



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

February 28, 1992

TO: ALL HOLDERS OF OPERATING LICENSES OR CONSTRUCTION PERMITS FOR NUCLEAR POWER PLANTS (EXCEPT YANKEE ATOMIC ELECTRIC COMPANY, LICENSEE FOR THE YANKEE NUCLEAR POWER STATION)

SUBJECT: REACTOR VESSEL STRUCTURAL INTEGRITY, 10 CFR 50.54(f)
(GENERIC LETTER 92-01)

The U.S. Nuclear Regulatory Commission (NRC) is issuing this generic letter to obtain information needed to assess compliance with requirements and commitments regarding reactor vessel integrity in view of certain concerns raised in the staff's review of reactor vessel integrity for the Yankee Nuclear Power Station. In Section 50.60(a) of Title 10 of the Code of Federal Regulations (10 CFR 50.60(a)), the NRC requires that licensees for all light water nuclear power reactors meet fracture toughness requirements and have a material surveillance program for the reactor coolant pressure boundary. These requirements are set forth in Appendices G and H to 10 CFR Part 50. In 10 CFR 50.60(b), where the requirements of Appendices G and H to 10 CFR Part 50 cannot be met, an exemption is necessary pursuant to 10 CFR 50.12. In 10 CFR 50.61 the NRC also provided fracture toughness requirements for protecting pressurized water reactors against pressurized thermal shock events. Licensees and permit holders have also made commitments in response to Generic Letter (GL) 88-11, "NRC Position on Radiation Embrittlement of Reactor Vessel Materials and its Impact on Plant Operations," to use the methodology in Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," to predict the effects of neutron irradiation as required by Paragraph V.A of 10 CFR Part 50, Appendix G. The 10 CFR 50.60 and 10 CFR 50.61 requirements and GL 88-11 are in the overall regulatory program to maintain the structural integrity of the reactor vessel. While reviewing the integrity of the reactor vessel at the Yankee Nuclear Power Station, the NRC staff raised concerns regarding the licensee's compliance with certain requirements and commitments.

This generic letter is part of a program to evaluate reactor vessel integrity and take regulatory actions, if needed, to ensure that licensees and permit holders are complying with 10 CFR 50.60 and 10 CFR 50.61, and are fulfilling commitments made in response to GL 88-11. Enclosure 1 is a discussion of the applicable regulatory requirements. The NRC is requiring information on compliance under the provisions of 10 CFR 50.54(f).

Assessment of Embrittlement for the Yankee Nuclear Power Station Reactor Vessel

In an effort to resolve concerns regarding the neutron embrittlement of the Yankee reactor vessel, the staff performed a safety assessment of the Yankee

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reactor vessel. The staff found that the licensee for the Yankee Nuclear Power Station might not be in compliance with 10 CFR 50.60 and had not properly completed the assessment required in 10 CFR 50.61. Further, the licensee for the Yankee Nuclear Power Station had incorrectly applied the methodology in Regulatory Guide 1.99, Revision 2.

The staff found that the Charpy upper shelf energy of the Yankee reactor vessel material could be as low as 35.5 foot-pounds which is less than the 50 foot-pound value required in Appendix G to 10 CFR Part 50. However, the licensee for the Yankee Nuclear Power Station had not performed the actions required in Paragraphs IV.A.1 or V.C of Appendix G to 10 CFR Part 50. Since then, the licensee has performed an analysis in accordance with Paragraph IV.A.1 of Appendix G to 10 CFR Part 50 using criteria being developed by the American Society of Mechanical Engineers (ASME) to demonstrate margins of safety equivalent to those in the ASME Code.

The NRC expressed a concern regarding compliance with the requirements of Appendix H to 10 CFR Part 50. Section E 185 of the American Society for Testing and Materials (ASTM) Code requires that the licensee take sample specimens from actual material used in fabricating the beltline of the reactor vessel. These surveillance materials shall include one heat of base metal, one butt weld, and one weld "heat affected zone." The licensee for the Yankee Nuclear Power Station terminated the material surveillance program in 1965. Therefore, the Yankee Nuclear Power Station had no material surveillance program on July 26, 1983, when Appendix H to 10 CFR Part 50 became effective. Further, the samples irradiated at Yankee Rowe before 1965 were comprised only of base metal.

The licensee for the Yankee Nuclear Power Station had used the methodology in Regulatory Guide 1.99, Revision 2, to predict the effects of neutron embrittlement. However, the staff found that the methodology in Regulatory Guide 1.99, Revision 2, was incorrectly applied by the licensee. The specific issues were (1) the irradiation temperature, (2) the chemistry composition of reactor vessel material, and (3) the results of the material surveillance program.

The irradiation temperature at the Yankee Nuclear Power Station is between 454 °F and 520 °F, which is below the nominal irradiation temperature of 550 °F used in developing Regulatory Guide 1.99, Revision 2. A lower irradiation temperature increases the effect of neutron embrittlement. The regulatory guide indicates that for irradiation temperatures less than 525 °F, embrittlement effects should be considered to be greater than predicted by the methods of the guide. Adjustments that were made by the licensee were insufficient to account for this effect.

The limited results of the surveillance program from the Yankee Nuclear Power Station indicated that the increase in the reference temperature exceeds the mean-plus-two standard deviations as predicted by the procedures in Regulatory Guide 1.99, Revision 2. The regulatory guide states that the licensee should use credible surveillance data to predict the increase in reference temperature resulting from neutron irradiation.

The staff implemented RG 1.99, Revision 2, by issuing GL 88-11. In committing to GL 88-11, licensees have committed to calculate radiation embrittlement in accordance with the procedures documented in RG 1.99, Revision 2. To meet the limitations in Section 1.3 of the regulatory guide, the licensee should consider the effects on irradiation embrittlement during core critical operation with irradiation temperatures less than 525 °F. Section 2 of the regulatory guide states that the licensees should consider the effects of the results from its surveillance capsules.

The Summer 1972 Addenda of the 1971 Edition of Section III of the ASME Boiler and Pressure Vessel Code are the earliest code requirements for testing materials to determine their unirradiated reference temperature. Since the Yankee reactor vessel was constructed to an ASME Code earlier than the Summer 1972, it had not been sufficiently tested to determine its unirradiated reference temperature. The licensee for the Yankee Nuclear Power Station extrapolated the available test results to determine an unirradiated reference temperature. The staff determined that the licensee's extrapolation was not conservative.

The chemical composition of the Yankee reactor vessel welds is unknown. The material's sensitivity to neutron embrittlement depends on its chemical content. The licensee assumed that the chemistry of its welds was equivalent to that of the BR-3 reactor vessel in Mol, Belgium. However, the licensee could not identify the heat number of the wire used to fabricate the Yankee welds. The licensee was assuming a chemical composition that was not based on its plant-specific information, since the chemical composition, in particular, the amount of copper, depends upon the heat number of the weld wire.

These factors prompted the staff to find that the licensee for the Yankee Nuclear Power Station had not considered plant-specific information in assessing compliance with 10 CFR 50.61. When plant-specific information is considered, the Yankee reactor vessel may have exceeded the screening criteria in 10 CFR 50.61. Since then, the licensee has performed a probabilistic fracture mechanics analysis in accordance with 10 CFR 50.61(b)(4) and the staff is continuing its review.

Upon conducting the Yankee Nuclear Power Station review, the staff became concerned that this may not be an isolated case regarding compliance with 10 CFR 50.60 and 10 CFR 50.61 and fulfillment of commitments made in response to GL 88-11. Thus, the staff is issuing this generic letter to obtain information to assess compliance with these regulations and fulfillment of commitments. The staff is continuing to pursue this concern with the Yankee Atomic Electric Company. Therefore, the Yankee Atomic Electric Company need not respond to this generic letter.

Required Information

Portions of the following information requested are not applicable to all addressees. The responses provided should, in these cases, indicate that the requested information is not applicable and why it is not applicable.

1. Certain addressees are requested to provide the following information regarding Appendix H to 10 CFR Part 50:

Addressees who do not have a surveillance program meeting ASTM E 185-73, -79, or -82 and who do not have an integrated surveillance program approved by the NRC (see Enclosure 2), are requested to describe actions taken or to be taken to ensure compliance with Appendix H to 10 CFR Part 50. Addressees who plan to revise the surveillance program to meet Appendix H to 10 CFR Part 50 are requested to indicate when the revised program will be submitted to the NRC staff for review. If the surveillance program is not to be revised to meet Appendix H to 10 CFR Part 50, addressees are requested to indicate when they plan to request an exemption from Appendix H to 10 CFR Part 50 under 10 CFR 50.60(b).

2. Certain addressees are requested to provide the following information regarding Appendix G to 10 CFR Part 50:

- a. Addressees of plants for which the Charpy upper shelf energy is predicted to be less than 50 foot-pounds at the end of their licenses using the guidance in Paragraphs C.1.2 or C.2.2 in Regulatory Guide 1.99, Revision 2, are requested to provide to the NRC the Charpy upper shelf energy predicted for December 16, 1991, and for the end of their current license for the limiting beltline weld and the plate or forging and are requested to describe the actions taken pursuant to Paragraphs IV.A.1 or V.C of Appendix G to 10 CFR Part 50.

- b. Addressees whose reactor vessels were constructed to an ASME Code earlier than the Summer 1972 Addenda of the 1971 Edition are requested to describe the consideration given to the following material properties in their evaluations performed pursuant to 10 CFR 50.61 and Paragraph III.A of 10 CFR Part 50, Appendix G:

- (1) the results from all Charpy and drop weight tests for all unirradiated beltline materials, the unirradiated reference temperature for each beltline material, and the method of determining the unirradiated reference temperature from the Charpy and drop weight test;
- (2) the heat treatment received by all beltline and surveillance materials;
- (3) the heat number for each beltline plate or forging and the heat number of wire and flux lot number used to fabricate each beltline weld;

- (4) the heat number for each surveillance plate or forging and the heat number of wire and flux lot number used to fabricate the surveillance weld;
 - (5) the chemical composition, in particular the weight in percent of copper, nickel, phosphorous, and sulfur for each beltline and surveillance material; and
 - (6) the heat number of the wire used for determining the weld metal chemical composition if different than Item (3) above.
3. Addressees are requested to provide the following information regarding COMMENTS made to respond to GL 88-11:
- a. How the embrittlement effects of operating at an irradiation temperature (cold leg or recirculation suction temperature) below 525 °F were considered. In particular licensees are requested to describe consideration given to determining the effect of lower irradiation temperature on the reference temperature and on the Charpy upper shelf energy.
 - b. How their surveillance results on the predicted amount of embrittlement were considered.
 - c. If a measured increase in reference temperature exceeds the mean-plus-two standard deviations predicted by Regulatory Guide 1.99, Revision 2, or if a measured decrease in Charpy upper shelf energy exceeds the value predicted using the guidance in Paragraph C.1.2 in Regulatory Guide 1.99, Revision 2, the licensee is requested to report the information and describe the effect of the surveillance results on the adjusted reference temperature and Charpy upper shelf energy for each beltline material as predicted for December 16, 1991, and for the end of its current license.

Reporting Requirements

Pursuant to Section 182a of the Atomic Energy Act of 1954, as amended, and 10 CFR 50.54(f), each addressee shall submit a letter within 120 days of the date of this generic letter providing the information described under "Required Information." The letter shall be addressed to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555, under oath or affirmation. A copy shall also be submitted to the appropriate Regional Administrator. This generic letter requests information that will enable the NRC to verify that the licensee is complying with its current licensing basis regarding reactor vessel fracture toughness and material surveillance for the reactor coolant pressure boundary. Accordingly, an evaluation justifying this information request is not necessary under 10 CFR 50.54(f).

Backfit Discussion

This generic letter requests information that will enable the NRC staff to determine whether licensees are complying with their prior commitments and any license conditions regarding 10 CFR 50.60, 10 CFR 50.61, and GL 88-11. The staff is not establishing a new position for such compliance in this generic letter. The staff is requesting information to verify that the licensee is complying with its previously established commitments and is not establishing any new position. Therefore, this generic letter does not constitute a backfit and no documented evaluation or backfit analysis need be prepared.

Request for Voluntary Submittal of Impact Data

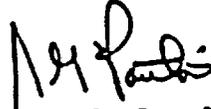
This request is covered by Office of Management and Budget Clearance Number 3150-0011, which expires May 31, 1994. The estimated average number of burden hours is 200 person hours for each addressee's response, including the time required to assess the requirements, search data sources, gather and analyze the data, and prepare the required letters. This estimated average number of burden hours pertains only to the identified response-related matters and does not include the time to implement the actions required by the regulations. Comments on the accuracy of this estimate and suggestions to reduce the burden may be directed to Ronald Minsk, Office of Information and Regulatory Affairs (3150-0011), NEOB-3019, Office of Management and Budget, Washington, DC 20503, and to the U.S. Nuclear Regulatory Commission, Information and Records Management Branch, Division of Information Support Services, Office of Information and Resources Management, Washington, DC 20555.

Although no specific request or requirement is intended, the following information would assist the NRC in evaluating the cost of complying with this generic letter:

- (1) the licensee staff's time and costs to perform requested inspections, corrective actions, and associated testing;
- (2) the licensee staff's time and costs to prepare the requested reports and documentation;
- (3) the additional short-term costs incurred to address the inspection findings such as the costs of the corrective actions or the costs of down time; and
- (4) an estimate of the additional long-term costs that will be incurred as a result of implementing commitments such as the estimated costs of conducting future inspections or increased maintenance.

If you have any questions about this matter, please contact one of the NRC technical contacts or the lead project manager listed below.

Sincerely,



James G. Partlow
Associate Director for Projects
Office of Nuclear Reactor Regulation

Enclosures:

1. Applicable Regulatory Requirements
2. Plants with Integrated Programs
3. List of Recently Issued
Generic Letters

Technical Contacts:

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(301) 504-2757

Lead Project Manager:

Daniel G. McDonald, NRR
(301) 504-1408

Regulatory Requirements Applicable to
Reactor Vessel Structural Integrity

10 CFR 50.60

Pursuant to 10 CFR 50.60, all light water nuclear power reactors must meet the fracture toughness and material surveillance program requirements for the reactor coolant pressure boundary set forth in Appendices G and H to 10 CFR Part 50.

The fracture toughness of the reactor coolant pressure boundary required by 10 CFR 50.60 is necessary to provide adequate margins of safety during any condition of normal plant operation, including anticipated operational occurrences and system hydrostatic tests. The material surveillance program required by 10 CFR 50.60 monitors changes in the fracture toughness properties of ferritic materials in the reactor vessel beltline region of light water nuclear power reactors resulting from exposure of these materials to neutron irradiation and the thermal environment. Under the program, fracture toughness test data are obtained from material specimens exposed in surveillance capsules, which are withdrawn periodically from the reactor vessel.

Appendix G to 10 CFR Part 50 requires that the reactor vessel beltline materials must have Charpy upper shelf energy of no less than 50 ft-lb throughout the life of the vessel. Otherwise, licensees are required to provide demonstration of equivalent margins of safety in accordance with Paragraph IV.A.1 of Appendix G to 10 CFR Part 50 or perform actions in accordance with Paragraph V.C of Appendix G to 10 CFR Part 50.

Appendix H to 10 CFR Part 50 requires the surveillance program to meet the American Society for Testing and Materials (ASTM) Standard E 185, "Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels." Further, Appendix H to 10 CFR Part 50 specifies the applicable edition of ASTM E 185. Appendix H to 10 CFR Part 50, as amended on July 26, 1983, requires that the part of the surveillance program conducted before the first capsule is withdrawn must meet the requirements of the 1973, the 1979, or the 1982 edition of ASTM E 185 that is current on the issue date of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code under which the reactor vessel was purchased. The licensee may also use later editions of ASTM E 185 which have been endorsed by the NRC. The test procedures and reporting requirements for each capsule withdrawal after July 26, 1983 must meet the requirements of the 1982 edition of ASTM E 185 to the extent practical for the configuration of the specimens in the capsule. The licensee may use either the 1973, the 1979, or the 1982 edition of ASTM E 185 for each capsule withdrawal before July 26, 1983.

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Licensees, especially those with reactor vessels purchased before ASTM issued the 1973 edition of ASTM E 185, may have surveillance programs that do not meet the requirements of Appendix H to 10 CFR Part 50 but may have alternative surveillance programs. The licensee may use these alternative surveillance programs in accordance with 10 CFR 50.60(b) if the licensee has been granted an exemption by the Commission under 10 CFR 50.12.

The licensee must monitor the test results from the material surveillance program. According to Paragraph III.C of Appendix H to 10 CFR Part 50, the results of the surveillance program may indicate that a technical specifications change is required, either in the pressure-temperature limits or in the operating procedures required to meet the limits.

10 CFR 50.61

Pursuant to 10 CFR 50.61, there are fracture toughness requirements for protection against pressurized thermal shock events for pressurized water reactors. Licensees are required to perform an assessment of the projected values of reference temperature. If the projected reference temperature exceeds the screening criteria established in 10 CFR 50.61, licensees are required to submit an analysis and schedule for such flux reduction programs as are reasonably practicable to avoid exceeding the screening criteria. If no reasonably practicable flux reduction program will avoid exceeding the screening criteria, licensees shall submit a safety analysis to determine what actions are necessary to prevent potential failure of the reactor vessel if continued operation beyond the screening criteria is allowed. In 10 CFR 50.61(b)(1), as amended effective June 14, 1991 (56 Fed Reg 22300 et. seq., May 15, 1991), licensees are required to submit their assessment by December 16, 1991, if the projected reference temperature will exceed the screening criteria before the expiration of the operating license.

Plant-specific information is required to be considered in assessing the level of neutron embrittlement as specified in 10 CFR 50.61(b)(3). This information includes but is not limited to the reactor vessel operating temperature and surveillance results.

Prediction of Irradiation Embrittlement

Paragraph V.A of Appendix G to 10 CFR Part 50 requires the prediction of the effects of neutron irradiation on reactor vessel materials. The extent of neutron embrittlement depends on the material properties, thermal environment, and results of the material surveillance program. In Generic Letter 88-11, "NRC Position on Radiation Embrittlement of Reactor Vessel Materials and its Impact on Plant Operations," the staff stated that it will use the guidance in Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," in estimating the embrittlement of the materials in the reactor vessel beltline. All licensees and permittees have responded to Generic Letter 88-11 committing to use the methodology in Regulatory Guide 1.99,

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Revision 2, in predicting the effects of neutron irradiation as required by Paragraph V.A of 10 CFR Part 50, Appendix G. The methodology in Regulatory Guide 1.99, Revision 2, is also the basis in 10 CFR 50.61 in projecting the reference temperature.

Enclosure 2

Plants With Integrated Surveillance Programs Approved By The NRC

Oconee Units 1, 2, and 3
Arkansas Nuclear One Unit 1
Rancho Seco
Three Mile Island Unit 1
Davis-Besse
Ginna
Point Beach Units 1 and 2
Surry Units 1 and 2
Turkey Point Units 3 and 4
Zion Units 1 and 2

February 25, 1992

The Committee to Review Generic Requirements recommended in favor of issuing this generic letter at its meeting number 211 held on November 26, 1991.

CR 92-01-00000 873

James G. Partlow
Associate Director for Projects
Office of Nuclear Reactor Regulation

Enclosure:
Generic Letter 92-01

cc w/enclosure:
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