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

April 17, 1997

MEMORANDUM TO: Document Control Desk
Document Management Branch
Division of Information Support Services
Office of Information Resources Management

FROM: James W. Shapaker
Events Assessment and Generic Communications Branch
Division of Reactor Program Management
Office of Nuclear Reactor Regulation

SUBJECT: DOCUMENTS ASSOCIATED WITH NRC GENERIC LETTER 97-01, DEGRADATION OF CONTROL ROD DRIVE MECHANISM AND OTHER VESSEL CLOSURE HEAD PENETRATIONS (TAC No. M95280)

The Materials and Chemical Engineering Branch (EMCB) in the Divisison of Engineering (DE) prepared the subject generic letter, which was issued on April 1, 1997, and given accession number 9703260336. There is material related to the subject generic letter that should be placed in the NRC Public Document Room and made available to the public. Therefore, by copy of this memorandum, I am providing the following documents to the NRC Public Document Room: (1) a copy of the published version of the subject generic letter, (2) a copy of the information paper (SECY-97-063) that was sent to the Commission, (3) a copy of each letter received in response to the notice of oportunity for public comment on the proposed generic letter that was published in the *Federal Register* on August 1, 1996, (4) a copy of the summary and resolution of public comments that were received, (5) a copy of the CRGR review package.

I request that you provide me with the Nuclear Documents System accession number for this memorandum. This information may be provided by telephone (415-1151) or by e-mail (JWS). In addition, please modify the appropriate NUDOCS entries to reflect the fact that the documents identified herein are related to Generic Letter 97-01.

Attachments:
As stated

DFD3/1

ID4R-5
Info/Generic Ltr

9704180036 970417
PDR I&E
MISC PDR

I&E

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

April 1, 1997

NRC GENERIC LETTER 97-01: DEGRADATION OF CONTROL ROD DRIVE MECHANISM NOZZLE AND OTHER VESSEL CLOSURE HEAD PENETRATIONS

Addressees

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

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 closure head penetrations and (2) require that all addressees provide to the NRC a written response to the requested information. The information requested is needed by the NRC staff to verify compliance with 10 CFR 50.55a and 10 CFR Part 50, Appendix A, GDC 14, and to determine whether an augmented inspection program, pursuant to 10 CFR 50.55a(g)(6)(ii), is required.

Background

Primary Water Stress Corrosion Cracking of Vessel Closure Head Penetrations

Most PWRs have Alloy 600 CRDM nozzle and other vessel head closure 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. (Figure 1 is for illustrative purposes only and is not intended to be indicative of every nuclear steam supply system (NSSS) vendor's CRDM design.) The weld between the nozzle top and bottom pieces 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 CRDM nozzle and other VHPs are basically the same for all PWRs worldwide, which use a U.S. design (except in Germany and Russia). The areas of interest for potential cracking are the weld between the nozzle and reactor vessel head, and the portion of the nozzle inside the reactor vessel head above the nozzle-to-vessel weld.

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 CRDM nozzle 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) [0.12 to 0.16 in.] between the nozzle and the sleeve.

~~9703260336~~ 12pp

Beginning in 1986, leaks have been reported in several Alloy 600 pressurizer instrument nozzles at both domestic and foreign reactors from several different NSSS vendors. 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. 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. Further, with the exception of the leak found at Bugey 3 during hydrostatic testing, the NRC staff is not aware of any failure of an Alloy 600 vessel closure head penetration during plant operation. The NRC staff issued Information Notice (IN) 90-10, "Primary Water Stress Corrosion Cracking (PWSCC) of Inconel 600," dated February 23, 1990, to inform the nuclear industry of the issue.

In September 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, Sweden, Switzerland, Spain, and Japan were performed, and additional VHPs with axial cracks were detected in several European plants. 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. European and Japanese utilities have taken steps to detect and mitigate the PWSCC damage and to detect the leakage at an early stage. European and Japanese utilities have inspected most of the CRDM nozzles and repaired the nozzles or replaced the vessel heads as appropriate. In Japan, the three most susceptible vessel heads are being replaced, even though no cracks were found in the nozzles of these heads. In France, Élecricité de France (EdF) is planning on replacing all vessel heads as a preventative measure. Inservice inspection of the upper head is now required in Sweden. Removable insulation on the vessel head and leakage monitoring systems are installed at French and Swedish plants for early detection of leakage.

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 closure 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.

Intergranular Attack of CRDM Penetration Nozzle at Zorita

In 1994, circumferential intergranular attack (IGA) associated with the weld between the inner surface of the reactor closure head and the CRDM penetration (usually referred to as the J-grove 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 bead 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 [10.57 U.S. gallons] of cation resin entered the reactor coolant system (RCS). In September 1981, a mixed bed demineralizer screen failed and between 200 to 320 liters [52.83 to 84.54 U.S. gallons] 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 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.

Westinghouse 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. Westinghouse 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. Westinghouse concluded that no immediate safety issue is involved and that the conclusions in its CRDM safety evaluation remain valid. Westinghouse 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 bead 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 detected 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 [1.81 in.], 16 mm [0.63 in.], and 6 to 8 mm [0.24 to 0.31 in.] in length, and the deepest flaw was 6.8 mm [0.27 in.] deep. The tip of the 46-mm [1.81 in.] 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, none of the PWR Owners Groups (i.e., WOG, B&WOG, or CEOG) 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 primarily an economic 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 primarily economic.

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 it is probable that VHPs at other plants contain similar axial cracks. 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 susceptibility 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 closure head provides the vital function of maintaining 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,

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 continues to believe that an integrated, long-term program, which includes periodic inspections and monitoring of VHPs, is necessary. This was the conclusion of the staff's November 19, 1993, safety evaluation, which stated, in part, "...the staff recommends that you consider enhanced leakage detection by visually examining the reactor vessel head until either inspections have been completed showing absence of cracking or on-line leakage detection is installed in the head area ... nondestructive examinations should be performed to ensure there is no unexpected cracking in domestic PWRs. These examinations do not have to be conducted immediately ... As the surveillance walkdowns proposed by NUMARC are not intended for detecting small leaks, it is conceivable that some affected PWRs could potentially operate with small undetected leakage at CRDM/CEDM penetrations. In this regard, the staff believes that it is prudent for NUMARC to consider the implementation of an enhanced leakage detection method for detecting small leaks during plant operation." 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 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.

Requested Information

The information requested in item 1 is needed by the NRC staff to verify compliance with 10 CFR 50.55a and 10 CFR Part 50, Appendix A, GDC 14, and to determine whether an augmented inspection program of the weld between the penetration nozzle and reactor vessel head as well as the portion of the nozzle above the weld is required, pursuant to 10 CFR 50.55a(g)(6)(ii), while the information requested in item 2 relates to the occurrence of resin bead intrusion in PWRs, such as occurred at Zorita.

Within 120 days of the date of this generic letter, each addressee is requested to provide a written report that includes the following information for its facility:

1. Regarding inspection activities:

   1.1 A description of all inspections of CRDM nozzle and other VHPs performed to the date of this generic letter, including the results of these inspections[1].

   1.2 If a plan has been developed to periodically inspect the CRDM nozzle and other VHPs:

      a. Provide the schedule for first, and subsequent, inspections of the CRDM nozzle and other VHPs, including the technical basis for this schedule.

      b. Provide the scope for the CRDM nozzle and other VHP inspections, including the total number of penetrations (and how many will be inspected), which penetrations have thermal sleeves, which are spares, and which are instrument or other penetrations.

   1.3 If a plan has not been developed to periodically inspect the CRDM nozzle and other VHPs, provide the analysis that supports why no augmented inspection is necessary.

   1.4 In light of the degradation of CRDM nozzle and other VHPs described above, provide the analysis that supports the selected course of action as listed in either 1.2 or 1.3, above. In particular, provide a description of all relevant data and/or tests used to develop crack initiation and crack growth models, the methods and data used to validate these models, the plant-specific inputs to these models, and how these models substantiate the susceptibility evaluation. Also, if an integrated industry inspection program is being relied on, provide a detailed description of this program.

2. Provide a description of any resin bead intrusions, 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:

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

   2.2 What were the durations of these intrusions?

   2.3 Does the plant's RCS water chemistry Technical Specifications follow the EPRI guidelines?

---

1 Those licensees that have previously submitted the requested information need not resubmit it, but may instead reference the appropriate correspondence in their response to this Generic Letter.

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

2.5 Identify any conductivity excursions which may be indicative of resin intrusions. Provide a technical assessment of each excursion and any followup actions.

2.6 Provide an 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

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

NRC staff will review the responses to this generic letter and if concerns are identified, affected addressees will be notified.

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 CRDM nozzles and other VHPs. 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) NUREG/CR-6245, "Assessment of Pressurized Water Reactor Control Rod Drive Mechanism Nozzle Cracking," dated October 1994.

(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.

Backfit Discussion

Under the provisions of Section 182a of the Atomic Energy Act of 1954, as amended, and 10 CFR 50.54(f), this generic letter transmits an information request for the purpose of verifying compliance with applicable existing regulatory requirements. Specifically, the requested information would enable the NRC staff to determine whether or not the licensees' 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 requested information is also needed to determine whether an augmented inspecttion program, pursuant to 10 CFR 50.55a(g)(6)(ii), is required to maintain public health and safety.

Additionally, no backfit is either intended or approved in the context of issuance of this generic letter. Therefore, the staff has not performed a backfit analysis.

Federal Register Notification

A notice of opportunity for public comment was published in the *Federal Register* (61 FR 40253) on August 1, 1996, and extended on August 22, 1996 (61 FR 43393). Comments were received from seven licensees, two industry organizations, and one Code Committee. 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.

Thomas T. Martin, 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. Figure 1. Typical Control Rod Drive Mechanism Nozzle
2. List of Recently Issued NRC Generic Letters

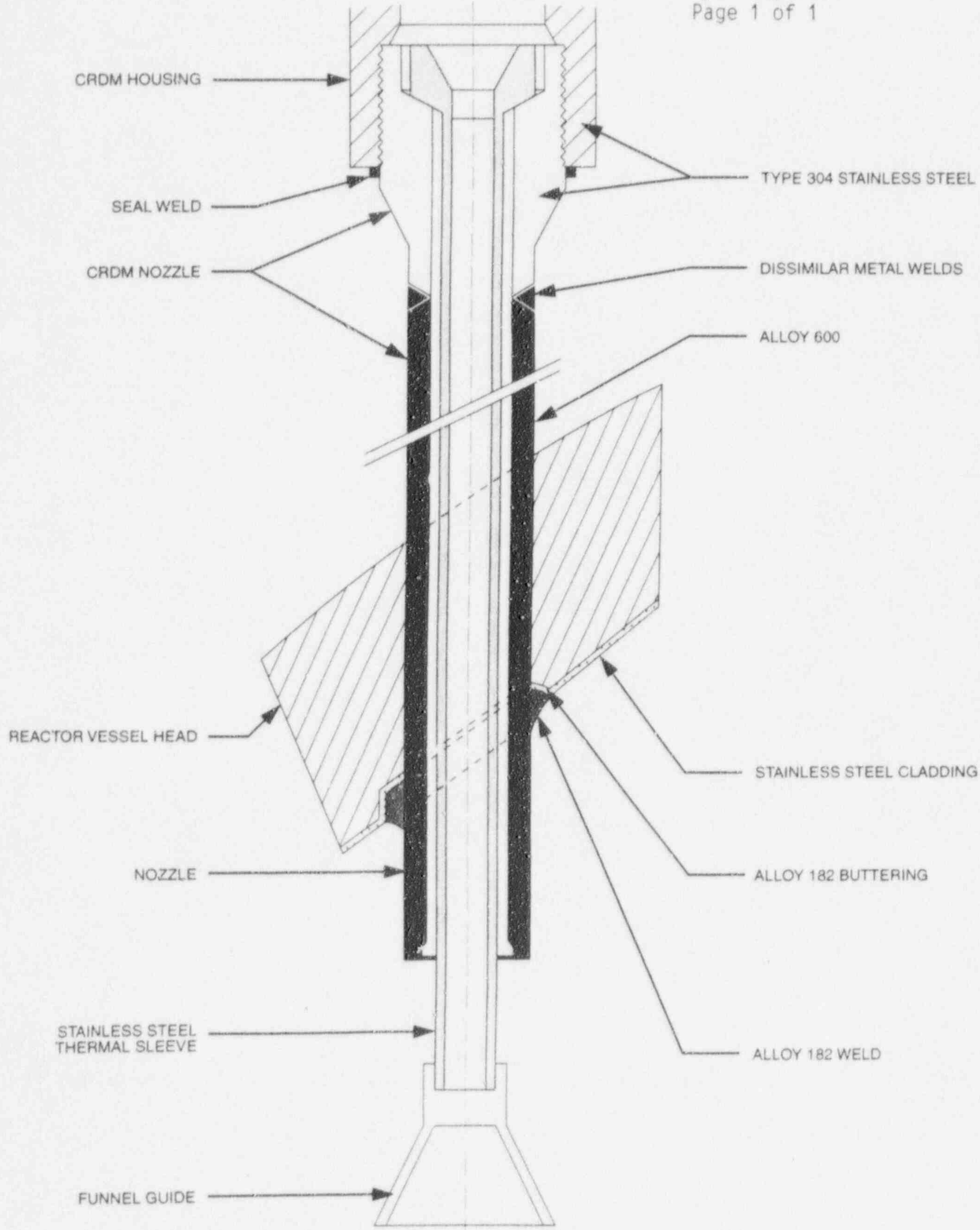

Figure 1. Typical control rod drive mechanism nozzle.

LIST OF RECENTLY ISSUED GENERIC LETTERS

Generic Letter	Subject	Date of Issuance	Issued To
95-06, SUPP. 1	CHANGES IN THE OPERATOR LICENSING PROGRAM	02/31/97	ALL HOLDERS OF OLs (EXCEPT THOSE LICENSEES OF PERMANENTLY SHUTDOWN REACTORS WHO ARE NO LONGER REQUIRED TO UTILIZE LICENSED REACTOR OPERATORS) FOR NPRs
96-07	INTERIM GUIDANCE ON TRANSPORTATION OF STEAM GENERATORS	12/05/96	ALL HOLDERS OF OLs AND DECOMMISSIONING FACILITIES WITH POSSESSION-ONLY LICENSES FOR PRESSURIZED-WATER NPRs
96-06	ASSURANCE OF EQUIPMENT OPERABILITY AND CONTAINMENT INTEGRITY DURING DESIGN-BASIS ACCIDENT CONDITIONS	11/13/96	ALL HOLDERS OF OLs FOR NPRs, EXCEPT FOR THOSE LICENSES THAT HAVE BEEN AMENDED TO POSSESSION-ONLY STATUS
96-05	PERIODIC VERIFICATION OF DESIGN-BASIS CAPABILITY OF SAFETY-RELATED MOTOR-OPERATED VALVES	09/18/96	ALL HOLDERS OF OLs (EXCEPT THOSE LICENSES THAT HAVE BEEN AMENDED TO POSSESSION-ONLY STATUS) OR CPs FOR NPRs
96-04	BORAFLEX DEGRADATION IN SPENT FUEL POOL STORAGE RACKS	06/26/96	ALL HOLDERS OF OLs FOR NPRs
95-09, SUPP. 1	MONITORING AND TRAINING OF SHIPPERS AND CARRIERS OF RADIOACTIVE MATERIALS	04/05/96	ALL U.S. NUCLEAR REGULATORY COMMISSION LICENSEES

OL = OPERATING LICENSE
CP = CONSTRUCTION PERMIT
NPR = NUCLEAR POWER REACTORS

POLICY ISSUE
(Information)

March 18, 1997 SECY-97-063

FOR: The Commissioners

FROM: L. Joseph Callan
Executive Director for Operations

SUBJECT: PROPOSED NRC GENERIC LETTER: "DEGRADATION OF CONTROL ROD DRIVE MECHANISM AND OTHER VESSEL CLOSURE HEAD PENETRATIONS"

PURPOSE:

To inform the Commission, in accordance with the guidance in the December 20, 1991, memorandum from Samuel J. Chilk to James M. Taylor regarding SECY-91-172, "Regulatory Impact Survey Report - Final," of the staff's intent to issue the subject generic letter. The generic letter requests holders of operating licenses for pressurized water reactors (PWRs), except those who have certified to a permanent cessation of operations, to describe their program for ensuring the timely inspection of PWR control rod drive mechanism (CRDM) and other vessel closure head penetrations (VHPs) and require that all addressees provide to the NRC a written response to the requested information. The information requested in the generic letter is needed by the NRC staff to verify compliance with 10 CFR 50.55a and 10 CFR Part 50, Appendix A, GDC 14, and to determine whether an augmented inspection program, pursuant to 10 CFR 50.55a(g)(6)(ii), is required. The generic letter also requests information related to the potential for resin bead intrusions, such as occurred at the Zorita plant in Spain.

A copy of the proposed generic letter is attached.

SUMMARY:

In September 1991, cracks were found in an Alloy 600 VHP in the reactor head of Bugey 3, a French PWR. Subsequently, most foreign countries have implemented VHP inspection programs, and VHP cracking was detected in several European plants. About 2 percent of the VHPs examined to date contain short, axial cracks. European and Japanese utilities have taken steps to detect and mitigate the PWSCC damage and to detect the leakage at an early stage at most of their plants. European and Japanese utilities have inspected most of the CRDM nozzles and repaired the nozzles or replaced the vessel heads as

CONTACT: C. E. Carpenter, Jr.
(301) 415-2169

NOTE: TO BE MADE PUBLICLY AVAILABLE IN 5 WORKING DAYS FROM THE DATE OF THIS PAPER

appropriate. In Japan, the three most susceptible vessel heads are being replaced, even though no cracks were found in the nozzles of these heads. Élecricité de France (EdF) is planning on replacing all vessel heads as a preventative measure. Inservice inspection of the upper head is now required in Sweden, and replacement of the Ringhals 2 vessel head was planned. Removable insulation on the vessel head and leakage monitoring systems are installed at French and Swedish plants for early detection of leakage.

Since the discovery of cracking at Bugey 3, the NRC staff has met with the PWR Owners Groups [Westinghouse Owners Group (WOG), the Combustion Engineering Owners Group (CEOG), and the Babcock & Wilcox Owners Group (B&WOG)] numerous times to discuss their respective programs for investigating PWSCC of Alloy 600 and to assess the possibility of cracking of VHPs in their respective plants. In February 1993, the PWR Owners Groups submitted safety assessments to the NRC on this issue. After reviewing the industry's safety assessments, the safety significance of the cracking that occurred, as well as the repair and replacement activities at the affected facilities, 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. This conclusion was based on the cracks being predominately axial in orientation and that circumferential cracking, which could lead to VHP severance, was not expected. Further, the industry had committed to perform inspections at three U.S. plants.

During 1994, four U.S. plants conducted VHP examinations and two found indications. For one of the units, it could not be shown that the cracking would not exceed ASME Code margins of safety and repairs had to be made. 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 detected. The inspection at the Oconee Nuclear Generating Station in the fall of 1994 revealed 20 indications in one penetration. 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, and repairs were needed. Some high-stress areas were examined on each outer ring CRDM penetration at North Anna Unit 1 during its spring 1996 refueling outage, and no indications were observed.

The NRC staff was informed that the PWR Owners Groups (i.e., WOG, B&WOG, CEOG) have developed a susceptibility model for VHPs and that all domestic PWRs have been modeled and the ranking has been given to each licensee. At present, none of the PWR Owners Groups has submitted its models and assessments to the NRC staff for review.

In 1994, circumferential intergranular attack (IGA) was discovered in the CRDM penetrations at Zorita, a Spanish reactor. The IGA, a different degradation mechanism than the PWSCC described above, is believed to have resulted from two cation resin bead ingress events, which resulted in high concentrations of sulfates. Four of the active and 16 of the 17 spare CRDM penetrations at Zorita showed axial and circumferential stress corrosion cracking and IGA. Westinghouse notified the PWROGs of the Zorita incident by issuing NSAL-94-028, which suggested that U.S. PWR plants review their RCS chemistry and other

operating records pertaining to sulfur ingress events. The results of any 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 bead intrusions have occurred at any U.S. PWR.

DISCUSSION:

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 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). Appropriate monitoring for degradation of the CRDM and other VHPs is important in the long term based on the possibility of (1) exceeding the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code 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.

The NRC Staff is not establishing a new position in this generic letter. The proposed generic letter is a request for information from licensees. It does not require that licensees perform inspections but it does request licensees to inform the NRC staff whether or not they plan to inspect and the basis for these plans. The information requested in the generic letter will allow the staff to determine if planned industry actions provide reasonable assurance of compliance with 10 CFR 50.55a and 10 CFR Part 50, Appendix A, GDC 14, and to determine if the imposition of an augmented inspection program, pursuant to 10 CFR 50.55a(g)(6)(ii), is required. Additionally, the information requested related to the potential for resin bead intrusions, such as occurred at Zorita, is needed to ascertain whether the potential exists for U.S. plants to have more severe cracking than observed to date.

A notice of opportunity for public comment was published in the *Federal Register* (61 FR 40253) on August 1, 1996, and extended on August 22, 1996 (61 FR 43393). Comments were received from seven licensees, two industry organizations, and one Code Committee. Copies of the staff evaluation of these comments have been made available in the public document room. The comments on the proposed generic letter focused on (1) the need for the generic letter, (2) editorial comments, and (3) backfit justification. The staff has revised the generic letter to clarify certain areas.

The proposed generic letter had formal review waived by the CRGR prior to its issuance for public comment. Following consideration of public comments, the staff sent the proposed generic letter to the CRGR with a description of the staff's response to public comments and of changes made to the proposed generic letter as a result of those comments. The revised proposed generic letter was reviewed by the CRGR during Meeting 299 on January 28, 1997, and endorsed by the CRGR during Meeting No. 300 on February 4, 1997. The staff incorporated comments made by the CRGR in those meetings.

The Office of the General Counsel has reviewed the proposed GL and has no legal objections.

The proposed generic letter is a "rule" for purposes of the Small Business Regulatory Enforcement Act (5 U.S.C., Chapter 8). The staff has received confirmation from the Office of Management and Budget that the generic letter is a non-major rule.

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

L. Joseph Callan
Executive Director
for Operations

Attachment:
Proposed NRC Generic Letter: "Degradation of Control Rod Drive Mechanism and Other Vessel Closure Head Penetrations"

DISTRIBUTION:
Commissioners
OGC
OCAA
OIG
OPA
OIP
OCA
ACRS
EDO
REGIONS
SECY

. DS09
J. Shapaker
G. Carpenter

61 FR 40253
Aug. 1, 1996

NEI

RECEIVED

NUCLEAR ENERGY INSTITUTE

1996 AUG -7 PM 3:08

RULES REVIEW
USNRC

(1)

Ralph E. Beedle
SENIOR VICE PRESIDENT AND
CHIEF NUCLEAR OFFICER
NUCLEAR GENERATION

August 6, 1996

Chief, Rules Review and Directives Branch
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

The purpose of this letter is to request an extension of the comment period for the "Proposed Generic Communication; Primary Water Stress Corrosion Cracking of Control Rod Drive Mechanism and Other Vessel Head Penetrations," (61 *Federal Register* 40253, August 1, 1996). We respectively request that the comment period be extended from September 3, 1996, to October 3, 1996.

The Nuclear Energy Institute, on behalf of the nuclear utility industry, will be developing comments on this draft generic letter for submittal to the NRC. We are requesting this comment period extension to permit sufficient time for industry representatives to assemble and develop comments. This extension will also allow sufficient time to transmit our draft comments to all nuclear utilities for consideration in developing and submitting plant-specific comments. We believe that a one-month extension is necessary to accommodate industrywide review of the proposed generic letter.

The appropriateness of this extension request is supported by the NRC staff's November 19, 1993, safety evaluation report (SER) conclusion that states primary water stress corrosion cracking of Alloy 600 head penetrations "is not a significant safety issue at this time as long as surveillance walkdowns [visual inspection] in accordance with GL-88-05 continue." Utilities are continuing the walkdowns. Furthermore, recent utility volumetric inspections results confirm that the PWR Owners Groups safety evaluations considered in the NRC staff's SER remain valid.

We also request that the NRC staff notify either Alex Marion (202-739-8080) or Kurt Cozens (202-739-8085) of the NEI staff of the disposition of this extension request.

Sincerely,

Ralph E. Beedle

REB/KOC/ead

c: Mr. C.E. (Gene) Carpenter, Jr. (NRR/DRPE/PDI-1)
Mr. Brian W. Sheron (NRR/DE)

9608090005 1P

J. Shapaker

Aug. 1, 1996

2

OPPD
Omaha Public Power District
444 South 16th Street Mall
Omaha NE 68102-2247

September 26, 1996
LIC-96-0122

U. S. Nuclear Regulatory Commission
Attn: Chief, Rules Review and Directives Branch
Mail Stop T-6D-69
Washington, D.C. 20555-0001

References: 1. Docket No. 50-285
2. Federal Register Volume 61, No. 149, dated August 1, 1996 (61 FR 40253)

Subject: Comments on Proposed Generic Communication Regarding Primary Water Stress Corrosion Cracking of Control Rod Drive Mechanism and Other Vessel Head Penetrations

The Omaha Public Power District (OPPD) has reviewed the proposed generic communications regarding primary water stress corrosion cracking of control rod drive mechanisms and other vessel head penetrations. OPPD, as the licensee for Fort Calhoun Station and a member of the Combustion Engineering Owner's Group (CEOG), has been monitoring this issue and has been involved in the development of the initial CEOG susceptibility assessments for the CEDM nozzles at CEOG plants. OPPD plans to participate in an updated assessment of the CEDM nozzles, which will incorporate the results of the CEDM testing performed at the Palisades nuclear plant.

OPPD has the following specific comments on the proposed Generic Letter:

1. It should be noted in the background information that no CEDM nozzles in any plants worldwide have failed during plant operation. Evidence of cracking has been revealed during planned inspections. As alluded to in the proposal, any through-wall cracking would be slow, result in detectable leakage, and provide an opportunity to take corrective action, because the leak rates of primary systems are tracked during the operation of all nuclear plants.

2. The proposed response period of 90 days may be insufficient given the recognized potential data collection difficulties and the fact that the owners groups may still be completing or updating their susceptibility assessments. We suggest that the NRC staff coordinate the issuance and response timing of the Generic Letter with the owners groups.

9610080152 2pp

Please contact me if you have any questions.

Sincerely,

James W. Tills
For

T. L. Patterson
Division Manager
Nuclear Operations

TCM/tcm

c: Winston & Strawn
L. J. Callan, NRC Regional Administrator, Region IV
L. R. Wharton, NRC Project Manager
W. C. Walker, NRC Senior Resident Inspector

J. Shapaker

Codes and Standards

Aug. 1, 1996

ASME International

Tel 212-705-8500
Fax 212-705-8501

345 East 47th Street
New York NY 10017-2392
USA

September 30, 1996

Mr. David Meyer
Chief, Rules Review and Directives Branch
US Nuclear Regulatory Commission
Washington, DC 20555-0001

Subj: Proposed Generic Communication: Primary Water Stress Corrosion Cracking of Control Rod Drive Mechanisms and Other Head Penetrations (61 Fed. Reg. 40253)

Dear Mr. Meyer:

Enclosed are comments resulting from review by individual members of the ASME BPVC Subcommittee on Nuclear Inservice Inspection. This review is not to be construed as a position or opinion on the subject document by ASME; rather, the enclosed comments are submitted as a constructive public service, and represent the opinion of individual committee members.

Yours truly,

George Fechter

George Fechter, Secretary
ASME BPVC Subcommittee on
Nuclear Inservice Inspection
(212)705-8018

cc with encl.
J. Perry
D. Landers
D. Canonico
T. Mawson
G. Eisenberg

~~9610080097~~ 3PP

COMMITTEE CORRESPONDENCE

committee: Section XI Subcommittee

address writer care of: ABB-Combustion Engineering
2000 Day Hill Rd.
Windsor, CT 06095-1521

subject: Comments on draft PWSCC Generic Letter

date: September 27, 1996

copy to: Westinghouse Energy Center
P.O. Box 355
Pittsburgh, PA 15230-0355

to: Mr. George Fechter
ASME Codes and Standards

We have reviewed the NRC "Proposed Generic Communication: Primary Water Stress Corrosion Cracking of Control Rod Drive Mechanisms and Other Head Penetrations"(61 Fed. Reg. 40253) and have a number of comments to bring to their attention.

Members of the ASME Section XI subcommittee are concerned that additional inspections are being imposed on the industry for what appears to be a non-safety issue. It is our concern that these additional inspection requirements are being justified based on the lack of action by Section XI on this subject. This is not the case at all.

The Section XI Subcommittee has been monitoring this issue closely since the condition was first identified in France, and it has been kept up to date by regular briefings of the Executive Committee at each meeting(four times per year). Their conclusion to date on this issue is that there are no safety concerns that cannot be addressed by the regular inspections for boron deposits already required. To date we have received no requests to add additional inspection requirements for the head penetrations from any of our members, which include representatives of industry, utilities, national laboratories, and the NRC.

The primary concern of the Section XI Subcommittee, as with all other committees in the ASME Boiler and Pressure Vessel Code, is safety. The principle focus of the Section XI Subcommittee is that the integrity of the reactor coolant pressure boundary is maintained. Industry experience has indicated that cracking that has occurred in the control rod drive penetrations is not a safety concern. After more than 5000 penetrations have been inspected worldwide, only one penetration has been found to have a through-wall crack from PWSCC.

The proposed generic letter comes as a surprise to the Subcommittee, and we respectfully request that this letter be forwarded to the NRC with our request that they remove any implications from the letter that Section XI has been ignoring this issue. Please address the NRC care of Mr David L. Meyer, Chief,Rules Review and Directives Branch, USNRC, Washington, DC 20555-0001.

Sincerely,

Owen Hedden

Owen Hedden, Chairman
Subcommittee on Inservice Inspection

Warren H. Bamford

Warren Bamford, Chairman
Subgroup on Evaluation Standards

cc: James A. Perry, BNCS
Gery M. Eisenberg, ASME
Domenic A. Canonico, ABB-CE
Gil Millman, USNRC
Tom Mawson, Northeast Utilities
Don Landers, Teledyne

0509
J. Shapaker

61 FR 40253
Aug. 1, 1996

Florida Power & Light Company, P.O. Box 14000, Juno Beach, FL 33408-0420

FPL

(4)

OCT 02 1996

L-96-249

Chief, Rules Review and Directives Branch
U.S. Nuclear Regulatory Commission
Mail Stop T-6D-69
Washington, DC 20555-0001

Subject: Proposed Generic Communication; Primary Water Stress Corrosion Cracking of Control Rod Drive Mechanism and Other Vessel Head Penetrations (61 FR 40253, dated August 1, 1996)
Notice of Opportunity for Public Comment

On August 1, 1996, the Nuclear Regulatory Commission published for public comment, "Proposed Generic Communication; Primary Water Stress Corrosion Cracking of Control Rod Drive Mechanism and Other Vessel Head Penetrations." The issuance of the proposed generic letter would request that addressees describe their program for ensuring the timely inspection of PWR control rod drive mechanism and other vessel head penetrations and require that all addressees provide a written response to the NRC regarding this generic letter. These comments are submitted on behalf of Florida Power & Light (FPL), a licensed operator of two nuclear power plant units in Dade County, Florida and two units in St. Lucie County, Florida.

The Nuclear Energy Institute (NEI) is providing comments on the proposed generic letter (GL) on behalf of the industry. FPL endorses the NEI comments. Additionally, the Nuclear Utility Backfitting and Reform Group (NUBARG) is providing comments on the proposed GL. FPL endorses the NUBARG comments.

FPL appreciates the opportunity to comment on the proposed GL.

Very truly yours,

W. H. Padmanos Jr.
for W. H. Bohlke
Vice President
Nuclear Engineering

9610160179 1P

0501 J. Shapaker Aug. 1, 1996

(5)

WINSTON & STRAWN

35 WEST WACKER DRIVE
CHICAGO, ILLINOIS 60601-9703

200 PARK AVENUE
NEW YORK, NY 10166-4193

1400 L STREET, N.W.
WASHINGTON, D.C. 20005-3502

(202) 371-5700

FACSIMILE (202) 371-5950

6 RUE DU CIRQUE
75008 PARIS, FRANCE

SULAYMANIYAH CENTER
RIYADH 11495, SAUDI ARABIA

43 RUE DU RHONE
1204 GENEVA, SWITZERLAND

October 3, 1996

VIA MESSENGER

U.S. Nuclear Regulatory Commission
Rules, Review and Directives Branch
Two White Flint
11545 Rockville Pike
Rockville, MD 20852-2738

Re: **Comments on Primary Water Stress Corrosion Cracking of Control Rod Drive Mechanism and Other Vessel Head Penetrations; Proposed Generic Communication: 61 Fed. Reg. 40,253 (August 1, 1996); 61 Fed. Reg. 43,393 (August 22, 1996)**

ATTN: Rules, Review and Directives Branch

On August 1, 1996, the Nuclear Regulatory Commission (NRC) issued the above-captioned proposed generic communication for public comment. Provided below are the comments of the Nuclear Utility Backfitting and Reform Group (NUBARG).[1] These comments concern the backfitting implications of the proposed generic communication. We support NEI's comments on the substantive aspects of the proposed Generic Letter.

The generic letter would require licensees to provide written responses, including a description of their programs for ensuring the timely inspection of PWR control rod drive mechanisms (CRDMs) and other vessel head penetrations (VHPs). The proposed generic letter states that "[i]f 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." 61 Fed. Reg. 40,253, 40,255 (1986). The Staff indicates that the integrated, long-term program that would include periodic inspections and monitoring is necessary in order to verify that the margins required by the

---

[1] NUBARG is a consortium of 16 nuclear utilities formed in the early 1980s, which participated actively in the development of the NRC's backfitting rule (10 C.F.R. § 50.109) in 1985, and which has closely monitored the NRC's application of the rule since that time.

9610080167 2pp

ASME Code continue to be satisfied, and to ensure that the safety significance of VHP cracking remains low. Id. In addition, the Staff believes that the program is needed to determine if the imposition of an augmented inspection program is required. Id.

The Staff asserts it is not going to perform a backfitting analysis because the proposed 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)." Id. at 40,256. In this regard, the Staff claims that it is "not establishing a new position for such compliance in this generic letter." Id.

We believe the Staff is required to perform a backfit analysis on the proposed imposition of the development of a plan to inspect the CRDM and other vessel head penetrations. Under 10 C.F.R. § 50.109, a backfit includes "the modification of or addition to . . . the procedures . . . required to . . . operate a facility; any of which may result from a new or amended provision in the Commission rules or the imposition of a [new] regulatory staff position. . . ." The backfit rule includes within its scope any means used by the NRC "to create an obligation upon licensees to change the . . . operation of a facility. . . ." 49 Fed. Reg. 47,034, 47,035 (1985). Imposition of this new inspection requirement goes beyond simply asking licensees to provide an information response. Instead, the new requirement to develop an inspection program is a modification or addition to the operational procedures resulting from the imposition of a new regulatory staff position. The Commission has indicated that a backfitting analysis should be performed in close cases. 50 Fed. Reg. 38,097, 38,102 (1985)("The Commission recognizes that there may be instances where it is not clear whether a backfit will follow an information request. Those cases should be resolved in favor of analysis."). In this case, a backfitting analysis is required under 10 C.F.R. § 50.109.

A backfitting analysis not only is legally required but also will ensure the protection of the public health and safety, as well as provide practical benefits to both licensees and the NRC Staff. Specifically, by performing a backfitting analysis, the Staff can ensure that the requested periodic inspections and monitoring activities are effective from a safety perspective and are cost beneficial. Moreover, by performing the analysis, the Staff can ensure that unnecessary downtime and adverse schedule impacts are avoided by licensees and that any resulting radiation exposure is assessed and minimized.

Sincerely,

Daniel F. Stenger
Kathryn M. Sutton

DS09
J. Shapaker

61FR 40253
Aug. 1, 1996

NEI

NUCLEAR ENERGY INSTITUTE

(6)

Ralph E. Beedle

September 30, 1996

Mr. David L. Meyer
Chief, Rules Review and Directives Branch
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Dear Mr. Meyer:

Enclosed are Nuclear Energy Institute (NEI)[1] comments on the "Proposed Generic Communication; Primary Water Stress Corrosion Cracking of Control Rod Drive Mechanism and Other Vessel Head Penetrations," (61 Fed. Reg. 40253, August 1, 1996). These comments were developed by an NEI task force comprised of representatives from utilities, PWR Owners Groups and EPRI. Additionally, these comments were forwarded to the industry for consideration by individual utility licensees in developing plant-specific comments.

NEI will continue to coordinate industry activities in managing primary water stress corrosion cracking (PWSCC) of vessel head penetrations. This coordination will involve EPRI and the PWR Owners Groups to ensure that necessary information is evaluated and communicated to utilities to support their decisions to conduct inspections. NEI continues to believe that the decision to conduct inspections rests with individual utility management after due consideration of susceptibility, evidence of boric acid deposition and economic risk. As in the past, NEI will continue to meet with NRC staff to discuss inspection results as they relate to the PWR Owners Group safety evaluations and inspection criteria, and the NRC's safety evaluation report. NEI believes this approach in managing this issue is appropriate and sufficient given the low safety concern. Therefore, NEI concludes that there is no technical or regulatory basis for this generic letter.

---

[1] NEI is the organization responsible for establishing unified nuclear industry policy on matters affecting the nuclear energy industry, including the regulatory aspects of generic operational and technical issues. NEI's members include all utilities licensed to operate commercial nuclear power plants in the United States, nuclear plant designers, major architect/engineering firms, fuel fabrication facilities, materials licensees, and other organizations and individuals involved in the nuclear energy industry.

~~9610070395~~ 10PP

eneral comments relating to the draft generic letter are provided in Enclosure 1 and are summarized as follows:

- The stated purpose of the draft generic letter is to determine if augmented inspections are warranted. However, the draft generic letter essentially requests licensees to define and commit to an augmented inspection program. If augmented inspections are determined by NRC to be necessary, then such inspections should be based on the safety significance of the vessel head penetrations experiencing primary water stress corrosion cracking, not whether or not licensees are performing augmented inspections.

- The NRC staff safety concerns have been addressed by the PWR Owners Groups' safety evaluations, which considered the possibility of through-wall cracks.

- The stated scope of the draft generic letter is primary water stress corrosion cracking. The resin intrusion at the Zorita Plant resulted in intergranular stress corrosion which is a different degradation mechanism. Since the Zorita resin intrusion was communicated to utility licensees by Information Notice 96-11, and new concerns have not been identified, it is not clear why the NRC staff is now requesting licensees to submit information on this topic.

Enclosure 2 provides detailed comments on the specific text of the proposed generic letter.

If you have questions concerning these comments, please contact Alex Marion (202-739-8080) or me.

Sincerely,

R. Beedle

Ralph E. Beedle

TET/AM/ead
Enclosures

c: C. E. (Gene) Carpenter, NRC/NRR
Brian Sheron, NRC/NRR
Jack Strosnider, NRC/NRR

GENERAL COMMENTS ON THE DRAFT GENERIC LETTER

1. Items 1 and 2 in the Required Information section essentially requests licensees to define and commit to an augmented inspection program. The stated purpose of the draft regulatory guide is to evaluate whether or not an augmented inspection program is necessary. The justification for the augmented inspection should be based on the safety significance of the vessel head penetration's (VHP) experiencing primary water stress corrosion cracking, not if licensees are currently performing augmented inspections.

2. On Page 10 of NUREG/CR-6245, Assessment of Pressurized Water Reactor Control Rod Drive Mechanism Nozzle Cracking, it states, "There are two major safety concerns associated with CRDM nozzle cracking. First, a crack could eventually lead to a rupture of the nozzle and, if the nozzle is severed, to ejection of the connected CRDM housing. Second, a through-wall crack would allow the borated reactor coolant to come in contact with the vessel head and cause boric acid corrosion of the low-alloy steel base metal." In the NRC staff's safety evaluation dated November 19, 1993, it states, "The primary safety concern associated with stress corrosion cracking in Alloy 600 is the potential for circumferential cracks. Extensive circumferential cracking could lead to ejection of a CRDM." These safety concerns were considered by the PWR Owners Group safety evaluations submitted to and accepted by the NRC staff. The draft generic letter has not identified any safety concerns that were not previously evaluated and dispositioned. Summaries of these safety evaluations are contained in NUREG/CR-6245 and the NEI's white paper titled, "Alloy 600 RPV Head Penetration Primary Water Stress Corrosion Cracking."

3. The second paragraph of the Discussion section states that the goal of the draft generic letter is to ". . .verify that the margins required by the ASME Code as specified in § 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 Par 50, Appendix A, GDC 14) is continued to be satisfied, ..." These goals are unique and separate from the stated purpose of the first paragraph in the Required Information section which states; "The information requested in Items 1 and 2, below, is required to determine if the imposition of an augmented inspection program is required, ..." Although not stated as such, the Discussions section appears to raise a question of compliance rather than determining if new regulatory requirements (augmented inspections per 10 CFR 50.55a(g)(6)(ii)) should be imposed. Utility licensees are presently in compliance with the requirements identified in the Discussion section based on the following:

- The design and fabrication of the reactor vessel heads satisfy all applicable ASME requirements.
- Only the welds that attach VHPs to the reactor head are within the scope of the inservice inspection requirements (ASME Section XI, Table IWB-2500-1, Examination Category B-E). As noted in NUREG/CR-6245, the VHP surface which could experience PWSCC is not expected to be within the scope of ASME inservice inspections. However, should inservice inspection identify indications, licensees will disposition them per the ASME Code.
- GDC-14 states, "The reactor coolant pressure boundary shall be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture." Licensees are meeting GDC-14 because:
  - The reactor vessel head was designed, fabricated, and erected to the ASME Code or other requirements approved by the NRC.
  - The PWR Owners Group safety evaluations, accepted by the NRC staff (dated November 19, 1993), addressed the potential for rapid crack propagation, gross rupture and abnormal leakage. These evaluations determined that PWSCC would either be arrested or would grow very slowly requiring years to obtain a critical length. Axial cracks require many years to obtain critical length. Circumferential cracking requires through-wall leakage and will take significantly more time than the 40-year licensed operating period. One conservative circumferential cracking evaluation estimated that it would take in excess of 90 years before gross failure would occur.
  - Licensees are presently performing inspections in accordance with NRC Generic Letter 88-05 to detect leakage that could occur during operation. If leakage is detected, repairs and corrective action will be performed. In addition, corrective action is required if leakage exceeds the Technical Specification criteria.
  - This approach to GDC compliance is consistent with the leak-before-break criteria applied to other primary piping systems.

4. The Required Information section asks licensees to summarize the inspections they have performed, define the inspections they plan to perform or justify why inspections are not being performed. The NRC staff witnessed the VHP inspections performed by licensees (five plants) and has received written reports on the results. Hence, this is a redundant request for those licensees who have already performed inspections and requests submittal of information that the NRC already has in its possession. In addition, the NEI white paper discussed the method by which licensees are managing this issue, i.e., future inspections will be performed based on information sharing, predictive methodologies and tools, inspection results, and development of mitigation and repair technologies.

5. The topic of the draft generic letter is "Primary Water Stress Corrosion Cracking of Control Rod Drive Mechanism and Other Vessel Head Penetrations." The inclusion of a different form of degradation (intergranular stress corrosion cracking due to resin intrusion) is not warranted. PWSCC of Alloy 600 is an time dependent degradation mechanism. Intergranular stress corrosion cracking due to resin intrusion is an abnormal operating event. Furthermore, of the over 5200 penetrations inspected worldwide, no evidence has been observed that suggests resin induced intergranular stress corrosion cracking has occurred in any reactor vessel other than Zorita. This is strong evidence that resin induced intergranular stress corrosion cracking is an outlier event that is not generic.

6. 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," that advised licensees of the Zorita resin intrusion and potential intergranular stress corrosion cracking. It is unclear why the NRC has revised their position concerning a request for submitted information (Required Information, Item 3), since no additional resin intrusions concerns have occurred since IN 96-11 was issued. The extra burden on licensees to respond to Item 3 of the Required Information section is not justified.

DETAILED COMMENTS ON THE DRAFT GENERIC LETTER

CMT #	SECTION	PARAGRAPH	COMMENT	CORRECTIONS
1.	General	--	It is unclear why the draft generic letter (GL) did not reference nor discuss in detail the evaluations and conclusions contained in NUREG/CR-6245. This document provides a balanced evaluation of the safety evaluations performed by the PWR Owners Groups.	It would be beneficial to contrast the conclusions of NUREG/CR-6245 to the draft GL's definition of "long-term safety concerns."
2.	General	--	The phrase "other vessel head penetrations" used throughout the draft generic letter should be clarified to read "other reactor vessel closure head penetrations".	Additional clarity may be gained if the phrase "other vessel head penetrations" is altered to read "other reactor vessel closure head penetrations."
3.	Background	1st	Figure 1 and text appear to only discuss CRDMs designed by Westinghouse.	A description of the CE and B&W penetration design features would be beneficial.
4.	Background	3rd	The first sentence states that in 1989 the emerging issue was identified, then the second sentence states leakage has occurred since 1986. This is not chronologically correct and is confusing.	A chronologically sequenced paragraph would be easier to understand.
5.	Background	4th	The Bugey-3 cracking was discovered in September 1991.	Proper dates should be used.
6.	Background	4th	In the second sentence, it states that the Japanese have "uncovered" VHPs with cracks. We are unaware of any available reference stating that the Japanese have detected PWSCC cracks in their VHPs. A source reference should be provided. It would be more precise to state that cracks were "detected" rather than "uncovered" in this paragraph and throughout the document's text.	Provide a reference or delete Japan from the list of countries which have identified PWSCC in their VHPs. Improved clarity would be achieved if the word "uncovered" is changed to "detected."
7.	Background	6th	Sub-item (3). NUREG/CR-6245 states that the leakage would be detected "long" before significant damage to the reactor vessel head would occur.	The draft GL and NUREG/CR-6245 would be consistent if the word "long" was added before the word "before."
8.	Background	6th	The last two sentences discuss manual NDE and do not relate to the remainder of the paragraph. The merits of manual NDE and automatic tooling are not the subject of the draft generic letter.	Delete the last two sentences from the paragraph.
9.	Background	7th	The purpose of the EPRI NDE demonstration was not to qualify tooling or operators, but was limited to the demonstration of an inspection system's ability to detect and size defects.	The EPRI activities would be better described if the term "qualification" was modified.
10.	Background	7th	This paragraph appears to justify the draft generic letter based on advances in inspection techniques rather than assess the safety significance of PWSCC. This implies that inspections	Delete this paragraph.

CMT #	SECTION	PARAGRAPH	COMMENT	CORRECTIONS
			should be required because industry has voluntarily developed improved inspection methods. The paragraph should focus on safety concerns.	
11.	Background	8th	The description of the Zorita event could be more precise.	A more precise statement would be: "During the 1994 outage at Zorita (a Spanish reactor), visual inspection of the reactor vessel head discovered boron deposits on a single vessel head penetration. A more thorough inspection of this penetration detected a crack approximately two inches below the bimetallic weld. An extensive investigation and root cause evaluation were performed. It was determined that the indications were caused by intergranular stress corrosion cracking initiated by cation resin intrusion."
12.	Background	8th	The Zorita concern was primarily with the response of sensitized material attacked by reduced sulfur species.	It would be more precise to refer to "attack by sulfur species on sensitized materials."
13.	Background	8th	First sentence. Inspections at the Zorita Plant did not identify circumferential cracks in the J-groove weld, but found a through-wall crack at or near the bimetallic weld. In the third sentence, "resin bed" should be "resin bead." The text would be better understood if the measurements were provided in English as well as metric units, i.e., "liters" and "gallons."	These changes would provide factual clarity.
14.	Background	9th	It is our understanding that the NRC staff has Zorita resin intrusion reports and data that are not publicly available. It is difficult to assess the significance of the Zorita resin intrusion without all available information. In previous communications with the NRC staff, we have been told that these reports have been provided to all PWR Owners Groups. However, inquiries made to the PWR Owners Groups have not supported this. We request the NRC staff to place all informa'ior on the Zorita resin intrusion into the Public Document Room, and provide the opportunity for industry to evaluate.	Related reports and data should be made available.
15.	Background	9th and 10th	To maintain the chronological order of events, the 9th and 10th paragraphs should be switched.	Chronological order of these paragraphs would be beneficial.
16.	Background	10th	The draft generic letter does not discuss the recent VHPs re-inspections performed at Oconee and D.C. Cook, nor the VHP	It would be beneficial to document the most recent inspection activities and results.

CMT #	SECTION	PARAGRAPH	COMMENT	CORRECTIONS
			repair at D.C. Cook.	
17.	Background	10th	The NRC states that they have not been provided with the WOG resin intrusion review. IN 96-11 does not require any specific action by licensees. Furthermore, Westinghouse NSAL-94-028 did not request licensees to provide a response back to Westinghouse and no WOG report has been prepared.	The statement in this paragraph should reflect the comment.
18.	Background	13th	The citation of Westinghouse, Framatome Technologies, and Combustion Engineering are incorrect. The citations should be the PWR Owners Groups; i.e., WOG, B&WOG, and the CEOG.	Use the correct citations.
19.	Background	14	This paragraph states that "(t)he 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 that the NRC staff did not agree that the issue was only economic. This is not a correct interpretation of the NEI white paper. The white paper documents the extensive safety evaluations developed by the PWR Owners Groups which addressed all the safety concerns identified by the NRC staff. The method discussed in the white paper to manage RPV head penetration cracking acknowledges that the issue is not an immediate safety concern and that leak-before-break will occur. Using this knowledge, the management methodology discussed provides a four step approach; of which one step evaluates the economic considerations.	An appropriate statement would reflect the Section VII white paper text. It is the NRC staff's prerogative to disagree with positions taken in the white paper. However, the NRC staff should identify those safety concerns have not been addressed by the NRC approved PWR Owners Groups safety evaluations.
20.	Discussion	1st	The sentence starting, " Further, if any significant ..." is an absolute statement which has not been technically justified in this document nor the references. It would be technically correct if the sentence was revised to read, "Further, if any significant resin intrusions have occurred at U.S. plants such as occurred at Zorita, the resultant chemistry condition in combination with stress may be significant."	This change provides clarity.
21.	Discussion	2nd	The sentence which starts, " Cracking in the VHPs ..." is potentially misleading. While cracking has occurred in 116 of the 5146 penetrations inspected, it has not been observed in the large majority of VHPs. PWSCC is an age related degradation mechanism which could occur some time in the future, many years beyond the initial or renewed license or never.	A more precise statement would be "Cracking occurred in a few VHPs and could occur in others at some future time. An existing crack may continue to grow, but could stop."
22.	Discussion	2nd	The paragraph states that the NRC staff considers the 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	The PWR Owners Group safety evaluations addressed the safety concerns identified by NRC staff.

CMT #	SECTION	PARAGRAPH	COMMENT	CORRECTIONS
			for plant safety. These safety concerns are addressed by the PWR Owners Groups safety evaluations. These were summarized on Page 10 of NUREG/CR-6245, "Assessment of Pressurized Water Reactor Control Rod Drive Mechanism Nozzle Cracking," which states that "There are two major safety concerns associated with CRDM nozzle cracking. First, a crack could eventually lead to a rupture of the nozzle and, if the nozzle is severed, to ejection of the connected CRDM housing. Second, a through-wall crack would allow the borated reactor coolant to come in contact with the vessel head and cause boric acid corrosion of the low-alloy steel base metal..." In addition, the NRC staff's safety evaluation dated November 19, 1993, states that "The primary safety concern associated with stress corrosion cracking in Alloy 600 is the potential for circumferential cracks. Extensive circumferential cracking could lead to ejection of a CRDM..." Since the PWR Owners Groups safety evaluations evaluated a through-wall crack and ejection of the connected CRDM housing, it appears that the two long term concerns identified by the draft GL are less severe than those already evaluated.	
23.	Required Information	1.2.a	The concept of scheduling augmented inspections is inconsistent with the concept of "long term safety concerns." Given that technical safety concerns have been addressed, requesting a "technical basis" for a schedule is unclear.	Provide clarity.
24.	Required Information	1.2.b	The required information is unnecessarily prescriptive (e.g., the direction of inspection (top or bottom) will not affect the quality of an inspection which a licensee may choose to perform, the presence of thermal sleeves, etc.)	Delete as this level of detail is not necessary.
25.	Required Information	2.	The first sentence states, "... include the susceptibility ranking of your plant and the factors used to determine this ranking." This phrase is redundant with the first part of the sentence which states, "A description of the evaluation methods and results used to assess the susceptibility of the CRDM and other VHPs in your plant to PWSCC, ..."	Delete the phrase "... include the susceptibility ranking of your plant and the factors used to determine this ranking."
26.	Required Information	2.	The susceptibility models were not used as input to the PWR Owners Groups safety evaluations that were submitted and approved by the NRC staff. The susceptibility models and subsequent rankings may be used by licensees to make economic	Since it is not possible to make a safety determination with the susceptibility rankings, this paragraph should be deleted.

CMT #	SECTION	PARAGRAPH	COMMENT	CORRECTIONS
			evaluations, but are not sufficiently precise to be used in a safety assessment that may be submitted to the NRC staff. In addition, it is unclear how the NRC staff will use such models to evaluate a safety concern.	
27.	Required Information	2.	This requested information implies that the GL 88-05 visual inspection is inadequate to detect boric acid deposits and which could be caused by PWSCC. This implication is not supported by operating history and safety evaluations: • The only through-wall VHPs cracks (Bugey and Zorita) were detected by visual inspections. • GL 88-05 visual inspections are considered acceptable for detecting PWSCC in the remainder of the reactor coolant system. • A conservative definition for "long term safety concern" implied by NUREG-CR-6245 would infer a minimum of nine years after the initiation of a PWSCC through wall leak. Boric acid deposited over this time period would be readily observed using the GL 88-05 visual inspections.	Boric acid deposits will be identified by the visual inspections recommended in Generic Letter 88-05.
28.	Required Information	3.	The intergranular stress corrosion cracking resulting from a Zorita type resin intrusion is a different mechanism than the primary water stress corrosion cracking (PWSCC). The resin intrusion cracking is a degradation mechanism caused by an abnormal operating event and is not a age-related degradation mechanism like PWSCC. Furthermore, the predictive tools for PWSCC are not capable of predicting resin intrusion. It is noted that the VHP inspections performed on over 5200 penetrations at 87 plants worldwide did not identify any other plant that exhibited intergranular stress corrosion cracking similar to that exhibited at Zorita.	The resin induced intergranular stress corrosion cracking is different than the stated scope and should be deleted.
29.	Required Information	3.4	The draft generic letter has not provided a basis for supplying information on chlorides, fluorides, oxygen, boron, or lithium. The Zorita experience has been linked to the sulfates, but to our knowledge the other chemistry species have not been linked.	Delete.

Post Office Box 1295
Birmingham, Alabama 35201
Telephone (205) 868-5131

Aug. 1, 1996

DS09
J. Shapaker

(7)

Dave Morey
Vice President
Farley Project

Southern Nuclear Operating Company
the southern electric system

September 30, 1996

Docket Nos. 50-348
50-364

Mr. David L. Meyer
Chief, Rules Review and Directives Branch
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Comments on Proposed Generic Communication
"Primary Water Stress Corrosion Cracking of Control Rod Drive Mechanism and Other Vessel Head Penetrations"
(61 Federal Register 40253 dated August 1, 1996)

Dear Sir:

Southern Nuclear Operating Company has reviewed the proposed rule "Primary Water Stress Corrosion Cracking of Control Rod Drive Mechanism and Other Vessel Head Penetrations," published in the Federal Register on August 1, 1996. In accordance with request for comments, Southern Nuclear Operating Company is in total agreement with the NEI comments which are to be provided to the NRC.

Respectfully submitted,

D. N. Morey

Dave Morey

DNM/TWS

9610160065 3PP

cc: Southern Nuclear Operating Company
R. D. Hill, Plant Manager

U. S. Nuclear Regulatory Commission, Washington, DC
J. I. Zimmerman, Licensing Project Manager, NRR

U. S. Nuclear Regulatory Commission, Region II
S. D. Ebneter, Regional Administrator
T. M. Ross, Senior Resident Inspector

Georgia Power Company
40 Inverness Center Parkway
Post Office Box 1295
Birmingham, Alabama 35201
Telephone 205 877 7122

DS09
J. Shapaker

61 FR 40253
Aug. 1, 1996

8

Georgia Power
the southern electric system

C. K. McCoy
Vice President, Nuclear
Vogtle Project

October 1, 1996

Docket Nos. 50-321 50-424
50-366 50-425

HL-5247
LCV-0885

Mr. David L. Meyer
Chief, Rules Review and Directives Branch
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Comments on Proposed Generic Communication
"Primary Water Stress Corrosion Cracking of Control Rod Drive Mechanism and Other Vessel Head Penetrations"
(61 Federal Register 40253 dated August 1, 1996)

Dear Sir:

Georgia Power Company has reviewed the proposed rule "Primary Water Stress Corrosion Cracking of Control Rod Drive Mechanism and Other Vessel Head Penetrations," published in the Federal Register on August 1, 1996. In accordance with request for comments, Georgia Power Company is in total agreement with the NEI comments which are to be provided to the NRC.

Should you have any questions, please advise.

Respectfully submitted,

C.K. McCoy

C. K. McCoy

CKM/TWS

9610160073 2pp.

cc: Georgia Power Company
J. T. Beckham, Jr., Vice President - Plant Hatch
J. B. Beasley, General Manager - Vogtle Electric Generating Plant
H. L. Sumner, Jr., General Manager - Plant Hatch

U. S. Nuclear Regulatory Commission, Washington, DC
K. N. Jabbour, Licensing Project Manager - Hatch
L. L. Wheeler, Licensing Project Manager, Vogtle

U. S. Nuclear Regulatory Commission, Region II
S. D. Ebneter, Regional Administrator
B. L. Holbrook, Senior Resident Inspector - Hatch
C. R. Ogle, Senior Resident Inspector - Vogtle

HL-5247
LCV-0885
REES File: G.03.19

0504
J. Shapaker

Florida Power & Light Company, P.O. Box 14000, Juno Beach, FL 33408-0420

FPL

61 FR 40253
Aug. 1, 1996

OCT 0 2 1996

L-96-249

9

Chief, Rules Review and Directives Branch
U.S. Nuclear Regulatory Commission
Mail Stop T-6D-69
Washington, DC 20555-0001

Subject: Proposed Generic Communication; Primary Water Stress Corrosion Cracking of Control Rod Drive Mechanism and Other Vessel Head Penetrations (61 FR 40253, dated August 1, 1996)
Notice of Opportunity for Public Comment

On August 1, 1996, the Nuclear Regulatory Commission published for public comment, "Proposed Generic Communication; Primary Water Stress Corrosion Cracking of Control Rod Drive Mechanism and Other Vessel Head Penetrations." The issuance of the proposed generic letter would request that addressees describe their program for ensuring the timely inspection of PWR control rod drive mechanism and other vessel head penetrations and require that all addressees provide a written response to the NRC regarding this generic letter. These comments are submitted on behalf of Florida Power & Light (FPL), a licensed operator of two nuclear power plant units in Dade County, Florida and two units in St. Lucie County, Florida.

The Nuclear Energy Institute (NEI) is providing comments on the proposed generic letter (GL) on behalf of the industry. FPL endorses the NEI comments. Additionally, the Nuclear Utility Backfitting and Reform Group (NUBARG) is providing comments on the proposed GL. FPL endorses the NUBARG comments.

FPL appreciates the opportunity to comment on the proposed GL.

Very truly yours,

W. N. Paduano Jr.
fo W. H. Bohlke
Vice President
Nuclear Engineering

9610160179 1P

J. Shapaker

Aug. 1, 1996

Log # TXX-96484 (10)
File # 883

01

USNRC

TU ELECTRIC

October 3, 1996

C. Lance Terry
Group Vice President

Chief, Rules Review and Directives Branch
U. S. Nuclear Regulatory Commission
Washington, DC 20555-0001

SUBJECT: COMANCHE PEAK STEAM ELECTRIC STATION (CPSES) ENDORSEMENT OF NEI COMMENT LETTER ON PROPOSED NRC GENERIC COMMUNICATION, PRIMARY WATER STRESS CORROSION CRACKING OF CONTROL ROD DRIVE MECHANISM AND OTHER VESSEL HEAD PENETRATIONS

REF:
1) 61 Federal Register 40253, Proposed Generic Communication, Primary Water Stress Corrosion Cracking of Control Rod Drive Mechanism and Other Vessel Head Penetrations, dated August 1, 1996

2) Nuclear Energy Institute (NEI) letter, addressed to Chief, Rules Review and Directives Branch, USNRC, dated October 3, 1996

Gentlemen:

In response to the Federal Register notice of August 1, 1996 (Reference 1) TU Electric is providing comments on the proposed NRC Generic Communication "Primary Water Stress Corrosion Cracking of Control Rod Drive Mechanism and Other Vessel Head Penetrations."

TU Electric has reviewed and endorses the NEI letter (Reference 2). TU Electric agrees with the NEI discussed issues, recommendations and rationale. TU Electric further agrees that no technical or regulatory basis exists for this generic communication and recommends that NRC provide due consideration of the NEI comments in the final evaluation of the proposed generic communication.

Sincerely,

C. L. Terry

By: J S Marshall
J. S. Marshall
Generic Licensing Manager

RTB/grp

c - R. E. Beedle, NEI

P. O. Box 1002 Glen Rose, Texas 76043

9610160185 P

J. Shapaker

Duquesne Light Company Beaver Valley Power Station
P.O. Box 4
Shippingport, PA 15077-0004

Aug. 1, 1996

11

SUSHIL C. JAIN
Division Vice President
Nuclear Services
Nuclear Power Division

October 4, 1996

(412) 393-5512
Fax (412) 643-8069

Mr. David L. Meyer
Chief, Rules Review and Directives Branch
U. S. Nuclear Regulatory Commission
Mail Stop T-6D-69
Washington, DC 20555-0001

Subject: Beaver Valley Power Station, Unit No. 1 and No. 2
BV-1 Docket No. 50-334, License No. DPR-66
BV-2 Docket No. 50-412, License No. NPF-73
Proposed Generic Communication, "Primary Water Stress Corrosion Cracking of Control Rod Drive Mechanism and Other Vessel Head Penetrations"

Dear Mr. Meyer:

Duquesne Light Company (DLC) is responsible for the operation of Beaver Valley Power Station Units 1 and 2. DLC has reviewed the proposed generic communication, "Primary Water Stress Corrosion Cracking of Control Rod Drive Mechanism and Other Vessel Head Penetrations," which was published in the Federal Register on August 1, 1996 (61 FR 40253). DLC hereby submits the following comments.

DLC endorses the comments provided by the Nuclear Energy Institute (NEI). The NEI comments identify the key issues which need to be considered. DLC concurs with NEI that there is no technical or regulatory basis for this generic letter.

Thank you for the opportunity to comment on this issue. If you have any questions on this submittal, please contact Mr. Roy K. Brosi, Manager, Nuclear Safety Department, (412) 393-5210.

Sincerely,

Sushil C. Jain

DELIVERING QUALITY ENERGY

9610160120 - 1P

DS01
J. Shapaker

61FR 40253
Aug. 1, 1996

12

VIRGINIA POWER

October 4, 1996

Chief, Rules Review and Directives Branch
U. S. Nuclear Regulatory Commission
Washington, DC 20555

Serial No. 96-076

Gentlemen:

COMMENTS ON PROPOSED GENERIC LETTER:
PRIMARY WATER STRESS CORROSION CRACKING OF CONTROL ROD DRIVE MECHANISMS AND OTHER VESSEL HEAD PENETRATIONS

On August 1, 1996, the NRC requested comments on the "Proposed Generic Communication; Primary Water Stress Corrosion Cracking of Control Rod Drive Mechanisms and Other Vessel Head Penetrations," (61 Fed. Reg. 40253, August 1, 1996).

We have reviewed the NRC proposed generic letter and fully endorse the Nuclear Energy Institute review comments provided in their letter dated September 30, 1996.

We appreciate the opportunity to make comments on this proposed generic letter. Should you have any additional questions, please feel free to contact us.

Very truly yours,

M. R. Kansler, Vice President
Nuclear Engineering & Services

Attachment
cc: Mr. Thomas E. Tipton
Vice President, Operations and Engineering
Nuclear Energy Institute
1776 Eye Street Suite 300
Washington, DC 20006-3706

9610160157