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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

(2) 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.

<u>Required Information</u>

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.

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