June 4, 2002

Mr. Valeri Tolstykh
Regulatory Activities Unit
Safety Assessment Section
Division of Nuclear Installation Safety
International Atomic Energy Agency
Wagramer Strasse 5
P.O. Box 100, A-1400
Vienna, Austria

Dear Mr. Tolstykh:

The following operating experience reports from United States reactors are enclosed for your consideration for including in the AIRS database:

- NRC BULLETIN 2001-01: CIRCUMFERENTIAL CRACKING OF REACTOR PRESSURE VESSEL HEAD PENETRATION NOZZLES

- NRC INFORMATION NOTICE 2002-11: RECENT EXPERIENCE WITH DEGRADATION OF REACTOR PRESSURE VESSEL HEAD

- NRC BULLETIN 2002-01: REACTOR PRESSURE VESSEL HEAD DEGRADATION AND REACTOR COOLANT PRESSURE BOUNDARY INTEGRITY

- NRC INFORMATION NOTICE 2002-13: POSSIBLE INDICATORS OF ONGOING REACTOR PRESSURE VESSEL HEAD DEGRADATION

Each report is being submitted in the following two media: (1) a hard copy of the input file for the AIRS database; and (2) a 3.5-inch HD diskette containing the input file for the AIRS database in Microsoft Word 6.0 format.

If you have any questions regarding these reports, please call Jerry Dozier of my staff. He can be reached at 301-415-1014.

Sincerely,

/RA/

William D. Beckner, Program Director
Operating Reactor Improvements Program
Division of Regulatory Improvement Programs
Office of Nuclear Reactor Regulation

Enclosures: As stated

cc w/Enclosures 1, 2 and 3:
Mr. Lennart Carlsson
Nuclear Safety Division
Nuclear Energy Agency
Organization for Economic
Cooperation and Development
Le Seine Saint Germain
12, Boulevard des Iles
92130, Issy-les-Moulineaux, France

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

August 3, 2001

NRC BULLETIN 2001-01: 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 the basis for concluding that 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 – particularly the large circumferential cracking identified in two CRDM nozzles at ONS3 – has raised concerns about the potential safety implications and prevalence of cracking in VHP nozzles in PWRs.

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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° 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 -140°F; they were then inserted into the closure head penetration, and the entire assembly was allowed to warm to room temperature (70°F 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 550°F) 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 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. 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 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 qualified visual examination should be able to reliably detect and accurately characterize leakage from cracking in VHP nozzles considering two characteristics. One characteristic is 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). Secondly, similar to the effective visual examination for moderate susceptibility plants, 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, 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 requested 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. your basis for concluding that 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, 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. your basis for concluding that 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. your basis for concluding that 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. Addressees are requested to provide the following information within 30 days after plant restart following the next refueling 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.

Backfit Discussion

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.

Federal Register Notification

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 subject to the Paperwork Reduction Act of 1995 (44 U.S.C. 3501 et seq.) These information collections were approved by the Office of Management and Budget, approval number 3150-0011.

The burden to the public for these mandatory information collections is 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 information collection. Send comments regarding this burden estimate or on any other aspect of these information collections, including suggestions for reducing the burden, to the Records Management Branch (T-6 E6), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by Internet electronic mail to BJS1@NRC.GOV; and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202 (3150-0011), Office of Management and Budget, 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.

/RA/

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

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Lead Project Manager: Jacob I. Zimmerman, NRR
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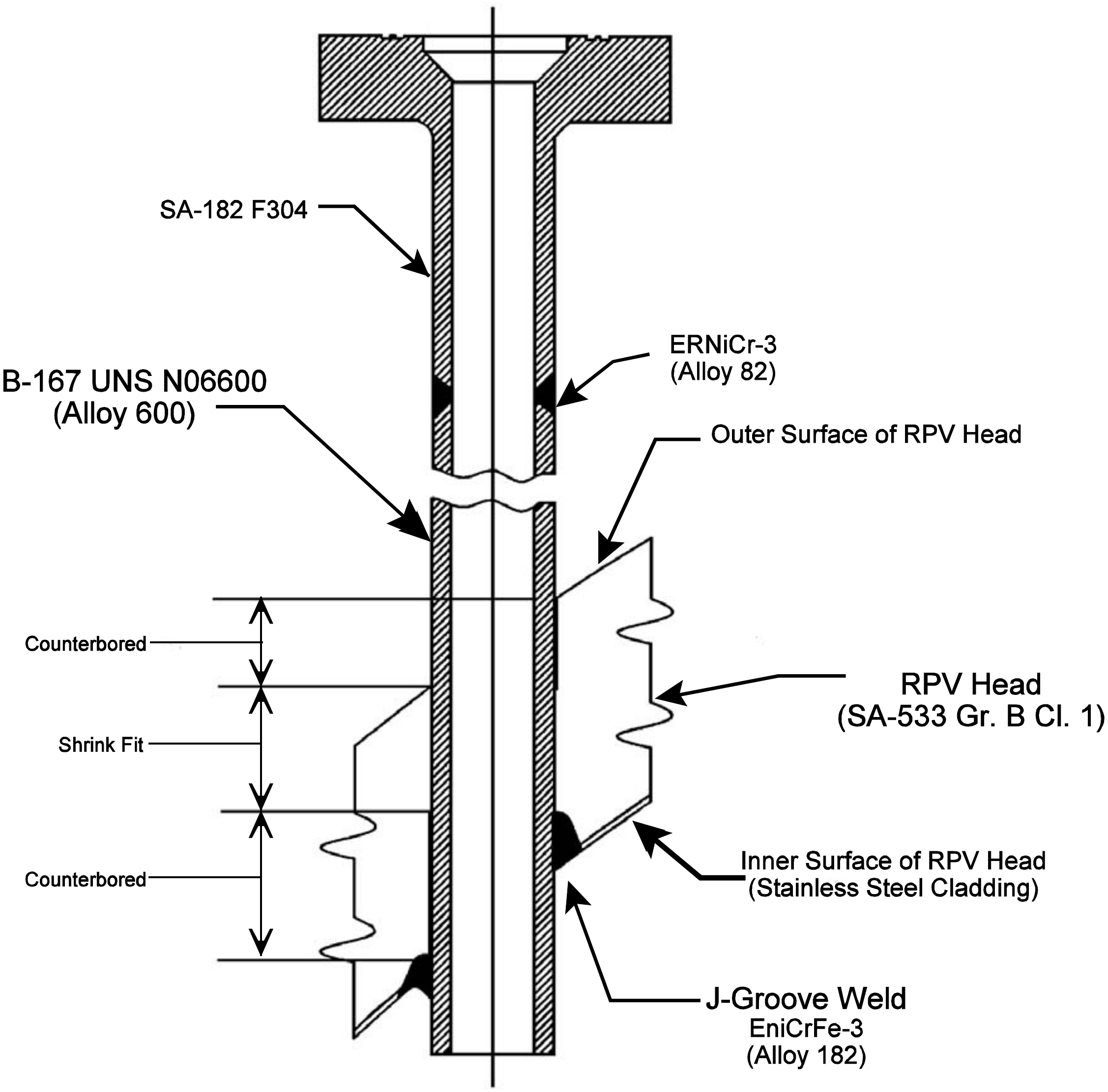
Attachment:
Schematic Figure of Typical CRDM Nozzle Penetration



Schematic Figure of Typical CRDM Nozzle Penetration

UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION
WASHINGTON, DC 20555-0001

March 12, 2002

NRC INFORMATION NOTICE 2002-11: RECENT EXPERIENCE WITH DEGRADATION OF REACTOR PRESSURE VESSEL HEAD

Addressees

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

Purpose

The U.S. Nuclear Regulatory Commission (NRC) is issuing this information notice to inform addressees about findings from recent inspections and examinations of the reactor pressure vessel (RPV) head at Davis-Besse Nuclear Power Station. It is expected that recipients will review the information for applicability to their facilities and consider actions, as appropriate, to avoid similar problems. However, suggestions contained in this information notice are not NRC requirements; therefore, no specific action or written response is required.

Description of Circumstances

On February 16, 2002, the Davis-Besse facility began a refueling outage that included inspection of the vessel head penetration (VHP) nozzles, which focused on the inspection of control rod drive mechanism (CRDM) nozzles, in accordance with the licensee's commitments to NRC Bulletin 2001-01, "Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles," which was issued on August 3, 2001. These inspections identified axial indications in three CRDM nozzles, which had resulted in pressure boundary leakage. Specifically, these indications were identified in CRDM nozzles 1, 2, and 3, which are located near the center of the RPV head. These findings were reported to the NRC on February 27, 2002, and supplemented on March 5 and March 9, 2002. The licensee decided to repair these three nozzles, as well as two other nozzles that had indications but had not resulted in pressure boundary leakage.

The repair process for these nozzles included roll expanding the CRDM nozzle material into the surrounding RPV head material, followed by machining along the axis of the CRDM nozzle to an elevation above the indications in the nozzle material. On March 6, 2002, the machining process on CRDM nozzle 3 was prematurely terminated and the machining apparatus was removed from the nozzle. During the removal process, nozzle 3 was mechanically agitated and subsequently displaced in the downhill direction (i.e., tipped away from the top of the RPV head) until its flange contacted the flange of the adjacent CRDM nozzle.

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To identify the cause of the CRDM nozzle displacement, the licensee began an investigation into the condition of the RPV head surrounding CRDM nozzle 3. This investigation included removing the CRDM nozzle from the RPV head, removing boric acid deposits from the top of the RPV head, and performing ultrasonic thickness measurements of the RPV head in the vicinity of CRDM nozzles 1, 2, and 3. Upon completing the boric acid removal on March 7, 2002, the licensee conducted a visual examination of the area, which identified a large cavity in the RPV head on the downhill side of CRDM nozzle 3. Followup characterization by ultrasonic testing indicated wastage of the low alloy steel RPV head material adjacent to the nozzle. The wastage area was found to extend approximately 5 inches downhill on the RPV head from the penetration for CRDM nozzle 3, with a width of approximately 4 to 5 inches at its widest part. The minimum remaining thickness of the RPV head in the wastage area was found to be approximately ⅜ inch. This thickness was attributed to the thickness of the stainless steel cladding on the inside surface of the RPV head, which is nominally ⅜ inch thick.

<u>Background</u>

The Davis-Besse Nuclear Power Station has an RPV head that is constructed from low alloy steel, fabricated in accordance with the American Society of Mechanical Engineers (ASME) specification SA-533, Grade B, Class1, and clad on the inside surface with stainless steel. Of those 69 VHP nozzles, 61 are used for CRDMs, 7 are spare (empty) nozzles, and 1 is used for the RPV head vent piping. Each of the 69 nozzles is approximately 4 inches in outside diameter, with a wall-thickness of approximately ⅝ inch. Each is constructed of Alloy 600 and is attached to the RPV head by a partial-penetration, J-groove weld using Alloy 82 and 182. The distance from the center of one nozzle to the center of the next is approximately 12 inches.

The vessel head is insulated with metal reflective insulation, which is located on a horizontal plane slightly above the RPV head (i.e., it is not in direct contact with the head). The minimum distance between the RPV head and the insulation is approximately 2 inches at the center (top) of the head. The CRDM nozzles pass from the RPV head through the insulation and terminate at flanges to which the CRDM housings are attached.

The limited gap between the insulation and the RPV head does not impede the performance of a visual inspection of the CRDM nozzles, as described in Bulletin 2001-01. This is because the top of the RPV head is surrounded by a service structure that has 18 openings (referred to as “weep holes”) near the bottom of the structure, through which small cameras can be inserted to facilitate visual inspections of the RPV head.

During refueling outages in 1998 and 2000, the licensee performed visual inspections of the RPV head surface that was accessible through the service structure weep holes. The scope of these visual inspections covered the bare metal of the RPV head to identify the presence of boric acid deposits, which would be indicative of primary coolant leakage. These inspections also included checking for leakage from any of the CRDM flanges, located above the insulation, in response to Generic Letter 88-05, “Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components,” which the NRC issued on March 17, 1988.

The visual inspections in 1998 showed an uneven layer of boric acid deposits scattered over the RPV head (including deposits near CRDM nozzle 3). The outside diameter of the CRDM nozzles had white streaks, which indicated to the licensee that the boric acid evident on the head flowed downward from leakage in the CRDM flanges.

During the refueling outage in 2000, the licensee also performed visual inspections of the CRDM flanges and nozzles. Above the RPV head insulation, those inspections revealed five CRDM flanges with evidence of leakage, including one flange that was the principal leakage point. Boric acid deposits on the vertical faces of three of these five flanges and the associated nozzles confirmed leakage from the flanges. Similarly, one of the other two leaking CRDM flanges had boric acid deposits between the flange and the insulation, which indicated leakage from the flange. All of these leaking flanges were repaired by replacing their gaskets. The faces of the flange that was the principal leakage point were also machined to ensure a better seal.

Visual inspections performed below the RPV head insulation during the 2000 refueling outage indicated some accumulation of boric acid deposits on the RPV head. These deposits were located beneath the leaking flanges, with clear evidence of downward flow from the flange area. No visible evidence of CRDM nozzle leakage (i.e., leakage from the gap between the nozzle and the RPV head) was detected. The licensee described that the RPV head area was cleaned with demineralized water to the greatest extent possible, while trying to maintain the dose as low as reasonably achievable (ALARA). Subsequent video inspection of the partially cleaned RPV head and nozzles was performed for future reference.

A subsequent review of the 1998 and 2000 inspection videotapes in 2001 confirmed that there was no evidence of leakage from the RPV head nozzles, although many areas of the RPV head were not accessible because of persistent boric acid deposits that the licensee did not clean because of ALARA issues (including the region around nozzle 3).

The inspections in 2002 did not reveal any visual evidence of flange leakage from above the RPV head. However, as discussed above, three CRDM nozzles had indications of cracking (identified by ultrasonic testing of the nozzles), which could result in leakage from the RPV to the top of the RPV head.

Discussion

The following documents describe reactor operating experience with boric acid corrosion of ferritic steel reactor coolant pressure boundary components in PWR plants:

- Information Notice 86-108, “Degradation of Reactor Coolant System Pressure Boundary Resulting from Boric Acid Corrosion,” issued December 29, 1986

- Information Notice 86-108, Supplement 1, “Degradation of Reactor Coolant System Pressure Boundary Resulting from Boric Acid Corrosion,” issued April 20, 1987

- Information Notice 86-108, Supplement 2, “Degradation of Reactor Coolant System Pressure Boundary Resulting from Boric Acid Corrosion,” issued November 19, 1987

- Information Notice 86-108, Supplement 3, “Degradation of Reactor Coolant System Pressure Boundary Resulting from Boric Acid Corrosion,” issued January 5, 1995

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

Several instances of boric acid corrosion discussed in these generic communications are associated with corrosion of the RPV head. NRC Information Notice 86-108, Supplement 1, for example, described an instance in which boric acid had severely corroded three of the RPV flange bolts, the control rod drive shroud support, and an instrument tube seal clamp. Similarly, NRC Information Notice 86-108, Supplement 2, described an instance in which boric acid resulted in nine pits in the surface of the RPV head, ranging in depth from 0.9 to 1 cm [approximately 0.4 inch] and ranging in diameter from 2.5 to 7.5 cm [1 to 3 inches].

As discussed in Information Notice 86-108, Supplement 2, the primary effect of boric acid leakage onto the ferritic steel RPV head is wastage or general dissolution of the material. Pitting, stress corrosion cracking (SCC), intergranular attack, and other forms of corrosion are not generally of concern in concentrated boric acid solutions at elevated temperatures such as those that may occur on the surface of the RPV head. The rate of general corrosion (wastage) of ferritic steel from boric acid varies and depends on several conditions, including whether the boric acid is dry or in solution. If the boric acid is dry (i.e., boric acid crystals), the corrosion rate is less severe; however, boric acid crystals are not completely benign to carbon steel. During operation, the temperature of the RPV head is sufficiently high that any leaking primary coolant would be expected to flash to steam, leaving behind dry boric acid crystals.

Given the wide range of conditions around reactor primary coolant leakage sites and the wide variation in boric acid corrosion rates, the deleterious effects of boric acid on ferritic steel components indicate the importance of minimizing boric acid leakage, detecting and correcting leaks in a timely manner, and promptly cleaning any boric acid residue.

The investigation of the causative conditions surrounding the degradation of the RPV head at Davis-Besse is continuing. Boric acid or other contaminants could be contributing factors. As discussed above, factors contributing to the degradation might also include the environment of the head during both operating and shutdown conditions (e.g., wet/dry), the duration for which the RPV head is exposed to boric acid, and the source of the boric acid (e.g., leakage from the CRDM nozzle or from sources above the RPV head such as CRDM flanges).

<u>Related Generic Communications</u>

Bulletin 2001-01, “Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles,” August 3, 2001.

Bulletin 82-02, “Degradation of Threaded Fasteners in the Reactor Coolant Pressure Boundary of PWR Plants,” June 2, 1982.

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

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

Information Notice 80-27, “Degradation of Reactor Coolant Pump Studs,” June 11, 1980.

Information Notice 82-06, “Failure of Steam Generator Primary Side Manway Closure Studs,” March 12, 1982.

Information Notice 86-108, "Degradation of Reactor Coolant System Pressure Boundary Resulting from Boric Acid Corrosion," December 29, 1986.

Information Notice 86-108, Supplement 1, "Degradation of Reactor Coolant System Pressure Boundary Resulting from Boric Acid Corrosion," April 20, 1987.

Information Notice 86-108, Supplement 2, "Degradation of Reactor Coolant System Pressure Boundary Resulting from Boric Acid Corrosion," November 19, 1987.

Information Notice 86-108, Supplement 3, "Degradation of Reactor Coolant System Pressure Boundary Resulting from Boric Acid Corrosion," January 5, 1995.

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

Information Notice 94-63, "Boric Acid Corrosion of Charging Pump Casing Caused by Cladding Cracks," August 30, 1994.

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 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.

This information notice does not require any specific action or written response. If you have any questions about the information in this notice, please contact one of the technical contacts listed below or the appropriate project manager in the NRC's Office of Nuclear Reactor Regulation (NRR).

/RA/
William D. Beckner, Program Director
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Division of Regulatory Improvement Programs
Office of Nuclear Reactor Regulation

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Attachment: List of Recently Issued NRC Information Notices

LIST OF RECENTLY ISSUED
NRC INFORMATION NOTICES

Information Notice No.	Subject	Date of Issuance	Issued to
2002-10	Nonconservative Water Level Setpoints on Steam Generators	03/07/2002	All holders of operating licenses for nuclear power reactors, except those who have permanently ceased operations and have certified that fuel has been permanently removed from the reactor.
2002-09	Potential for Top Nozzle Separation and Dropping of Certain Type of Westinghouse Fuel Assembly	02/13/2002	All holders of operating licenses for nuclear power reactors, and non-power reactors and holders of licenses for permanently shutdown facilities with fuel onsite.
2002-08	Pump Shaft Damage Due to Excessive Hardness of Shaft Sleeve	01/30/2002	All holders of operating licenses for nuclear power reactors, except those who have permanently ceased operations and have certified that fuel has been permanently removed from the reactor.
2002-07	Use of Sodium Hypochlorite for Cleaning Diesel Fuel Oil Supply Tanks	01/28/2002	All holders of operating licenses for nuclear power except those who have ceased operations and have certified that fuel has been permanently removed from the reactor vessel.
2002-06	Design Vulnerability in BWR Reactor Vessel Level Instrumentation Backfill Modification	01/18/2002	All holders of operating licenses or construction permits for boiling water reactors (BWRs).
2002-05	Foreign Material in Standby Liquid Control Storage Tanks	01/17/2002	All holders of licenses for nuclear power reactors.
2002-04	Wire Degradation at Breaker Cubicle Door Hinges	01/10/2002	All holders of operating licenses for nuclear power reactors.

OL = Operating License
CP = Construction Permit

OMB Control No.: 3150-0012

UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION
WASHINGTON, DC 20555-0001

March 18, 2002

NRC BULLETIN 2002-01: REACTOR PRESSURE VESSEL HEAD DEGRADATION AND REACTOR COOLANT PRESSURE BOUNDARY INTEGRITY

Addressees

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

Purpose

The U.S. Nuclear Regulatory Commission (NRC) is issuing this bulletin to require pressurized-water reactor (PWR) addressees to submit:

(1) information related to the integrity of the reactor coolant pressure boundary including the reactor pressure vessel head and the extent to which inspections have been undertaken to satisfy applicable regulatory requirements, and

(1) the basis for concluding that plants satisfy applicable regulatory requirements related to the structural integrity of the reactor coolant pressure boundary and future inspections will ensure continued compliance with applicable regulatory requirements, and

(3) a written response to the NRC in accordance with the provisions of Title 10, Section 50.54(f), of the *Code of Federal Regulations* (10 CFR 50.54(f)) if they are unable to provide the information or they can not meet the requested completion dates.

Background

On August 3, 2001, the NRC issued Bulletin 2001-01, “Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles” (ADAMS Accession Number ML012080284). That bulletin described instances of cracked and leaking Alloy 600 reactor pressure vessel head penetration nozzles, including control rod drive mechanism and thermocouple nozzles. In response to that bulletin, pressurized-water reactor licensees provided their plans for inspecting their reactor pressure vessel head penetrations and/or the outside surface of the reactor pressure vessel head to determine whether the nozzles were leaking. Some plants have completed these inspections.

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In conducting these inspections at the Davis-Besse Nuclear Power Station in February and March 2002, the licensee identified three control rod drive mechanism nozzles with indications of axial cracking that resulted in reactor coolant pressure boundary leakage. One of these three control rod drive mechanism nozzles also had a circumferential indication which was not through-wall, and therefore, did not result in reactor coolant pressure boundary leakage. These were not unexpected findings, given the high susceptibility of the Davis-Besse plant to vessel head penetration nozzle cracking (as described in NRC Bulletin 2001-01). These axial indications were identified in control rod drive mechanism nozzles 1, 2, and 3, which are located near the center of the reactor pressure vessel head. Because of these indications, the licensee decided to repair control rod drive mechanism nozzles 1, 2, and 3, as well as two other nozzles that had indications but had not resulted in reactor coolant pressure boundary leakage.

The repair process for these nozzles included roll expanding the control rod drive mechanism nozzle material into the surrounding reactor pressure vessel head material, followed by machining along the axis of the control rod drive mechanism nozzle to an elevation above the indications in the nozzle material. On March 6, 2002, the machining process on control rod drive mechanism nozzle 3 was prematurely terminated and the machining apparatus was removed from the nozzle. During the removal process, control rod drive mechanism nozzle 3 was mechanically agitated and subsequently displaced, or tipped, in the downhill direction (away from its vertical position on top of the dome-shaped reactor pressure vessel head) until its flange contacted the flange of the adjacent control rod drive mechanism nozzle.

To identify the cause of the control rod drive mechanism nozzle displacement, the licensee began an investigation into the condition of the reactor pressure vessel head surrounding control rod drive mechanism nozzle 3. This investigation included removing the nozzle and boric acid deposits from the reactor pressure vessel head, and ultrasonically measuring the thickness of the reactor pressure vessel head in the vicinity of control rod drive mechanism nozzles 1, 2, and 3. Upon completing the boric acid removal on March 7, 2002, the licensee conducted a visual examination of the area, which identified a cavity in the reactor pressure vessel head on the downhill side of control rod drive mechanism nozzle 3 (i.e., the lowest portion of the nozzle extending out of the reactor pressure vessel head). Follow-up characterization by ultrasonic testing indicated thinning of the reactor pressure vessel head material adjacent to the nozzle. The thinned area was initially estimated to extend approximately 5 inches from the penetration for control rod drive mechanism nozzle 3; however, from more recent results, the thinned area extends approximately 7 inches from the nozzle at the stainless steel cladding, indicating the degradation was more severe at the bottom of the cavity than on the top. The width of the exposed area was approximately 4 to 5 inches at its widest part. The minimum remaining thickness of the reactor pressure vessel head in the thinned area was found to be approximately 3/8-inch. This thickness was attributed to the thickness of the stainless steel cladding on the inside surface of the reactor pressure vessel head, which is nominally 3/8-inch thick.

NRC Information Notice 2002-11, “Recent Experience with Degradation of Reactor Pressure Vessel Head,” dated March 12, 2002, provides additional detail concerning the Davis-Besse inspection findings, the design and configuration of the Davis-Besse reactor pressure vessel head and service structure, and past inspections.

Since the NRC issued Information Notice 2002-11, additional information has become available concerning the condition of the reactor pressure vessel head at Davis-Besse. Specifically, the 3/8-inch stainless steel cladding near control rod drive mechanism nozzle 3 was found to be deflected upwards by about 1/8-inch over a 4-inch distance, indicating that the material had yielded. This is significant because the 3/8-inch cladding had essentially become the reactor coolant pressure boundary near the affected nozzle after the base material of the reactor pressure vessel head had degraded.

In addition, two areas of less severe thinning have been detected near control rod drive mechanism nozzle 2. At the time this bulletin was being prepared, it was not known whether these two areas were connected because one was detected on the outer surface of the reactor pressure vessel head and the other was detected at the inner surface. In addition, the dimensions of these areas were not known at the time this bulletin was being prepared. On the basis of preliminary information, the affected area appeared to be much smaller in size than the area located near control rod drive mechanism nozzle 3.

The investigation of the causative conditions surrounding the degradation of the reactor pressure vessel head at Davis-Besse is continuing. Boric acid or other contaminants could be contributing factors, as could steam jet cutting caused by leakage from the nozzle. Other factors contributing to the degradation might include the environment (e.g., wet/dry) surrounding the reactor pressure vessel head during both operating and shutdown conditions, the duration for which the reactor pressure vessel head was exposed to boric acid, and the source of the boric acid (e.g., leakage from cracks in the reactor pressure vessel head penetration nozzle or from sources above the reactor pressure vessel head such as control rod drive mechanism flanges).

Discussion

The reactor pressure vessel head is an integral part of the reactor coolant pressure boundary, and its integrity is important to the safe operation of the plant. The recent identification of thinning of the reactor pressure vessel head at Davis-Besse raises questions regarding licensees' practices for identifying and resolving degradation of the reactor coolant pressure boundary, including licensees' models for assessing corrosion that is caused by contaminants such as boric acid in the operating environment of the reactor pressure vessel head, or erosion that is caused by flow through a through-wall defect in a vessel head penetration nozzle.

As indicated above, the investigation of the causative conditions surrounding the degradation of the reactor pressure vessel head at Davis-Besse is continuing. An evaluation of the available information leads to several observations. First, the base metal of the reactor pressure vessel head degraded near leaking nozzles. Second, the reactor pressure vessel head has had boric acid deposits in the vicinity of the degraded areas for at least the past several years; that is, the deposits were not fully removed during the last several refueling outages. Third, some of the boric acid deposits on the top of the reactor pressure vessel head came from leaking control rod drive mechanism flanges, as discussed in NRC Information Notice 2002-11. Evaluations are on-going on whether similar degradation could occur (1) with just deposits and/or contaminants on the reactor pressure vessel head (i.e., without a leaking nozzle), (2) with just a leaking nozzle (i.e., without deposits and/or contaminants on the reactor pressure vessel head), or (3) whether both conditions are necessary to cause the observed degree of degradation. That is, the interaction between these two conditions and their respective influences in initiating the degradation of the reactor pressure vessel head is still being evaluated.

Although the root cause is still under investigation, preliminary assessments indicate that boric acid was a contributor. Corrosion of ferritic material, such as the base metal of the reactor pressure vessel head, is well documented in the list of related generic communications identified in this bulletin. In response to NRC Generic Letter 88-05, “Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants,” dated March 17, 1988, licensees committed to implement a systematic program to monitor locations where boric acid leakage could occur, and to implement measures to prevent degradation of the reactor coolant pressure boundary by boric acid corrosion.

Historically, these programs have assumed that there is only a small potential for wastage of the reactor pressure vessel head attributable to leakage of primary coolant through the vessel head penetration nozzles. The supporting analyses assumed that coolant escaping from a penetration would flash to steam, leaving behind deposits of boric acid crystals. Typically, these crystals are assumed to accumulate on the reactor pressure vessel head; however, such deposits are assumed to cause minimal corrosion while the reactor is operating because the temperature of the reactor pressure vessel head is above 500 F during operation, and dry boric acid crystals are not very corrosive. Therefore, wastage is typically expected to occur only during outages when the boric acid could be in solution, such as when the temperature of the reactor pressure vessel head falls below 212 F. However, the findings at Davis-Besse bring into question the reliability of this model.

As indicated above, one of the contributing factors to the observed degradation could be the presence of boric acid deposits on the top of the reactor pressure vessel head. The procedures for determining whether these deposits could be present on the top of the reactor pressure vessel head are plant-specific because they are contingent on plant-specific design characteristics. For example, some plants have the reactor pressure vessel head insulation sufficiently offset from the head itself, in order to allow effective visual examination (as discussed in Bulletin 2001-01). Other plants have the insulation offset from the reactor pressure vessel head, but in a contour matching that of the head itself, in a design that requires special tooling and procedures to perform an effective visual examination. Still other plants have the reactor pressure vessel head insulation directly adjacent or attached to the head itself, in a design that potentially requires the removal of the insulation to permit an effective visual examination.

Plants for which limited data are available from direct visual inspection must use another method to determine whether boric acid deposits could be on the top of the reactor pressure vessel head. One method includes assessing whether boric acid (1) has leaked from locations above the reactor pressure vessel head, (2) has penetrated the insulation by flowing through the insulation or through gaps in the insulation, and (3) has precipitated onto the reactor pressure vessel head or has allowed precipitants to fall onto the reactor pressure vessel head.

One of the other factors suspected of contributing to the degradation observed at Davis-Besse is the presence of a leaking reactor pressure vessel head penetration nozzle. The integrity of reactor pressure vessel head penetration nozzles is discussed in NRC Bulletin 2001-01.

That bulletin discusses an industry model for assessing the susceptibility of plants to primary water stress corrosion cracking at the reactor pressure vessel head penetration nozzles. The industry's susceptibility ranking model has limitations, such as large uncertainties and the inability to predict when cracking will occur. Nonetheless, this model does provide a starting point for assessing the potential for cracking of reactor pressure vessel head penetration nozzles in pressurized water reactor plants.

Inspections performed to date at plants with high and moderate susceptibility have generally confirmed the ability of the model to predict a plant's relative susceptibilities; however, a plant with a ranking of 14.3 effective full-power years from the Oconee 3 condition (at the time when circumferential cracking was identified at Oconee 3 in March 2001) identified three nozzles with cracking; other plants with fewer effective full-power years from the Oconee 3 condition did not identify cracking.

Several plants have repaired nozzles with through-wall degradation (i.e., nozzles that leaked). Results from these inspections do not appear to indicate the presence of a degraded area in the reactor pressure vessel base metal. However, the extent to which the inspection techniques used would have detected such an area or the degree to which attention was placed on identifying this form of degradation, varies from plant to plant. Some inspection and repair methods may not have been capable of identifying the presence of a void in the carbon steel head adjacent to the cladding interface.

The NRC has developed Web pages to keep the public informed of generic activities related to Alloy 600 cracking and reactor pressure vessel head degradation:

http://www.nrc.gov/reactors/operating/ops-experience/alloy600.html
http://www.nrc.gov/reactors/operating/ops-experience/vessel-head-degradation.html

These Web pages provide 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 these Web pages as new information becomes available.

Applicable Regulatory Requirements

Several provisions of the NRC regulations and plant operating licenses (Technical Specifications) pertain to reactor coolant pressure boundary integrity. 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 degradation of the reactor coolant pressure boundary.

The applicable GDC include GDC 14 (Reactor Coolant Pressure Boundary), GDC 31 (Fracture Prevention of Reactor Coolant Pressure Boundary), and GDC 32 (Inspection of Reactor Coolant Pressure Boundary). GDC 14 specifies that the reactor coolant pressure boundary (RCPB) has an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture. GDC 31 specifies that the probability of rapidly propagating fracture of the RCPB be minimized. 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 degradation are not consistent with this GDC.

NRC regulations in 10 CFR 50.55a state that the American Society of Mechanical Engineers (ASME) Class 1 components (which includes the reactor coolant pressure boundary) must meet the requirements of Section XI of the ASME Boiler and Pressure Vessel Code. Various portions of the ASME Code address reactor coolant pressure boundary inspection. For example, Table IWA-2500-1 of Section XI of the ASME Code provides examination requirements for reactor pressure vessel head penetration 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 degradation of the reactor pressure vessel head penetration 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 V (Instructions, Procedures, and Drawings) 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 the reactor coolant pressure boundary are activities that should be documented in accordance with these requirements.

Criterion IX (Control of Special Processes) 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 reactor coolant pressure boundary for the degradation observed at Davis-Besse, special requirements for visual examination and/or ultrasonic testing would generally require the use of qualified visual and ultrasonic testing methods. Such methods are ones that a plant-specific analysis has demonstrated would result in the reliable detection of degradation prior to a loss of specified reactor coolant pressure boundary margins of safety. The analysis would have to consider, for example, the as-built configuration of the system and the capability to reliably detect and accurately characterize flaws or degradation, and contributing factors such as the presence of insulation, preexisting deposits, and other factors that could interfere with the detection of degradation.

Criterion XVI (Corrective Action) 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 degradation of the reactor coolant pressure boundary, the root cause determination is important for understanding the nature of the degradation present and the required actions to mitigate future degradation. These actions could include proactive inspections and repair of degraded portions of the reactor coolant pressure boundary.

Plant technical specifications pertain to this issue insofar as they do not allow operation with known reactor coolant system pressure boundary leakage.

Generic Letter 88-05, "Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants," pertains to this issue in that the staff concluded that in the absence of a program for addressing the corrosive effects of reactor coolant system leakage, compliance with General Design Criteria 14, 30, and 31 cannot be ensured.

Required Information

1. Within 15 days of the date of this bulletin, all PWR addressees are required to provide the following:

A. a summary of the reactor pressure vessel head inspection and maintenance programs that have been implemented at your plant,

B. an evaluation of the ability of your inspection and maintenance programs to identify degradation of the reactor pressure vessel head including, thinning, pitting, or other forms of degradation such as the degradation of the reactor pressure vessel head observed at Davis-Besse,

C. a description of any conditions identified (chemical deposits, head degradation) through the inspection and maintenance programs described in 1.A that could have led to degradation and the corrective actions taken to address such conditions,

D. your schedule, plans, and basis for future inspections of the reactor pressure vessel head and penetration nozzles. This should include the inspection method(s), scope, frequency, qualification requirements, and acceptance criteria, and

E. your conclusion regarding whether there is reasonable assurance that regulatory requirements are currently being met (see the Applicable Regulatory Requirements, above). This discussion should also explain your basis for concluding that the inspections discussed in response to Item 1.D will provide reasonable assurance that these regulatory requirements will continue to be met. Include the following specific information in this discussion:

(1) If your evaluation does not support the conclusion that there is reasonable assurance that regulatory requirements are being met, discuss your plans for plant shutdown and inspection.

(2) If your evaluation supports the conclusion that there is reasonable assurance that regulatory requirements are being met, provide your basis for concluding that all regulatory requirements discussed in the Applicable Regulatory Requirements section will continue to be met until the inspections are performed.

2. Within 30 days after plant restart following the next inspection of the reactor pressure vessel head to identify any degradation, all PWR addressees are required to submit to the NRC the following information:

A. the inspection scope (if different than that provided in response to Item 1.D.) and results, including the location, size, and nature of any degradation detected,

B. the corrective actions taken and the root cause of the degradation.

3. Within 60 days of the date of this bulletin, all PWR addressees are required to submit to the NRC the following information related to the remainder of the reactor coolant pressure boundary:

A. the basis for concluding that your boric acid inspection program is providing reasonable assurance of compliance with the applicable regulatory requirements discussed in Generic Letter 88-05 and this bulletin. If a documented basis does not exist, provide your plans, if any, for a review of your programs.

The information required in Item 1.A, 1.B, and 1.C, should address:

- the material condition of the reactor pressure vessel head as determined through direct visual examinations dating back to the last time the entire reactor pressure vessel head was visually inspected to the bare metal. Include the date of the last 100 percent bare metal inspection, the results of that examination, and the extent and results of visual examinations conducted since the last 100 percent bare metal inspection. If no 100 percent bare metal inspection has ever been conducted, indicate so in your response.

- any leaks of boric acid or any other corrosive material onto the reactor pressure vessel head or insulation since the last 100 percent bare metal inspection (the results of which were provided in responding to 1.C). Include the extent to which boric acid deposits or other corrosive materials were removed from the reactor pressure vessel head, the length of time this material was left on the reactor pressure vessel head (and whether it is still on the reactor pressure vessel head), and the condition of the head following removal of the deposits. Also include a discussion of your program for preventing corrosion of the reactor pressure vessel head and the location of the leaks relative to any nozzle with through-wall cracks. If leakage was onto the insulation, discuss whether the leakage could have permeated the insulation or flowed through gaps in the insulation (e.g., around nozzles) such that deposits accumulated on the reactor pressure vessel head.

- the leakage integrity of the reactor pressure vessel head penetration nozzles. Include a summary of inspections performed (including scope and extent) to detect cracking and/or degradation of the vessel penetration weld or nozzle base metal, whether the inspection plan included any examination that could identify a potential cavity behind the reactor pressure vessel head nozzle, and if so, the potential for the inspection method used to accurately and reliably detect a cavity in the reactor pressure vessel head near the penetration nozzles (including the basis for this conclusion), particularly in cases where a leakage path has existed (i.e., even if the nozzle has been repaired). For repaired nozzles, the description should include the scope and results from the post-repair inspections.

Required Response

In accordance with 10 CFR 50.54(f), in order to determine whether any license should be modified, suspended, or revoked, each PWR 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 7 days of the date of this bulletin, a PWR addressee is required to submit a written response if they are unable to provide the information or they can not meet the requested completion dates. The PWR addressee must address 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, 11555 Rockville Pike, Rockville, MD 20852, 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

Extensive degradation of the reactor coolant pressure boundary including leakage violates NRC regulations and plant technical specifications. Degradation of the reactor pressure vessel head or other portions of the reactor coolant pressure boundary can pose a significant safety risk if permitted to progress to the point that their integrity is in question and the risk of a loss of coolant accident 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 degradation of the reactor pressure vessel head and/or other portions of the reactor coolant pressure boundary. Such regulatory actions could include regulatory requirements for augmented inspection programs under 10 CFR 50.55a(g)(6)(ii) or additional generic communication.

The NRC staff is interacting with the industry on the implications of the degradation observed at Davis-Besse. The NRC staff will continue to assess additional information it receives on this subject in determining the need for, and to guide the development of, additional regulatory actions to address degradation of the reactor pressure vessel head and/or other portions of the reactor coolant pressure boundary.

Related Generic Communications

- Information Notice 2002-11: “Recent Experience with Degradation of Reactor Pressure Vessel Head,” March 12, 2002. [ADAMS Accession No. ML020700556]

- Bulletin 2001-01: “Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles,” August 3, 2001. [ADAMS Accession No. ML012080284]

- 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 86-108, Supplement 3, “Degradation of Reactor Coolant System Pressure Boundary Resulting from Boric Acid Corrosion,” January 5, 1995.

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

- Information Notice 94-63, “Boric Acid Corrosion of Charging Pump Casing Caused by Cladding Cracks,” August 30, 1994.

- 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.

- Information Notice 86-108, Supplement 2, “Degradation of Reactor Coolant System Pressure Boundary Resulting from Boric Acid Corrosion,” November 19, 1987.

- Information Notice 86-108, Supplement 1, “Degradation of Reactor Coolant System Pressure Boundary Resulting from Boric Acid Corrosion,” April 20, 1987.

- Information Notice 86-108, “Degradation of Reactor Coolant System Pressure Boundary Resulting from Boric Acid Corrosion,” December 29, 1986.

- Bulletin 82-02, “Degradation of Threaded Fasteners in the Reactor Coolant Pressure Boundary of PWR Plants,” June 2, 1982.

- Information Notice 82-06, “Failure of Steam Generator Primary Side Manway Closure Studs,” March 12, 1982.

- Information Notice 80-27, “Degradation of Reactor Coolant Pump Studs,” June 11, 1980.

Backfit Discussion

Under the provisions of Section 182a of the Atomic Energy Act of 1954, as amended, and 10 CFR 50.54(f), this bulletin 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 required information will enable the NRC staff to determine whether current inspection and maintenance practices for the detection of degradation of the reactor coolant pressure boundary at reactor facilities (similar to that observed at Davis-Besse) provides reasonable assurance that reactor coolant pressure boundary integrity is being maintained. The required information will also enable the NRC staff to determine whether PWR addressee inspection and maintenance practices need to be augmented to ensure that the safety significance of this form of degradation remains low. No backfit is either intended or approved by the issuance of this bulletin, and the staff has not performed a backfit analysis.

Federal Register Notification

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 degradation of the reactor pressure vessel head at Davis-Besse. Degradation of this extent has not been postulated or identified in PWRs. As the resolution of this matter progresses, the opportunity for public involvement will be provided.

Small Business Regulatory Enforcement Fairness Act

The NRC has determined that this action is not subject to the Small Business Regulatory enforcement Fairness Act of 1996.

Paperwork Reduction Act Statement

This bulletin contains an information collection that is subject to the Paperwork Reduction Act of 1995 (44 U.S.C. 3501 et seq.). This information collection was approved by the Office of Management and Budget, clearance number 3150-0012, which expires July 31, 2003. The burden to the public for this mandatory information collection is estimated to average 135 hours per response, including the time for reviewing instructions, searching existing data sources, gathering and maintaining the data needed, and completing and reviewing the information collection. Send comments regarding this burden estimate or any other aspect of this information collection, including suggestions for reducing the burden, to the Records Management Branch (T-6 E6), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by Internet electronic mail at INFOCOLLECTS@NRC.GOV; and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202, (3150-0012), Office of Management and Budget, 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 one of the persons listed below or the appropriate Office of Nuclear Reactor Regulation (NRR) project manager.

/ RA /

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

Technical Contact: Kenneth J. Karwoski, NRR
301-415-2752
E-mail: kjk1@nrc.gov

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION
WASHINGTON, DC 20555-0001

April 4, 2002

NRC INFORMATION NOTICE 2002-13: POSSIBLE INDICATORS OF ONGOING REACTOR PRESSURE VESSEL HEAD DEGRADATION

ADDRESSEES

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

PURPOSE

The U.S. Nuclear Regulatory Commission (NRC) is issuing this information notice on recent Davis-Besse experience to alert addressees to possible indicators of reactor coolant pressure boundary degradation including degradation of the reactor pressure vessel (RPV) head material. The NRC anticipates that recipients will review this information for applicability to their facilities and consider taking appropriate actions. However, the suggestions contained in this information notice do not constitute NRC requirements and, therefore, no specific action or written response is required.

DESCRIPTION OF CIRCUMSTANCES

The Davis-Besse nuclear power plant recently discovered a significant cavity in the RPV head on the downhill side of control rod drive nozzle number 3 and some head wastage behind nozzle number 2. In response, the NRC issued Information Notice 2002-11, “Recent Experience With Degradation of Reactor Pressure Vessel Head,” on March 12, 2002, and Bulletin 2002-01, “Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity,” on March 18, 2002. NRC also sent an Augmented Inspection Team (AIT) to the plant to investigate the circumstances of the degradation of the RPV head material. Through the AIT, several possible indicators of reactor coolant pressure boundary degradation such as was observed at Davis-Besse were identified. These indicators include unidentified reactor coolant system (RCS) leakage and containment air cooler (CAC) and radiation element (RE) filter fouling.

Until 1998, RCS unidentified leakage at Davis-Besse was normally less than 0.1 gallons per minute (gpm). In October 1998, the licensee removed the rupture disks downstream of the pressurizer relief valves and bypassed a drain line that collected leakage from the relief valves in the quench tank (identified leakage). As a result, all leakage past the relief valves was vented directly into the containment atmosphere and collected in the sump, increasing the unidentified leakage to approximately 0.8 gpm. In May 1999, the licensee reinstalled the rupture disks and reconnected the drain line; however, the RCS unidentified leakage was only reduced to approximately 0.2 gpm (or approximately 0.1 gpm higher than normal). This elevated level of unidentified leakage was attributed by the licensee to control rod drive mechanism (CRDM) flange leakage since the plant had a past history of flange leakage.

The Davis-Besse CACs control containment temperature and humidity. In November 1998, the licensee identified increased CAC fouling caused by boron deposits. The licensee attributed the increase in CAC fouling to the venting of the pressurizer relief valve leakage directly to containment caused by the October 1998 modification discussed previously. The CACs were cleaned many times between November 1998 and May 1999. In May 1999, the licensee reinstalled the rupture disks and reconnected the drain line. After that modification, the licensee cleaned the CACs again in June and July 1999. At that time, the licensee noticed that the boric acid deposits removed from CAC number 1 exhibited a rust-like color. The licensee attributed the discoloration to migration of the surface corrosion on the CACs into the boric acid deposits and to the aging of the boric acid deposits. After the spring 2000 refueling outage, deposits again began to form on the CACs. Between June 2000 and May 2001, the licensee cleaned the CACs eight times. No further CAC cleaning was needed until the current outage when the licensee reported that fifteen 5-gallon buckets of boric acid were removed from the CAC ductwork and plenum. A flow from the CACs also resulted in boric acid deposits elsewhere within containment including on service water piping, stairwells, and other areas of low ventilation.

Davis-Besse also has REs that are two identical air sampling systems in containment. The RE filters accumulate particulates and may need to be changed to ensure acceptable system operation. Licensee records correlate RE filter changes with past RCS leakage increases. In March 1999, RE filter clogging from boric acid deposits was identified and attributed to the pressurizer relief valve modification discussed previously. In November 1999, after identifying yellowish brown deposits in the filters, the licensee obtained a chemical analysis of the filter particulates which identified the presence of ferric oxide in addition to boric acid crystals. Around this time, the licensee began changing the filters every one-to-three weeks. By November 1999, the frequency of filter changes had again increased.

DISCUSSION

RCS leakage, boron deposits, and corrosion products like ferric oxide in CACs and RE filters may indicate degradation of the reactor coolant pressure boundary materials. These indicators do not provide clear evidence of the degradation; however, they may provide an opportunity for licensees to suspect that degradation is ongoing. The NRC understands that the indications at Davis-Besse were sometimes complicated by other events (e.g., flange leaks). Nonetheless, in combination with other indicators, they may provide insights into whether degradation of the reactor coolant pressure boundary materials is occurring.

The information in this notice is, in part, based on preliminary information. The safety significance and generic implications of the information justify NRC's urgency to issue this information notice.

This information notice does not require any specific action or written response. If you have any questions about the information in this notice, please contact one of the technical contacts listed below or the appropriate project manager from the NRC's Office of Nuclear Reactor Regulation.

/RA/
William D. Beckner, Program Director
Operating Reactor Improvements Program
Division of Regulatory Improvement Programs
Office of Nuclear Reactor Regulation

Technical contacts: Ian Jung, NRR
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Attachment: List of Recently Issued NRC Information Notices

LIST OF RECENTLY ISSUED
NRC INFORMATION NOTICES

Information Notice No.	Subject	Date of Issuance	Issued to
99-28, Supp 1	Recall of Star Brand Fire Protection Sprinkler Heads	03/22/2002	All holders of licenses for nuclear power, research, and test reactors and fuel cycle facilities.
2002-12	Submerged Safety-Related Electrical Cables	03/21/2002	All holders of operating licenses or construction permits for nuclear power reactors
2002-11	Recent Experience with Degradation of Reactor Pressure Vessel Head	03/12/2002	All holders of operating licenses for pressurized-water reactors (PWRs), except those who have permanently ceased operations and have certified that fuel has been permanently removed from the reactor.
2002-10	Nonconservative Water Level Setpoints on Steam Generators	03/07/2002	All holders of operating licenses for nuclear power reactors, except those who have permanently ceased operations and have certified that fuel has been permanently removed from the reactor.
2002-09	Potential for Top Nozzle Separation and Dropping of Certain Type of Westinghouse Fuel Assembly	02/13/2002	All holders of operating licenses for nuclear power reactors, and non-power reactors and holders of licenses for permanently shutdown facilities with fuel onsite.
2002-08	Pump Shaft Damage Due to Excessive Hardness of Shaft Sleeve	01/30/2002	All holders of operating licenses for nuclear power reactors, except those who have permanently ceased operations and have certified that fuel has been permanently removed from the reactor.

INCIDENT REPORTING SYSTEM

IRS NO.	EVENT DATE	DATE RECEIVED
	Various	

EVENT TITLE

NRC BULLETIN 2001-01: CIRCUMFERENTIAL CRACKING OF REACTOR PRESSURE VESSEL HEAD PENETRATION NOZZLES

COUNTRY	PLANT AND UNIT	REACTOR TYPE
USA		PWR

INITIAL STATUS	RATED POWER (MWe NET)
N/A	N/A

DESIGNER	1st COMMERCIAL OPERATION

ABSTRACT

This IRS report discusses the U.S. Nuclear Regulatory Commission's request that PWR licensees provide information related to the structural integrity of reactor vessel head penetration (VHP) nozzles for their respective plants. This request was made as a result of recent discoveries of cracked and leaking Alloy 600 VHP nozzles, including control rod drive mechanism and thermocouple nozzles, at four pressurized water reactors (PWRs). These discoveries have raised concerns about the structural integrity of VHP nozzles throughout the PWR industry.

Please refer to the dictionary of codes corresponding to each of the sections below and to the coding guidelines manual.

1.	**Reporting Categories:**	1.2.2		
2.	**Plant Status Prior to the Event:**	2.0		
3.	**Failed/Affected Systems:**	3.AC		
4.	**Failed/Affected Components:**	4.2.5		
5.	**Cause of the Event:**	5.1.1.1		
6.	**Effects on Operation:**	6.0		
7.	**Characteristics of the Incident:**	7.2		
8.	**Nature of Failure or Error:**	8.0		
9.	**Nature of Recovery Actions:**	9.0		

INCIDENT REPORTING SYSTEM

IRS NO.	EVENT DATE	DATE RECEIVED
	02/02/16	

EVENT TITLE

NRC INFORMATION NOTICE 2002-11: RECENT EXPERIENCE WITH DEGRADATION OF REACTOR PRESSURE VESSEL HEAD

COUNTRY	PLANT AND UNIT	REACTOR TYPE
USA	Generic	PWR

INITIAL STATUS	RATED POWER (MWe NET)
N/A	N/A

DESIGNER (WEST, GE, CE, B&W)	1st COMMERCIAL OPERATION
	N/A

ABSTRACT

This IRS report discusses findings from inspections and examinations of the reactor pressure vessel head at the Davis Besse Nuclear Power Station.

Please refer to the dictionary of codes corresponding to each of the sections below and to the coding guidelines manual.

1.	Reporting Categories:	1.2		
2.	Plant Status Prior to the Event:	2.0		
3.	Failed/Affected Systems:	3.AC		
4.	Failed/Affected Components:	4.2.5		
5.	Cause of the Event:	5.1.1.1		
6.	Effects on Operation:	6.0		
7.	Characteristics of the Incident:	7.2		
8.	Nature of Failure or Error:	8.0		
9.	Nature of Recovery Actions:	9.0		

INCIDENT REPORTING SYSTEM

IRS NO.	EVENT DATE	DATE RECEIVED
	N/A	

EVENT TITLE

NRC BULLETIN 2002-01: REACTOR PRESSURE VESSEL HEAD DEGRADATION AND REACTOR COOLANT PRESSURE BOUNDARY INTEGRITY

COUNTRY	PLANT AND UNIT	REACTOR TYPE
USA	Generic	PWR

INITIAL STATUS	RATED POWER (MWe NET)
N/A	N/A

DESIGNER	1st COMMERCIAL OPERATION
(WEST, GE, CE, B&W)	N/A

ABSTRACT

This IRS report discusses the U.S. Nuclear Regulatory Commission's request that PWR licensees provide information related to the integrity of the reactor coolant pressure boundary including the reactor pressure vessel head and the extent to which inspections have been undertaken to satisfy applicable regulatory requirements.

Please refer to the dictionary of codes corresponding to each of the sections below and to the coding guidelines manual.

1.	Reporting Categories:	1.2		
2.	Plant Status Prior to the Event:	2.0		
3.	Failed/Affected Systems:	3.AC		
4.	Failed/Affected Components:	4.2.5		
5.	Cause of the Event:	5.1.1.1		
6.	Effects on Operation:	6.0		
7.	Characteristics of the Incident:	7.2		
8.	Nature of Failure or Error:	8.0		
9.	Nature of Recovery Actions:	9.0		

INCIDENT REPORTING SYSTEM

IRS NO.	EVENT DATE	DATE RECEIVED
	N/A	

EVENT TITLE

NRC INFORMATION NOTICE 2002-13: POSSIBLE INDICATORS OF ONGOING REACTOR PRESSURE VESSEL HEAD DEGRADATION

COUNTRY	PLANT AND UNIT	REACTOR TYPE
USA	Generic	PWR

INITIAL STATUS	RATED POWER (MWe NET)
N/A	N/A

DESIGNER	1st COMMERCIAL OPERATION
(WEST, GE, CE, B&W)	N/A

ABSTRACT

This IRS report discusses recently identified indicators which may alert reactor operators to reactor coolant pressure boundary degradation including degradation of the reactor pressure vessel head material.

Please refer to the dictionary of codes corresponding to each of the sections below and to the coding guidelines manual.

1.	**Reporting Categories:**	1.2		
2.	**Plant Status Prior to the Event:**	2.0		
3.	**Failed/Affected Systems:**	3.AC		
4.	**Failed/Affected Components:**	4.2.5		
5.	**Cause of the Event:**	5.1.1.1		
6.	**Effects on Operation:**	6.0		
7.	**Characteristics of the Incident:**	7.2		
8.	**Nature of Failure or Error:**	8.0		
9.	**Nature of Recovery Actions:**	9.0		