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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION OF REQUEST FOR RELIEF NO. IR-28, FIRST 10-YEAR INTERVAL NORTHEAST NUCLEAR ENERGY COMPANY MILLSTONE NUCLEAR POWER STATION, UNIT NO. 3 DOCKET NUMBER 50-423

1.0 INTRODUCTION

The Technical Specifications for Millstone Unit 3, state that the inservice inspection of the American Society of Mechanical Engineers (ASME) Code Class 1, 2, and 3 components shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable addenda as required by Title 10 of the Code of Federal Regulations (10 CFR), Part 50, Section 50.55a. Pursuant to 10 CFR 50.55a(g)(6)(i), if the licensee has determined that conformance with certain Code requirements is impractical for its facility, the Commission may grant relief as it determines is authorized by law and will not endanger life or property or the common defense and security and is otherwise in the public interest giving due consideration to the burden upon the licensee that could result if the requirement were imposed on the facility.

Pursuant to 10 CFR 50.55a(g)(4), ASME Code Class 1, 2, and 3 components (including supports) shall meet the requirements, except the design and access provisions and the pre-service examination requirements, set forth in the 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. The regulations require that inservice examination of components and system pressure tests conducted during the first 10-year inservice inspection (ISI) interval and subsequent intervals comply with the requirements in the latest edition and addenda of Section XI of the ASME Code incorporated by reference in 10 CFR 50.55a(b) 12 months prior to the start of the 120-month interval, subject to the limitations and modifications listed therein.

By letter dated September 17, 1999, Northeast Nuclear Energy Company (NNECO) requested relief, pursuant to 10 CFR 50.55a(g), from the requirements of the ASME Code, Section XI for Millstone Unit 3. Specifically, NNECO requested relief pursuant to 10 CFR 50.55a(g)(5)(iii), from performing the VT-3 visual examination of the Millstone Unit 3 reactor pressure vessel supports to the extent required by the Code for Class 1 supports, other than piping, subject to visual examination (Request for Relief IR-28). The request applies to the inaccessible portions of the supports that are encased in permanent insulation.

Enclosure

2.0 EVALUATION

The information provided by NNECO in support of its request for relief from Code requirements has been evaluated and the basis for disposition is documented below. The Code of record for the Millstone Nuclear Power Station, Unit No. 3, first 10-year ISI interval, which began April 23, 1986, is the 1983 Edition through Summer 1983 Addenda of Section XI of the ASME Boiler and Pressure Vessel Code. The Code requires that for Class 1 supports, other than piping supports, the settings of guides and stops, alignment of supports, and proper assembly of support items be included within the scope of the VT-3 visual examination. As an alternative to the Code, Code Case N-491 has been approved by the NRC for general use by licensees in Regulatory Guide 1.147, "Inservice Inspection Code Case Acceptability."

2.1 Request for Relief IR-28 - First Inspection Interval - Reactor Pressure Vessel Supports

Component Identification:

Code Class:	1
Examination Category:	F-A
Item Number:	F1.40
Component Numbers:	RVS-1, RVS-2, RVS-3, and RVS-4

Code Requirement:

ASME Section XI, Code Case N-491, Table 2500-1 requires that Class 1 supports, other than piping supports, be subject to a visual VT-3 examination once every interval.

Code Relief Request:

Pursuant to 10 CFR 50.55a(g)(5)(iii), relief is requested from performing the visual examination of the subject supports to the extent required by the Code. This relief request applies to the inaccessible portions of the supports that are encased in permanent insulation.

Basis for Requesting Relief:

The Millstone Unit No. 3 reactor vessel has four supports that are located under two cold leg nozzles and two hot leg nozzles. The support assembly at each of these nozzles consists of a nozzle pad and steel plates positioned between a steel support structure that is welded to the neutron shield tank as shown in Figure 1. The support is designed to function as a vertical restraint, loaded in compression and would not

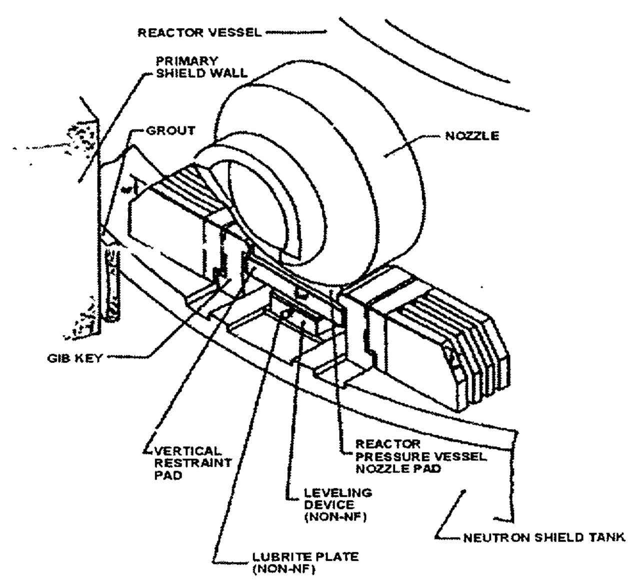

Figure 1
Reactor Vessel Support Assembly

be subject to typical failure mechanisms associated with stress at rigid connections and loosened or degraded fasteners. The majority of each support is encased in the permanent insulation panels of the reactor vessel and vessel nozzles. Portions of the steel support structure and associated welds are accessible for a limited visual VT-3 examination.

When performing the examination during the last refueling outage (RFO6), NNECO determined that the supports are located in a congested, confined space below the permanent refueling cavity seal ring. The area can only be accessed through four seal ring hatches. In addition to difficult access, the radiation levels in the area are approximately 100mr - 150mr per hour. It is estimated that the removal and reinstallation of the permanent insulation in this confined space would result in additional exposure of approximately 4.0 man-rem. Additionally, it is anticipated that modification and/or removal of the permanent cavity seal ring would be required to support access for this project. Based on the access restrictions, high radiation levels and support design, relief is requested from performing the visual VT-3 examination on the inaccessible portions of these supports for Millstone Unit No. 3, first 10-year inspection interval (April 23, 1986 through October 23, 1999).

Proposed Alternative Examinations:

A limited visual VT-3 exam was performed satisfactorily on the accessible portions of the subject supports to the maximum extent practical with the insulation in place including examination of the insulation for any evidence of disturbance or degradation which may be attributed to abnormal support disturbance. This examination will be performed once per interval as currently required by the ASME Code.

2.2 Staff Evaluation

The Code requires that Class 1 supports, other than piping supports, be subject to a 100 percent VT-3 visual examination once every interval. However, due to access restrictions, the support design, and high local radiation levels, as an alternative during RFO6 the licensee performed a limited VT-3 visual examination on the accessible portions of the supports including examination of the insulation for any evidence of disturbance or degradation attributable to abnormal support disturbance. Because of the design of the support and the significant thermal loads experienced by the support during transitions between shutdown and operating conditions, it is reasonable to conclude that problems with the alignment or assembly of the supports would result in degradation of the insulation surrounding the supports. Also, portions of the steel structure and associated welds are visible for VT-3 visual examination. Examination of these areas provides further information regarding the condition of the supports. On the basis of the above considerations, the staff concludes that the licensee's proposed alternative examination provides reasonable assurance of continued structural integrity of the reactor pressure vessel supports.

3.0 CONCLUSION

The NRC staff has evaluated the licensee's submittal and concludes that pursuant to 10 CFR 50.55a(a)(6)(i) NNECO has demonstrated that performing the visual examination to the extent required by the Code of the reactor pressure vessel supports is impractical and that the alternative examination provides reasonable assurance of structural integrity. Therefore, relief is granted and the alternative requirement imposed for the first 10-year inspection interval pursuant to 10 CFR 50.55a(g)(6)(i). The relief granted is authorized by law, will not endanger life or property or the common defense and security, and is otherwise in the public interest giving due consideration to the burden upon the licensee that could result if the requirements were imposed on the facility.

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Date: November 3, 1999