generator internals. The NRC staff will continue to monitor information on tube support plate and tube bundle wrapper damage as it becomes available from foreign authorities.

This letter also alerts addressees to the importance of performing comprehensive examinations of steam generator internals to ensure steam generator tube structural integrity is maintained in accordance with the requirements of Appendix B to 10 CFR Part 50. Criterion XI of Appendix B, ''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 the applicable design documents. The applicable steam generator tube design documents include General Design Criteria (GDCs) 14, 15, 30, 31, and 32 of 10 CFR Part 50, Appendix A and Section III of the ASME Boiler and Pressure Vessel code. Criterion XVI of Appendix B, ''Corrective Action,'' requires in part that measures be established to assure that conditions adverse to quality are promptly identified and corrected.

Requested Information

Within 60 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) Discussion of the program in place, if any, to detect degradation of steam generator internals and a description of the inspection plans, including the inspection scope, frequency, methods, equipment and criteria, and plans for corrective action in the event degradation is found.

The discussion should include the following information:

(a) Whether past inspection records at the facility have been reviewed for indications of tube support plate signal anomalies from eddy current testing of the steam generator tubes that may be indicative of support plate damage or ligament cracking. If the addressee has performed such a review, include a discussion of the findings.

(b) Whether visual or video camera inspections on the secondary side of the steam generators have been performed at the facility to provide information on the condition of steam generator internals (e.g., support plates, tube bundle wrappers, or other components). If the addressee has performed such inspections, include a discussion of the findings.

(c) Whether degradation of steam generator internals has been detected at the facility, and how the degradation was assessed and dispositioned.

(2) If the addressee currently has no program in place to detect degradation of steam generator internals, the written response should include a discussion of the plans for establishing such a program, or a justification as to why no such program is needed.

Addressees are encouraged to work closely with industry groups on the coordination of inspections, evaluations, and repair options for all types of steam generator degradation that may be found.

The NRC is aware that the industry has developed generic industry guidance on performing steam generator inspections, and that this guidance is continually being updated. If an addressee intends to follow the guidance developed by the industry for this issue, reference to the relevant generic guidance documents is acceptable, and encouraged, as part of the response, as long as the referenced documents have been officially submitted to the NRC. However, additional plant-specific information will be needed.

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 responses 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).

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 will enable the NRC staff to determine whether or not the condition of the addressees' steam generator internals comply and conform with the current licensing basis for their respective facilities. In particular, it would help ascertain whether or not the regulatory requirements pursuant to Appendix B to 10 CFR Part 50 are met, namely, (1) Criterion XI, ''Test Control,'' concerning the establishment of effective test programs for systems, structures and components, and (2) Criterion XVI, ''Corrective Action,'' which requires that measures shall be established to assure that conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and nonconformances are promptly identified and corrected. 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.

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

For the Nuclear Regulatory Commission.

David B. Matthews,

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

[FR Doc. 96–33250 Filed 12–30–96; 8:45 am]

BILLING CODE 7590–01–P

Proposed Generic Communication; Steam Generator Tube Inspection Techniques (M96401)

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 steam generator tube inspection practices at pressurized-water reactor facilities. The purpose of the proposed generic letter is to (1) emphasize to 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) request certain information from addressees to verify whether or not steam generator tube inservice inspection practices comply and conform with the current licensing basis for their respective facilities. 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 December 17, 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 January 30, 1997. 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: Phillip J. Rush, (301) 415–2790.

SUPPLEMENTARY INFORMATION:

NRC Generic Letter 96–XX: Steam Generator Tube Inspection Techniques

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) 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) request certain information from addressees to verify whether or not steam generator tube inservice inspection practices comply and conform with the current licensing basis for their respective facilities.

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 a requirement under Title 10 of the Code of Federal Regulations, Part 50 (10 CFR Part 50), Appendix A. Specifically, 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 eddy-current methods are a proven technique for detecting tube degradation, there has been only limited success in demonstrating the capability to accurately depth size indications from the eddy-current signals. 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. Through recent inspections and discussions of eddy-current practice with various licensees, the NRC has become aware that several utilities are allowing degraded steam generator tubes to remain in service on the basis of estimates of IGA and SCC degradation depths using eddy-current methods.

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 mode of degradation, e.g., cracklike, IGA, corrosion induced thinning, or wear and the orientation for cracklike degradation, and (3) accurately sizing the depth of the indication. 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., probe) 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 the structural and leakage integrity margins for steam generator tubes provided in the General Design Criteria (Appendix A to 10 CFR Part 50) are satisfied, the NRC has concluded that it is appropriate for the addressees to review the types of steam generator tube degradation that are being left in service based on sizing, the inspection method being used to perform the sizing for each type of degradation, and the technical basis for the acceptability of each inspection method.

Requested Information

Within 60 days of the date of this generic letter, all addressees are requested to provide the following information: (1) Whether it is their practice to leave steam generator tubes with defects in service, based on sizing, and (2) if the response to item (1) is affirmative, those licensees are requested to submit a written report that includes, for each type of steam generator degradation mechanism, a description of the associated nondestructive examination method being used and the technical basis for the acceptability of the technique used.

Required Response

Within 30 days of the date of this generic letter, addressees are required to submit a written response indicating: (a) Whether or not the requested information will be submitted, and (b) whether or not the requested information will be submitted within the requested time period. Addressees who respond in the affirmative to item (1) under Requested Information and 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 written material 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).

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). The information requested will enable the NRC staff to determine whether addressees' steam generator tube inspection practices comply and conform with the current licensing basis for their respective facilities. In particular, it would help 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. 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.

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

For the Nuclear Regulatory Commission.

David B. Matthews,

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

[FR Doc. 96–33248 Filed 12–30–96; 8:45 am]

BILLING CODE 7590–01–P

Proposed Generic Communication; Effectiveness of Ultrasonic Testing Systems in Inservice Inspection Programs

AGENCY: Nuclear Regulatory Commission.

ACTION: Notice of opportunity for public comment.

SUMMARY: The Nuclear Regulatory Commission (NRC) is proposing to issue a generic letter to determine if addressees are taking appropriate action to qualify future ultrasonic testing (UT) examinations. The purpose of the proposed generic letter is to (1) alert addressees to the importance of using equipment, procedures, and examiners (UT systems) capable of reliably detecting and sizing flaws in the performance of comprehensive examinations of reactor vessels and piping, (2) notify addressees about enhancements in UT systems and the significance of these enhancements in plant-specific inservice inspection (ISI) programs, (3) request that all addressees describe the extent to which their piping and reactor pressure vessel ISI activities are being qualified consistent with the objectives of Appendix VIII to Section XI of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), and (4) require that all addressees send to the NRC a written response to this generic letter relating to the actions and information requested in this letter. The