comments to the working group's consideration orally, or in writing, at times specified by the working group, Co-Chairs. Seating at the working group meetings will be on a first-come, first-served basis.

MEETING ANNOUNCEMENTS: No meeting is scheduled at this time. Announcements for the first, and subsequent, meetings will be made through the NRC's Meeting Announcement system. The meeting announcement system can be reached three ways:

1. Voice: 800–952–9674.
2. Electronic Bulletin Board: 800–952–9676.
3. Electronic Bulletin Board at FedWorld: 800–303–9672.

Dated at Rockville, Maryland, this 23rd day of July, 1996.

For the U.S. Nuclear Regulatory Commission.

Richard L. Bangart,

Director, Office of State Programs.

[FR Doc. 96–19587 Filed 7–31–96; 8:45 am]

BILLING CODE 7590–01–P

Proposed Generic Communication; Primary Water Stress Corrosion Cracking of Control Rod Drive Mechanism and Other Vessel Head Penetrations

AGENCY: Nuclear Regulatory Commission.

ACTION: Notice of opportunity for public comment.

SUMMARY: The Nuclear Regulatory Commission (NRC) is proposing to issue a generic letter concerning primary water stress corrosion cracking in control rod drive mechanisms and other vessel head penetrations of nuclear power reactors. The purpose of the proposed generic letter is to (1) request that addressees describe their program for ensuring the timely inspection of PWR control rod drive mechanism (CRDM) and other vessel head penetrations and (2) require that all addressees provide to the NRC a written response to this generic letter. The NRC is seeking comment from interested parties regarding both the technical and regulatory aspects of the proposed generic letter presented under the Supplementary Information heading.

The proposed generic letter was endorsed by the Committee to Review Generic Requirements (CRGR) on July 25, 1996. The relevant information that was sent to the CRGR will be placed in the NRC Public Document Room. The NRC will consider comments received from interested parties in the final evaluation of the proposed generic letter. The NRC's final evaluation will include a review of the technical position and, as appropriate, an analysis of the value/impact on licensees. Should this generic letter be issued by the NRC, it will become available for public inspection in the NRC Public Document Room.

DATES: Comment period expires September 3, 1996. Comments submitted after this date will be considered if it is practical to do so, but assurance of consideration cannot be given except for comments received on or before this date.

ADDRESSES: Submit written comments to Chief, Rules Review and Directives Branch, U.S. Nuclear Regulatory Commission, Mail Stop T–6D–69, Washington, DC 20555–0001. Written comments may also be delivered to 11545 Rockville Pike, Rockville, Maryland, from 7:30 am to 4:15 pm, Federal workdays. Copies of written comments received may be examined at the NRC Public Document Room, 2120 L Street, N.W. (Lower Level), Washington, D.C.

FOR FURTHER INFORMATION CONTACT: C. E. (Gene) Carpenter (301) 415–2169.

SUPPLEMENTARY INFORMATION:

Generic Letter 96–##: Primary Water Stress Corrosion Cracking of Control Rod Drive Mechanism and Other Vessel Head Penetrations (TACS No. M95280)

Addressees

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

Purpose

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

Background

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

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

The NRC staff identified primary water stress corrosion cracking (PWSCC) as an emerging technical issue to the Commission in 1989, after cracking was noted in Alloy 600 pressurizer heater sleeve penetrations at a domestic PWR facility. Other leaks have occurred since 1986 in several Alloy 600 pressurizer instrument nozzles at both domestic and foreign reactors from several different nuclear steam supply system vendors. The NRC staff reviewed the safety significance of the cracking that occurred, as well as the repair and replacement activities at the affected facilities. The NRC staff determined that the cracking was not of immediate safety significance because the cracks were axial, had a low growth rate, were in a material with an extremely high flaw tolerance (high fracture toughness) and, accordingly, were unlikely to propagate very far. These factors also demonstrated that any cracking would result in detectable leakage and the opportunity to take corrective action before a penetration would fail. The NRC staff issued Information Notice 90–10, ''Primary Water Stress Corrosion Cracking (PWSCC) of Inconel 600,'' dated February 23, 1990, to inform the nuclear industry of the issue.

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

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

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

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

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

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

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

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

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

The NRC staff was informed during a meeting on August 24, 1995, that Westinghouse had developed a susceptibility model for VHPs based on a number of factors, including operating temperature, years of power operation, method of fabrication of the VHP, microstructure of the VHP, and the location of the VHP on the head. Each time a plant's VHPs are inspected, the inspection results are incorporated into the model. All domestic Westinghouse PWRs have been modeled and the ranking has been given to each licensee.

In addition, the NRC staff was informed that Framatome Technologies, Inc. [FTI, formerly Babcock & Wilcox (B&W)], also developed a susceptibility model for CRDM penetration nozzles and other VHPs in B&W reactor vessel designs. All domestic B&W PWRs have been modeled and the ranking has been given to each B&W licensee. The NRC staff was further informed that Combustion Engineering (CE) had performed an initial susceptibility assessment for the CE PWRs. At present, neither Westinghouse, FTI, nor CE has submitted its models and assessments to the NRC staff for review.

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

Discussion

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

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

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

Required Information

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

Addressees are required to provide the following information:

1. Regarding inspection activities:

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

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

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

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

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

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

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

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

3.2 What were the durations of these intrusions?

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

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

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

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

Required Response

All addressees shall submit in writing the information identified above within 90 days from the date of this letter.

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

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

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

Related Generic Communications

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

(2) 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

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

Dated at Rockville, Maryland, this 26th day of July, 1996.

For the Nuclear Regulatory Commission.

Elinor G. Adensam,

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

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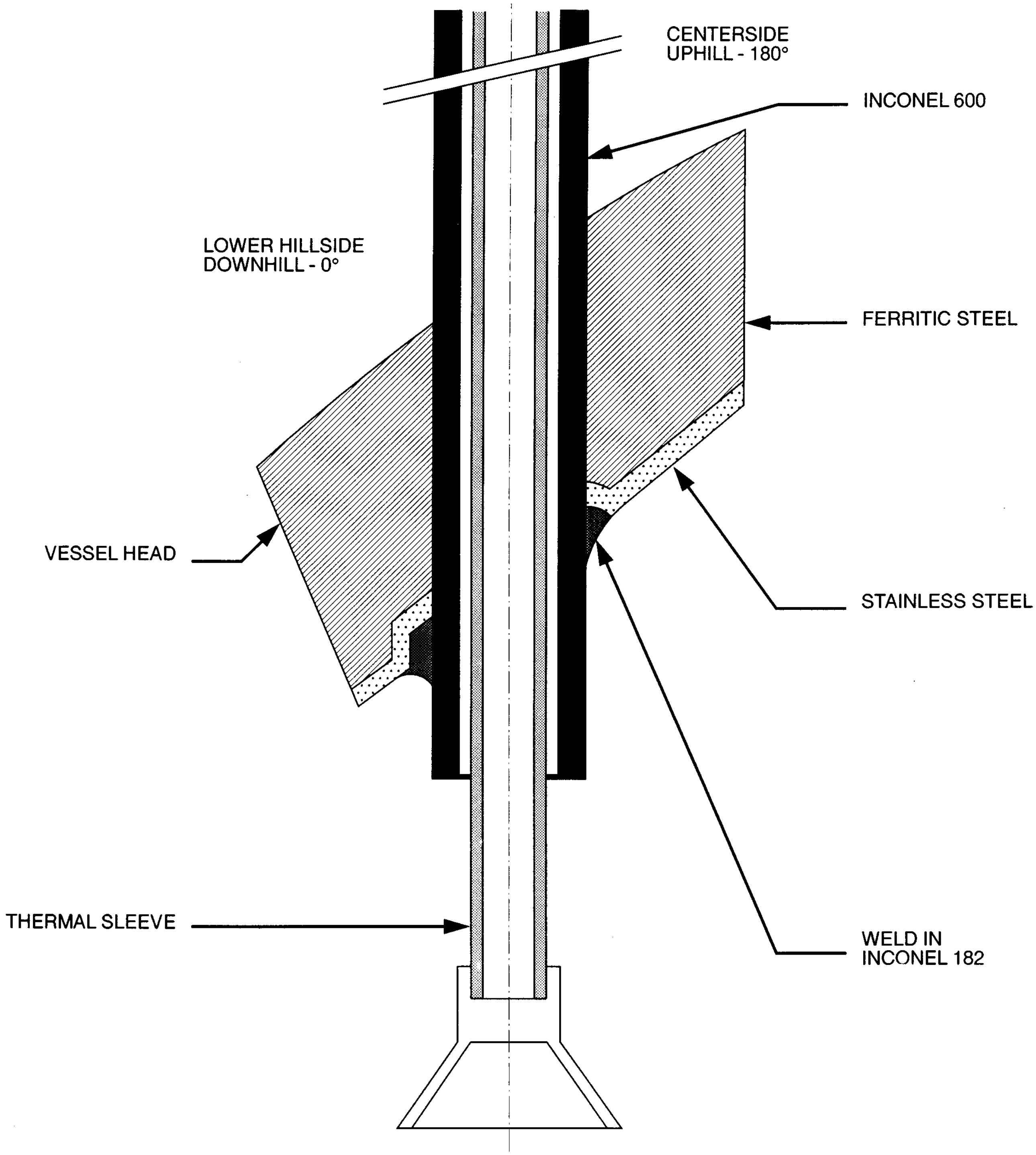

Figure 1. Head Penetration and Vessel Assembly

[FR Doc. 96–19588 Filed 7–31–96; 8:45 am]
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