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.

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.

<u>Discussion</u>

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.

<u>Requested Information</u>

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?

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[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 inspect- tion 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.

signed by

Thomas T. Martin, Director
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Attachments:
1. Figure 1. Typical Control Rod Drive Mechanism Nozzle

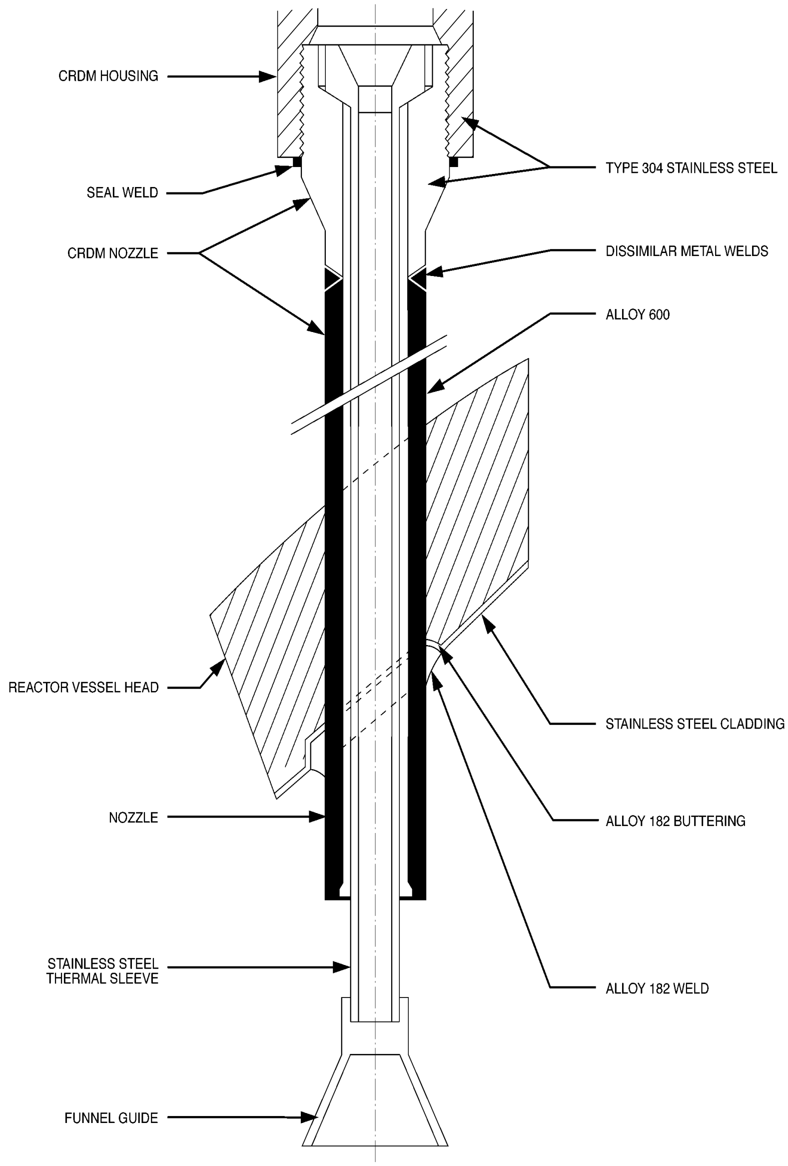

Figure 1. Typical control rod drive mechanism nozzle.