July 24, 2000

Mr. S. E. Scace - Director
Nuclear Oversight and Regulatory Affairs
c/o Mr. David A. Smith
Northeast Nuclear Energy Company
P. O. Box 128
Waterford, CT 06385-0128

SUBJECT: SAFETY EVALUATION OF RELIEF REQUESTS ASSOCIATED WITH THE FIRST AND SECOND 10-YEAR INTERVAL OF THE INSERVICE INSPECTION (ISI) PLAN, MILLSTONE NUCLEAR POWER STATION, UNIT NO. 3 (TAC NO. MA5446)

Dear Mr. Scace:

By letter dated April 27, 1999, as supplemented by letters dated September 17 and November 4, 1999, and April 12 and June 21, 2000, you requested, pursuant to 10 CFR 50.55a, that the U.S. Nuclear Regulatory Commission (NRC) grant relief from and authorize alternatives to certain provisions of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI, for for Millstone Nuclear Power Station, Unit No. 3. Specifically, you submitted your Second 10-Year Interval ISI Program Plan Requests for Relief Nos. IR-2-01 through IR-2-09 and First 10-Year Interval ISI Program Plan Requests for Relief No. IR-28.

The NRC staff, with technical assistance from its contractor, the Idaho National Engineering and Environmental Laboratory (INEEL), has reviewed the proposed relief requests. The staff adopts the evaluations and recommendations for granting relief or authorizing alternatives contained in the enclosed INEEL Technical Letter Report (TLR) for IR-28, IR-2-01, IR-2-03 through IR-2-05, and IR-2-07 through IR-2-09. INEEL rejected Request for Relief IR-2-06, “Utilization of Code Case N-566.” Subsequently, you submitted a revised relief request based on Code Case N-566-1. The NRC staff found this relief request acceptable in its safety evaluation. Also, INEEL did not evaluate Request for Relief IR-2-02. The NRC staff evaluated this relief request in its safety evaluation. Specifically, the NRC staff concludes that for Requests for Relief:

a. No. IR-28: Pursuant to 10 CFR 50.55a(g)(6)(i), you requested relief from performing 100% of the Code-required VT-3 visual examinations for the reactor vessel supports because of access restrictions, local high radiation levels, and support design. You proposed to perform a VT-3 visual examination of the accessible portions of the subject supports to the maximum extent practical with the insulation in place. In addition, the insulation will be examined for any evidence of disturbance or degradation which may be attributed to abnormal support disturbance. The Code requirements are impractical and the examinations performed will provide reasonable assurance of continued structural integrity of the supports. Therefore, relief is granted pursuant to 10 CFR 50.55a(g)(6)(i).

b. No. IR-2-01: Pursuant to 10 CFR 50.55a(a)(3)(i), you proposed use of Code Cases N-389-1, N-416-1, N-491-1, N-521, N-522, and N-524 as alternatives to the requirements of the ASME Boiler and Pressure Vessel Code. The NRC staff recently reviewed these code cases and found them acceptable for general use, with specific conditions as applicable, as evidenced by incorporation into Regulatory Guide (RG) 1.147, *Inservice Inspection Code Case Acceptability*, Revision 12, (May 1999). Therefore, these code cases with the applicable conditions specified in RG 1.147 are approved for use during the second 10-year interval.

c. No. IR-2-02: Pursuant to 10 CFR 50.55a(a)(3)(i), you proposed utilization of the requirements set forth in Technical Specification (TS) 4.7.10 as alternatives to the requirements of the ASME Boiler and Pressure Vessel Code. The staff concludes that the proposed alternative to use the requirements set forth in TS 4.7.10 for the snubber program provides an acceptable level of quality and safety. Therefore, the proposed alternative is authorized pursuant to 10 CFR 50.55a(a)(3)(i) for the second 10-year interval.

d. No. IR-2-03: Pursuant to 10 CFR 50.55a(g)(5)(iii), you proposed performance of a full volumetric examination in lieu of the Code-required surface examination of nozzle to safe-end and safe-end to pipe butt welds greater than 4-inch nominal pipe size on the reactor pressure vessel. Due to the design of the reactor vessel nozzles, which have a permanent mirror type insulation installed covering the outside diameter surface welds, to require you to perform a surface examination would be a burden with no compensating increase in the level of quality and safety. Therefore, the proposed alternative is authorized pursuant to 10 CFR 50.55a(a)(3)(ii) for the second 10-year interval.

e. No. IR-2-04: Pursuant to 10 CFR 50.55a(g)(5)(iii), you proposed an alternative to the scheduling requirements of Table IWB-2412-1. Your proposed alternative is to perform the Code-required examinations of eight (of nine) Examination Category B-H integrally welded attachments during the second inspection period. Requiring licensee personnel to enter high radiation fields for the purpose of insulation removal and examination on three separate occasions, is a hardship with no compensating increase in level of quality and safety. Therefore, the proposed alternative is authorized pursuant to 10 CFR 50.55a(a)(3)(ii) for the second 10-year interval.

f. No. IR-2-05: Pursuant to 10 CFR 50.55a(g)(5)(iii), you requested relief from performing the volumetric examination of the inner radius of the Main Steam Nozzles. Based on the design of the steam generator main steam outlet nozzle inner radius sections it has been concluded that the Code-required volumetric examination is impractical and the licensee’s proposed alternative provides reasonable assurance of structural integrity of the subject components. Therefore, relief is granted pursuant to 10 CFR 50.55a(g)(6)(i).

g. No. IR-2-06: Pursuant to 10 CFR 50.55a(a)(3)(i), you proposed use of Code Case N-566-1, as an alternative to the requirements of the ASME Boiler and Pressure Vessel Code for bolted connections. The use of Code Case N-566-1 as an alternative to the Code requirements provides an acceptable level of quality and safety. Therefore, the

proposed alternative is authorized pursuant to 10 CFR 50.55a(a)(3)(i) for the second 10-year interval or until such time as Code Case N-566-1 is referenced in a future revision of 10 CFR 50.55a. At that time, if you intend to continue to implement Code Case N-566-1 you are to follow all provisions in the Code Case with limitations or conditions specified in 10 CFR 50.55a.

h. No. IR-2-07: Pursuant to 10 CFR 50.55a(a)(3)(i), you proposed use of Code Case N-535, as an alternative to the requirements of the ASME Boiler and Pressure Vessel Code for components which are classified as ASME Code Class 1, 2, or 3. The proposed alternative provides an acceptable level of quality and safety. Therefore, the proposed alternative is authorized pursuant to 10 CFR 50.55a(a)(3)(i) for the second 10-year interval.

i. No. IR-2-08: Pursuant to 10 CFR 50.55a(a)(3)(ii), you proposed use of Code Case N-533 as an alternative to the requirements of the ASME Boiler and Pressure Vessel Code which requires removal of insulation from bolted connections for VT-2 visual examination when the systems are borated for the purpose of controlling reactivity. Compliance with the Code requirements for Class 1 systems would result in a burden without a compensating increase in the level of quality and safety. Furthermore, the proposed alternative provides reasonable assurance of continued leakage integrity for Class 1 bolted connections. Therefore, the proposed alternative is authorized pursuant to 10 CFR 50.55a(a)(3)(ii) for the second 10-year interval.

j. No. IR-2-09: Pursuant to 10 CFR 50.55a(a)(3)(i), you proposed use of Code Case N-546 as an alternative to the requirements of the ASME Boiler and Pressure Vessel Code which requires personnel performing examinations to be qualified by examination and certified in accordance with SNT-TC-1A. Based on a review of this Code Case the staff concludes that the proposed alternative provides an acceptable level of quality and safety. Therefore, the proposed alternative is authorized pursuant to 10 CFR 50.55a(a)(3)(i) for the second 10-year interval.

Our detailed evaluation and conclusions are documented in the enclosed safety evaluation. This completes our effort on these requests, and therefore, we are, closing out TAC No. MA5446. If you have any questions regarding this matter, please contact the Millstone Unit 3 Project Manager, Victor Nerses, at (301) 415-1484.

Sincerely,

/RA/

James W. Clifford, Chief, Section 2
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-423

Enclosure: Safety Evaluation

cc w/encl: See next page

proposed alternative is authorized pursuant to 10 CFR 50.55a(a)(3)(i) for the second 10-year interval or until such time as Code Case N-566-1 is referenced in a future revision of 10 CFR 50.55a. At that time, if you intend to continue to implement Code Case N-566-1 you are to follow all provisions in the Code Case with limitations or conditions specified in 10 CFR 50.55a.

h. No. IR-2-07: Pursuant to 10 CFR 50.55a(a)(3)(i), you proposed use of Code Case N-535, as an alternative to the requirements of the ASME Boiler and Pressure Vessel Code for components which are classified as ASME Code Class 1, 2, or 3. The proposed alternative provides an acceptable level of quality and safety. Therefore, the proposed alternative is authorized pursuant to 10 CFR 50.55a(a)(3)(i) for the second 10-year interval.

i. No. IR-2-08: Pursuant to 10 CFR 50.55a(a)(3)(ii), you proposed use of Code Case N-533 as an alternative to the requirements of the ASME Boiler and Pressure Vessel Code which requires removal of insulation from bolted connections for VT-2 visual examination when the systems are borated for the purpose of controlling reactivity. Compliance with the Code requirements for Class 1 systems would result in a burden without a compensating increase in the level of quality and safety. Furthermore, the proposed alternative provides reasonable assurance of continued leakage integrity for Class 1 bolted connections. Therefore, the proposed alternative is authorized pursuant to 10 CFR 50.55a(a)(3)(ii) for the second 10-year interval.

j. No. IR-2-09: Pursuant to 10 CFR 50.55a(a)(3)(i), you proposed use of Code Case N-546 as an alternative to the requirements of the ASME Boiler and Pressure Vessel Code which requires personnel performing examinations to be qualified by examination and certified in accordance with SNT-TC-1A. Based on a review of this Code Case the staff concludes that the proposed alternative provides an acceptable level of quality and safety. Therefore, the proposed alternative is authorized pursuant to 10 CFR 50.55a(a)(3)(i) for the second 10-year interval.

Our detailed evaluation and conclusions are documented in the enclosed safety evaluation. This completes our effort on these requests, and therefore, we are, closing out TAC No. MA5446. If you have any questions regarding this matter, please contact the Millstone Unit 3 Project Manager, Victor Nerses, at (301) 415-1484.

Sincerely,
/RA/
James W. Clifford, Chief, Section 2
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-423
Enclosure: Safety Evaluation
cc w/encl: See next page

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

FOR FIRST 10-YEAR INTERVAL INSERVICE INSPECTION REQUEST FOR RELIEF IR-28

AND SECOND 10-YEAR INTERVAL INSERVICE INSPECTION

REQUESTS FOR RELIEF IR-2-01 THROUGH IR-2-09

NORTHEAST NUCLEAR ENERGY COMPANY, ET AL.

MILLSTONE NUCLEAR POWER STATION, UNIT NO. 3

DOCKET NO. 50-423

1.0 INTRODUCTION

Inservice inspection (ISI) of the American Society of Mechanical Engineers (ASME) Code Class 1, 2, and 3 components is performed in accordance with Section XI of the ASME Boiler and Pressure Vessel (B&PV) Code and applicable addenda as required by 10 CFR 50.55a(g), except where specific written relief has been granted by the Commission pursuant to 10 CFR 50.55a(g)(6)(i). Title 10 of the Code of Federal Regulations (10 CFR) Section 50.55a(a)(3) states that alternatives to the requirements of paragraph (g) may be used, when authorized by the U.S. Nuclear Regulatory Commission (NRC), if (i) the proposed alternatives would provide an acceptable level of quality and safety or (ii) compliance with the specified requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

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 (ISI) 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 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. The Code of record for the Millstone Nuclear Power Station, Unit No. 3 first 10-year ISI interval is the 1983 Edition through 1983 Summer Addenda of the ASME Boiler and Pressure Vessel Code and the Code of record for the Millstone Nuclear Power Station, Unit No. 3 second 10-year ISI interval is the 1989 Edition of the ASME Boiler and Pressure Vessel Code.

Pursuant to 10 CFR 50.55a(g)(5), if the licensee determines that conformance with an examination requirement of Section XI of the ASME Code is not practical for its facility,

information will be submitted to the Commission in support of that determination and a request must be made for relief from the ASME Code requirement. After evaluation of the determination, pursuant to 10 CFR 50.55a(g)(6)(i), the Commission may grant relief and/or may impose alternative requirements that are determined to be authorized by law, will not endanger life, property, or the common defense and security, and are otherwise in the public interest, giving due consideration to the burden upon the licensee that could result if the requirements were imposed.

2.0 EVALUATION

The Materials and Chemical Engineering Branch, with technical assistance from Idaho National Engineering and Environmental Laboratory (INEEL), has reviewed the information concerning Inservice Inspection (ISI) program Requests for Relief IR-28 (1st Interval) and IR-2-01, IR-2-03 through IR-2-09 (2nd Interval) submitted for the first and second 10-year intervals for Millstone Nuclear Power Station, Unit No. 3 in Northeast Nuclear Energy Company's (the licensee) letters dated April 27, 1999, as supplemented, September 17, 1999, November 4, April 12, and June 21, 2000. The Mechanical Engineering Branch has reviewed Request for Relief IR-2-02 (2nd Interval) submitted for the second 10-year interval for Millstone Nuclear Power Station, Unit No. 3. Its evaluation is contained in this safety evaluation.

The NRC staff adopts the evaluations and recommendations for granting relief or authorizing alternatives contained in the Technical Letter Report (TLR) for IR-28, IR-2-01, IR-2-03 through IR-2-05, and IR-2-07 through IR-2-09, included in the Attachment prepared by INEEL.

For the Millstone Nuclear Power Station, Unit No. 3 relief is granted from the inspection requirements which have been determined to be impractical to perform. Alternatives are authorized where an alternative provides an acceptable level of quality and safety, or where compliance would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

The ISI Request for Relief IR-28 was granted for the closure of the first 10-year ISI interval, which concluded on April 22, 1996, for Millstone Nuclear Power Station, Unit No. 3. The ISI Requests for Relief IR-2-01 through IR-2-09, are granted or authorized for the second 10-year ISI interval, which began on April 23, 1996, for Millstone Nuclear Power Station, Unit No. 3.

During discussions with the licensee, the staff determined that Relief Request IR-2-06, "Utilization of Code Case N-566," contained in the April 27, 1999, submittal did not adequately address the staff's concerns with the use of this Code Case and, therefore, was being rejected by INEEL. As a result, by letter dated June 21, 2000, the licensee submitted a revised Relief Request IR-2-06. This request was for use of Code Case N-566-1, "Corrective Action for Leakage at Bolted Connections, Section XI, Division I." The NRC staff's evaluation of this revised request is included in this safety evaluation.

2.1 Request for Relief RR-IR-2-02, Snubber Examination and Testing

Code Requirement: Pursuant to 10 CFR 50.55a(b)(2), components (including supports) which are classified as ASME Code Class 1, 2, and 3 must meet the requirements set forth in the 1989 Edition of the ASME Boiler and Pressure Vessel Code. IWF-5300 calls for inservice examinations for snubbers to be performed in accordance with the first Addenda to ASME/ANSI IM-1987, Part 4, published in 1988 (OMa-1988), using the VT-3 examination method described in IWA-2213.

Licensee's Proposed Alternative: In lieu of the requirements of Paragraph IWA-5300, the licensee proposed utilizing the requirements as set forth in Technical Specification 4.7.10 as alternatives to the requirements of the ASME Boiler and Pressure Vessel Code specified in 10 CFR 50.55a(g)(4).

Evaluation: OMa-1988 imposes surveillance requirements for visual inspection of snubbers. A visual inspection is the observation of the condition of the installed snubbers to identify those that are damaged, degraded, or inoperable as caused by physical means, leakage, corrosion, or environmental exposure. Because the current schedule for snubber visual inspections, as specified by the Code, is based only on the number of inoperable snubbers found during the previous inspection, irrespective of the size of the snubber population, the visual inspection schedule is excessively restrictive. A significant amount of resources must be spent, and plant personnel subjected to unnecessary radiological exposure to comply with the visual examination requirements.

To alleviate this situation, the licensee developed an alternate schedule for visual inspections that maintains the same confidence level as the existing schedule and generally will allow performance of inspections and corrective actions during plant outages. This schedule is given in Table 4.7-2 of TS 4.7.10.b, which is based on the number of unacceptable snubbers found during the previous inspection in proportion to the sizes of the various snubber populations or categories. This alternative visual inspection schedule is similar to the recommendations on snubber visual inspection frequencies, as contained in Generic Letter 90-09, titled "Alternative Requirements for Snubber Visual Inspection Intervals and Corrective Actions." The alternative snubber program provides the necessary assurance for snubber operability and visual examination requirements to fulfill the ASME Section XI Code requirements without duplicating the inspections.

The licensee stated that while the schedule of examinations is to be determined from TS 4.7.10, the examinations are still to be performed using VT-3 visual examination certified personnel. The staff found the above alternative program, as provided in the TS, to be acceptable.

Based on the information provided by the licensee, the staff determined that the licensee has presented an adequate justification for relief from the requirements of ASME Code 1989 Edition, Section XI, IWF-5300 (which references OMa-4), with regard to visual examination of the Millstone 3 snubbers. The staff has determined that the proposed alternative use of the Millstone 3 TS for the snubber activity would provide an acceptable level of quality and safety. Therefore, pursuant to 10 CFR 50.55a(a)(3)(i), the licensee's request for relief for the second 10-year interval of the Millstone Nuclear Power Station, Unit No. 3 ISI program is authorized.

2.2 Request for Relief RR-IR-2-06, Paragraph IWA-5250(a)(2), Corrective Measures for Leakage from Bolted Connections

Code Requirement: If leakage occurs at a bolted connection, Paragraph IWA-5250(a)(2) requires that the bolting be removed, VT-3 visual examined for corrosion, and evaluated in accordance with IWA-3100.

Licensee's Proposed Alternative: In lieu of the requirements of Paragraph IWA-5250(a)(2), the licensee proposed performing an evaluation in accordance with Code Case N-566-1 "Corrective Action for Leakage Identified at Bolted Connections Sections, Division 1" dated February 15, 1999, in the event of leakage from a bolted connection. The licensee stated:

> The alternative rules set forth in Code Case N-566-1 will be used for the engineering evaluation of joint integrity. A VT-1 visual examination of the accessible bolting surfaces, with the bolting in-place under tension, shall be performed.

Licensee's Basis for Proposed Alternative (as stated):

> A number of problems have been identified with this Code requirement:
>
> - IWA-5250(a) directs that a VT-3 be performed of the removed bolt in accordance with IWA-3100. IWA-3100 does not contain acceptance criteria for VT-3 of bolting.
> - The Code does not require that the leakage be stopped; therefore, after pulling and examining the bolt, the leakage may continue.
> - Removing one bolt at a time, the leakage may become even worse than originally found,
> - The Code does not address integrity of the joint.
> - Bolts can be damaged when being removed.
> - The Code requires removing the bolting even if the leakage is minor, can be monitored, or if there is no corrosion concern. This can impact startup, cause hardship, and increase exposure without a commensurate increase in safety.
>
>   A Special Task Group within the ASME BPVC [Boiler and Pressure Vessel Code] Section XI Subcommittee has addressed through-wall and mechanical joint leakage. They have concluded that structural integrity, does not imply leak tightness. IWB-3142.4 allows acceptance of relevant conditions by analytical evaluation. It is felt that this can be applied to leakage from mechanical connections. Therefore, Code Case N-566-1 allowed for the engineering evaluation of the integrity of bolted joint connections.

> Since compliance with the existing Code requirement results in hardship or unusual difficulty without a compensating increase in the level of quality and safety, and the alternative in the Code Case provides a level of quality and safety equivalent to other components evaluated under IWB-3142.4, authorization to utilize Code Case N-566-1 is requested pursuant to 10 CFR 50.55a(a)(3)(i).

Evaluation: The Code requires that all bolts be removed from leaking bolted connections and that the bolts be VT-3 visual examined for corrosion and evaluated in accordance with IWA-3100. The Code requirements provide assurance that bolting corroded by system leakage will be detected and that corrective actions will be taken. However, the Code requirements are often unnecessarily conservative since corrosion is dependent on other factors beyond system leakage. Additionally, removal and examination of all bolts may not be necessary to assure continued integrity of the bolted connection.

In lieu of these requirements, the licensee has proposed to implement Code Case N-566-1 "Corrective Action for Leakage Identified at Bolted Connections Sections, Division 1" dated February 15, 1999, which requires in part that an engineering evaluation be performed to determine the need for additional examinations of the bolts considering the criteria listed as follows:

The requirements of Code Case N-566-1 below shall be met:

(a) The leakage shall be stopped, and the bolting and component material shall be evaluated for joint integrity as described in (c) below.

(b) If the leakage is not stopped, the joint shall be evaluated in accordance with IWB-3142.4 for joint integrity. This evaluation shall include the considerations listed in (c) below.

(c) The evaluation of (a) and (b) above is to determine the susceptibility of the bolting to corrosion and failure. This evaluation shall include the following:
   1. the number and service age of the bolts;
   2. bolt and component material;
   3. corrosiveness of process fluid;
   4. leakage location and system function;
   5. leakage history at the connection or other system components
   6. visual evidence of corrosion at the assembled connection

If the evaluation determines that examination is required, the licensee proposed that the bolt closest to the leak be removed and VT-1 examined. The bolt will be evaluated per IWA-3100 and requires that the evaluation of flaws are in accordance with IWB-3000, IWC-3000, and IWD-3000 for Class 1, 2, and 3 pressure retaining components, respectively. The staff determined that removal and VT-1 examination of the bolt closest to the leak is a reasonable alternative since degradation of this bolt is most likely, and would be representative of the worst case condition of the other bolts in the subject connection.

Based on the items included in the evaluation process, the staff has determined that the evaluation proposed by the licensee presents a sound engineering approach. In addition, if the initial evaluation indicates the need for a more detailed analysis, the bolt closest to the source

of leakage will be removed, VT-1 visually examined, and evaluated in accordance with IWA-3100(a). The VT-1 examination criteria are more stringent than the simple corrosion evaluation described in IWA-5250. Therefore, pursuant to 10 CFR 50.55a(a)(3)(i), the licensee's proposed alternative is authorized for the second interval at Millstone Nuclear Power Station, Unit No. 3 or until such time as the Code Case is referenced in a future revision of 10 CFR 50.55a. At that time, if the licensee intends to continue to implement this Code Case, the licensee is to follow all provisions in the Code Case with limitations or conditions specified in 10 CFR 50.55a, if any.

3.0 CONCLUSION

The Millstone Nuclear Power Station, Unit No. 3, requests for relief from the Code requirements have been reviewed by the NRC staff with the assistance of its contractor, INEEL. The TLR provides INEEL's evaluation of these relief requests with the exception of IR-2-02 which INEEL did not evaluate. The staff has reviewed the TLR and concurs with the evaluations and recommendations for granting or authorizing alternatives for IR-28, IR-2-01, IR-2-03 through IR-2-05, and IR-2-07 through IR-2-09.

The licensee's proposed alternative in Request for Relief IR-2-02 was reviewed by the NRC staff and is authorized for the second 10-year interval because the staff has determined that the licensee has presented adequate justification for relief from the requirements of ASME Code 1989 Edition, Section XI, IWF-5300 (which references OMa-4), with regard to visual examination of the Millstone Nuclear Power Station Unit No. 3 snubbers. In addition, the proposed alternative use of the Millstone Nuclear Power Station, Unit No. 3 TS for the snubber activity would provide an acceptable level of quality and safety. Therefore, pursuant to 10 CFR 50.55a(a)(3)(i), the licensee's proposed alternative is authorized for the second 10-year interval of the Millstone Nuclear Power Station, Unit No. 3 ISI program.

The licensee's proposed alternative in Request for Relief IR-2-06 was reviewed by the NRC staff and is authorized for the second interval at Millstone Nuclear Power Station, Unit 3 or until such time as the Code Case is referenced in a future revision of 10 CFR 50.55a. At that time, if the licensee intends to continue to implement this Code Case, the licensee is to follow all provisions in the Code Case with limitations or conditions specified in 10 CFR 50.55a, if any. Therefore, pursuant to 10 CFR 50.55a(a)(3)(i), the licensee's proposed alternative is authorized for the second 10-year interval of the Millstone Nuclear Power Station, Unit No. 3 ISI program.

The staff concludes that the relief requests, as evaluated by this Safety Evaluation provide reasonable assurance of structural integrity of the subject components in the licensee's requests for relief. The staff has determined that granting relief pursuant to 10 CFR 50.55a (g)(6)(i) and authorizing alternatives pursuant to 10 CFR 50.55a(a)(3)(i) or (a)(3)(ii) is authorized by law and will not endanger life or property, or the common defense and security and is otherwise in the public interest.

Principal Contributors: A. Lee
T. McLellan

Date: July 24, 2000

TECHNICAL LETTER REPORT
ON
THE FIRST 10-YEAR INTERVAL INSERVICE INSPECTION
REQUEST FOR RELIEF IR-28
AND
THE SECOND TEN-YEAR INTERVAL INSERVICE INSPECTION
REQUESTS FOR RELIEF IR-2-01 THROUGH IR-2-09
FOR
NORTHEAST NUCLEAR ENERGY COMPANY
MILLSTONE NUCLEAR POWER STATION, UNIT NO. 3
DOCKET NUMBER: 50-423

1. INTRODUCTION

By letter dated April 27, 1999, the licensee, Northeast Nuclear Energy Company, submitted requests for relief IR-2-01 through IR-2-09 for the Millstone Nuclear Power Station, Unit 3, second 10-year inservice inspection (ISI) interval. Also by letter dated September 17, 1999, the licensee submitted request for relief IR-28 for the Millstone Nuclear Power Station, Unit 3, first 10-year inservice inspection (ISI) interval. In response to an NRC Request for Additional Information (RAI), the licensee provided clarification for these requests in a letter dated April 12, 2000. The Idaho National Engineering and Environmental Laboratory (INEEL) staff's evaluation of the subject requests for relief is in the following section.

2. EVALUATION

The information provided by Northeast Nuclear Energy Company in support of the requests for relief from Code requirements has been evaluated and the bases for disposition are documented below. The Code of record for the Millstone Nuclear Power Station, Unit 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 of record for the Millstone Nuclear Power Station, Unit 3, second 10-year ISI interval, which began April 23, 1999, is the 1989 Edition of Section XI of the ASME Boiler and Pressure Vessel Code.

2.1 Request for Relief IR-28 (First 10-year Interval), Examination Category F-A, Item F1.40, Examination of Reactor Vessel Supports

Code Requirement: ASME Section XI, Code Case N-491, Table 2500-1, Examination Category F-A, Item F1.40, requires 100% VT-3 visual examination of all Class 1, 2, and 3 supports other than piping supports.

Licensee's Code Relief Request: The licensee requested relief from performing 100% of the Code-required VT-3 visual examinations for the reactor vessel supports.

Licensee's Basis for Relief Request (as stated):
"The Millstone Unit 3 Reactor Vessel has four supports which 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

ATTACHMENT

steel support structure that is welded to the neutron shield tank as shown in Figure 1[1]. The support is designed to function as a vertical restraint, loaded in compression and would not 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.

"During the performance of the examination during the last refuel outage (RF06) it was 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, 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 of the inaccessible portions of these supports for Millstone Unit 3, first ten year inspection interval (April 23, 1986 through October 23, 1999)."

<u>Licensee's Proposed Alternative Examination</u> (as stated):
"A limited visual VT-3 exam was performed satisfactory 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."

<u>Evaluation:</u> Code Case N-491 (implemented for the first interval) requires a VT-3 visual examination of 100% of the reactor vessel supports. However, due to access restrictions, the support design, and high local radiation levels, the licensee proposed to perform a limited visual examination.

The support assembly at each of the reactor vessel nozzles consists of a nozzle pad and steel plates positioned between a steel support structure that is welded to the neutron shield tank. Additionally, the majority of each support is encased in permanent insulation panels of the reactor vessel and vessel nozzles. These design features make the VT-3 visual examination of the reactor vessel supports impractical to perform to the extent required by the Code. To meet the Code requirements, the permanent insulation would have to be removed and redesigned to allow access for complete examination. Imposition of this requirement would create a considerable burden on the licensee.

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1 Figures, drawings and attachments supplied by the licensee are not included in this report.

The licensee has proposed to perform a VT-3 visual examination of the accessible portions of the subject supports to the maximum extent practical with the insulation in place. In addition, the insulation will be examined for any evidence of disturbance or degradation which may be attributed to abnormal support disturbance. These examinations will provide reasonable assurance of continued structural integrity of the supports.

Due to the design of the reactor vessel supports and permanent insulation, it is impractical to perform the VT-3 visual examination to the extent required by the Code. The licensee's alternative will provide reasonable assurance of structural integrity; therefore, it is recommended that relief be granted pursuant to 10 CFR 50.55a(g)(6)(i).

2.2 <u>Request for Relief IR-2-01, Use of Code Cases N-389-1, N-416-1, N-491-1, N-521, N-522 and N-524</u>

<u>Code Requirement:</u> Components (including supports) which are classified as ASME Code Class 1, Class 2, and Class 3 must meet various requirements set forth in the 1989 Edition of the ASME Boiler and Pressure Vessel Code.

<u>Licensee's Proposed Alternative:</u> In accordance with 10 CFR 50.55a(a)(3)(i), authorization is sought to use Code Cases N-389-1, N-416-1, N-491-1, N-521, N-522 and N-524 as alternatives to the requirements of the ASME Boiler and Pressure Vessel Code.

<u>Licensee's Basis for Proposed Alternative</u> (as stated):
"The alternative rules set forth in Code Cases N-389-1, N-416-1, N-491-1, N-521, N-522 and N-524 will be used, with all additional conditions set forth in Regulatory Guide 1.47, Revision 12."

<u>Evaluation:</u> The NRC staff recently reviewed Code Cases N-389-1, N-416-1, N-491-1, N-521, N-522 and N-524, and found the code cases acceptable for general use, with specific conditions as applicable, as evidenced by incorporation into Regulatory Guide 1.147, *Inservice Inspection Code Case Acceptability*, Revision 12, (May 1999). Therefore, Code Cases N-389-1, N-416-1, N-491-1, N-521, N-522 and N-524, with the applicable conditions specified in Regulatory Guide 1.47, should be approved for use during the second 10-year interval at Millstone Nuclear Power Station, Unit 3.

2.3 <u>Request for Relief IR-2-03, Examination Category B-F & B-J, Items B5.10 and B9.11, Pressure Retaining Dissimilar Metal Welds in Vessel Nozzles and Pressure Retaining Welds in Piping</u>

<u>Code Requirement</u>: Examination Category B-F & B-J, Items B5.10 and B9.11, require both a surface and volumetric examination, as defined by Table IWB-2500-1, for pressure retaining dissimilar metal welds in vessel nozzles and piping.

<u>Licensee's Proposed Alternative</u>: In accordance with 10 CFR 50.55a(a)(3)(ii), the licensee proposed to perform a full volumetric examination in lieu of the Code-required surface examination of nozzle to safe-end and safe-end to pipe butt welds greater than

4-inch NPS on the reactor pressure vessel. These welds are listed in the licensee's table below.

<u>Licensee's Basis for the Proposed Alternative</u> (as stated):
"The Millstone Unit 3 reactor vessel is a four-loop PWR. There are eight nozzle to safe-ends and eight safe-end to pipe butt welds for items B5.10 and B9.11, in the in-service inspection program. The purpose of this relief is to perform a full volumetric ultrasonic examination in lieu of a surface examination in order to reduce exposure. The reactor vessel nozzles have a permanent mirror type insulation installed which covers the OD surface of the welds. The exposure received from the removal and reinstallation of this insulation and performing the surface examination is estimated to be 2.5 man-rem per weld.

"The ultrasonic examination of these welds will be performed during the third inspection period from the ID surface using the remote immersion method. NNECO believes that any failure of the weld will be induced from the ID surface and that the ultrasonic examination from this surface shall be sufficient to detect any indications.

"NNECO believes that the proposed ultrasonic examination will prove to be adequate to detect flaws in the listed items. NNECO believes that the ultrasonic exam will assist in reducing the exposure rates, without losing the ability to detect flaws in the reactor vessel nozzle to safe-end and safe-end to pipe welds.

"A similar relief request (IR-9) was approved for the First Interval in NRC Letter A10880 dated March 3, 1993, contingent upon demonstration of the adequacy of the alternative. This demonstration was performed with the NRC in attendance in March 1995. IR-9 has been revised to reflect the actual state of the art technique used and was resubmitted for the First Interval in NNECO Letter B10880, dated December 21,1998.

"The welds listed in below receive a full volumetric examination from the ID surface as shown in the following table. This included 40º refracted longitudinal wave exam search units manipulated and scanned in four directions to obtain full volumetric coverage. This alternative volumetric examination was demonstrated with NRC in attendance during the first interval.

"NNECO will perform visual examination during system leakage tests as required by Section XI and Code Case N-498-1."

Zone Number	Weld Identification	Code Category	Material Type	Technique
12	302-121-C	B-F	29" SA508/CL2 SA182/F316	40º refracted longitudinal wave
12	301-121-C	B-F	27½" SA182/F316 SA508/CL2	40º refracted longitudinal wave

Zone Number	Weld Identification	Code Category	Material Type	Technique
13	302-121-D	B-F	29" SA508/CL2 SA182/F316	40º refracted longitudinal wave
13	301-121-D	B-F	27½" SA182/F316 SA508/CL2	40º refracted longitudinal wave
14	302-121-A	B-F	29" SA508/CL2 SA182/F316	40º refracted longitudinal wave
14	301-121-A	B-F	27½" SA182/F316 SA508/CL2	40º refracted longitudinal wave
15	302-121-B	B-F	29" SA508/CL2 SA182/F316	40º refracted longitudinal wave
15	301-121-B	B-F	27½" SA182/F316 SA508/CL2	40º refracted longitudinal wave

Evaluation: Millstone Unit 3 is a four-loop pressurized water reactor. There are eight nozzle to safe-ends and eight safe-end to pipe butt welds included in Code Items B5.10 and B9.11, required to be examined by the inservice inspection program. The Code requires both an OD surface and a volumetric examination, as defined by Table IWB-2500-1, for these pressure retaining dissimilar metal welds. The reactor vessel nozzles have a permanent mirror type insulation installed which covers the OD surface of the welds. The exposure received from the removal and reinstallation of this insulation and performing the surface examination is estimated to be approximately 2.5 man-rem per weld. This represents a significant hardship for the licensee.

As an alternative, the licensee has proposed to eliminate the surface examination, but expand the required volumetric examination to cover 100% of the wall thickness in the weld areas. This expanded volume would cover the entire cross-section of the weld, shown in Figure IWB-2500-8 of the 1989 Code Edition, and would be conducted from the inside diameter using an automated technique. This technique was approved for use during the first interval provided that the actual procedure and instrumentation were demonstrated to adequately detect OD-connected flaws. This demonstration was performed with the NRC in attendance in March 1995. Relief Request IR-9 was later revised to reflect the actual state-of-the-art technique used and was resubmitted for the first interval in NNECO Letter B10880, dated December 21,1998.

The proposed examination has been demonstrated to adequately detect OD generated flaws and provides reasonable assurance of the continued structural integrity of the components mentioned above. Due to the design of the reactor vessel nozzles, which have a permanent mirror type insulation installed covering the OD surface of the welds, to require the licensee to perform a surface examination would be a burden with no compensating increase in quality and safety. Therefore, it is recommended that the alternative be authorized pursuant to 10 CFR 50.55a(a)(3)(ii). The licensee should note that if ultrasonic techniques used for this alternative are modified as a result of ASME Appendix VIII Supplement(s) being developed for this application, and subsequently invoked by regulation, the impact of new performance demonstration parameters upon

the original demonstration should be evaluated. If necessary, further demonstration of the effectiveness of the examination to detect and characterize OD-connected flaws may be required.

2.4 Request for Relief IR-2-04, Examination Category B-H, Item B8.20, Integrally Welded Attachments

Code Requirement: Examination Category B-H, Item B8.20, requires 100% examination, as defined by Figures IWB-2500-13, -14, or -15, as applicable, for integral attachments to vessels in accordance with the percentage scheduling requirements of Table IWB-2412-1.

Licensee's Proposed Alternative: In accordance with 10 CFR 50.55a(a)(3)(ii), the licensee proposed an alternative to the scheduling requirements of Table IWB-2412-1. The licensee's proposed alternative is to perform the Code-required examinations of eight (of nine) Examination Category B-H integrally welded attachments during the second inspection period.

Licensee's Basis for Proposed Alternative (as stated):
"Millstone Unit 3 has nine integrally welded attachments for examination category B-H. Eight of these welded attachments provide component support for the pressurizer and are located near the top head. In order to meet the scheduling requirement of Table IWB-2412-1 for B-H category, the pressurizer would have to have its insulation removed three times in the ten-year interval, resulting in unnecessary personnel exposure.

"Millstone Unit 3 ISI program currently schedules most of the pressurizer volumetric surface and visual examinations during one refueling outage, in order to remove this insulation once. The high exposure levels in the pressurizer cubicle makes it prudent from an ALARA stand point to do this.

"A similar relief request (IR-18) was approved for the First Interval in NRC Letter A10880 dated March 3, 1993.

"Examinations of the eight integral attachments for the pressurizer support brackets were performed in accordance with Figure IWB-2500-15 during the second inspection period of the first interval. No indications were found at the time of the examinations. Examinations of the eight integral attachments will be performed during the second period of the second interval."

Evaluation: Paragraph IWB-2412-1 requires that examinations be nearly equally distributed among inspection periods. The licensee's proposed alternative is to perform the Code-required examinations of eight (of nine) Examination Category B-H integrally welded attachments during the second inspection period.

The subject integrally welded attachments are all located near the top head of the pressurizer. In order to examine all welds in accordance with the Code, the insulation would have to be removed three separate times to allow one-third of the examinations to

be performed each inspection period. The intent of the Code is to distribute examinations among components to ensure that generic failure mechanisms are detected. Considering the high radiation levels in the pressurizer cubicle, removal of the insulation on the pressurizer three separate times during an interval would cause excessive personnel exposure. This does not represent best ALARA principles. Further, there is only one other Category B-H integral attachment weld on a different component at Millstone Unit 3, so distribution of examinations among other Class 1 vessels over the 10-year interval is not feasible.

Examinations of the subject eight integral attachments for the pressurizer support brackets were performed in accordance with Figure IWB-2500-15 during the second inspection period of the first interval. No indications were found at the time of these examinations. Examinations of the eight integral attachments will be performed during the second period of the second interval, so no more than 10 Code years will elapse between examinations. Therefore, reasonable assurance of the continued structural integrity is maintained. Requiring licensee personnel to enter high radiation fields for the purpose of insulation removal and examination on three separate occasions, is a hardship with no compensating increase in quality or safety. Therefore, it is recommended that the alternative be authorized pursuant to 10 CFR 50.55a(a)(3)(ii).

2.5 <u>Request for Relief IR-2-05, Examination Category C-B, Item C2.22, Pressure Retaining Nozzle Welds in Vessels</u>

<u>Code Requirement</u>: Examination Category C-B, Item C2.22, requires 100% volumetric examination, as defined by Figure IWC-2500-4 (a) or (b), for the inner radius of pressure retaining nozzles in vessels.

<u>Licensee's Code Relief Request</u>: In accordance with 10 CFR 50.55a(g)(5)(iii), the licensee requested relief from performing the volumetric examination of the inner radius of the Main Steam Nozzles.

<u>Licensee's Basis for Relief Request</u> (as stated):
"NNECO believes that, in evaluating the referenced ASME Code, the concern is that the inside radius of the Main Steam Nozzle is considered susceptible to flaw initiation and growth due to high thermal and mechanical stresses associated with the vessel and connected piping systems. In the case of the Millstone Unit 3 steam generator nozzles, the nozzle is a one-piece forging containing a set of seven holes bored parallel to the nozzle centerline. NNECO believes that this nozzle design does not fall under the criteria of a "radiused nozzle" as defined in Figure IWC-2500-4. Since the ligaments between the holes distribute the loads throughout the nozzle forging, the primary stress resulting from the pressure is significantly lower than those for a typical nozzle since the plate ligaments provide both reinforcement and additional insulation by directing the flow away from the nozzle inner radius. Furthermore, the local stresses resulting from the thermal gradients in the inner radius are expected to be low, since the nozzle is only exposed to saturated steam resulting in a low transfer between the nozzle and the coolant.

“The design of the nozzle precludes visual examination of the nozzle inside radius even if access to this area were possible. Access to the steam generator main steam nozzle from inside the steam generator is restricted by the upper deck plate and moisture separators; therefore, visual examinations from inside the steam generator are impractical.

“The unique design of the steam generator main steam nozzle results in low stresses when compared to nozzles with typical inner radius configurations. Stresses in the nozzle inner radius region are less than 68 percent of the ASME Code allowable for each design condition. This was determined by review of a proprietary Westinghouse Steam Generator Stress Report. In addition, the stresses are considerably lower than those of other nozzles, such as steam generator feedwater nozzles.

“A similar relief request (IR-19) was approved for the First Interval in NRC Letter A12289, dated May 4, 1995.”

Licensee’s Proposed Alternative Examination (as stated):
“NNECO will perform visual examinations during the system leakage test as required by Section XI and Code Case N-498-1.”

Evaluation: The Code requires 100% volumetric examination of Class 2 pressure vessel nozzle inside radius sections. In the case of Millstone Unit 3 steam generator nozzles, the nozzle is designed as a solid one-piece forging containing a set of seven holes bored parallel to the nozzle centerline. The geometry of this nozzle design, with the bored flow restrictor holes, does not result in an actual inner radius, and therefore, no meaningful examination can be performed. Access to the main steam nozzles from inside the steam generators is restricted by the upper deck plate and moisture separators; therefore, visual examinations from inside the steam generators are impractical.

It is also noted that the unique design of the steam generator main steam outlet nozzles results in low stresses when compared to nozzles with typical inner radius configurations. The licensee stated that stresses in the nozzle inner radius region are less than 68 percent of the ASME Code allowable for each design condition. This was determined by a review of the Westinghouse Steam Generator Stress Report (a proprietary report). In addition, it was stated that the stresses are considerably lower than those of other nozzles, such as the steam generator feedwater nozzle.

Based on review of the information provided, it has been concluded that the Code-required volumetric examination of the steam generator main steam outlet nozzle inner radius sections is impractical at Millstone Nuclear Power Station, Unit 3. The licensee’s proposed VT-2 visual examination, in combination with the conservative nozzle design and other higher stressed nozzle inner radii being examined on other components, should provide reasonable assurance of continued structural integrity. This design does not entail a radiused nozzle as described in Figure IWC-2500-4, but instead has several “corners”, corresponding to each bored hole. As a result, the design of the nozzle is not applicable to the Code requirement and compliance with the Code requirement is not

practical; therefore, this relief request should be granted pursuant to 10 CFR 50.55a(g)(6)(i).

2.6 Request for Relief IR-2-06, Implement Requirements of Code Case N-566, *Corrective Action for Leakage Identified at Bolted Connections*

Code Requirement: Components (including supports) which are classified as ASME Code Class 1, Class 2, or Class 3 must meet the requirements set forth in the 1989 Edition of the ASME Boiler and Pressure Vessel Code. IWA-5250(a)(2) requires all bolting to be removed for examination and evaluation when leakage occurs at a bolted connection.

Licensee's Proposed Alternative: Authorization is sought to utilize Code Case N-566, *Corrective Action for Leakage Identified at Bolted Connections,* as an alternative to the requirements of the ASME Boiler and Pressure Vessel Code.

Licensee's Basis for Proposed Alternative (as stated):
"A number of problems have been identified with this Code requirement:

- IWA5250(a) directs that a VT-3 be performed of the removed bolt in accordance with IWA-3100. IWA-3100 does not contain acceptance criteria for VT-3 of bolting.
- The Code does not require that the leakage be stopped; therefore, after pulling and examining the bolt, the leakage may continue.
- Removing one bolt at a time, the leakage may become even worse than originally found.
- The Code does not address integrity of the joint.
- Bolts can be damaged when being removed.
- The Code requires removing the bolting even if the leakage is minor, can be monitored, or if there is no corrosion concern. This can impact startup, cause hardship, and increase exposure without a commensurate increase in safety.

"A Special Task Group within the ASME BPVC Section XI Subcommittee has addressed through-wall and mechanical joint leakage. They have concluded that structural integrity does not imply leak tightness. IWB-3142.4 allows acceptance of relevant conditions by analytical evaluation. It is felt that this can be applied to leakage from mechanical connections.

"Since compliance with the existing Code requirement results in hardship or unusual difficulty without a compensating increase in the level of quality and safety, and the alternative in the Code Case provides a level of quality and safety equivalent to other components evaluated under IWB-3142.4, authorization to utilize Code Case N-566 is requested pursuant to 10CFR50.55a(a)(3)(iii).

"The alternative rules set forth in Code Case N-566 will be used."

Evaluation: In accordance with IWA-5250(a)(2) of the 1989 Edition of ASME XI, if leakage occurs at a bolted connection, the bolting shall be removed, VT-3 visually examined for corrosion, and evaluated in accordance with IWA-3100. In lieu of this requirement, the licensee's proposed alternative is to implement the requirements of Code Case N-566, *Corrective Action for Leakage Identified at Bolted Connections.*

Code Case N-566 requires that leakage be stopped, and the bolting and component material be reviewed for joint integrity. However, the Code Case does not define specific and detailed requirements to be included in the "review" to determine joint integrity.

Code Case N-566 also requires that, if the leakage is not stopped, the joint integrity be evaluated in accordance with IWB-3142.4, *Acceptance by Analytical Evaluation*. This type of Code evaluation should include consideration of the number and condition of bolts, leaking medium, bolt and component material, system function, and long-term leakage monitoring. However, an IWB-3142.4 evaluation presupposes that the bolting has been examined, and all unacceptable relevant conditions accounted for in the analytical evaluation. Since the bolting is not being removed, the Code Case does not provide a comprehensive evaluation process for ensuring joint integrity. Consequently, the INEEL staff does not believe that Code Case N-566, as written, provides an acceptable level of quality and safety. Therefore, it is recommended that the licensee's proposed alternative not be authorized.

2.7 Request for Relief IR-2-07, Proposed Alternative to use Code Case N-535, *Alternative Requirements for Inservice Inspection Intervals, Section XI, Division 1*

Code Requirement: Components (including supports) which are classified as ASME Code Class 1, Class 2, or Class 3 must meet the requirements set forth in the 1989 Edition of the ASME Boiler and Pressure Vessel Code. Paragraph IWA-2430 defines the Inspection Intervals.

Licensee's Proposed Alternative: In accordance with 10 CFR 50.55a(a)(3)(i), the licensee proposed that the alternative rules set forth in Code Case N-535 be used.

Licensee's Basis for Proposed Alternative (as stated):
"While IWA-2430(d) allows extension of an interval, such an extension may only be needed for certain components. Other components scheduled for the first period of the successive interval may need to be inspected during the same outage. This Code Case allows the extension of an interval without changing the start or end dates of successive intervals. While it is allowed to have two intervals open concurrently, examinations performed as part of the extended interval are prohibited from being credited in the successive interval."

Evaluation: Paragraph IWA-2432 requires that successive inspection intervals be comprised of 10 years following the previous interval except as modified by Paragraph IWA-2430(d). Paragraph IWA-2430(d) allows an interval to be extended or reduced by as much as one year to coincide with an outage, thus changing the length of an interval. The licensee proposed to apply the requirements of Code Case N-535 for the

scheduling of intervals and examinations of Code Class 1, 2, and 3 piping and components. This is in lieu of the existing Code requirements of Paragraph IWA-2432, as modified by Paragraph IWA-2430(d).

Code Case N-535 consists of the following requirements:

1. Allows a one-year extension of an inspection interval. Also, a successive interval may start prior to the end of the previous interval that was extended;
2. Prohibits examinations performed as part of the extended interval from being credited in the successive interval;
3. Allows an inspection period to be extended or reduced to coincide with an outage; and
4. Requires examination records to identify in what interval the examination was performed.

Part (1) above of N-535 is the only change from current ASME Section XI philosophy. The one-year extension is independent of the plant operating cycle and two intervals can be open concurrently during that year. Although slightly different from current Code requirements, this code case does not change the number of examinations, acceptance criteria, or any other Code requirement, with the possible exception of distribution of examinations. However, this change would be slight and, therefore, Code Case N-535 provides an acceptable level of quality and safety.

For these reasons, it is recommended that the proposed alternative be authorized, pursuant to 10 CFR 50.55a(a)(3)(i). Use of Code Case N-535 should be authorized until such time as the code case is published in a future revision of Regulatory Guide 1.147. At that time, if the licensee intends to continue to implement this code case, the licensee is to follow all provisions in Code Case N-535 with limitations issued in Regulatory Guide 1.147, if any.

2.8 Request for Relief IR-2-08, Utilization of Code Case N-533, *Alternative Requirements for VT-2 Visual Examination of Class 1 Insulated Pressure-Retaining Bolted Connections*

Code Requirement: Paragraph IWA-5242(a) requires removal of insulation from bolted connections for VT-2 visual examination when the systems are borated for the purpose of controlling reactivity.

Licensee's Proposed Alternative: Pursuant to 10CFR50.55a(3)(ii), authorization is sought to use Code Case N-533 as an alternative to the requirements of the Code.

Licensee's Basis for Proposed Alternative (as stated):
"Code Case N-533 provides an equivalent alternative to the requirements of IWA-5242(a). The purpose of IWA-5242(a) is to facilitate the VT-2 examination for borated water leakage that could result in corrosion of bolting materials. Code Case N-533 permits the removal of insulation from the subject bolted connections during the outage without pressurization of the bolted connection. Borated water leakage, either current or previous to the examination would be

evident by the accumulation of boron crystals, which are required to be evaluated in accordance with IWA-5250. The Class 1 system pressure test and VT-2 examination are then performed near the end of the refueling outage without the removal of insulation.

"The requirement of IWA-5242(a) places an undue burden and hardship on the facility during the performance of Class 1 system leakage tests. The Class 1 system leakage test is typically performed at the completion of each refueling outage with the unit in hot standby and the reactor coolant system at normal operating temperature and pressure. The Class 1 system leakage test is one of the last activities performed as a prerequisite to restart of the unit. The requirements of IWA-5242(a) to perform VT–2 examination of system borated for the purpose of controlling reactivity with the insulation removed from the bolted connections involves personnel access to the unit in hot standby mode for the purpose of reinstallation of thermal insulation and removal of scaffolding. These activities jeopardize the safety of personnel during the performance of the VT-2 examination, reinstallation of the insulation, and removal of the scaffolding due to heat stress and the potential for burns from contact with the un-insulated hot components. Insulation replacement activities also delay startup and require the unit to be in hot standby for an extended duration."

<u>Evaluation</u>: Paragraph IWA-5242(a) requires the removal of all insulation from pressure-retaining bolted connections in systems borated for the purpose of controlling reactivity when performing VT-2 visual examinations during system pressure tests. The licensee has proposed to implement the requirements of Code Case N-533, *Alternative Requirements for VT-2 Visual Examination of Class 1 Insulated Pressure-Retaining Bolted Connections*, for Class 1 bolted connections in borated systems. This code case allows the VT-2 visual examination to be performed in conjunction with startup following a 4-hour hold time at operating pressure with the insulation in place. A VT-2 visual examination is then subsequently performed each refueling outage during cold shutdown with the insulation removed. The system need not be pressurized to perform the subsequent examination. Requiring the licensee to remove insulation during the Class 1 system pressure test would create a safety hazard due to elevated temperatures, and would also result in excess radiation exposure to plant personnel. Therefore, the requirements of IWA-5242(a) would create an undue burden on the licensee.

For Class 1 systems, the licensee's proposed alternative provides a prudent approach for ensuring the leakage integrity of systems borated for the purpose of controlling reactivity. First, the 4-hour hold time allows any significant leakage to penetrate the insulation. Second, by removing the insulation each refueling outage, the licensee will be able to detect minor leakage indicated by the presence of boron crystals or residue. This two-phase approach provides reasonable assurance of the continued leakage integrity of Class 1 bolted connections in borated systems.

Based on the evaluation above, it is concluded that compliance with the Code requirements for Class 1 systems would result in a burden without a compensating increase in the level of quality and safety. Furthermore, the licensee's proposed

alternative provides reasonable assurance of continued leakage integrity for Class 1 bolted connections. Therefore, pursuant to 10 CFR 50.55a(a)(3)(ii), it is recommended that the licensee's proposed alternative be authorized for Class 1 systems. Use of Code Case N-533 should be authorized until such time as the code case is published in a future revision of Regulatory Guide 1.147. At that time, if the licensee intends to continue to implement this code case, the licensee is to follow all provisions in Code Case N-533 with limitations issued in Regulatory Guide 1.147, if any.

2.9 Request for Relief IR-2-09, Use of Code Case N-546, *Alternative Requirements for Qualification of VT-2 Examination Personnel, Section XI, Division 1*

Code Requirement: ASME Code, Section XI, IWA-2313 requires personnel performing examinations to be qualified by examination and certified in accordance with SNT-TC-1A. Level I and II personnel shall be re-certified by qualification examinations every three years. Level III personnel shall be re-certified by qualification examination every five years.

Licensee's Proposed Alternative: In accordance with 10 CFR 50.55a(a)(3)(i), the licensee proposed to use Code Case N-546 as an alternative to the requirements of the ASME Boiler and Pressure Vessel Code.

Licensee's Basis for Proposed Alternative (as stated):
"Code Case N-546 provides an alternative to the requirements of IWA-2300 for the qualification of VT-2 visual examination personnel, which maintains an equivalent level of quality and safety. The purpose of IWA-2300 is to provide requirements to ensure personnel performing NDE activities are appropriately qualified and certified. The following parameters are typically utilized for personnel qualification and certification:

1. Demonstrated experience and knowledge and understanding of the method of examination, VT-2 in this case.
2. Visual acuity

"Code Case N-546 addresses these parameters:

(a) At least 40 hours plant walkdown experience, such as that gained by licensed and nonlicensed operators, local leak rate personnel, systems engineers, and inspection and nondestructive examination personnel.
(b) At least four hours of training on Section XI requirements and plant specific procedures for VT-2 visual examination.
(c) Vision test requirements of IWA-2321, 1995 Edition."

Evaluation: The Code requires that VT-2 visual examination personnel be qualified to levels of competency comparable to those identified in SNT-TC-1A. The Code also requires that these examination personnel be qualified for near and far distance vision acuity. In lieu of the Code requirements, the licensee proposed to implement Code Case N-546 for personnel performing VT-2 visual examinations; this Code Case includes the following requirements:

1. At least 40 hours plant walk-down experience, such as that gained by licensed and non-licensed operators, local leak rate personnel, system engineers, and inspection and nondestructive examination personnel.

2. At least four hours of training on Section XI requirements and plant-specific procedures for VT-2 visual examination.

3. Vision test requirements of IWA-2321, 1995 Edition.

The qualification requirements in Code Case N-546 are not significantly different from those of the Code for VT-2 visual examiner certification. Licensed and non-licensed operators, local leak rate personnel, system engineers, and inspection and nondestructive examination personnel typically have a sound working knowledge of plant components and piping layouts. This knowledge makes them acceptable candidates for performing VT-2 visual examinations.

In addition to meeting the requirements contained in Code Case N-546, the licensee has committed to use procedural guidelines for consistent, quality VT-2 visual examinations, and to verify and maintain records of the qualification of persons selected to perform VT-2 visual examinations. Based on a review of Code Case N-546 and the additional commitments made by the licensee, the proposed alternative to the Code requirements will provide an acceptable level of quality and safety. Therefore, it is recommended that the licensee's request to implement Code Case N-546 with the additional commitments be authorized pursuant to 10 CFR 50.55a(a)(3)(i). Use of Code Case N-546 should be authorized until such time as the code case is published in a future revision of Regulatory Guide 1.147. At that time, if the licensee intends to continue to implement this code case, the licensee is to follow all provisions in Code Case N-546 with limitations issued in Regulatory Guide 1.147, if any.

3.0 CONCLUSION

The INEEL staff evaluated the licensee’s submittal and concluded that certain inservice examinations cannot be performed to the extent required by the Code at the Millstone Nuclear Power Station, Unit No. 3. For Requests for Relief Nos. IR-28 and IR-2-05, it is concluded that the Code coverage requirements are impractical. Therefore, it is recommended that relief be granted pursuant to 10 CFR 50.55a(g)(6)(i).

For Requests for Relief Nos. IR-2-07 and IR-2-09, the INEEL concludes that the licensee’s proposed alternative will provide an acceptable level of quality and safety, and should be authorized pursuant to 10 CFR 50.55a(a)(3)(i) for the second ISI interval.

For Requests for Relief Nos. IR-2-03, IR-2-04, and IR-2-08, the licensee has demonstrated that the Code examination coverage requirements would result in a burden without a compensating increase in the level of quality and safety; therefore, it is recommended that the licensee’s proposed alternative be authorized pursuant to 10 CFR 50.55a(a)(3)(ii) for the second ISI interval.

For Request for Relief IR-2-06, the INEEL staff does not believe that Code Case N-566, as written, provides an acceptable level of quality and safety. Therefore, it is recommended that the licensee's proposed alternative not be authorized.

For Request for relief IR-2-01, the licensee's alternative is to adopt several code cases published in Regulatory Guide (RG) 1.147, Revision12. The licensee has committed to perform the requirements of these code cases in their entirety, and to meet any additional conditions, as applicable, specified in RG 1.147, Rev.12. The NRC has found these code cases acceptable for general use in RG 1.147, Rev.12; it is recommended that use of these code cases be approved.

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