December 3, 1997

FOR: The Commissioners

FROM: L. Joseph Callan /s/
Executive Director for Operations

SUBJECT: PROPOSED NRC GENERIC LETTER, "STEAM GENERATOR TUBE INSPECTION TECHNIQUES"

PURPOSE:

To inform the Commission, in accordance with the guidance in a memorandum dated December 20, 1991, 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 attached generic letter. The purpose of the generic letter is to (1) emphasize to licensees the importance of performing steam generator tube inservice inspections using qualified techniques in accordance with the requirements of Appendix B to 10 CFR Part 50 and (2) require certain information from licensees to determine whether they are in compliance with the current licensing basis for their respective facilities, given their steam generator tube inservice inspection practices.

DISCUSSION:

The structural and leakage integrity of steam generator tubing is maintained through several defense-in-depth measures, including inservice inspection, tube repair criteria, primary-to-secondary leak rate monitoring, water chemistry, operator training, and analyses to ensure that safety objectives are met. The degraded tubes must be removed or repaired if detected indications (flaws) exceed 40 percent of the nominal tube wall thickness as required in plant technical specifications. The indications are detected by periodic inspections using qualified nondestructive testing as required by Criterion IX in Appendix B to 10 CFR Part 50. Eddy current technology, one method of nondestructive testing, is the primary means used by the industry to assess the condition of steam generator tubing.

The eddy current inspection technique correlates the depth and length of an indication to signal responses received by probes passing through the inside of the tube. Although the eddy current method is a proven technique for detecting the length of indications, there has been limited success in demonstrating its capability to accurately measure the depth of certain typesof steam generator tube indications. Specifically, indications caused by intergranular attack (IGA) and stress-corrosion cracking are difficult to size with eddy current techniques because of a number of complicating variables, such as oxide deposits, material properties and geometry, crack morphology, human factors, data analysis, and data acquisition practices. In one recent instance, a licensee sized the depths of IGA indications and removed from service those tubes with IGA indications exceeding the 40 percent through-wall repair limit. Data from subsequent destructive examinations of several degraded tube specimens removed from the licensee's steam generators during the outage indicated that the estimated through-wall extent of degradation in these specimens, based on eddy current, was significantly less than the true depth of the IGA indications.

To verify compliance with Appendix B to 10 CFR Part 50 and the plant technical specifications and to maintain a reasonable level of assurance that structural and leakage integrity margins for steam generator tubes are satisfied, the generic letter requests licensees to submit a written response stating whether they leave steam generator tubes with indications in service based on sizing the depth of confirmed indications. If a depth sizing is used, licensees should submit a description of the associated eddy current method being used and the technical basis for the acceptability of the technique used.

The NRC staff is not establishing a new position in this generic letter. The generic letter only requests information from licensees under the provisions of Section 182a of the Atomic Energy Act of 1954, as amended, and 10 CFR 50.54(f).

A notice of opportunity for public comment was published in the Federal Register (61 FR 69118) on December 31, 1996. Comments were received from one industry organization and one licensee. The comments on the proposed generic letter focused on (1) the need for the generic letter, (2) the length of the period in which licensees could respond to the generic letter, and (3) editorial comments. The staff has evaluated these concerns and made appropriate changes to the generic letter. Copies of the comment letters that were received are available in the Public Document Room (PDR). A copy of the staff's evaluation of the comments is available in the NRC Central Files and will be made available in the PDR after the generic letter is issued.

The Committee To Review Generic Requirements (CRGR) reviewed the proposed generic letter during Meeting Number 296 on November 19, 1996, and formally endorsed the final proposed generic letter in Meeting Number 309 on August 5, 1997. The staff incorporated comments made by the CRGR in those meetings.

The Office of the General Counsel reviewed this generic letter and has no legal objection to its contents.

The staff intends to issue this generic letter 5 working days after the date of this information paper.

L. Joseph Callan
Executive Director for Operations

CONTACT: Phillip Rush, NRR
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Attachment: Proposed NRC Generic Letter, "Steam Generator Tube Inspection Techniques"

ATTACHMENT

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

NRC GENERIC LETTER 97-XX: STEAM GENERATOR TUBE INSPECTION TECHNIQUES

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) emphasize to the addressees the importance of performing steam generator tube inservice inspections using qualified techniques in accordance with the requirements of Appendix B to 10 CFR Part 50, and (2) require certain information from addressees to determine whether they are in compliance with the current licensing basis for their respective facilities given their steam generator tube inservice inspection practices.

Background

Steam generator tubing constitutes a significant portion of the reactor coolant pressure boundary (RCPB). The design of the RCPB for structural and leakage integrity is addressed in either Title 10 of the Code of Federal Regulations, Part 50 (10 CFR Part 50), Appendix A or the licensing basis of a facility. The General Design Criteria (GDC) of Appendix A state that the RCPB shall "have an extremely low probability of abnormal leakage" (GDC 14), "shall be designed with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation" (GDC 15), and "shall be designed to permit periodic inspection and testing of important areas and features to assess their structural and leaktight integrity" (GDC 32).

Once a plant is in operation, licensees are required by their technical specifications to perform periodic inservice inspections of the steam generator tubing and to repair or remove from service all tubes with degradation exceeding the tube repair limits. Eddy-current inspection techniques are the primary means by which licensees assess the condition of the steam generator tubes. Such inspections are an important component of the defense-in-depth measures to ensure the structural and leaktight integrity of the steam generator tubes.

The NRC issued Generic Letter (GL) 95-03, "Circumferential Cracking of Steam Generator Tubes," on April 28, 1995. One of the purposes of GL 95-03 was to emphasize the importance of utilizing qualified inspection techniques and equipment capable of reliably detecting steam generator tube degradation.

Criterion IX, "Control of Special Processes," contained in Appendix B to 10 CFR Part 50 states, in part, that "measures shall be established to assure that special processes, including ... nondestructive testing, are controlled and accomplished by qualified personnel using qualified procedures." Although the main focus of GL 95-03 was to address circumferential steam generator tube cracking, the requirement of using qualified inspection techniques applies to all inspections for all forms of tube degradation.

Criterion XI, "Test Control," requires, in part, that a test program be established to assure that all testing required to demonstrate that structures, systems, and components will perform satisfactorily in service is identified and performed in accordance with written test procedures which incorporate the requirements and acceptance limits contained in applicable design documents.

Licensees have traditionally relied upon eddy-current inspection techniques to assess the condition of their steam generator tubes. Although the eddy-current method is a proven technique for detecting tube degradation, the ability to depth size indications is possible only for specific modes of degradation. Specifically, tube degradation from intergranular attack (IGA) and stress corrosion cracking (SCC), major modes of steam generator tube degradation, are difficult to size with eddy current inspection techniques because of a number of complicating variables. In one recent instance, a licensee employed a technique to size the depths of IGA tube degradation based on tube specimens removed from two plants. However, pulled tube data analyzed after the initial application of the technique indicated that the method did not adequately estimate the true depth of the indications consistent with the criteria established for qualifying the sizing technique.

Discussion

(1) Evaluation of Recent Inspection Experience

In general, plant technical specifications require the removal from service or the repair of those steam generator tubes with degradation exceeding 40 percent of the nominal tube-wall thickness. Criterion IX in Appendix B to 10 CFR Part 50 requires that nondestructive testing be completed using qualified procedures. Therefore, licensees must be able to demonstrate through the qualification process that an inspection technique used for sizing steam generator tube indications can measure the through-wall penetration of cracks and other forms of degradation with an accuracy commensurate with the "bases" of the tube repair limits in the technical specifications.

Theoretically, there is a relationship between the depth of penetration of a defect and the eddy-current signal response; in practice, however, the relationship between signal voltage or phase angle and the degradation depth is influenced by many other variables. Oxide deposits, variability of tube material properties and geometry, degradation morphology, human factors, and eddy-current data analysis and acquisition practices are some of the factors that can significantly alter a depth estimation of steam generator tube degradation. The NRC is aware that the depth of several specific forms of volumetric steam generator tube degradation can be sized with a reasonable degree of accuracy; however, qualifying techniques for sizing of some forms of degradation, e.g., IGA and SCC, has been problematic.

In order to successfully disposition steam generator tube degradation in accordance with the repair limits in the technical specifications and Appendix B to 10 CFR Part 50, the inspection process must be capable of (1) detecting indications of tube degradation, (2) characterizing the indications, e.g., cracklike, IGA, manufacturing burnish mark, or wear and the orientation for cracklike degradation, and (3) accurately sizing the depth of degradation. The term "inspection process" refers to the use of one or a combination of nondestructive inspection techniques to evaluate a specific mode of steam generator tube degradation. This evaluation could potentially include three inspection methods (e.g., eddy current probes)--one for detection, one for characterization, and a third to size the indication. However, the successful qualification of the inspection process requires a qualification of each method (i.e., probes, cables, software, etc.) for the mode of degradation being evaluated in the steam generator tube examinations. Experience has demonstrated that for effective qualification the data set demonstrating the capability of the inspection process should consist, to the extent practical, of service-degraded tube specimens (i.e., specimens removed from operating steam generators), supplemented, as necessary, by tube specimens containing flaws fabricated using alternative methods provided that the nondestructive examination parameter responses from these flaws are fully consistent with actual inservice degradation of the same flaw geometry.

(2) Safety Assessment

Steam generator tube degradation is managed through a combination of several defense-in-depth measures including inservice inspection, tube repair criteria, primary-to-secondary leak rate monitoring, water chemistry, operator training, and analyses to ensure safety objectives are met. In addition, on the basis of NRC conclusions regarding the potential consequences of steam generator tube failure events in NUREG-0844, "NRC Integrated Program for the Resolution of Unresolved Safety Issues A-3, A-4, and A-5 Regarding Steam Generator Tube Integrity," the risk from the potential rupture of one or more tubes is small. However, since tube ruptures represent a failure of one of the principal fission product boundaries and present a pathway for a release to the environment bypassing the containment, all reasonable precautions should be taken to prevent such an occurrence.

To verify compliance with Appendix B to 10 CFR Part 50 and the technical specifications, and to maintain a reasonable level of assurance that structural and leakage integrity margins for steam generator tubes are satisfied, the NRC has concluded that it is appropriate for the addressees to review the types of steam generator tube indications that are being left in service based on sizing, the inspection method being used to perform the sizing for each type of indication, and the technical basis for the acceptability of each inspection method.

Required Information

Within 90 days of the date of this generic letter, addressees are required to submit a written response that includes the following information:

(1) whether it is their practice to leave steam generator tubes with indications in service based on sizing,

(2) if the response to item (1) is affirmative, those licensees should submit a written report that includes, for each type of indication, a description of the associated nondestructive examination method being used and the technical basis for the acceptability of the technique used.

Address the required written information to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, D.C. 20555-0001, under oath or affirmation under the provisions of Section 182a, Atomic Energy Act of 1954, as amended, and 10 CFR 50.54(f). This information will enable the Commission to determine whether a license should be modified, suspended or revoked. In addition, submit a copy of the written information to the appropriate regional administrator.

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

Backfit Discussion

This generic letter only requests 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). This generic letter does not constitute a backfit as defined in 10 CFR 50.109(a)(1) since it does not impose modifications of or additions to structures, systems or components or to design or operation of an addressee's facility. It also does not impose an interpretation of the Commission's rules that is either new or different from a previous staff position. The staff, therefore, has not performed a backfit analysis.

Reason for Information Request

This generic letter transmits an information request pursuant to the provisions of Section 182a of the Atomic Energy Act of 1954, as amended, and 10 CFR 50.54(f) for the purpose of verifying compliance with applicable regulatory requirements. Specifically, the requested information will enable the NRC staff to determine whether addressees, given their steam generator tube inspection practices, are in compliance with current licensing basis for their respective facilities. In particular, this information will help to ascertain whether or not the regulatory requirements pursuant to Appendix B to 10 CFR

Part 50, namely, Criterion IX, "Control of Special Processes," and Criterion XI, "Test Control," are met.

Federal Register Notification

A notice of opportunity for public comment was published in the *Federal Register* (61 FR 69118) on December 31, 1996. Comments were received from one industry organization and one licensee. 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 September 30, 2000.

The public reporting burden for this collection of information is estimated to average 75 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, or by Internet electronic mail at BJS1@NRC.GOV; 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 the technical contact listed below or the appropriate Office of Nuclear Reactor Regulation (NRR) project manager.

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

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Attachment: List of Recently Issued NRC Generic Letters