allow leakage from the RCPB (i.e., 0 gpm pressure boundary leakage allowed), and only allow 1 gpm of unidentified reactor coolant system (RCS) leakage. ITS action statements on RCS leakage require licensees to bring within required limits within 4 hours of detection or else bring the reactor to hot shutdown within the following 6 hours and cold shutdown within the following 36 hours. Plant specific versions of PWR technical specifications have corresponding limiting condition of operation and action statement requirements that are equivalent in intent to those in the ITS. Thus, for those cases that document recordable leakage from the RCPB as a result of a CRDM nozzle leak, any decision to cite licensees for a violation of the technical specification leakage requirements will be dependent on whether the staff can establish within a reasonable doubt the time of occurrence of the pressure boundary leakage, and whether the licensee could have reasonable taken measures to detect the leakage prior to discovery. In this case, a policy decision will have to be made as to whether the information and issues raised in Bulletin are sufficient to have provided a given licensee with an opportunity to take measures to detect the leakage prior to the time the leakage was actually discovered.