POLICY ISSUE
INFORMATION

July 27, 2001 SECY-01-0141

FOR: The Commissioners

FROM: William D. Travers
Executive Director for Operations

SUBJECT: PROPOSED BULLETIN: "CIRCUMFERENTIAL CRACKING OF REACTOR PRESSURE VESSEL HEAD PENETRATION NOZZLES"

PURPOSE:

To inform the Commission of the staff's intent to issue the attached bulletin to address the structural integrity of reactor pressure vessel head penetration (VHP) nozzles in pressurized water reactors (PWRs).

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.

DISCUSSION:

Circumferential cracking of VHP nozzles would pose a safety concern if permitted to progress to the point that a loss of coolant accident occurs. The potential ejection of a VHP nozzle could results in additional deleterious effects such as potential damage to adjacent rods.

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Recent experience with VHP nozzle cracking also raises concerns regarding compliance with NRC requirements. The safety and regulatory concerns that provide the basis for the information request are discussed in the bulletin.

After the initial finding of significant circumferential cracking at Oconee Nuclear Station Unit 3, the staff held a public meeting with the Electric Power Research Institute (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. However, the staff determined that additional information is needed to assess the issue.

The proposed bulletin requests information from licensees related to the structural integrity of the reactor pressure VHP nozzles for their facilities, including the extent of VHP nozzle leakage and cracking that has been found to date, the inspections and repairs that have been undertaken to satisfy applicable regulatory requirements, future plans to inspect VHP nozzles, and a description of how future inspection plans will ensure compliance with applicable regulatory requirements.

Under the provisions of Section 182a of the Atomic Energy Act of 1954, as amended, and 10 CFR 50.54(f), the proposed bulletin transmits an information request for the purpose of verifying licensee compliance with existing regulatory requirements. Specifically, the requested information will enable the staff to determine whether current inspection practices for the detection of cracking in the VHP nozzles at reactor facilities provide reasonable assurance that reactor coolant pressure boundary integrity is being maintained. In accordance with the agency's risk-informed, performance-based process, the staff will use the information to quantify the risk associated with VHP failure and to determine whether licensee inspection practices need to be augmented to ensure that the likelihood of VHP failure remains low in the longer term.

The staff briefed the Committee to Review Generic Requirements (CRGR) on the proposed bulletin during a special meeting (No. 364) on July 2, 2001, and incorporated the comments made by the CRGR members. A memorandum dated July 19, 2001, with the final draft of the bulletin attached, was forwarded to the CRGR requesting review and endorsement. The CRGR reviewed the proposed final bulletin and endorses its release. In its endorsement letter, the CRGR states that the proposed bulletin does not impose a backfit, and thus the staff has not performed a backfit analysis.

The staff briefed the Advisory Committee on Reactor Safeguards (ACRS) Materials & Metallurgy and Plant Operations Subcommittees on July 10, 2001, and the ACRS full committee on July 11, 2001. In its letter dated July 23, 2001, the ACRS indicated that the decision to issue a bulletin addressing the recent incidents of circumferential cracking of CRDM nozzles in United States PWRs is timely and appropriate.

The staff will perform an initial assessment of the information submitted in accordance with the bulletin by December 31, 2001, to determine the need for further regulatory action, such as additional generic communication or rulemaking. Should the staff determine that additional generic communication or rulemaking is necessary, the staff will notify the Commission by early 2002.

Continued communication with the industry and other stakeholders is needed to effectively resolve the VHP nozzle cracking issue. Using the criteria provided in the bulletin and the initial industry analysis presented in topical report EPRI Report TP-1001491, Part 2, “PWR Materials Reliability Program Interim Alloy 600 Safety Assessments for US PWR Plants (MRP-44), Part 2: Reactor Vessel Top Head Penetrations,” a population of 14 PWR plants would be identified as either exhibiting high susceptibility to PWSCC or having experienced PWSCC in their VHP nozzles. An updated evaluation by the industry using licensee corrections to the RPV head operating temperature data adjusts this population to a combined eleven plants in these two groupings of plants. The NRC staff plans to hold a public meeting with the licensees for these plants and other stakeholders in early August to discuss the information request in the bulletin and staff expectations regarding licensee responses to the bulletin. The staff also will continue to meet with the industry and other stakeholders, including international organizations, to discuss technical issues such as the qualification of inspection methods developed by the industry.

Although VHP nozzle cracking is an issue for plants in the initial 40-year license period, the staff and the ACRS have also considered this issue in the context of license renewal. All necessary actions resulting from the resolution will carry forward as part of the licensing basis for the renewed operating period. The ACRS addressed the renewal aspects of this issue in its May 18, 2001, letter to the Commission on the ANO-1 license renewal. The ACRS recommendations on the immediate actions to address CRDM cracking will be provided in a forthcoming letter to the Commission.

The Office of the General Counsel has reviewed this bulletin and has no legal objections to its content.

The staff intends to issue this bulletin approximately five working days after the date of this information paper.

/RA/

William D. Travers
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for Operations

Attachment:
Proposed NRC Bulletin: Circumferential Cracking
of Reactor Pressure Vessel Head Penetration Nozzles”

OMB Control No.: 3150-0012

UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION
WASHINGTON, D.C. 20555-0001

July xx, 2001

NRC BULLETIN 2001-XX: CIRCUMFERENTIAL CRACKING OF REACTOR PRESSURE VESSEL HEAD PENETRATION NOZZLES

Addressees

All holders of operating licenses for pressurized water nuclear power reactors, except those who have ceased operations and have certified that fuel has been permanently removed from the reactor vessel.

Purpose

The U.S. Nuclear Regulatory Commission (NRC) is issuing this bulletin to:

(1) request that addressees provide information related to the structural integrity of the reactor pressure vessel head penetration (VHP) nozzles for their respective facilities, including the extent of VHP nozzle leakage and cracking that has been found to date, the inspections and repairs that have been undertaken to satisfy applicable regulatory requirements, and how their plans for future inspections will ensure compliance with applicable regulatory requirements, and

(2) require that all addressees provide to the NRC a written response in accordance with the provisions of 10 CFR 50.54(f).

Background

The recent discoveries of cracked and leaking Alloy 600 VHP nozzles, including control rod drive mechanism (CRDM) and thermocouple nozzles, at four pressurized water reactors (PWRs) have raised concerns about the structural integrity of VHP nozzles throughout the PWR industry. Nozzle cracking at Oconee Nuclear Station Unit 1 (ONS1) in November 2000 and Arkansas Nuclear One Unit 1 (ANO1) in February 2001 was limited to axial cracking, an occurrence deemed to be of limited safety concern in the NRC staff's generic safety evaluation on the cracking of VHP nozzles, dated November 19, 1993. However, the discovery of circumferential cracking at Oconee Nuclear Station Unit 3 (ONS3) in February 2001 and Oconee Nuclear Station Unit 2 (ONS2) in April 2001 **S** particularly the large circumferential cracking identified in two CRDM nozzles at ONS3 **S** has raised concerns about the potential safety implications and prevalence of cracking in VHP nozzles in PWRs.

Attachment

As described in NRC Information Notice (IN) 2001-05, “Through-Wall Circumferential Cracking of Reactor Pressure Vessel Head Control Rod Drive Mechanism Penetration Nozzles at Oconee Nuclear Station, Unit 3,” dated April 30, 2001, Duke Energy Corporation (the licensee) performed a visual examination (VT-2) on the outer surface of the reactor pressure vessel (RPV) head at ONS3 to inspect for indications of borated water leakage, as part of normal surveillance during a planned maintenance outage. This visual examination followed cleaning of the RPV head during the prior outage to remove all existing boric acid deposits (from other sources such as leaking CRDM flanges) that could mask the identification of subsequent deposits that would be indicative of new or ongoing leakage. The VT-2 examination revealed small amounts of boric acid deposits (less than 1 cubic inch) at locations where the CRDM nozzles exit the RPV head for 9 of the 69 CRDM nozzles. Subsequent nondestructive examination (NDE) identified 47 recordable crack indications in the 9 degraded CRDM nozzles. The licensee initially characterized these flaws as being axial and a part of the RPV pressure boundary, or below-the-weld circumferential indications (which are not part of the RPV pressure boundary), and initiated repairs of the degraded areas.

Subsequent dye-penetrant testing (PT) of the repaired areas revealed the presence of additional indications in two of the nine degraded nozzles. While repairing the indications in these two nozzles, the licensee found that each nozzle had a circumferential crack that extended about 165^{o} around the nozzle, above the weld (i.e., at a location that is part of the RPV pressure boundary). Further investigation and metallurgical examination identified that these cracks had initiated from the outside diameter (OD) of the CRDM nozzles. The circumferential crack in one of the nozzles was through-wall, and the crack in the other nozzle had pin hole indications on the nozzle inside diameter (ID). These cracks followed the contour of the weld profile.

The licensee stated that pre-repair ultrasonic testing (UT) examinations had identified indications in these areas, but that these indications had been misinterpreted as inconsequential craze cracking with unusual characteristics. The characterizations of these two nozzle indications were subsequently revised following the initial post-repair PT examinations. The licensee concluded that the root cause of the CRDM nozzle cracking was primary water stress corrosion cracking (PWSCC). The cracking initiated at the OD of the nozzles after cracking of the J-groove weld (see below) or adjacent heat-affected zone metal permitted coolant leakage into the annular region between the CRDM nozzle and the RPV head. This conclusion was based on metallurgical examinations, crack location and orientation, and finite element analyses.

The CRDM nozzles at ONS3 are approximately 5 feet long and are J-groove welded to the inner radius of the RPV head, with the lower end of each nozzle extending about 6 inches below the inside of the RPV head (see Attachment). The nozzles are constructed from 4-inch OD Alloy 600 Inconel procured in accordance with the requirements of Specification SB-167 to the 1965 Edition, including Addenda through the Summer 1967 Addenda, of Section II of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code. The weld preparation for the installation of each nozzle in the RPV head was accomplished by machining and buttering the J-groove with Alloy 182 weld metal. The RPV head was subsequently stress relieved and then the final machining of the CRDM penetrations, including the counterbore, was accomplished. Each nozzle was then machined to final dimensions to assure the appropriate design interference fit between the RPV head bore and the OD of the nozzle. The interference fit of the CRDM nozzles was made using a shrink fit process to install the CRDM nozzles. In this process, the nozzles were cooled to at least -140EF; they were then inserted into the closure head penetration, and the entire assembly was allowed to warm to room temperature (70EF minimum). The CRDM nozzles were tack welded and then permanently welded to the closure head using Alloy 182 weld metal. The manual shielded metal arc welding (SMAW) process was used for both the tack weld and the J-groove weld. During weld buildup, the weld was ground and PT inspected at each 9/32 inch of the weld. The final weld surface was ground and PT inspected.

The design and fabrication process for the VHPs in all PWR plants is similar to that described for ONS3.

Since the issuance of NRC IN 2001-05, circumferential cracking was identified in another CRDM nozzle, at ONS2. During a visual examination of the RPV head, Duke Energy Corporation identified boric acid deposits in the vicinity of four CRDM nozzles at ONS2. Subsequent UT examination identified a single CRDM nozzle with one OD-initiated circumferential crack, having a crack depth of 0.070 inch (-11% through-wall) and a length of 1.26 inches (-10% of the circumference).

Cracking due to PWSCC in PWR CRDM nozzles and other VHP nozzles fabricated from Alloy 600 is not a new issue; axial cracking in the CRDM nozzles has been identified since the late 1980s. In addition, numerous small-bore Alloy 600 nozzles and pressurizer heater sleeves have experienced leaks attributable to PWSCC. Generally, these components are exposed to high temperatures (greater than 550EF) and a primary water environment. However, circumferential cracking from the nozzle OD to the ID, above the weld, and cracking of the J-groove weld have not been previously identified in PWRs.

As described in Generic Letter (GL) 97-01, “Degradation of Control Rod Drive Mechanism Nozzle and Other Vessel Closure Head Penetrations,” dated April 1, 1997, an action plan was implemented by the NRC staff in 1991 to address PWSCC of Alloy 600 VHP nozzles at all operating U.S. PWRs. After reviewing safety assessments submitted by the industry and examining overseas inspection findings, the NRC staff concluded in its generic safety evaluation that CRDM nozzle and weld cracking in PWRs was not an immediate safety concern. The basis for this conclusion was that if PWSCC occurred (1) the cracks would be predominately axial in orientation, (2) the axial cracks would result in detectable leakage before catastrophic failure (with the expectation that CRDM nozzle cracking would result in a substantial volume of leaking coolant) and (3) the expected large amount of leakage would be detected during visual examinations performed as part of surveillance walkdown inspections before significant damage to the RPV head occurred. The safety evaluation identified concerns about potential circumferential cracking (which would need to be addressed on a plant-specific basis) as a consequence of high residual stresses resulting from initial manufacture and the impact of tube straightening that may have been needed after welding. The safety evaluation also noted the need for enhanced leakage monitoring.

The generic responses of licensees to GL 97-01 were predicated on the development of susceptibility ranking models to relate the operating conditions (in particular the operating temperature and time) for each plant to the plant's relative susceptibility to PWSCC. The generic responses committed to surface examinations of the VHP nozzles at the plants identified as having the highest relative susceptibility ranking. Consistent with the expectations expressed by the NRC staff in GL 97-01, the surface examinations conducted prior to November 2000 identified only limited axial cracking, and circumferential cracking below the weld in the base metal of CRDM nozzles, but no circumferential cracking above the nozzle welds and no cracking in the Alloy 182 welds.

Discussion

The recent identification of circumferential cracking in CRDM nozzles at ONS2 and ONS3, along with axial cracking in the J-groove welds at these two units and at ONS1 and ANO1, has resulted in the staff reassessing its conclusion in GL 97-01 that cracking of VHP nozzles is not an immediate safety concern. Specifically, the findings indicate that circumferential cracks outside of the J-groove welds can occur, in contrast to an earlier conclusion that the cracks would be predominantly axial in orientation. The findings indicate that cracking of the J-groove weld metal can precede cracking of the base metal. These findings raise questions regarding the industry approach, developed in generic responses to GL 97-01, that utilizes PWSCC susceptibility modeling based on the base metal conditions and do not consider those of the weld metal. In addition, the presence of circumferential cracking at ONS3, where only a small amount of boric acid residue indicated a problem, calls into question the adequacy of current visual examinations for detecting either axial or circumferential cracking in VHP nozzles. This is especially significant if prior existing boric acid deposits on the RPV head mask the identification of new deposits. Also, the presence of insulation on the RPV head or other impediments may restrict an effective visual examination. As a remedial measure, the RPV head may have to be cleaned at a prior outage for effective identification of new deposits from VHP nozzle cracking if new deposits cannot be discriminated from existing deposits from other sources. However, the NRC staff believes that boric acid deposits that cannot be dispositioned as coming from another source should be considered, as a conservative assumption, to be from VHP nozzles, and appropriate corrective actions may be necessary. In addition, the use of special tooling or procedures may be required to provide assurance that the visual examinations will be effective in detecting the relevant conditions.

One function of VHP nozzles is to maintain the reactor coolant system pressure boundary. The CRDM nozzles support and guide the control rods, and, therefore, are relied upon in shutting down the reactor. Cracking of CRDM nozzles and welds is a degradation of the reactor coolant system pressure boundary. Industry experience has shown that Alloy 600 is susceptible to stress corrosion cracking. Further, the findings at ONS2 and ONS3 highlight the possible existence of a more aggressive environment in the CRDM housing annulus following through-wall leakage; potentially highly concentrated borated primary water could become oxygenated in this annulus and possibly cause increased propensity for the initiation of cracking and higher crack growth rates.

The cracking identified at ONS2 and ONS3 reinforces the importance of conducting effective examinations of the RPV upper head area (e.g., visual under-the-insulation examinations of the penetrations for evidence of borated water leakage, or volumetric examinations of the CRDM nozzles), and using appropriate NDE methods (such as PT, UT, and eddy-current testing) to adequately characterize cracks. Because of plant-specific design characteristics, there is no uniform way to perform effective visual examinations of the RPV head at PWR facilities. Some plants have the head insulation sufficiently offset from the RPV head to permit an effective visual examination. Other plants have the insulation offset from the head but in a contour matching that of the head, requiring special tooling and procedures to perform an effective visual examination. Still other plants have insulation directly adjacent to or attached to the RPV head, potentially requiring the removal of the insulation to permit an effective visual examination. Several licensees have recently performed expanded VT-2 examinations using remote devices to inspect between the RPV head and the insulation. One aspect of conducting effective visual examinations that is common to all PWR plants is the need to successfully distinguish boric acid deposits originating with VHP nozzle cracking from deposits that are attributable to other sources.

For boric acid deposits from CRDM nozzle cracks to be detectable at the outer surface of the RPV head, sufficient reactor coolant has to leak through the primary pressure boundary into the annulus between the CRDM nozzle and the RPV head base metal, propagate up the annulus, and finally emerge onto the outer surface of the RPV head. Since PWSCC cracks in Alloy 600 and Alloy 182 welds are very tight, leakage from axial cracks in the nozzle and their associated welds is expected to be small. In addition, possible restraint of pressure-induced bending of circumferential cracks in CRDM nozzles could minimize the leakage available even from CRDM nozzles with large circumferential cracks, as evidenced by small boric acid deposits identified at ONS3. As described in Electric Power Research Institute (EPRI) Report TP-1001491, Part 2, “PWR Materials Reliability Program Interim Alloy 600 Safety Assessments for US PWR Plants (MRP-44), Part 2: Reactor Vessel Top Head Penetrations” (referred to as “the MRP-44, Part 2, report”), the majority of CRDM nozzles are installed into the RPV head with an interference fit at room temperature, with 43 plants having specified interference fit ranges greater than those at ONS and ANO1. Should these interference fits persist at plant operating conditions, they could provide an impediment to the flow of coolant leakage up the annulus and thereby limit the amount of deposit available on the RPV head for detection by visual examination.

The recently identified CRDM nozzle degradation phenomena raise several issues regarding the resolution approach taken in GL 97-01:

(1) Cracking of Alloy 182 weld metal has been identified in CRDM nozzle J-groove welds for the first time. This finding raises an issue regarding the adequacy of cracking susceptibility models based only on the base metal conditions.

(2) The identification of cracking at ANO1 raises an issue regarding the adequacy of the industry's GL 97-01 susceptibility model. ANO1 cracking was predicted to be more than 15 effective full power years (EFPY) beyond January 1, 1997, from reaching the same conditions as the limiting plant, based on the susceptibility models used by the industry to address base metal cracking in response to GL 97-01.

(3) Circumferential cracking of CRDM nozzles, located outside of any structural retaining welds, has been identified for the first time. This finding raises concerns about the potential for rapidly propagating failure of CRDM nozzles and control rod ejection, causing a loss of coolant accident (LOCA).

(4) Circumferential cracking from the CRDM nozzle OD to the ID has been identified for the first time. This finding raises concerns about increased consequences of secondary effects of leakage from relatively benign axial cracks.

(5) Circumferential cracking of CRDM nozzles was identified by the presence of relatively small amounts of boric acid deposits. This finding increases the need for more effective inspection methods to detect the presence of degradation in CRDM nozzles before the nozzle integrity is compromised.

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 plants. 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 approach taken in the MRP-44, Part 2, report uses an assessment of the relative susceptibility of each PWR to OD-initiated or weld 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 equivalent to ONS3 at the time the above-weld circumferential cracks were identified in early 2001. To address the experience at ONS, the report 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.

The NRC staff provided questions to the MRP on various aspects of the MRP-44, Part 2, report in a letter dated June 22, 2001; the MRP provided responses in a letter dated June 29, 2001. These questions addressed aspects of the proposed industry treatment that the NRC staff did not agree with. Two specific areas of concern are (1) the finding that nozzle leaks are detectable on all vessel heads, and (2) the lack of consideration of an applicable crack growth rate for the VHP nozzle cracking situation (including a conclusion in the MRP responses that the appropriate crack growth rate for OD cracking of VHP nozzles is represented by data from a primary water environment). The issue of detectibility of nozzle leaks in any particular plant is difficult to address due to a need for plant-specific as-built geometries, such as measured dimensions on CRDM nozzles and RPV penetrations to characterize the interference fit population for a particular RPV head. In addition, there is a need to provide a sufficiently detailed model of the RPV head and expected through-wall crack characteristics, such as surface roughness and crack tightness, to provide assurance that any nozzles with through-wall cracking will provide sufficient leakage to the RPV head surface such that residual deposits of boric acid will provide a detectable condition for the visual examination. An inability to provide assurance of a detectable residual deposit or to discriminate prior existing boric acid deposits caused by non-safety-significant sources from boric acid deposits caused by CRDM nozzle cracking could limit the effectiveness of visual examinations.

Because visual examination of the RPV head or volumetric examination of the VHP nozzles occurs only periodically (generally at a scheduled refueling outage), the issue of crack growth rate in VHP nozzles is an important consideration in providing assurance that VHP nozzles will maintain their structural integrity between examination opportunities. In particular, crack growth should be low enough to ensure that VHP nozzles which are determined to be unflawed during an examination do not have critical flaw sizes prior to the next scheduled examination.

From the results of the susceptibility ranking model proposed in Appendix B to MRP-44, Part 2, the population of PWR plants can be divided into several subpopulations with similar characteristics:

- 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.

Although the industry susceptibility ranking model has limitations, such as large uncertainties and no predictive capability, the model does provide a starting point for assessing the potential for VHP nozzle cracking in PWR plants.

The following paragraphs characterize the gradation of inspection effort for the subpopulations of plants noted above. Nevertheless, addressees should 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. In addition, since inspection and repair activities can potentially result in large personnel exposures, licensees should ensure that all activities related to the inspection of VHP nozzles and the repair of identified degradation are planned and implemented to keep personnel exposures as low as reasonably achievable (ALARA), consistent with the NRC ALARA policy.

For the subpopulation of plants considered to have a low susceptibility to PWSCC, based upon a susceptibility ranking of more than 30 EFPY from the ONS3 condition, the anticipated low likelihood of PWSCC degradation at these facilities indicates that enhanced examination beyond the current requirements is not necessary at the present time because there is a low likelihood that the enhanced examination would provide additional evidence of the propensity for PWSCC in VHP nozzles.

For the subpopulation of plants considered to have a moderate susceptibility to PWSCC based upon a susceptibility ranking of more than 5 EFPY but less than 30 EFPY from the ONS3 condition, an effective visual examination, at a minimum, of 100% of the VHP nozzles that is capable of detecting and discriminating small amounts of boric acid deposits from VHP nozzle leaks, such as were identified at ONS2 and ONS3, may be sufficient to provide reasonable confidence that PWSCC degradation would be identified prior to posing an undue risk.

For the subpopulation of plants considered to have a high susceptibility to PWSCC based upon a susceptibility ranking of less than 5 EFPY from the ONS3 condition, the possibility of VHP nozzle cracking at one of these facilities indicates the need to use a qualified visual examination of 100% of the VHP nozzles. This visual examination should be able to reliably detect and accurately characterize any leakage from cracking in VHP nozzles. This assurance could be provided through a plant-specific demonstration that any VHP nozzle exhibiting through-wall cracking will provide sufficient leakage to the RPV head surface (based on the as-built configuration of the VHPs), and that the effectiveness of the visual examination is not compromised by the presence of insulation, preexisting deposits on the RPV 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% 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 subpopulation of plants which have already identified the existence of PWSCC in the CRDM nozzles (for example, through the detection of boric acid deposits), there is a sufficient likelihood that the cracking of VHP nozzles will continue to occur as the facilities continue to operate. 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) may be appropriate to provide evidence of the structural integrity of the VHP nozzles.

The NRC has developed a Web page to keep the public informed of generic activities on PWR Alloy 600 weld cracking (http://www.nrc.gov/NRC/REACTOR/ALLOY-600/index.html). This page provides links to information regarding the cracking identified to date, along with documentation of NRC interactions with industry (industry submittals, meeting notices, presentation materials, and meeting summaries). The NRC will continue to update this Web page as new information becomes available.

Applicable Regulatory Requirements

Several provisions of the NRC regulations and plant operating licenses (Technical Specifications) pertain to the issue of VHP nozzle cracking. The general design criteria (GDC) for nuclear power plants (Appendix A to 10 CFR Part 50), or, as appropriate, similar requirements in the licensing basis for a reactor facility, the requirements of 10 CFR 50.55a, and the quality assurance criteria of Appendix B to 10 CFR Part 50 provide the bases and requirements for NRC staff assessment of the potential for and consequences of VHP nozzle cracking.

The applicable GDC include GDC 14, GDC 31, and GDC 32. GDC 14 specifies that the reactor coolant pressure boundary (RCPB) have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture; the presence of cracked and leaking VHP nozzles is not consistent with this GDC. GDC 31 specifies that the probability of rapidly propagating fracture of the RCPB be minimized; the presence of cracked and leaking VHP nozzles is not consistent with this GDC. GDC 32 specifies that components which are part of the RCPB have the capability of being periodically inspected to assess their structural and leaktight integrity; inspection practices that do not permit reliable detection of VHP nozzle cracking are not consistent with this GDC.

NRC regulations at 10 CFR 50.55a state that ASME Class 1 components (which include VHP nozzles) must meet the requirements of Section XI of the ASME Boiler and Pressure Vessel Code. Table IWA-2500-1 of Section XI of the ASME Code provides examination requirements for VHP nozzles and references IWB-3522 for acceptance standards. IWB-3522.1(c) and (d) specify that conditions requiring correction include the detection of leakage from insulated components and discoloration or accumulated residues on the surfaces of components, insulation, or floor areas which may reveal evidence of borated water leakage, with leakage defined as “the through-wall leakage that penetrates the pressure retaining membrane.” Therefore, 10 CFR 50.55a, through its reference to the ASME Code, does not permit through-wall cracking of VHP nozzles.

For through-wall leakage identified by visual examinations in accordance with the ASME Code, acceptance standards for the identified degradation are provided in IWB-3142. Specifically, supplemental examination (by surface or volumetric examination), corrective measures or repairs, analytical evaluation, and replacement provide methods for determining the acceptability of degraded components.

Criterion IX of Appendix B to 10 CFR Part 50 states that special processes, including nondestructive testing, shall be controlled and accomplished by qualified personnel using qualified procedures in accordance with applicable codes, standards, specifications, criteria, and other special requirements. Within the context of providing assurance of the structural integrity of VHP nozzles, special requirements for visual examination would generally require the use of a qualified visual examination method. Such a method is one that a plant-specific analysis has demonstrated will result in sufficient leakage to the RPV head surface for a through-wall crack in a VHP nozzle, and that the resultant leakage provides a detectable deposit on the RPV head. The analysis would have to consider, for example, the as-built configuration of the VHPs and the capability to reliably detect and accurately characterize the source of the leakage, considering the presence of insulation, preexisting deposits on the RPV head, and other factors that could interfere with the detection of leakage. Similarly, special requirements for volumetric examination would generally require the use of a qualified volumetric examination method, for example, one that has a demonstrated capability to reliably detect cracking on the OD of the VHP nozzle above the J-groove weld.

Criterion V of Appendix B to 10 CFR Part 50 states that activities affecting quality shall be prescribed by documented instructions, procedures, or drawings, of a type appropriate to the circumstances and shall be accomplished in accordance with these instructions, procedures, or drawings. Criterion V further states that instructions, procedures, or drawings shall include appropriate quantitative or qualitative acceptance criteria for determining that important activities have been satisfactorily accomplished. Visual and volumetric examinations of VHP nozzles are activities that should be documented in accordance with these requirements.

Criterion XVI of Appendix B to 10 CFR Part 50 states that measures shall be established to assure that conditions adverse to quality are promptly identified and corrected. For significant conditions adverse to quality, the measures taken shall include root cause determination and corrective action to preclude repetition of the adverse conditions. For cracking of VHP nozzles, the root cause determination is important to understanding the nature of the degradation present and the required actions to mitigate future cracking. These actions could include proactive inspections and repair of degraded VHP nozzles.

Plant technical specifications pertain to the issue of VHP nozzle cracking insofar as they require no through-wall reactor coolant system leakage.

<u>Requested Information</u>

This bulletin requests addressees to submit information. Addressees who choose to utilize the analyses provided in the MRP-44, Part 2, report or similar analyses need to consider the NRC staff questions relative to this report (provided to the MRP by letter dated June 22, 2001) when preparing their plant-specific responses to the requested information. Addressees should note that the NRC staff has found that the industry response to these questions (provided by letter dated June 29, 2001) does not provide a sufficient basis for resolving the relevant technical issues and that additional information will be necessary to support the plant-specific evaluations.

Addressees are requested to provide the information within 30 days of the date of this bulletin (except for Item 5).

1. All addressees are requested to provide the following information:

   a. The plant-specific susceptibility ranking for your plant(s) (including all data used to determine each ranking) using the PWSCC susceptibility model described in Appendix B to the MRP-44, Part 2, report.

   b. A description of the VHP nozzles in your plant(s), including the number, type, inside and outside diameter, materials of construction, and the minimum distance between VHP nozzles.

   c. A description of the RPV head insulation type and configuration.

   d. A description of the VHP nozzle and RPV head inspections (type, scope, qualification requirements, and acceptance criteria) that have been performed at your plant(s) in the past 4 years, and the findings. Include a description of any limitations (insulation or other impediments) to accessibility of the bare metal of the RPV head for visual examinations.

   e. A description of the configuration of the missile shield, the CRDM housings and their support/restraint system, and all components, structures, and cabling from the top of the RPV head up to the missile shield. Include the elevations of these items relative to the bottom of the missile shield.

2. If your plant has previously experienced either leakage from or cracking in VHP nozzles, addressees are requested to provide the following information:

   a. A description of the extent of VHP nozzle leakage and cracking detected at your plant, including the number, location, size, and nature of each crack detected.

   b. A description of the additional or supplemental inspections (type, scope, qualification requirements, and acceptance criteria), repairs, and other corrective actions you have taken in response to identified cracking to satisfy applicable regulatory requirements.

   c. Your plans for future inspections (type, scope, qualification requirements, and acceptance criteria) and the schedule.

   d. How the inspections identified in 2.c will assure that regulatory requirements are met (see Applicable Regulatory Requirements section). Include the following specific information in this discussion:

      (1) If your future inspection plans do not include performing inspections before December 31, 2001, provide your basis for concluding that the regulatory requirements discussed in the Applicable Regulatory Requirements section will continue to be met until the inspections are performed.

      (2) If your future inspection plans do not include volumetric examination of all VHP nozzles, provide your basis for concluding that the regulatory requirements discussed in the Applicable Regulatory Requirements section will be satisfied.

3. If the susceptibility ranking for your plant is within 5 EFPY of ONS3, provide the following information:

   a. Your plans for future inspections (type, scope, qualification requirements, and acceptance criteria) and the schedule.

   b. How the inspections identified in 3.a. will assure that regulatory requirements are met (see Applicable Regulatory Requirements section). Include the following specific information in this discussion:

      (1) If your future inspection plans do not include performing inspections before December 31, 2001, provide your basis for concluding that the regulatory requirements discussed in the Applicable Regulatory Requirements section will continue to be met until the inspections are performed.

      (2) If your future inspection plans include only visual inspections, discuss the corrective actions that will be taken, including alternative inspection methods (for example, volumetric examination), if leakage is detected.

4. If the susceptibility ranking for your plant is greater than 5 EFPY and less than 30 EFPY of ONS3, addressees are requested to provide the following information:

   a. Your plans for future inspections (type, scope, qualification requirements, and acceptance criteria) and the schedule.

   b. How the inspections identified in 4.a will assure that regulatory requirements are met (see Applicable Regulatory Requirements section). Include the following specific information in this discussion:

      (1) If your future inspection plans do not include a qualified visual examination at the next scheduled refueling outage, provide your basis for concluding that the regulatory requirements discussed in the Applicable Regulatory Requirements section will continue to be met until the inspections are performed.

      (2) The corrective actions that will be taken, including alternative inspection methods (for example, volumetric examination), if leakage is detected.

5. Addresses are requested to provide the following information within 30 days after plant restart following the next refueling or scheduled maintenance outage:

   a. A description of the extent of VHP nozzle leakage and cracking detected at your plant, including the number, location, size, and nature of each crack detected

b. If cracking is identified, a description of the inspections (type, scope, qualification requirements, and acceptance criteria), repairs, and other corrective actions you have taken to satisfy applicable regulatory requirements. This information is requested only if there are any changes from prior information submitted in accordance with this bulletin.

Required Response

In accordance with 10 CFR 50.54(f), in order to determine whether any license should be modified, suspended, or revoked, each addressee is required to respond as described below. This information is sought to verify licensee compliance with the current licensing basis for the facilities covered by this bulletin.

Within 30 days of the date of this bulletin, each addressee is required to submit a written response indicating (1) whether the requested information will be submitted and (2) whether the requested information will be submitted within the requested time period. Addressees who choose not to submit the requested information, or are unable to satisfy the requested completion date, must describe in their response any alternative course of action they propose to take, including the basis for the acceptability of the proposed alternative course of action.

The required written response should be addressed to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001, under oath or affirmation under the provisions of Section 182a of the Atomic Energy Act of 1954, as amended, and 10 CFR 50. 54(f). In addition, submit a copy of the response to the appropriate regional administrator.

Reasons for Information Request

Through-wall cracking of VHP nozzles violates NRC regulations and plant technical specifications. Circumferential cracking of VHP nozzles can pose a safety risk if permitted to progress to the point that nozzle integrity is in question and the risk of a loss of coolant accident or probability of a VHP nozzle ejection increases. This information request is necessary to permit the assessment of plant-specific compliance with NRC regulations. This information will also be used by the NRC staff to determine the need for and to guide the development of additional regulatory actions to address cracking in VHP nozzles. Such regulatory actions could include regulatory requirements for augmented inspection programs under 10 CFR 55a(g)(6)(ii) or additional generic communication.

Related Generic Communications

- Information Notice 2001-05, “Through-Wall Circumferential Cracking of Reactor Pressure Vessel Head Control Rod Drive Mechanism Penetration Nozzles at Oconee Nuclear Station, Unit 3,” April 30, 2001. [ADAMS Accession No. ML011160588]

- Generic Letter 97-01, “Degradation of Control Rod Drive Mechanism Nozzle and Other Vessel Closure Head Penetrations,” April 1, 1997.

- Information Notice 96-11, “Ingress of Demineralizer Resins Increases Potential for Stress Corrosion Cracking of Control Rod Drive Mechanism Penetrations,” February 14, 1996.

- Information Notice 90-10, “Primary Water Stress Corrosion Cracking of INCONEL 600,” February 23, 1990.

- Generic Letter 88-05, “Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants,” March 17, 1988.

- NUREG/CR-6245, ”Assessment of Pressurized Water Reactor Control Rod Drive Mechanism Nozzle Cracking,” October 1994.

<u>Backfit Discussion</u>

Under the provisions of Section 182a of the Atomic Energy Act of 1954, as amended, and 10 CFR 50.54(f), this generic letter transmits an information request for the purpose of verifying compliance with existing applicable regulatory requirements (see the Applicable Regulatory Requirements section of this bulletin). Specifically, the requested information will enable the NRC staff to determine whether current inspection practices for the detection of cracking in the VHP nozzles at reactor facilities provide reasonable confidence that reactor coolant pressure boundary integrity is being maintained. The requested information will also enable the NRC staff to determine whether addressee inspection practices need to be augmented to ensure that the safety significance of VHP nozzle cracking remains low. No backfit is either intended or approved by the issuance of this bulletin, and the staff has not performed a backfit analysis.

<u>*Federal Register* Notification</u>

A notice of opportunity for public comment on this bulletin was not published in the *Federal Register* because the NRC staff is requesting information from power reactor licensees on an expedited basis for the purpose of assessing compliance with existing applicable regulatory requirements and the need for subsequent regulatory action. This bulletin was prompted by the discovery of circumferential cracking in CRDM nozzles (above the nozzle-to-vessel head weld) from the OD to the ID and cracking in the J-groove weld metal itself. Both of these phenomena have not been previously identified in PWRs. As the resolution of this matter progresses, the opportunity for public involvement will be provided.

Paperwork Reduction Act Statement

This bulletin contains information collections that are covered by the Office of Management and Budget, approval number 3150-0012. The burden of this mandatory information collection is estimated to average 140 hours per response, including the time for reviewing instructions, searching existing data sources, gathering and maintaining the data needed, and completing and reviewing the collection of information. Send comments regarding this burden estimate or on any aspect of this collection of information, including suggestions for reducing this burden, to the U.S. Nuclear Regulatory Commission, Information and Records Management Branch, T-6E6, Washington, DC 20555-0001, or by Internet electronic mail to BJS1@NRC.GOV ; and to the Office of Management and Budget, Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202 (3150-0012), Washington, DC 20503.

Public Protection Notification

If a means used to impose an information collection does not display a currently valid OMB control number, the NRC may not conduct or sponsor, and a person is not required to respond to, the information collection.

If you have any questions about this matter, please contact the technical contact listed below or the appropriate Office of Nuclear Reactor Regulation (NRR) project manager.

David B. Matthews, Director
Division of Regulatory Improvement Programs
Office of Nuclear Reactor Regulation

Technical Contact: Allen L. Hiser, Jr., NRR
301-415-1034
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Lead Project Manager: Jacob I. Zimmerman, NRR
301-415-2426
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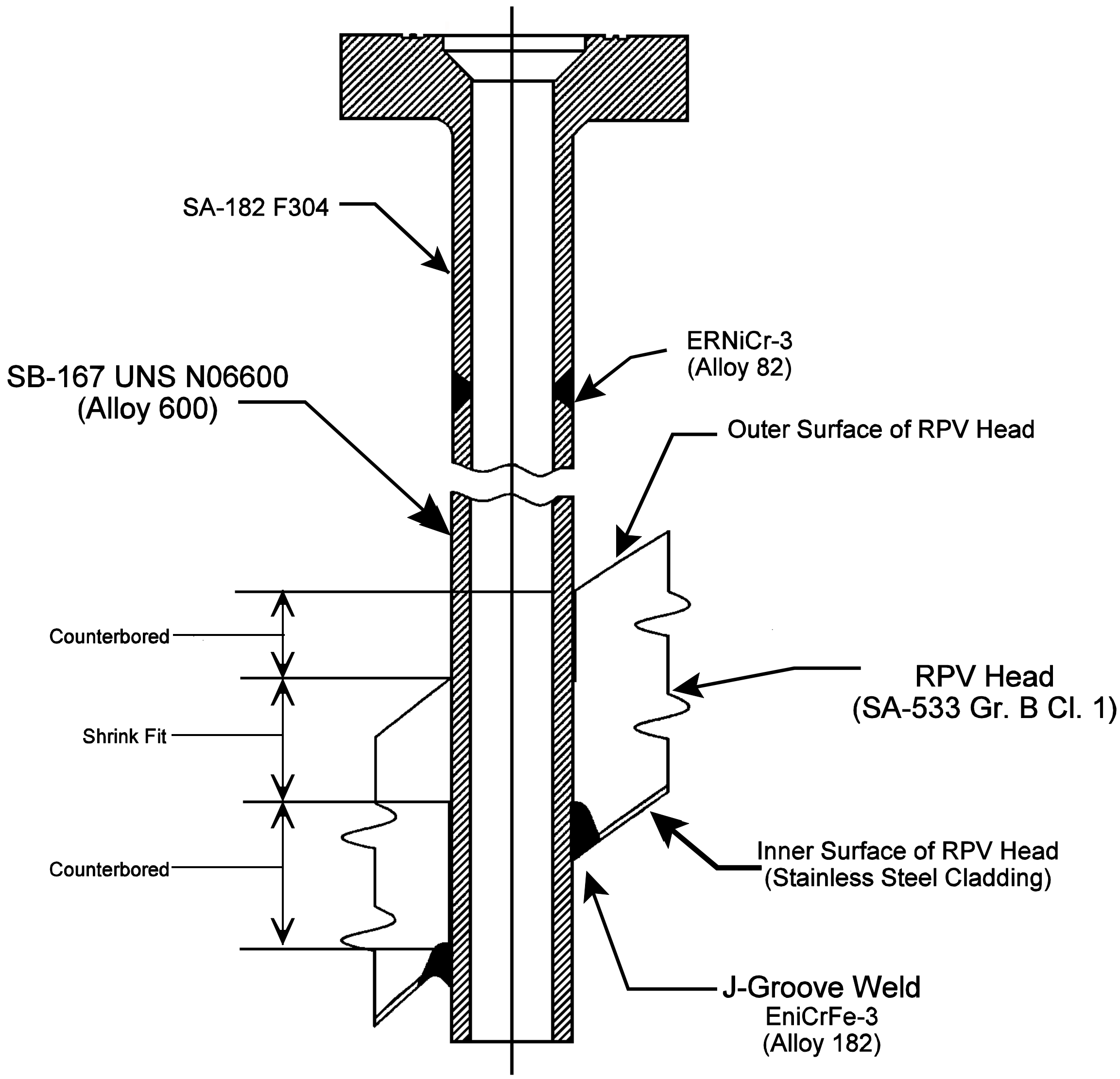
Attachment:
Schematic Figure of Typical CRDM Nozzle Penetration



Schematic Figure of Typical CRDM Nozzle Penetration