PDR per D. Hoover

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

July 25, 1996

MEMORANDUM TO: Frank J. Miraglia, Jr., Deputy Director
Office of Nuclear Reactor Regulation

FROM: Edward L. Jordan, Chairman
Committee to Review Generic Requirements

SUBJECT: YOUR MEMORANDUM OF JULY 19, 1996, ON CRDM CRACKING

The subject memo to the CRGR requested that a formal review of a proposed GL on the subject of CRDM cracking is not warranted, because the GL involves no new or revised regulatory requirements. The letter would be issued for comment. The CRGR has no objection to issuance of the letter for comment, without CRGR review. We would like to review the final version, however.

There is some inconsistency in the rationale. At step (ix) of the CRGR information it is stated that NRC is not requesting any new actions. Rather, it is requesting the information already gathered. Yet, the GL action 1.2.a makes it clear that the results for subsequent inspections are wanted (and clearly this information must be gathered in the future). Also, the backfit discussion notes that the NRC needs to determine whether to impose augmented requirements in order to maintain public health and safety. Perhaps a more appropriate phrase is to determine compliance with regulatory requirements.

I have attached a copy of a report prepared by INEL under AEOD sponsorship, "Assessment of Pressurized Water Reactor Control Rod Drive Mechanism Nozzle Cracking," NUREG/CR-6245 (October 1994). I believe that it would be a useful annex to send along with the GL, for information.

cc: E. Rossi
R. Barrett
CRGR Members
J. Conran
R. Tripathi

DF03
1/1

300060

RD-8-3 Pipe Crack
X OEM-7 CRGR

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

July 19, 1996

MEMORANDUM TO: Edward L. Jordan, Chairman
Committee to Review Generic Requirements

FROM: Frank J. Miraglia, Jr., Deputy Director
Office of Nuclear Reactor Regulation

Frank J Miraglia

SUBJECT: REQUEST FOR ENDORSEMENT, WITHOUT FORMAL REVIEW, OF THE PROPOSED GENERIC LETTER ENTITLED "PRIMARY WATER STRESS CORROSION CRACKING OF CONTROL ROD DRIVE MECHANISM AND OTHER VESSEL HEAD PENETRATIONS" (TAC NO. M95280)

The Office of Nuclear Reactor Regulation (NRR) requests that the Committee to Review Generic Requirements (CRGR) endorse without formal review the subject proposed Generic Letter (GL). Following endorsement, the proposed GL will be published in the *Federal Register* for public comment.

NRR believes that formal review of the proposed GL is not warranted because the GL involves no new or revised regulatory requirements. The purpose of the GL is to request addressees to describe their program for ensuring the timely inspection of the PWR CRDM and other vessel head penetrations. The availability of the information derived from these monitoring and inspection programs will aid NRC staff in assessing whether addressees are in compliance with existing rules and regulations.

Attachment 1 is the GL as proposed by the staff. The NRC is issuing this GL to (1) require addressees to describe their program for assuring timely inspection of pressurized water reactor (PWR) control rod drive mechanism (CRDM) and other vessel head penetrations, and (2) require that all addressees provide to the NRC a written response to this Generic Letter relating to the requested information. The staff considers this GL to be Category 2.

Attachment 2 is the response to the questions contained in Section IV.B of the CRGR Charter. The responses to these questions document the justification for the required responses regulated by 10 CFR 50.54(f).

A notice of opportunity for public comment on the proposed GL will be published in the *Federal Register* prior to issuing the proposed GL. The PWR Owners Groups are aware of the NRC staff's concern regarding this issue, and have informed the NRC staff that they are taking appropriate actions to preclude a safety issue from developing. However, cracking in the vessel head penetrations (VHPs) has occurred and is expected to continue to occur as plants age. Therefore, while the NRC staff has concluded that VHP cracking does not pose a safety concern in the near term, the NRC staff considers cracking of VHPs to be a safety concern for the long term based on the possibility of: (1) exceeding the ASME Code margins if the cracks are

Contact: C. E. Carpenter, NRR
415-2169

sufficiently deep and continue to propagate during subsequent operating cycles; and, (2) eliminating a layer of defense-in-depth for plant safety. The information collected will enable the staff to verify that the margins required by the ASME Code, as specified in Section 50.55a of Title 10 of the *Code of Federal Regulations* (10 CFR 50.55a) are met, that the guidance of General Design Criterion 14 of Appendix A to 10 CFR Part 50 (10 CFR Part 50, Appendix A, GDC 14) continues to be satisfied, and to ensure that the safety significance of VHP cracking remains low, the NRC staff requires licensees to submit information to assess compliance with the above stated requirements. The NRC staff finds that the requested information is also needed to determine if the imposition of an augmented inspection program, pursuant to 10 CFR 50.55a(g)(6)(ii), is required to maintain public health and safety. The staff is not establishing a new position for such compliance in this GL.

The Office of the General Counsel reviewed this GL and has no legal objections. Furthermore, OGC has determined that the proposed GL is not a "Rule" under the provisions of the Small Business Regulatory Enforcement Fairness (SBREF) Act (*see* 5 U.S.C., Chapter 8) enacted on March 29, 1996.

The GL is sponsored by Brian W. Sheron, Director, Division of Engineering.

Attachments:
1. Proposed Generic Letter, titled "Primary Water Stress Corrosion Cracking of Control Rod Drive Mechanism and Other Vessel Head Penetrations"
2. Response to CRGR Charter Questions

Distribution: see next page

DOCUMENT NAME: G:\CARPENTR\CRDMCRGR.MEM * See Previous Concurrence
To receive a copy of this document, indicate in the box: "C" = Copy without attachment/enclosure "E" = Copy with attachment/enclosure "N" = No copy

EMCB:LPM	E	EMCB:SC	E	EMCB:BC	E	DE:D	
CECarpenter*		KRWichman*		JRStrosnider*		BWSheron*	
05/22/96		05/22/96		05/28/96		05/30/96	
OGC		DRPM:D(A)		ADT:AD		NRR:DD	
LClark*		BKGrimes*		ACThadani		FJMiraglia	
06/06/96		06/24/96		7/9/96		7/9/96	

RECORD COPY

#7 7/10

7/2

sufficiently deep and continue to propagate during subsequent operating cycles; and, (2) eliminating a layer of defense-in-depth for plant safety. The information collected will enable the staff to verify that the margins required by the ASME Code, as specified in Section 50.55a of Title 10 of the *Code of Federal Regulations* (10 CFR 50.55a) are met, that the guidance of General Design Criterion 14 of Appendix A to 10 CFR Part 50 (10 CFR Part 50, Appendix A, GDC 14) continues to be satisfied, and to ensure that the safety significance of VHP cracking remains low, the NRC staff requires licensees to submit information to assess compliance with the above stated requirements. The NRC staff finds that the requested information is also needed to determine if the imposition of an augmented inspection program, pursuant to 10 CFR 50.55a(g)(6)(ii), is required to maintain public health and safety. The staff is not establishing a new position for such compliance in this GL.

The Office of the General Counsel reviewed this GL and has no legal objections. Furthermore, OGC has determined that the proposed GL is not a "Rule" under the provisions of the Small Business Regulatory Enforcement Fairness (SBREF) Act (*see* 5 U.S.C., Chapter 8) enacted on March 29, 1996.

The GL is sponsored by Brian W. Sheron, Director, Division of Engineering.

Attachments:
1. Proposed Generic Letter, titled "Primary Water Stress Corrosion Cracking of Control Rod Drive Mechanism and Other Vessel Head Penetrations"
2. Response to CRGR Charter Questions

Distribution:
PUBLIC
PD I-1
EMCB Reading
OGC

DOCUMENT NAME: G:\CARPENTR\CRDMCRGR.MEM * See Previous Concurrence
To receive a copy of this document, indicate in the box: "**C**" = Copy without attachment/enclosure "**E**" = Copy with attachment/enclosure "**N**" = No copy

EMCB:LPM	E	EMCB:SC	E	EMCB:BC	E	DE:D	E
CECarpenter*		KRWichman*		JRStrosnider*		BWSheron*	
05/22/96		05/22/96		05/28/96		05/30/96	
OGC	E	DRPM:D(A)	E	ADT:AD	E	NRR:DD	E
LClark*		BKGrimes*		ACThadani*		FJMiraglia*	
06/10/96		06/24/96		07/09/96		07/19/96	

RECORD COPY

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

July 19, 1996

GENERIC LETTER 96-##: PRIMARY WATER STRESS CORROSION CRACKING OF CONTROL ROD DRIVE MECHANISM AND OTHER VESSEL HEAD PENETRATIONS (TAC NO. M95280)

Addressees

All holders of operating licerses for pressurized water reactors (PWRs), except those licenses that have been amended to possession-only status.

Purpose

The U.S. Nuclear Regulatory Commission (NRC) is issuing this generic letter to (1) request addressees to describe their program for ensuring the timely inspection of PWR control rod drive mechanism (CRDM) and other vessel head penetrations and (2) require that all addressees provide to the NRC a written response to this generic letter relating to the requested information.

Background

Most PWRs have Alloy 600 CRDM nozzle and other vessel head penetrations (VHPs) that extend above the reactor pressure vessel head. The stainless steel housing of the CRDM is screwed and seal-welded onto the top of the nozzle penetration, as shown in Figure 1. The weld between the nozzle and the housing is a dissimilar metal weld, which is also called a bimetallic weld. The nozzles protrude below the vessel head, thus exposing the inside surface of the nozzles to reactor coolant. The control rod drive (CRD) nozzles and other VHPs are basically the same for all PWRs worldwide, which use a U.S. design (except in Germany and Russia).

Generally, there are 36 to 78 nozzles distributed over the low-alloy steel head. The vessel head is semi-spherical and the head penetrations are vertical so that the CRD nozzles and other VHPs are not perpendicular to the vessel surface except at the center. The uphill side (toward the center of the head) is called the 180-degree location and the downhill side (toward the outer periphery of the head) is called the 0-degree location. Most nozzles have a thermal sleeve with a conical guide at the bottom end and a small gap (3- to 4-mm) between the nozzle and the sleeve.

The NRC staff identified primary water stress corrosion cracking (PWSCC) as an emerging technical issue to the Commission in 1989, after cracking was noted in Alloy 600 pressurizer heater sleeve penetrations at a domestic PWR facility. Other leaks have occurred since 1986 in several Alloy 600 pressurizer instrument nozzles at both domestic and foreign reactors from several different nuclear steam supply system vendors. The NRC staff reviewed the safety significance of the cracking that occurred, as well as the repair

and replacement activities at the affected facilities. The NRC staff determined that the cracking was not of immediate safety significance because the cracks were axial, had a low growth rate, were in a material with an extremely high flaw tolerance (high fracture toughness) and, accordingly, were unlikely to propagate very far. These factors also demonstrated that any cracking would result in detectable leakage and the opportunity to take corrective action before a penetration would fail. The NRC staff issued Information Notice 90-10, "Primary Water Stress Corrosion Cracking (PWSCC) of Inconel 600," dated February 23, 1990, to inform the nuclear industry of the issue.

In December 1991, cracks were found in an Alloy 600 VHP in the reactor head at Bugey 3, a French PWR. Examinations in PWRs in France, Belgium, Switzerland, Sweden, Spain, and Japan have uncovered additional VHPs with axial cracks. About 2 percent of the VHPs examined to date contain short, axial cracks. Close examination of the VHP that leaked at Bugey 3 revealed very minor incipient secondary circumferential cracking of the VHP.

An action plan was implemented by the NRC staff in 1991 to address PWSCC of Alloy 600 VHPs at all U.S. PWRs. As explained more fully below, this action plan included a review of the safety assessments by the PWR Owners Groups, the development of VHP mock-ups by the Electric Power Research Institute (EPRI), the qualification of inspectors on the VHP mock-ups by EPRI, the review of proposed generic acceptance criteria from the Nuclear Utility Management and Resource Council (NUMARC) [now the Nuclear Energy Institute (NEI)], and VHP inspections. As part of this action plan, the NRC staff met with the Westinghouse Owners Group (WOG) on January 7, 1992, the Combustion Engineering Owners Group (CEOG) on March 25, 1992, and the Babcock & Wilcox Owners Group (B&WOG) on May 12, 1992, to discuss their respective programs for investigating PWSCC of Alloy 600 and to assess the possibility of cracking of VHPs in their respective plants since all of the plants have Alloy 600 VHPs. Subsequently, the NRC staff asked NUMARC to coordinate future industry actions because the issue was applicable to all PWRs. Meetings were held with NUMARC/NEI and the PWR Owner's Groups on the issue on August 18 and November 20, 1992, March 3, 1993, December 1, 1994, and August 24, 1995. Summaries of these meetings are available in the Commission's Public Document Room, 2120 L Street, N.W., Washington, D.C. 20555.

Each of the PWR Owners Groups submitted safety assessments, dated February 1993, through NUMARC to the NRC on this issue. After reviewing the industry's safety assessments and examining the overseas inspection findings, the NRC staff concluded in a safety evaluation dated November 19, 1993, that VHP cracking was not an immediate safety concern. The bases for this conclusion were that if PWSCC occurred at VHPs (1) the cracks would be predominately axial in orientation, (2) the cracks would result in detectable leakage before catastrophic failure, and (3) the leakage would be detected during visual examinations performed as part of surveillance walkdown inspections before significant damage to the reactor vessel head would occur. In addition, the NRC staff had concerns related to unnecessary occupational radiation exposures associated with eddy current or other forms of nondestructive examinations (NDEs), if performed manually. Field experience in foreign countries has

shown that occupational radiation exposures can be significantly reduced by using remotely controlled or automatic equipment to conduct the inspections.

In 1993, the nuclear industry developed remotely operated inservice inspection equipment and repair tools that reduced radiation exposure. Techniques and procedures developed by two vendors were successfully demonstrated in a blind qualification protocol developed and administered by the EPRI NDE Center. In the demonstrations, examinations by rotating and saber eddy current and ultrasonics showed a high probability of detection of the flaws which were also sized within reasonable uncertainty bounds. The qualification testing also demonstrated that personnel qualified through the EPRI program can reliably detect PWSCC in CRDM nozzles.

In 1994, circumferential intergranular attack (IGA) associated with the J-groove weld in one of the CRDM penetrations was discovered at Zorita, a Spanish reactor. This IGA is a different degradation mechanism than the PWSCC described above. It is believed to have resulted from the combination of ion exchange resin bed intrusions, which resulted in high concentrations of sulfates. Zorita has 37 CRDM penetrations, of which 20 are active penetrations and 17 are spare penetrations. Sixteen of the 17 spare penetrations showed stress corrosion cracking and IGA. The cracks were both axial and circumferential. Four of the active CRDM penetrations had significant cracking with axial and circumferential cracks. Two cation resin ingress events occurred at Zorita. In August 1980, 40 liters of cation resin entered the reactor coolant system (RCS). In September 1981, a mixed bed demineralizer screen failed and between 200 to 320 liters of resin entered the RCS. The coolant conductivity remained high for at least 4 months after the ingress. The increase in conductivity was attributed to locally high concentrations of sulfates. Sulfates were found around the crack areas and on the fracture surfaces. It is important to note that sulfate cracking can occur in regions that are not subject to significant applied or residual stresses.

The NRC staff issued Information Notice (IN) 96-11, "Ingress of Demineralizer Resins Increases Potential for Stress Corrosion Cracking of Control Rod Drive Mechanism Penetrations," dated February 14, 1996, to alert addressees to the increased likelihood of sulfate-driven stress corrosion cracking of PWR CRDMs and other VHPs if demineralizer resins contaminate the RCS.

The Westinghouse staff notified the WOG plants, the B&WOG plants, and the CEOG plants of the Zorita incident by issuing NSAL-94-028. Westinghouse reported that no other plant had been found worldwide that had experienced cracking similar to that at the Zorita plant. The Westinghouse staff further reported that U.S. plants monitor RCS conductivity on a routine basis, follow the EPRI guidelines on primary water chemistry, and monitor for sulfate three times a week. The Westinghouse staff concluded that no immediate safety issue is involved and that the conclusions in its CRDM safety evaluation remain valid. The Westinghouse staff suggested that U.S. PWR plants review their RCS chemistry and other operating records pertaining to sulfur ingress events. The results of this review have not been reported to the NRC staff, and the

NRC staff does not have sufficient information to ascertain whether any significant primary system resin bed intrusions have occurred at any U.S. PWR.

The first U.S. inspection of VHPs took place in the spring of 1994 at the Point Beach Nuclear Generating Station, and no indications were uncovered in any of its 49 CRDM penetrations. The eddy current inspection at the Oconee Nuclear Generating Station in the fall of 1994 revealed 20 indications in one penetration. Ultrasonic testing (UT) did not reveal the depth of these indications because they were shallow. UT cannot accurately size defects that are less than one mil deep (0.03 mm). These indications may be associated with the original fabrication and may not grow; however, they will be reexamined during the next refueling outage. A limited examination of eight in-core instrumentation penetrations conducted at the Palisades plant found no cracking. An examination of the CRDM penetrations at the D. C. Cook plant in the fall of 1994 revealed three clustered indications in one penetration. The indications were 46 mm, 16 mm, and 6 to 8 mm in length, and the deepest flaw was 6.8 mm deep. The tip of the 46-mm flaw was just below the J-groove weld.

Virginia Electric and Power Company inspected North Anna Unit 1 during its spring 1996 refueling outage. Some high-stress areas (e.g., upper and lower hillsides) were examined on each outer ring CRDM penetrations and no indications were observed using eddy current testing.

The NRC staff was informed during a meeting on August 24, 1995, that Westinghouse had developed a susceptibility model for VHPs based on a number of factors, including operating temperature, years of power operation, method of fabrication of the VHP, microstructure of the VHP, and the location of the VHP on the head. Each time a plant's VHPs are inspected, the inspection results are incorporated into the model. All domestic Westinghouse PWRs have been modeled and the ranking has been given to each licensee. In addition, the NRC staff was informed that Framatome Technologies, Inc. [FTI, formerly Babcock & Wilcox (B&W)], also developed a susceptibility model for CRDM penetration nozzles and other VHPs in B&W reactor vessel designs. All domestic B&W PWRs have been modeled and the ranking has been given to each B&W licensee. The NRC staff was further informed that Combustion Engineering (CE) had performed an initial susceptibility assessment for the CE PWRs. At present, neither Westinghouse, FTI, nor CE has submitted its models and assessments to the NRC staff for review.

By letter dated March 5, 1996, NEI submitted a white paper entitled "Alloy 600 RPV Head Penetration Primary Stress Corrosion Cracking," which reviews the significance of PWSCC in PWR VHPs and describes how the industry is managing the issue. The program outlined in the NEI white paper is based on the assumption that the issue is an economic one rather than a safety issue, and describes an economic decision tool to be used by PWR licensees to evaluate the probability of a VHP developing a crack or a through-wall leak during a plant's lifetime. This information would then be used by a PWR licensee to evaluate the need to conduct a VHP inspection at their plant. The NRC staff informed NEI in the several meetings listed above that it did not agree with NEI that the issue was only economic. Inspections have shown that cracking has initiated in some U.S. plants, and the industry has not provided

sufficient technical justification regarding susceptibility of the CRDM and other VHPs to PWSCC to justify an inspection plan based on economic considerations alone.

Discussion

The results of domestic VHP inspections are consistent with the February 1993 analyses by the PWR Owners Groups, the NRC staff safety evaluation report dated November 19, 1993, and the PWSCC found in the CRDMs in European reactors. On the basis of the results of the first five inspections of U.S. PWRs, the PWR Owner's Groups' analyses, and the European experience, the NRC staff has determined that there is a high probability that VHPs at other plants may contain similar axial cracks caused by PWSCC. Further, if any significant resin intrusions have occurred at U.S. PWRs such as occurred at Zorita, residual stresses are sufficient to cause circumferential intergranular stress corrosion cracking (IGSCC).

After considering this information, the NRC staff has concluded that VHP cracking does not pose an immediate or near term safety concern. Further, the NRC staff recognizes that the scope and timing of inspections may vary for different plants depending on their individual suceptibility to this form of degradation. In the long term, however, degradation of the CRDM and other VHPs is an important safety consideration that warrants further evaluation. The vessel head provides the vital function of maintaining a reactor pressure boundary. Cracking in the VHPs has occurred and is expected to continue to occur as plants age. The NRC staff considers cracking of VHPs to be a safety concern for the long term based on the possibility of (1) exceeding the American Society of Mechanical Engineers (ASME) Code for margins if the cracks are sufficiently deep and continue to propagate during subsequent operating cycles, and (2) eliminating a layer of defense in depth for plant safety. Therefore, in order to verify that the margins required by the ASME Code, as specified in Section 50.55a of Title 10 of the *Code of Federal Regulations* (10 CFR 50.55a) are met, that the guidance of General Design Criterion 14 of Appendix A to 10 CFR Part 50 (10 CFR Part 50, Appendix A, GDC 14) is continued to be satisfied, and to ensure that the safety significance of VHP cracking remains low, the NRC staff believes that an integrated, long-term program, which includes periodic inspections and monitoring, is necessary. In addition, the NRC staff finds that the requested information is also needed to determine if the imposition of an augmented inspection program, pursuant to 10 CFR 50.55a(g)(6)(ii), is required to maintain public health and safety.

The NRC staff recognizes that individual PWR licensees may wish to determine their inspection activities based on an integrated industry inspection program (i.e., B&WOG, CEOG, WOG, or some subset thereof), to take advantage of inspection results from other plants that have similar susceptibilities. The NRC staff does not wish to discourage such group actions but notes that such an integrated industry inspection program must have a well-founded technical basis that justifies the relationship between the plants and the planned implementation schedule.

Required Information

The information requested by items 1 and 2, below, is required by the NRC staff to determine if the imposition of an augmented inspection program is required, while the information requested by item 3 relates to the potential for domestic resin intrusions, such as occurred at Zorita.

Addressees are required to provide the following information:

1. Regarding inspection activities:

   1.1 A description of all inspections of CRDMs and other vessel head penetrations performed to the date of this generic letter, including the results of these inspections.

   1.2 If you have developed a plan to periodically inspect the CRDM and other vessel head penetrations:

      a. Your schedule for first, and subsequent, inspections of the CRDM and other vessel head penetrations, including the technical basis for your schedule.

      b. Your scope for the CRDM and other vessel head penetration inspections, including whether you plan to inspect from the top or bottom of the head, the total number of penetrations (and how many will be inspected), and which penetrations have thermal sleeves, which are spares, and which are instrument or other penetrations.

   1.3 If you have not developed a plan to periodically inspect the CRDM and other vessel head penetrations, provide your technical or safety basis for not periodically inspecting your VHPs; or, your schedule for developing such a plan and the basis for that schedule.

2. A description of the evaluation methods and results used to assess the susceptibility of the CRDM and other VHPs in your plant to PWSCC, including the susceptibility ranking of your plant and the factors used to determine this ranking. Other than or in addition to the boric acid visual examination (*see* Generic Letter 88-05, "Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants," dated March 17, 1988), include a description of all relevant data and/or tests used to develop crack initiation and crack growth models, and the methods and data used to validate these models. Include a statement explaining the applicability of these models to the VHP cracking issue. Also, if you are relying on any integrated industry inspection program, provide a detailed description of this program.

3. A description of any resin intrusions in your plant, as described in IN 96-11, that have exceeded the current EPRI PWR Primary Water

Chemistry Guidelines recommendations for primary water sulfate levels, including the following information:

3.1 Were the intrusions cation, anion, or mixed bed?

3.2 What were the durations of these intrusions?

3.3 Do your RCS water chemistry Technical Specifications follow the EPRI guidelines?

3.4 Identify any RCS chemistry excursions that exceed your plant administrative limits for the following species: sulfates, chlorides or fluorides, oxygen, boron, and lithium.

3.5 Identify any conductivity excursions which may be indicative of resin intrusions, provide your technical assessment of each excursion and your followup actions.

3.6 Provide your assessment of the potential for any of these intrusions to result in a significant increase in the probability for IGA of VHPs and any associated plan for inspections.

Required Response

All addressees are required to submit a written response with the information requested above within 90 days from the date of this letter.

Any inspection results that do not satisfy the acceptance criteria identified in the NRC staff's safety assessment dated November 16, 1993, should be reported to the NRC staff prior to plant restart.

Address the required written reports to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, D.C. 20555, under oath or affirmation under the provisions of Section 182a, Atomic Energy Act of 1954, as amended, and 10 CFR 50.54(f). In addition, submit a copy to the appropriate regional administrator.

The NRC recognizes the potential difficulties (number and types of sources, age of records, proprietary data, etc.) that licensees may encounter while ascertaining whether they have all of the data pertinent to the evaluation of their CRDMs and other vessel head penetrations. For this reason, the above time periods are allowed for the responses.

Related Generic Communications

(1) Information Notice 90-10, "Primary Water Stress Corrosion Cracking (PWSCC) of Inconel 600," dated February 23, 1990.

(2) Information Notice 96-11, "Ingress of Demineralizer Resins Increases Potential for Stress Corrosion Cracking of Control Rod Drive Mechanism Penetrations," dated February 14, 1996.

Backfit Discussion

This generic letter only requires information from the addressees under the provisions of Section 182a of the Atomic Energy Act of 1954, as amended, and 10 CFR 50.54(f). Therefore, the staff has not performed a backfit analysis. The information collected will enable the staff to verify that the margins required by the ASME Code, as specified in Section 50.55a of Title 10 of the *Code of Federal Regulations* (10 CFR 50.55a) are met, that the guidance of General Design Criterion 14 of Appendix A to 10 CFR Part 50 (10 CFR Part 50, Appendix A, GDC 14) continues to be satisfied, and to ensure that the safety significance of VHP cracking remains low, the NRC staff requires licensees to submit information to assess compliance with the above stated requirements. The NRC staff finds that the requested information is also needed to determine if the imposition of an augmented inspection program, pursuant to 10 CFR 50.55a(g)(6)(ii), is required to maintain public health and safety. The staff is not establishing a new position for such compliance in this generic letter. Therefore, this generic letter does not constitute a backfit and no documented evaluation or backfit analysis need be prepared.

Federal Register Notification

A notice of opportunity for public comment was published in the *Federal Register* (**XX FR XXXXX**) on **date**. Comments were received from **[Indicate the number of commentors by type (e.g., ten licensees, three industry organizations, two public interest groups, and five individuals)]**. Copies of the staff evaluation of these comments have been made available in the public document room.

Paperwork Reduction Act Statement

This generic letter 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, which expires July 31, 1997.

The public reporting burden for this collection of information is estimated to average 80 hours per response, including the time for reviewing instructions, searching existing data sources, gathering and maintaining the data needed, and completing and reviewing the collection of information. The U.S. Nuclear Regulatory Commission is seeking public comment on the potential impact of the collection of information contained in the generic letter and on the following issues:

1. Is the proposed collection of information necessary for the proper performance of the functions of the NRC, including whether the information will have practical utility?

2. Is the estimate of burden accurate?

3. Is there a way to enhance the quality, utility, and clarity of the information to be collected?

4. How can the burden of the collection of information be minimized, including the use of automated collection techniques?

Send comments on any aspect of this collection of information, including suggestions for reducing this burden, to the Information and Records Management Branch, T-6 F33, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202 (3150-0011), Office of Management and Budget, Washington, DC 20503.

The NRC may not conduct or sponsor, and a person is not required to respond to, a collection of information unless it displays a currently valid OMB control number.

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

Brian K. Grimes, Acting Director
Division of Reactor Program Management
Office of Nuclear Reactor Regulation

Technical contacts: Keith R. Wichman
(301) 415-2757
e-mail: krw@nrc.gov

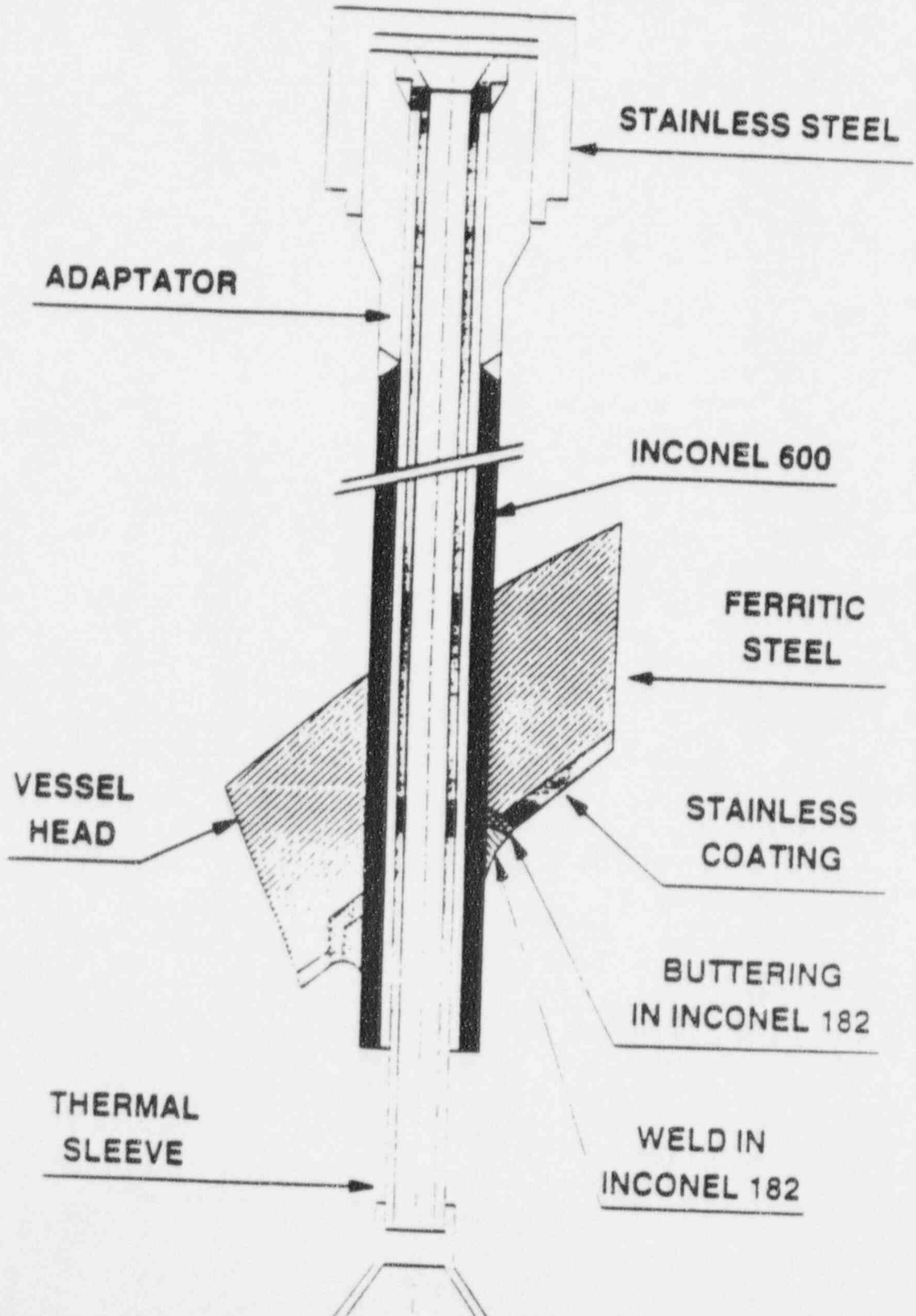
James Medoff
(301) 415-2715
e-mail: jxm@nrc.gov

Lead Project Manager: C. E. Carpenter, Jr.
(301) 415-2169
e-mail: cec@nrc.gov

Attachments:
1. References
2. List of Recently Issued NRC Generic Letters

DSIN

LATCH HOUSING OF CONTROL ROD DRIVE MECANISM



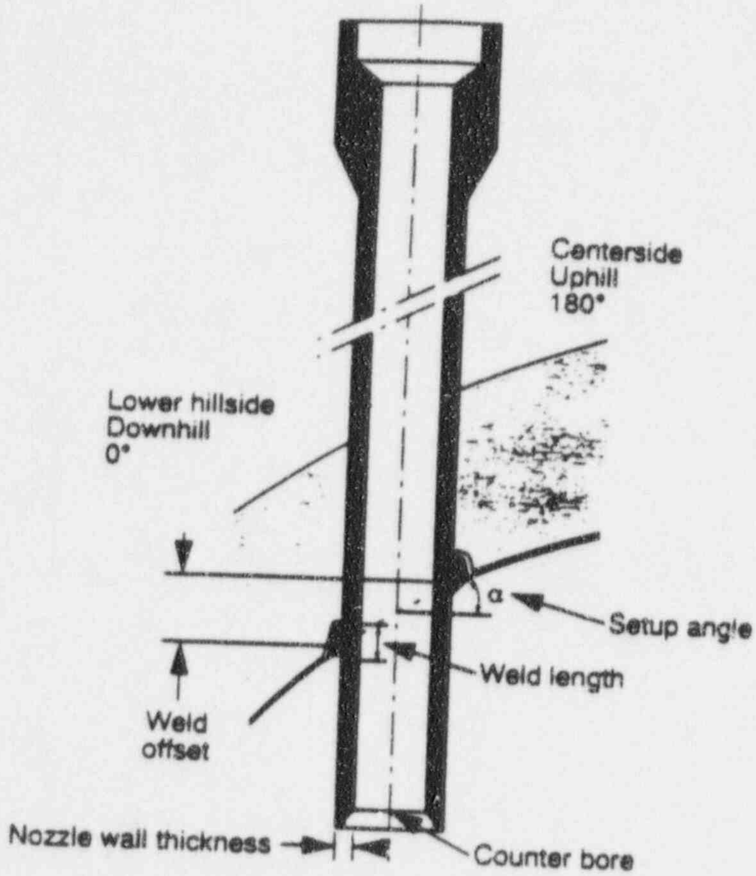

Centerside
Uphill
180°
Lower hillside
Downhill
0°
α
Setup angle
Weld length
Weld
offset
Nozzle wall thickness
Counter bore

Figure 4

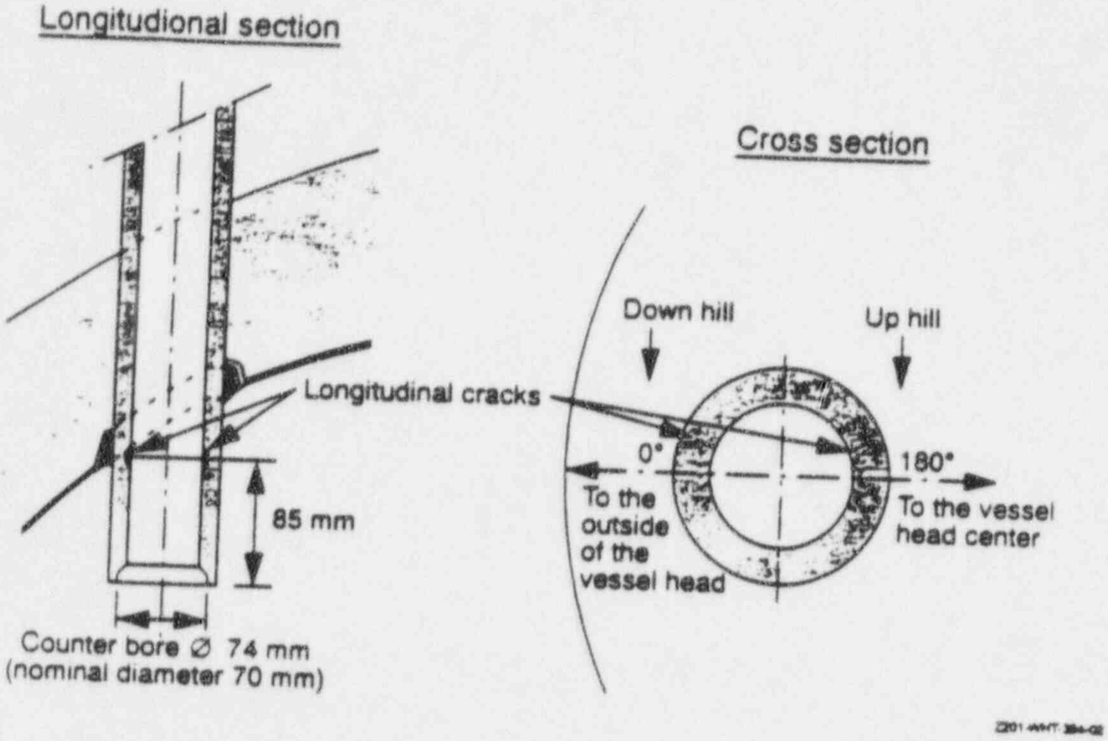
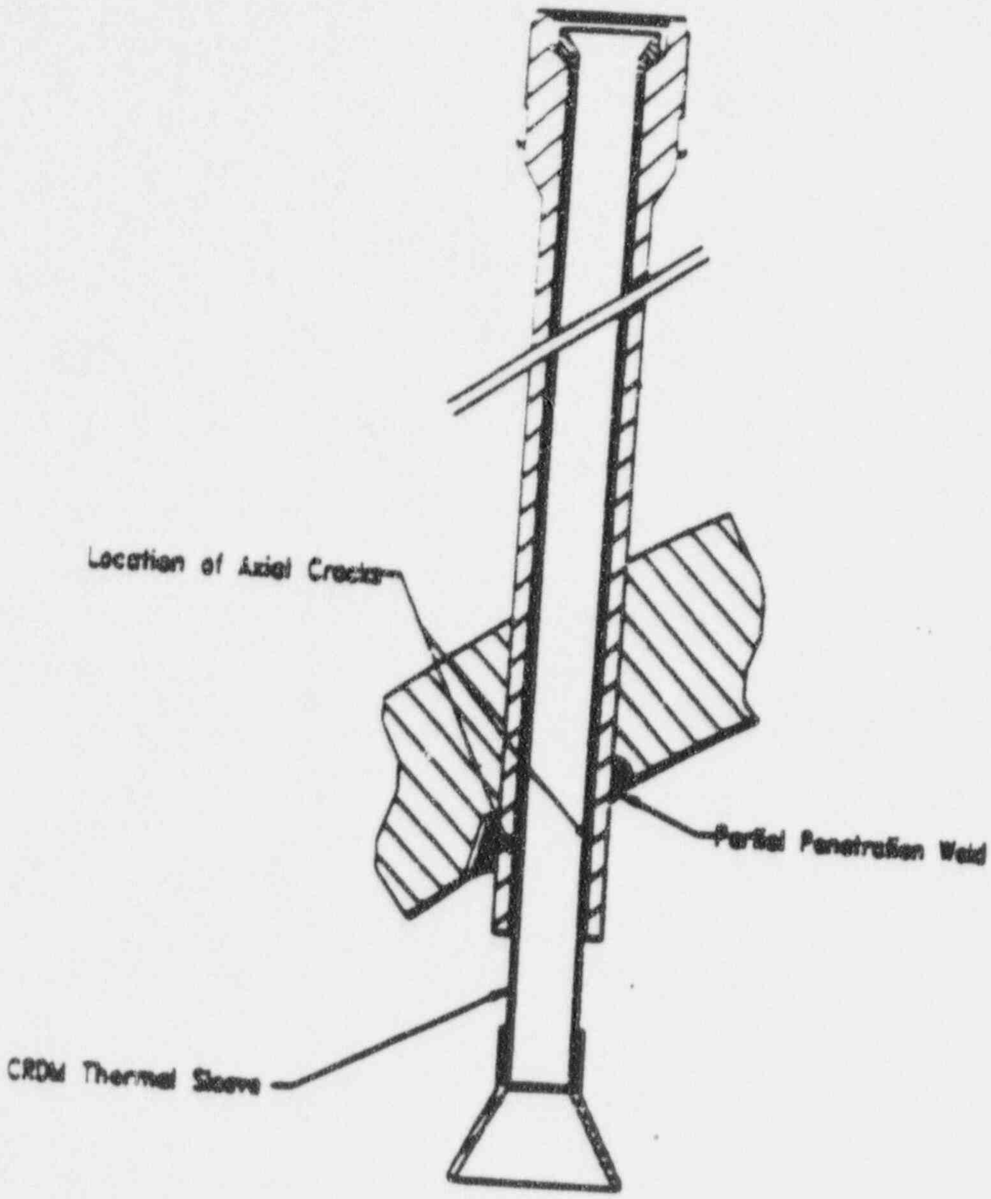

Longitudional section
Longitudinal cracks
85 mm
Counter bore Ø 74 mm
(nominal diameter 70 mm)
Cross section
Down hill
Up hill
0°
180°
To the outside of the vessel head
To the vessel head center

Figure 3



Head Penetration and Vessel Assembly
Figure 3.4-1

CRGR REVIEW PACKAGE

PROPOSED ACTION: Issue a generic letter on the primary water stress corrosion cracking of control rod drive mechanism and other vessel head penetrations.

CATEGORY: 2

RESPONSE TO REQUIREMENTS FOR CONTENT OF PACKAGE SUBMITTED FOR CRGR REVIEW

(i) **The proposed generic requirement or staff position as it is proposed to be sent out to licensees. Where the objective or intended result of a proposed generic requirement or staff position can be achieved by setting a readily quantifiable standard that has an unambiguous relationship to a readily measurable quantity and is enforceable, the proposed requirement should merely specify the objective or result to be attained, rather than prescribing to the licensee how the objective or result is to be attained.**

The information requested by items 1 and 2, below, is required by the NRC staff to determine if the imposition of an augmented inspection program is required, while the information requested by item 3 relates to the potential for domestic resin intrusions, such as occurred at Zorita.

Addressees are required to provide the following information:

1. Regarding inspection activities:

   1.1 A description of all inspections of CRDMs and other vessel head penetrations performed to the date of this generic letter, including the results of these inspections.

   1.2 If you have developed a plan to periodically inspect the CRDM and other vessel head penetrations:

      a. Your schedule for first, and subsequent, inspections of the CRDM and other vessel head penetrations, including the technical basis for your schedule.

      b. Your scope for the CRDM and other vessel head penetration inspections, including whether you plan to inspect from the top or bottom of the head, the total number of penetrations (and how many will be inspected), and which penetrations have thermal sleeves, which are spares, and which are instrument or other penetrations.

   1.3 If you have <u>not</u> developed a plan to periodically inspect the CRDM and other vessel head penetrations, provide your technical or safety basis for not periodically inspecting your VHPs; or, your schedule for developing such a plan and the basis for that schedule.

2. A description of the evaluation methods and results used to assess the susceptibility of the CRDM and other VHPs in your plant to PWSCC, including the susceptibility ranking of your plant and the factors used to determine this ranking. Other than or in addition to the boric acid visual examination (*see* Generic Letter 88-05, "Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants," dated March 17, 1988), include a description of all relevant data and/or tests used to develop crack initiation and crack growth models, and the methods and data used to validate these models. Include a statement explaining the applicability of these models to the VHP cracking issue. Also, if you are relying on any integrated industry inspection program, provide a detailed description of this program.

3. A description of any resin intrusions in your plant, as described in IN 96-11, that have exceeded the current EPRI PWR Primary Water Chemistry Guidelines recommendations for primary water sulfate levels, including the following information:

   3.1 Were the intrusions cation, anion, or mixed bed?

   3.2 What were the durations of these intrusions?

   3.3 Do your RCS water chemistry Technical Specifications follow the EPRI guidelines?

   3.4 Identify any RCS chemistry excursions that exceed your plant administrative limits for the following species: sulfates, chlorides or fluorides, oxygen, boron, and lithium.

   3.5 Identify any conductivity excursions which may be indicative of resin intrusions, provide your technical assessment of each excursion and your followup actions.

   3.6 Provide your assessment of the potential for any of these intrusions to result in a significant increase in the probability for IGA of VHPs and any associated plan for inspections.

(ii) **Draft staff papers or other underlying staff documents supporting the requirements or staff positions. (A copy of all materials referenced in the document shall be made available upon request to the CRGR staff. Any Committee member may request CRGR staff to obtain a copy of any reference material for his or her use.)**

(1) Information Notice 90-10, "Primary Water Stress Corrosion Cracking (PWSCC) of Inconel 600," dated February 23, 1990.

(2) NRC staff safety evaluation, "Potential Reactor Vessel Head Adaptor Tube Cracking," dated November 19, 1993

(3) Information Notice 96-11, "Ingress of Demineralizer Resins Increases Potential for Stress Corrosion Cracking of Control Rod Drive Mechanism Penetrations," dated February 14, 1996.

(iii) **Each proposed requirement or staff position shall contain the sponsoring office's position as to whether the proposal would increase requirements or staff positions, implement existing requirements or staff positions, or would relax or reduce existing requirements or staff positions.**

The results of domestic VHP inspections are consistent with the February 1993 analyses by the PWR Owners Groups, the NRC staff safety evaluation report dated November 19, 1993, and the PWSCC found in the CRDMs in European reactors. On the basis of the results of the first five inspections of U.S. PWRs, the PWR Owner's Groups' analyses, and the European experience, the NRC staff has determined that there is a high probability that VHPs at other plants may contain similar axial cracks caused by PWSCC. Further, if any significant resin intrusions have occurred at U.S. PWRs such as occurred at Zorita, residual stresses are sufficient to cause circumferential intergranular stress corrosion cracking (IGSCC).

After considering this information, the NRC staff has concluded that VHP cracking does not pose an immediate or near term safety concern. Further, the NRC staff recognizes that the scope and timing of inspections may vary for different plants depending on their individual suceptibility to this form of degradation. In the long term, however, degradation of the CRDM and other VHPs is an important safety consideration that warrants further evaluation. The vessel head provides the vital function of maintaining a reactor pressure boundary. Cracking in the VHPs has occurred and is expected to continue to occur as plants age. The NRC staff considers cracking of VHPs to be a safety concern for the long term based on the possibility of (1) exceeding the American Society of Mechanical Engineers (ASME) Code for margins if the cracks are sufficiently deep and continue to propagate during subsequent operating cycles, and (2) eliminating a layer of defense in depth for plant safety. Therefore, in order to verify that the margins required by the ASME Code, as specified in Section 50.55a of Title 10 of the *Code of Federal Regulations* (10 CFR 50.55a) are met, that the guidance of General Design Criterion 14 of Appendix A to 10 CFR Part 50 (10 CFR Part 50, Appendix A, GDC 14) is continued to be satisfied, and to ensure that the safety significance of VHP cracking remains low, the NRC staff believes that an integrated, long-term program, which includes periodic inspections and monitoring, is necessary. In addition, the NRC staff finds that the requested information is also needed to determine if the imposition of an augmented inspection program, pursuant to 10 CFR 50.55a(g)(6)(ii), is required to maintain public health and safety.

The NRC staff recognizes that individual PWR licensees may wish to determine their inspection activities based on an integrated industry inspection program (i.e., B&WOG, CEOG, WOG, or some subset thereof), to take advantage of inspection results from other plants that have similar susceptibilities. The NRC staff does not wish to discourage such group actions but notes that such an integrated industry inspection program must have a well-founded technical basis that justifies the relationship between the plants and the planned implementation schedule.

(iv) The proposed method of implementation with the concurrence (and any comments) of OGC on the method proposed. The concurrence of affected program offices or an explanation of any nonconcurrences.

See attached concurrence page.

(v) Regulatory analyses conforming to the directives and guidance of NUREG/BR-0058 and NUREG/CR-3568. (This does not apply for backfits that ensure compliance or ensure, define, or redefine adequate protection. In these cases a documented evaluation is required as discussed in IV.B.(ix).)

Not applicable

(vi) Identification of the category of reactor plants to which the generic requirement or staff position is to apply (that is, whether it is to apply to new plants only, new OLs only, OLs after a certain date, OLs before a certain date, all OLs, all plants under construction, all plants, all water reactors, all PWRs only, some vendor types, some vintage types such as BWR 6 and 4, jet pump and nonjet pump plants, etc.).

All holders of operating licenses for pressurized water reactors (PWRs), except those licenses that have been amended to possession-only status.

(vii) For backfits other than compliance or adequate protection backfits, a backfit analysis as defined in 10 CFR 50.109. The backfit analysis shall include, for each category of reactor plants, an evaluation which demonstrates how the action should be prioritized and scheduled in light of other ongoing regulatory activities. The backfit analysis shall document for consideration information available concerning any of the following factors as may be appropriate and any other information relevant and material to the proposed action:

(a) Statement of the specific objectives that the proposed action is designed to achieve;

Not applicable.

(b) General description of the activity that would be required by the licensee or applicant in order to complete the action;

Not applicable.

(c) Potential change in the risk to the public from the accidental release of radioactive material;

Not applicable.

(d) Potential impact on radiological exposure of facility employees and other onsite workers;

Not applicable.

(e) Installation and continuing costs associated with the action, including the cost of facility downtime or the cost of construction delay;

Not applicable.

(f) The potential safety impact of changes in plant or operational complexity, including the relationship of proposed and existing regulatory requirements and staff positions;

Not applicable.

(g) The estimated resource burden on the NRC associated with the proposed action and the availability of resources;

Not applicable.

(h) The potential impact of differences in facility type, design, or age on the relevancy and practicality of the proposed action;

Not applicable.

(i) Whether the proposed action is interim or final, and if interim, the justification for imposing the proposed action on an interim basis;

Not applicable.

(j) How the action should be prioritized and scheduled in light of other ongoing regulatory activities. The following information may be appropriate in this regard:

1. The proposed priority or schedule,
2. A summary of the current backlog of existing requirements awaiting implementation,
3. An assessment of whether implementation of existing requirements should be deferred as a result, and
4. Any other information that may be considered appropriate with regard to priority, schedule, or cumulative impact. For example, could implementation be delayed pending public comment?

Not applicable.

(viii) For each backfit analyzed pursuant to 10 CFR 50.109(a)(2) (i.e., not adequate protection backfits and not compliance backfits), the proposing Office Director's determination, together with the rational for the determination based on the consideration of paragraph (i) and (vii) above, that:

(a) There is a substantial increase in the overall protection of public health and safety or the common defense and security to be derived from the proposal; and

(b) The direct and indirect costs of implementation, for the facilities affected, are justified in view of this increased protection.

Not applicable.

(ix) For adequate protection or compliance backfits evaluated pursuant to 10 CFR 50.109(a)(4)

(a) a documented evaluation consisting of:
(1) the objectives of the modification
(2) the reasons for the modification
(3) the basis for invoking the compliance or adequate protection exemption.

(b) in addition, for actions that were immediately effective (and therefore issued without prior CRGR review as discussed in III.C) the evaluation shall document the safety significance and appropriateness of the action taken and (if applicable) consideration of how costs contributed to selecting the solution among various acceptable alternatives.

Not applicable. The proposed generic letter is a request for information only. The NRC staff is not requesting any new actions from the PWR licensees; rather, the proposed generic letter is requesting the PWR licensees to provide to the NRC information that the PWROGs has already told the NRC staff it has gathered, but has not shared with the NRC to date.

(x) For each evaluation conducted for proposed relaxations or decreases in current requirements or staff positions, the proposing Office Director's determination, together with the rationale for the determination based on the considerations or paragraphs (i) through (vii) above, that:

(a) Public health and safety and the common defense and security would be adequately protected if the proposed reduction in requirements or positions were implemented, and

(b) The cost savings attributed to the action would be substantial enough to justify taking the action.

Not applicable.

(xi) For each request for information under 10 CFR 50.54(f) (which is not subject to exception as discussed in III.A) an evaluation that includes at least the following elements:

(a) A problem statement that describes the need for the information in terms of potential safety benefit.

The NRC staff was informed during a meeting on August 24, 1995, that Westinghouse had developed a susceptibility model for VHPs based on a number of factors, including operating temperature, years of power operation, method of fabrication of the VHP, microstructure of the VHP, and the location of the VHP on the head. Each time a plant's VHPs are inspected, the inspection results are incorporated into the model. All domestic Westinghouse PWRs have been modeled and the ranking has been given to each licensee. In addition, the NRC staff was informed that Framatome Technologies, Inc. [FTI, formerly Babcock & Wilcox (B&W)], also developed a susceptibility model for CRDM penetration nozzles and other VHPs in B&W reactor vessel designs. All domestic B&W PWRs have been modeled and the ranking has been given to each B&W licensee. The NRC staff was further informed that Combustion Engineering (CE) had performed an initial susceptibility assessment for the CE PWRs. At present, neither Westinghouse, FTI, nor CE has submitted its models and assessments to the NRC staff for review.

The results of domestic VHP inspections are consistent with the February 1993 analyses by the PWR Owners Groups, the NRC staff safety evaluation report dated November 19, 1993, and the PWSCC found in the CRDMs in European reactors. On the basis of the results of the first five inspections of U.S. PWRs, the PWR Owner's Groups' analyses, and the European experience, the NRC staff has determined that there is a high probability that VHPs at other plants may contain similar axial cracks caused by PWSCC. Further, if any significant resin intrusions have occurred at U.S. PWRs such as occurred at Zorita, residual stresses are sufficient to cause circumferential intergranular stress corrosion cracking (IGSCC).

After considering this information, the NRC staff has concluded that VHP cracking does not pose an immediate or near term safety concern. Further, the NRC staff recognizes that the scope and timing of inspections may vary for different plants depending on their individual suceptibility to this form of degradation. In the long term, however, degradation of the CRDM and other VHPs is an important safety consideration that warrants further evaluation. The vessel head provides the vital function of maintaining a reactor pressure boundary. Cracking in the VHPs has occurred and is expected to continue to occur as plants age. The NRC staff considers cracking of VHPs to be a safety concern for the long term based on the possibility of (1) exceeding the American Society of Mechanical Engineers (ASME) Code for margins if the cracks are sufficiently deep and continue to propagate during subsequent operating cycles, and (2) eliminating a layer of defense in depth for plant safety. Therefore, in order to verify that the margins required by the ASME Code, as specified in Section 50.55a of Title 10 of the *Code of Federal Regulations* (10 CFR 50.55a) are met, that the guidance of General Design Criterion 14 of Appendix A to 10 CFR Part 50 (10 CFR Part 50, Appendix A, GDC 14) is continued to be satisfied, and to ensure that the safety significance of VHP cracking remains low, the NRC staff believes that an integrated, long-term program, which includes periodic inspections and monitoring, is necessary. In addition, the NRC staff finds that the requested information is also needed to determine if the imposition of an augmented inspection program, pursuant to

10 CFR 50.55a(g)(6)(ii), is required to maintain public health and safety.

The NRC staff recognizes that individual PWR licensees may wish to determine their inspection activities based on an integrated industry inspection program (i.e., B&WOG, CEOG, WOG, or some subset thereof), to take advantage of inspection results from other plants that have similar susceptibilities. The NRC staff does not wish to discourage such group actions but notes that such an integrated industry inspection program must have a well-founded technical basis that justifies the relationship between the plants and the planned implementation schedule.

(b) The licensee actions required and the cost to develop a response to the information request.

The information requested by items 1 and 2, below, is required by the NRC staff to determine if the imposition of an augmented inspection program is required, while the information requested by item 3 relates to the potential for domestic resin intrusions, such as occurred at Zorita.

Addressees are required to provide the following information:

1. Regarding inspection activities:

   1.1 A description of all inspections of CRDMs and other vessel head penetrations performed to the date of this generic letter, including the results of these inspections.

   1.2 If you have developed a plan to periodically inspect the CRDM and other vessel head penetrations:

      a. Your schedule for first, and subsequent, inspections of the CRDM and other vessel head penetrations, including the technical basis for your schedule.

      b. Your scope for the CRDM and other vessel head penetration inspections, including whether you plan to inspect from the top or bottom of the head, the total number of penetrations (and how many will be inspected), and which penetrations have thermal sleeves, which are spares, and which are instrument or other penetrations.

   1.3 If you have <u>not</u> developed a plan to periodically inspect the CRDM and other vessel head penetrations, provide your technical or safety basis for not periodically inspecting your VHPs; or, your schedule for developing such a plan and the basis for that schedule.

2. A description of the evaluation methods and results used to assess the susceptibility of the CRDM and other VHPs in your plant to PWSCC, including the susceptibility ranking of your plant and the

factors used to determine this ranking. Other than or in addition to the boric acid visual examination (*see* Generic Letter 88-05, "Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants," dated March 17, 1988), include a description of all relevant data and/or tests used to develop crack initiation and crack growth models, and the methods and data used to validate these models. Include a statement explaining the applicability of these models to the VHP cracking issue. Also, if you are relying on any integrated industry inspection program, provide a detailed description of this program.

3. A description of any resin intrusions in your plant, as described in IN 96-11, that have exceeded the current EPRI PWR Primary Water Chemistry Guidelines recommendations for primary water sulfate levels, including the following information:

   3.1 Were the intrusions cation, anion, or mixed bed?

   3.2 What were the durations of these intrusions?

   3.3 Do your RCS water chemistry Technical Specifications follow the EPRI guidelines?

   3.4 Identify any RCS chemistry excursions that exceed your plant administrative limits for the following species: sulfates, chlorides or fluorides, oxygen, boron, and lithium.

   3.5 Identify any conductivity excursions which may be indicative of resin intrusions, provide your technical assessment of each excursion and your followup actions.

   3.6 Provide your assessment of the potential for any of these intrusions to result in a significant increase in the probability for IGA of VHPs and any associated plan for inspections.

All addressees are required to submit a written response with the information requested above within 90 days from the date of this letter.

Any inspection results that do <u>not</u> satisfy the acceptance criteria identified in the NRC staff's safety assessment dated November 16, 1993, should be reported to the NRC staff prior to plant restart.

The public reporting burden for this collection of information is estimated to average 80 hours per response, including the time for reviewing instructions, searching existing data sources, gathering and maintaining the data needed, and completing and reviewing the collection of information.

The cost estimated for the collection of information is estimated to average $8000.00 ($100/hour expended).

(c) An anticipated schedule for NRC use of the information.

The NRC staff plans to make immediate use of the requested information and will develop an action plan for future generic actions based on the information collected by this proposed generic letter.

(d) A statement affirming that the request does not impose new requirements on the licensee, other than for the requested information.

Because the proposed generic letter only requests information from the PWR licensees, and the requested information has already been collected by the licensees (as stated by the PWR Owners Groups to the NRC during the meeting on August 24, 1995), the proposed generic letter does not impose new requirements on the licensees, other than submission of the requested information.

(xii) An assessment of how the proposed action relates to the Commission's Safety Goal Policy Statement.

The NRC staff feels that the proposed Generic Letter has no impact on the Commission's Safety Goal Policy Statement since it is only requesting information.