50-336

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

January 21, 1997

Mr. Bruce D. Kenyon
President - Nuclear Group
Northeast Utilities Service Company
c/o Mr. Richard T. Laudenat
Director - Nuclear Licensing Services
P.O. Box 128
Waterford, CT 06385

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION REGARDING THE MILLSTONE NUCLEAR POWER STATION, UNIT NO. 2, THIRD 10-YEAR INTERVAL INSERVICE INSPECTION PROGRAM PLAN AND ASSOCIATED REQUESTS FOR RELIEF (TAC NO. M96200)

Dear Mr. Kenyon:

By letter dated July 2, 1996, Northeast Utilities provided its third 10-year interval inservice inspection program plan and associated requests for relief from the American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Section XI, requirements for the Millstone Nuclear Power Station, Unit No. 2. The NRC staff, with assistance from its contractor, Idaho National Engineering Laboratory (INEL), is evaluating the submittal.

Enclosed is a request for additional information (RAI). The information is required in order for the NRC staff to complete its review. We request that a response be provided within 60 days from receipt of this RAI to be consistent with the NRC staff's inservice inspection program plan review schedule. In addition, to expedite the review process, please send a copy of your response to NRC's contractor, INEL, at the following address:

Michael T. Anderson
INEL Research Center
2151 North Boulevard
P.O. Box 1625
Idaho Falls, Idaho 83415-2209

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If you have any questions relating to this RAI, please contact me at (301) 415-1408.

Sincerely,

Daniel G. McDonald Jr.

Daniel G. McDonald, Sr. Project Manager
Special Projects Office - Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-336

Enclosure: Request for Additional Information

cc w/encl: See next page

If you have any questions relating to this RAI, please contact me at (301) 415-1408.

Sincerely,

Original signed by:

Daniel G. McDonald, Sr. Project Manager
Special Projects Office - Licensing
Office of Nuclear Reactor Regulation

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Enclosure: Request for Additional Information

cc w/encl: See next page

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REQUEST FOR ADDITIONAL INFORMATION

THIRD 10-YEAR INTERVAL INSERVICE INSPECTION PROGRAM PLAN

MILLSTONE NUCLEAR POWER STATION, UNIT NO. 2

NORTHEAST NUCLEAR ENERGY COMPANY

DOCKET NUMBER 50-336

1. Scope/Status of Review

Throughout the service life of a water-cooled nuclear power facility, 10 CFR 50.55a(g)(4) requires that components (including supports) that are classified as American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code Class 1, Class 2, and Class 3 meet the requirements, except design and access provisions and preservice examination requirements, set forth in ASME Code Section XI, *Rules for Inservice Inspection of Nuclear Power Plant Components*, to the extent practical within the limitations of design, geometry, and materials of construction of the components. This section of the regulations also requires that inservice examinations of components and system pressure tests conducted during successive 120-month inspection intervals comply with the requirements in the latest edition and addenda of the Code incorporated by reference in the *Code of Federal Regulations* (CFR) 10 CFR 50.55a(b) on the date 12 months prior to the start of the 120-month interval, subject to the limitations and modifications listed therein. The components (including supports) may meet requirements set forth in subsequent editions and addenda of the Code that are incorporated by reference in 10 CFR 50.55a(b) subject to the limitations and modifications listed therein. The licensee, Northeast Nuclear Energy Company, has prepared the *Millstone Nuclear Power Station, Unit 2, Third 10-Year Interval Inservice Inspection Program*, Revision 2, to meet the requirements of the 1989 Edition of Section XI of the ASME Code.

The Nuclear Regulatory Commission (NRC) is responsible for the review and disposition of requests related to inservice inspection requirements of ASME Section XI and CFR. It has been determined by the NRC that, when submitting proposed alternatives or requests for relief to ASME/CFR requirements, it is imperative that appropriate paragraphs in the Code of Federal Regulation, 10 CFR 50.55a, are cited.

The staff has reviewed the Northeast Nuclear Energy Company, *Millstone Nuclear Power Station, Unit 2, Third 10-Year Interval Inservice Inspection Program*, Revision 2, submitted by letter dated July 2, 1996, and the requests from the ASME/CFR requirements.

2. Additional Information Required

Based on the above review, the staff has concluded that additional information and/or clarification is required to complete the review of the Internal Inservice Inspection Program Plan.

A. Address the degree of compliance with augmented examinations that have been established by the NRC when added assurance of structural

Enclosure

reliability is deemed necessary. Examples of documents that address augmented examinations are:

(1) Branch Technical Position MEB 3-1, *High Energy Fluid Systems, Protection Against Postulated Piping Failures in Fluid Systems Outside Containment*;

(2) Regulatory Guide 1.150, *Ultrasonic Testing of Reactor Vessel Welds During Preservice and Inservice Examinations*.

Discuss these and any other augmented examinations that may have been incorporated in the *Millstone Nuclear Power Station, Unit 2, Third 10-Year Interval Inservice Inspection Program*, Revision 2.

B. Discuss any plans or schedules for examination of a sample of the welds in the Residual Heat Removal (RHR), Emergency Core Cooling (ECC), and Containment Heat Removal (CHR) systems to assure the continued integrity of thin-wall piping. Paragraph 10 CFR 50.55a(b)(2)(iv) requires that certain ASME Code Class 2 piping welds in these systems be examined. These systems are critical to the safe shutdown of the plant and should not be completely excluded from inservice volumetric examination based on piping wall thickness. The staff has previously determined for similar plants that a 7.5% augmented volumetric sample of thin-wall welds constitutes an acceptable resolution.

C. Provide the staff with the status of the augmented reactor pressure vessel examinations required by the 10 CFR 50.55a(g)(6)(ii)(A), effective September 8, 1992, and provide a technical discussion describing how the regulation was/will be implemented at Millstone Nuclear Power Plant, Unit 2. Include in the discussion a description of the approach and any specialized techniques or equipment that was/will be used to complete the required augmented examination.

D. Review RR-89-04, -05, -06, -08, and -16 and cite appropriate paragraphs of 10 CFR 50.55a as applicable. If revised or additional requests for relief are submitted, ensure that the appropriate reference to CFR is included.

The licensee must state the specific paragraph of the Regulations under which each proposed alternative or request for relief is submitted. The licensee should review the current submittal(s), and provide the required references to ensure that each proposed alternative or request for relief is evaluated in accordance with the appropriate criteria, as discussed below.

A licensee may propose an alternative to CFR or Code requirements in accordance with 10 CFR 50.55a(a)(3)(i) or 10 CFR 50.55a(a)(3)(ii). When submitting a proposed alternative, the licensee must specify the appropriate regulatory basis. Under 10 CFR 50.55a(a)(3)(i), the proposed alternative must be shown to provide an acceptable level of quality and safety, i.e., essentially be equivalent to the original requirement in terms of quality and safety. Under 10 CFR 50.55a(a)(3)(ii), the licensee must show that compliance with the original requirement results in a hardship or unusual difficulty without a compensating increase in the level of quality and safety. Examples of hardship and/or unusual difficulty include, but are not limited to, excessive radiation exposure, disassembly of components solely to provide access for examinations, and development of sophisticated tooling that would result in only minimal increases in examination coverage.

A licensee may submit a request for relief from ASME requirements. In accordance with 10 CFR 50.55a(g)(5)(iii), if a licensee determines that conformance with certain ASME Code requirements is impractical for its facility, the licensee shall notify the Commission and submit, as specified in §50.4, information to support that determination. When a licensee determines that an inservice inspection requirement is impractical, e.g., the system would have to be redesigned or a component would have to be replaced to enable inspection, the licensee should cite this portion of CFR to support the criteria for evaluation. The NRC may, giving due consideration to the burden placed on the licensee, impose an alternative examination requirement.

E. Describe the action the licensee proposes to take regarding the apparent conflict between the Code requirement that a percentage of examinations for each Examination Category be completed each period and the licensee's statement under Section 2, Inservice Inspection Program Table, Note 6, that RPV stud examinations (Examination Category B-G-1) will be deferred until the end of the third interval. The stated position is considered unacceptable without relief from the Code requirements.

F. Describe the action the licensee proposes to take regarding the apparent conflict between the Code requirement (IWB-2420 and IWC-2420) that the sequence of component examinations established during the first interval be repeated during successive examination intervals and the licensee's statement in Section 4.01-1 that the exam schedule for some examination areas may be changed. Note that approval to deviate from established schedules is required.

G. Describe the action the licensee proposes to take regarding the apparent conflict between the Code requirement that a percentage of examinations be performed each period and the licensee's statement in Section 4.01-3 that the steam generator exams would be performed on or near a ten-year interval. Considering that the Code requires a percentage of these examinations be performed each period and that these are replacement steam generators, it is considered unacceptable to defer these examinations to the end of the interval.

H. State whether testing of the Service Water, Relief Valve Inlet Piping and associated piping will be in compliance with the Code. In Section 6.03-3, the licensee stated that Request for Relief RR-89-15 is applicable for this piping. However Request for Relief RR-89-15 has been withdrawn. Explain the discrepancy.

I. Verify that the requirements of Appendix VI will be satisfied for bolting examinations. (In Section 8, the licensee referenced Code Case

N-457, *Qualification Specimen Notch Location for Ultrasonic Examination of Bolts and Studs*, for bolting examinations.)

J. Review these Requests for Relief and verify that the Safety Evaluations allow the use of the Code Cases listed in Section 8.01-3, -4 for the third interval. These Code Cases were proposed as alternatives to Code requirements under separate submittals. If these Code Cases are not approved specifically for the third interval, the licensee is required to submit them for review and approval.

K. Provide the actual coverage(s) that can be obtained for the Steam Generator Nozzle to Vessel Welds in Relief Request RR-89-10.

L. Provide the actual coverage(s) that can be obtained for the Steam Generator circumferential head welds in Relief Request RR-89-14.

M. Describe the action the licensee proposes to take regarding the apparent conflict between the Code requirement for insulation removal from bolted connections in systems borated for the purpose of controlling reactivity and the licensee's proposal to remove insulation each refueling outage only from connections with carbon steel bolting. In Relief Request RR-89-17, the licensee has proposed, as an alternative, to remove insulation only from connections with carbon steel bolting. The Code does not have an exception to the insulation removal requirement based on bolting material. Some austenitic bolting materials have been found to be susceptible to primary water stress corrosion. As a result, the NRC has found it unacceptable to exclude austenitic bolted connections from insulation removal.

N. Describe the action the licensee proposes to take regarding the apparent conflict between the Code Item B7.80 requirement of a VT-1 visual examination only when the component is disassembled and the licensee's request for relief from VT-1 visual examination of Control Element Drive Mechanism (CEDM) Greylock bolting. Is it the intent of Relief Request RR-89-18 to obtain relief from the VT-1 visual examination when the component is disassembled? It should be noted that the NRC has determined that Case N-547 is not acceptable for use without a commitment from the licensee to perform a VT-1 visual examination on bolting prior to replacement, unless the bolts are replaced with new ones.

O. Verify that there are no requests for relief in addition to those submitted. If additional requests for relief are required, the licensee should submit them for staff review.

The schedule for timely completion of this review requires that the licensee provide, by the requested date, the above requested information and/or clarification with regard to the *Millstone Nuclear Power Station, Unit 2, Third 10-Year Interval Inservice Inspection Program*, Revision 2.