Jerrold G. Dewease
Vice President Operations Support
Entergy Operations, Inc.
P. O. Box 31995
Jackson, MS 39266-1995

January 9, 1997

SUBJECT: EVALUATION OF ENTERGY OPERATION, INC., REQUEST FOR AUTHORIZATION TO UPDATE INSERVICE INSPECTION PROGRAMS TO THE 1992 AND PORTIONS OF THE 1993 ASME BOILER AND PRESSURE VESSEL CODE, SECTION XI FOR ARKANSAS NUCLEAR ONE, UNITS 1 AND 2, GRAND GULF NUCLEAR STATION, RIVER BEND STATION, AND WATERFORD STEAM ELECTRIC STATION, UNIT 3 (TAC NOS. M94472, M94471, M94454, M94473, AND M94488)

Dear Mr. Dewease:

On December 12, 1996, we issued the approval of the inservice inspection (ISI) program plans for the Entergy sites in the above subject. In our approval, we deferred our review of your proposal to exclude Appendix VIII on ultrasonic examinations of the 1992 Edition of ASME Section XI and to instead, follow the requirements in Appendix I of the 1989 Code Edition. On December 31, 1996, the Nuclear Regulatory Commission published in the Federal Register (61 FR 69117) a proposed generic communication on the effectiveness of ultrasonic testing systems in ISI programs. This notice is for opportunity for public comment. The notice and proposed generic communication are enclosed for your information.

Entergy Operations, Inc. will be expected to respond to the generic communication when it is issued in final and to revise your ISI programs accordingly. Our review will conclude with the evaluation of your response. If there are any questions on this matter, please let us know.

Sincerely,

ORIGINAL SIGNED BY:
William D. Beckner, Director
Project Directorate IV-1
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Docket Nos. 50-313, 50-368, 50-416
50-458, and 50-382

Enclosure: Notice and Proposed Generic Communication

cc w/encl: See next page

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

January 9, 1997

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Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Docket Nos. 50-313, 50-368, 50-416
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Enclosure: Notice and Proposed Generic Communication

cc w/encl: See next page

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Vol. 61 No. 252 Tuesday, December 31, 1996 p 69117 (Notice)
Nuclear Regulatory Commission
NOTICES
Generic letters:
Ultrasonic testing systems in inservice inspection programs; effectiveness, 69117 12/31/96

NUCLEAR REGULATORY COMMISSION

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 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 19, 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, DC.

FOR FURTHER INFORMATION CONTACT: Donald G. Naujock (301) 415-2767.

SUPPLEMENTARY INFORMATION:

Generic Letter 96-XX: Effectiveness of Ultrasonic Testing Systems In Inservice Inspection Programs

Addressees

All holders of operating licenses or construction permits for nuclear power reactors, 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) Alert addressees to the importance of using equipment, procedures, and examiners capable of reliably detecting and sizing flaws in the performance of comprehensive examinations of reactor vessels and piping, (2) notify addressees about enhancements in ultrasonic testing (UT) systems (Note: As used in this document, ''UT systems'' refers to the equipment, procedures, or examiners involved in the ultrasonic examination) 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 (Note: ''Consistent with the objectives of Appendix VIII'' means in close conformance with Appendix VIII criteria, even though the Appendix has not been formally incorporated into the regulations as a requirement.) 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.

Background

In the 1970s, operating experience and industry tests indicated a need for improving UT procedures to consistently and reliably detect and characterize flaws during ISI of reactor vessel welds. Also noted was the need for more definitive reporting of results and for more descriptive requirements for essential variables associated with ultrasonic examinations. That need was satisfied with the issuance of Regulatory Guide (RG) 1.150, Revision 1, ''Ultrasonic Testing of Reactor Vessel Welds During Preservice and Inservice Examinations,'' in February 1983. RG 1.150 was incorporated into the technical specifications of many plants.

As the nuclear industry gained more operating experience, the need for improving ISI capabilities became apparent. For example, in the late 1970s, thermal fatigue cracks were found on the inner-blend radius of nozzle-to-vessel surfaces in boiling-water reactor (BWR) feedwater and control rod drive return line nozzles. The NRC staff recommended, in NUREG-0619, ''BWR Feedwater Nozzle and Control Rod Drive Return Line Nozzle Cracking,'' dated November 1980, that licensees develop ISI programs to search for cracks in the inner-blend radii using dye-penetrant, visual, and ultrasonic examinations. The NRC staff recognized the potential for improvements to UT systems, and stated in NUREG-0619 that demonstrated improvements could be used as the basis for modifying the inspection criteria.

Also in the late 1970s, intergranular stress corrosion cracking (IGSCC) was identified in austenitic stainless steel piping. The NRC staff recommended in NUREG-0313, ''Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping,'' dated July 1977, and in subsequent revisions published in July 1980 and January 1988, that a program be established to conduct formal IGSCC performance demonstration testing for UT examiners.

The regulatory guide and NUREG reports were issued as guidance in detecting flaws and in preventing the conditions that could lead to unacceptable flaws.

The need for additional guidance related to performing UT in ISI programs, that were based on requirements in Section XI of the ASME Code, prompted a reexamination of the effectiveness of UT as it was being applied through the ASME Code. The conventional (amplitude-based) UT requirements in the ASME Code establish minimum acceptable inspection standards. In the 1970s and 1980s, the nuclear industry tested UT systems extensively to identify the critical aspects of an effective UT inspection program that would provide a high reliability for detection and characterization of flaws. In the mid-1980s, the NRC and the nuclear industry recognized that the reliability of UT in ISI programs could be significantly improved through performance-demonstration qualification of nondestructive examination equipment, procedures, and examiners.

In 1984, the NRC entered into an agreement, known as the IGSCC Coordination Plan, with the Boiling Water Reactor Owners' Group (BWROG) and the Electric Power Research Institute (EPRI) to coordinate selected activities in regard to training and qualification of personnel using UT to examine piping weldments. As part of the IGSCC Coordination Plan, EPRI administered IGSCC performance demonstration tests to personnel seeking UT qualifications in IGSCC detection and characterization in piping systems.

The nuclear industry set about changing ASME Code requirements for UT from the current minimum inspection standards to inspection standards with performance-based qualifications. The performance-based qualifications would also produce uniform acceptance criteria for evaluating new technology and addressing new forms of degradation. The efforts of the industry to develop performance-based qualification criteria culminated with the publication of Appendix VIII to Section XI of the ASME Code, which was published in the 1989 Addenda. Appendix VIII, ''Performance Demonstration for Ultrasonic Examination Systems,'' contains detailed requirements for UT performance demonstrations that include statistically based acceptance criteria to detect and size flaws.

The NRC has initiated rulemaking to amend 10 CFR 50.55a to reference Section XI of the ASME Code up to and including the 1995 Edition. After completion of rulemaking, Appendix VIII to Section XI will become a requirement for all licensees. The final rule incorporating Appendix VIII is expected to be issued around July 1998.

Description of Circumstances

Appendix VIII is based on the qualification of equipment, procedures, and examiners using performance demonstrations; whereas, existing requirements in the 1989 (and earlier) Edition of Section XI of the ASME Code are prescriptive, minimum inspection standards. A performance-based qualification program encourages development of improved methods for detecting and characterizing flaws, and facilitates implementing the methods with a defined testing curriculum. The performance demonstrations require that equipment, procedures, and examiners be tested on flawed and notched materials and configurations similar to those found in actual conditions. The nuclear industry created the Performance Demonstration Initiative (PDI) in 1991 to manage implementation of the performance demonstration criteria of Appendix VIII (Note: The PDI activities have been assessed by the NRC staff, as described in the letter from J. Strosnider (NRC) to B. Sheffel (PDI) dated March 6, 1996, and have been found to provide a significantly improved method for qualification of equipment, procedures, and examiners. Overall, the NRC staff found that PDI has established and is in the process of executing a well-planned and effective program to test UT technicians on selected portions of Appendix VIII. Accordingly, the NRC staff finds that UT procedures qualified under the PDI program using performance demonstration methods provide an acceptable level of quality and safety.)

Because performance demonstrations test the ability of equipment, procedures, and examiners to detect and size flaws, the demonstrations raise the performance threshold for examiners conducting ultrasonic inspections. For example, a sampling of individuals tested in the different piping examinations under the PDI program revealed that 22% of them did not satisfy the screening criteria for detection of flaws; 41% did not satisfy the screening criteria for length-sizing; 67% did not satisfy the screening criteria for depth measurement; and 49% did not satisfy the screening criteria for IGSCC. These percentages are based on a sampling that included retests. The PDI tests ensure that the equipment must have adequate sensitivity, the procedures must have sufficient detail, and the individuals must be sufficiently skilled in order to successfully qualify under the PDI program.

The improvements in UT techniques performed using Appendix VIII criteria became apparent in 1993 during the reactor pressure vessel shell weld augmented examination at the Browns Ferry Nuclear Power Plant, Unit 3, and in 1995 during the inspection of piping systems for IGSCC at the Millstone Nuclear Power Station, Unit 1. At Browns Ferry, the equipment, procedures, and examiners were qualified consistent with the objectives of Appendix VIII. The examination revealed 15 flaws that did not meet the ASME Code, Section XI, Subarticle IWB-3500 acceptance criteria and that required further evaluation. Of the 15 flaws, only 3 would have been recordable using conventional Section XI minimum inspection standards and RG 1.150 criteria, and only 2 of the 3 flaws would have required an analytical evaluation in accordance with Section XI, Subarticle IWB-3600.

This experience indicates that flaws large enough to require analytical evaluation might not be detected using current UT standards.

Millstone Unit 1 inspectors performed an augmented UT examination for IGSCC in the welds in reactor system piping. The licensee used a newly developed ultrasonic transducer technology to supplement the original examinations. Before the examination, UT examiners from Millstone who were qualified under the IGSCC Coordination Plan demonstrated the adequacy of the new transducer technology by successfully passing the Appendix VIII performance demonstration test administered through the PDI program. During the augmented examination, the UT inspection personnel examined 264 of the 411 pipe welds and found that 35 welds had cracks. A review of examination records from 1984 through 1994 revealed 211 indications that were previously considered by Level III inspectors to be nonmetallurgical or geometric indications. During the 1995 inspection, 14 of the indications previously identified as nonmetallurgical or geometric were identified as flaws; 3 of these flaws developed through-wall leaks when they were mechanically buffed in preparation for repair by the NRC-approved overlay process. The Appendix VIII qualification by Millstone inspectors using normal IGSCC UT procedures increased the licensee's reliability in detection of IGSCC. The additionally demonstrated capability of the new transducer technology under the PDI-administered program clearly increased the level of confidence in the new transducer technology used to identify previous errors made in flaw disposition.

Although, the above experiences clearly depict the need for improvement by using performance demonstration methods in performing UT examinations of reactor vessels and piping, it should be noted that a safety concern does not exist which would warrant immediate backfitting of Appendix VIII in advance of the rulemaking that has been initiated. The staff has reached this conclusion based on consideration of defense-in-depth measures, Code margins in component design, and leakage monitoring systems. In addition, the staff has been requiring for some time now that selected inspections be performed using performance-based qualified techniques (e.g., IGSCC piping inspections).

Regulatory Requirements

10 CFR 50.55a requires that systems and components of boiling-water and pressurized-water reactors conform to the requirements of the ASME Code, Sections III and XI.

Appendix A to 10 CFR Part 50 Criterion 14 requires that the reactor coolant pressure boundary shall be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture.

Criterion XVI of Appendix B to 10 CFR Part 50 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. In the case of significant conditions adverse to quality, the measures shall assure that the cause of the condition is determined and corrective action taken to preclude repetition. The identification of the significant condition adverse to quality, the cause of the condition, and the corrective action taken shall be documented and reported to appropriate levels of management.

Criterion II of Appendix B to 10 CFR Part 50 requires, in part, that a quality assurance program shall take into account the need for special controls, processes, test equipment, tools, and skills to attain the required quality and the need for verification of quality by inspection and test. It also requires that the program provide for indoctrination and training of personnel performing activities affecting quality, as necessary to assure that suitable proficiency is achieved and maintained.

Discussion

The qualification statistics from PDI discussed above and the issuance of the regulatory guide and staff reports highlight the fact that some UT systems satisfying ASME Code, Section XI amplitude-based UT requirements are less effective in identifying and characterizing certain types of flaws. The experiences at Browns Ferry Unit 3 and Millstone Unit 1 highlight the significant improvements in the effectiveness of UT systems when equipment, procedures, and examiners are qualified through a performance-demonstration program. Therefore, a significant improvement is gained in the effectiveness of UT systems qualified through performance demonstrations (e.g., Appendix VIII) over those satisfying conventional Section XI amplitude-based UT requirements.

The early and accurate detection of flaws in plants is important for maintaining the structural integrity and ensuring the safety function of safety-related systems and components. As plants age, improved reliability in inspection methods, more flexibility in utilizing advanced technology, and a better ability to detect new forms of degradation gain increased importance in ISI programs. The nuclear industry recognizes Appendix VIII as an improvement over the current ISI requirements, and the NRC staff finds that Appendix VIII criteria, as implemented by the PDI program, provide UT results that generally are superior to those of the 1989 (and earlier) Edition of Section XI of the ASME Code. The NRC staff finds that implementation of Appendix VIII criteria enhances the reliability of inspections and provides a significant improvement in the methods used to satisfy existing regulatory requirements and assure plant safety.

Some licensees have already submitted requests to utilize Appendix VIII performance demonstrations as an alternative examination for selective ASME Code, Section XI requirements. Licensees have also submitted requests to the staff to use Appendix VIII criteria in lieu of criteria in Regulatory Guide 1.150. Some licensees are using Appendix VIII concepts in developing alternatives to the IGSCC Coordination Plan, and the NRC staff has already approved the use of either the PDI program or the original IGSCC program for IGSCC qualification of examiners

(Note: Letter from W. T. Russell (NRC) to K. P. Donovan (Chairman, Boiling Water Reactor Owners' Group), ''Transition From the IGSCC Qualification Program to the Performance Demonstration Initiative Program,'' March 1, 1996.)

In conclusion, the NRC staff has determined that using only existing ISI requirements for performing UT examinations might not provide reasonable assurance that flaws can be reliably detected and sized in certain areas. The staff considers cracks and flaws in the reactor vessel and other safety-related components to be a concern when the possibility exists for flaws exceeding the ASME Code, Section XI allowable flaw sizes not being reliably detected or sized. Adequate safety exists through defense-in-depth measures, leakage monitoring systems, and Code margins in component design; however, significant improvement in the ability to reliably detect and size flaws in reactor vessels and piping can be achieved using performance demonstration methods. In order to assess whether the margins required by the ASME Code, Section XI are adequately maintained and to ensure compliance with the applicable existing requirements identified above, the NRC has concluded that it is appropriate to request certain actions and information from the addressees, as indicated below.

Requested Actions

In consideration of the information and concerns addressed above, each addressee is requested to perform an evaluation to determine whether its current ISI program ensures that flaws in the reactor vessel and safety-related piping are reliably detected and sized.

If it is determined that flaws in the reactor vessel and safety-related piping cannot be reliably detected and sized, each addressee is expected to take appropriate corrective action in future inspections, in accordance with the requirements of Criteria II and XVI of Appendix B to 10 CFR Part 50, to improve the capability to reliably detect and size flaws.

Requested Information

Within 90 days of the date of this generic letter, addressees are requested to submit a written summary report that includes the following:

1. A brief description of the addressee's evaluation of its ISI program, its determination regarding the capability of its current program to reliably detect and size flaws, and corrective actions taken (if any) in response to the requested actions above.

2. If the addressee is not using and does not plan to use the criteria in Appendix VIII of the ASME Code Section XI or other performance-based methods for the qualification of ISI activities, then provide a discussion of any plans for ensuring the effectiveness of current UT systems in detecting and sizing flaws in the reactor vessel and safety-related piping.

3. If the addressee is using or plans to use Appendix VIII for the qualification of ISI activities, then discuss the extent to which the equipment, procedures, and examiners in your ISI program for the reactor vessel and safety-related piping are (or will be) qualified using Appendix VIII criteria or other performance-based methods. Include in this discussion a description of any alternate examination methods (i.e., IWA-2240 of ASME Code Section XI) in your ISI program that use Appendix VIII or other performance-based examination methods as allowed in applicable sections of 10 CFR 50.55a for inspecting the

reactor vessel and safety-related piping.

Required Response

Within 30 days of the date of this generic letter, addressees are required to submit a written response indicating: (1) Whether or not the requested actions will be completed, (2) whether or not the requested information will be submitted, and (3) whether or not the requested information will be submitted within the requested time period.

Addressees who choose not to complete the requested actions, or 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 establishing the acceptability of the proposed alternative course of action. [For addressees that fail to have or implement appropriate qualification methods for future UT examinations where subsequent inspections find previously unidentified or improperly dispositioned flaws, the staff will consider whether such circumstances (a) are the result of failing to adequately take into account the need for special controls, skills and training needed to ensure suitable proficiency in the conduct of UT examinations contrary to the requirements of Criterion II, Quality Assurance Program, of Appendix B ''Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants,'' of 10 CFR Part 50; and/or (b) represent inadequate corrective action for known inadequacies contrary to the requirements of Criterion XVI, Corrective Action, of Appendix B, of 10 CFR Part 50.]

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). In addition, send a copy to the appropriate regional administrator.

Related Generic Communications

(1) Information Notice 96-32, ''Implementation of 10 CFR 50.55a(g)(6)(ii)(A), Augmented Examination of Reactor Vessel,'' June 5, 1996.

(2) Information Notice 93-20, ''Thermal Fatigue Cracking of Feedwater Piping to Steam Generators,'' March 24, 1993.

(3) Generic Letter 88-01, ''NRC Position on IGSCC in BWR Austenitic Stainless Steel Piping,'' January 25, 1988.

Backfit Discussion

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) to determine if licensees are taking appropriate action to qualify future UT examinations. To the extent that the actions requested in this letter may result in corrective actions taken by addressees that are considered backfits, the backfits are justified under the compliance exception of the backfit rule, i.e., 10 CFR 50.109 (a)(4)(i).

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. 96-33249 Filed 12-30-96; 8:45 am]
BILLING CODE 7590-01-P

The Contents entry for this article reads as follows: