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

April 11, 1997

50-245

The Honorable Joseph I. Lieberman
United States Senate
Washington, DC 20510

Dear Senator Lieberman:

I am responding to your letter of February 26, 1997, in which you forwarded letters dated October 25, 1996, and December 18, 1996, from your constituent, Mr. Joseph M. Stankoski, of Southington, Connecticut. You also forwarded a copy of a letter the staff of the U.S. Nuclear Regulatory Commission wrote to Mr. Stankoski on November 26, 1996, on which Mr. Stankoski had written comments.

In your letter, you state that Mr. Stankoski is concerned about any regulatory relief that may be granted to Northeast Utilities, specifically with regard to nondestructive testing of the critical welds in the piping between the Millstone Unit 3 reactor vessel and the reactor coolant pumps. Your constituent contends that these welds can be examined by x-ray but that Northeast Utilities does not support this approach because it would have to drain the water in the pipes — a process that would be time-consuming and costly for the utility. Your constituent further states that the Millstone Unit 1 reactor pressure vessel welds that are not accessible for inspection could be monitored using acoustic emission techniques.

One of the concerns raised in Mr. Stankoski's letter of December 18, 1996, involved the use of centrifugally cast stainless steel (CCSS) piping in the reactor coolant system and the fact that CCSS does not readily lend itself to examination by ultrasonic testing (UT). Westinghouse-designed nuclear steam supply systems use CCSS for the primary loop piping for the reactor coolant system. CCSS was chosen at Millstone Unit 3, in part, because of its metallurgical properties in this environment. CCSS is a highly resistant material to corrosion and corrosion-related forms of cracking. The piping design requirements as well as the CCSS material used in the design of the primary loop piping result in high resistance to stress corrosion cracking and to fatigue crack propagation. Operating experience with primary coolant piping fabricated from this type of material has been very good. In fact, as discussed in our November 26, 1996, letter to Mr. Stankoski, the American Society of Mechanical Engineers is considering supplementing its requirements on cast material through issuance of a Code Case. The Code Case, if technically acceptable to the NRC staff, could be endorsed by the NRC and listed in Regulatory Guide 1.147, "Inservice Inspection Code Case Acceptability, ASME Section XI, Division I," through a formal regulatory process, including the opportunity for public comments. This is not to be confused with a relief request, as stated by Mr. Stankoski. Relief requests are further discussed on Page 3 herein.

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Section 50.55a(g) of Title 10 of the Code of Federal Regulations (10 CFR 50.55a(g)) requires nuclear power facility piping and components to meet the applicable requirements of Section XI of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (hereafter referred to as the ASME

Code). The ASME Code requires inservice inspection (ISI) of ASME Class 1, 2, and 3 components. The preferred method for ISI Class 1 and 2 component examination is UT. UT is preferred, in part, because the primary loops do not have to be emptied of water to perform the examination. However, CCSS introduces a different obstacle for UT because the grain size of this steel is large. The large grains cause the ultrasonic sound to attenuate and/or divert from the intended angle of inspection. This phenomenon makes the use of UT difficult in this application; however, UT can still provide useful information.

With regard to Millstone Unit 3, in Inspection Report 50-423/85-22, dated August 13, 1985, the NRC staff reviewed the licensee's UT method for examining CCSS and stated that it believed a significant indication (flaw) could have been detected using the licensee's method. However, exact sizing and location of indications still would remain a problem when performing UT on this type of material, and, if such a situation arose, the licensee would have to account for the uncertainty in assessing the integrity of the piping. It should be noted that no such indications have been detected. Further, Northeast Utilities has not requested relief from ASME Code examination requirements for CCSS in the Millstone Unit 3 reactor coolant system. Your constituent's contentions that these welds could be examined by x-ray is correct. However, this method has its drawbacks. Radiography (x-ray) requires the licensee to drain the reactor coolant loops, which raises the possibility of introducing oxygen into the primary system which can increase corrosion potential and introduces risks associated with draining and refilling the system. Radiography would be difficult in this application since the reactor coolant piping is already radioactive, which creates personnel exposure concerns, and a large x-ray-producing source would be needed. Further, x-rays are most sensitive to three-dimensional type defects and are not very effective for detecting crack-like defects. X-ray examination would only detect larger flaws, which would most likely be found using UT. Therefore, although radiography could be used, the possible increase in detection capability would be offset by the negative aspects listed above.

Furthermore, a "leak-before-break" analysis was provided by the licensee which the staff reviewed and approved in 1985. This analysis concluded that leakage through a postulated through-wall degradation of the CCSS primary loop piping would be detected by leak detection systems, and the plant shut down, before the through-wall degradation could lead to complete piping failure even under the most severe piping loads that could occur during the life of the plant. It should be noted that the conclusions of this analysis did not rely on performing inspections of the piping, but these inspections are continuing to provide additional assurance that integrity of the piping is maintained.

In Mr. Stankoski's letter of December 16, 1996, he raised two other issues I would like to address. First, Mr. Stankoski appears to be equating notices of enforcement discretion, or NOEDs, with relief requests from ASME Code requirements. In Mr. Stankoski's letter, he quotes your letter to him which states that you were pleased that the NRC has sharply curtailed its policy relating to granting temporary exemptions to facilities from safety and other

regulations. I believe you were discussing NOEDs. Mr. Stankoski, however, provides examples of relief requests from ASME Code requirements that the NRC grants.

NOEDs may be used by the NRC staff when circumstances arise in which licensee compliance with a Technical Specification Limiting Condition for Operation or with other license conditions would involve (1) unnecessary plant transient or performance of testing, inspection, or system realignment that is inappropriate with the specific plant conditions or (2) unnecessary delays in plant startup without a corresponding health and safety benefit. You are correct in stating that the NRC will only grant NOEDs in those very limited circumstances in which safety is not an issue.

Consistent with 10 CFR 50.55a(a)(3), alternatives to ASME Code requirements may be used by nuclear licensees when authorized by the Director of the Office of Nuclear Reactor Regulation if the proposed alternatives to the requirements are such that they are shown to provide an acceptable level of quality and safety in lieu of the ASME Code requirements, or if compliance with the ASME Code requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety. These are commonly referred to as relief requests and continue to be authorized by the NRC when the licensee can justify that the above regulatory criteria are met. The NRC staff prepares safety evaluations supporting such relief requests. These safety evaluations are available in the NRC's public document room.

Second, concerning the relief request for the Millstone Unit 1 reactor vessel welds, in Inspection Report 50-245/96-05, dated August 15, 1996, the staff stated that in conformance with the guidance offered by the NRC in an information notice, Northeast Utilities would be required to seek relief from the requirements of ASME Code, Section XI, 1989 Edition. In Information Notice 96-32, "Implementation of 10 CFR 50.55a(g)(6)(ii)(A), Augmented Examination of Reactor Vessel," the staff alerted licensees to the new requirements for an augmented examination of reactor vessels. The rule requires licensees to implement, before the time required by normal updating of the ISI program, provisions in the 1989 Edition of the ASME Code, Section XI, to examine "essentially 100%" of the length of all reactor vessel shell welds. "Essentially 100%" examination is defined in paragraph (A)(2) as more than 90 percent of the examination volume of each weld. In its letter to Mr. Stankoski dated November 26, 1996, the staff erroneously stated that the licensee had submitted a request for relief from the ASME Code requirements on accessibility of the reactor pressure vessel welds. In conversations with NRC, the licensee told the staff of plans to reexamine these welds before the end of the current 10-year ISI cycle with new procedures and equipment. If the licensee is still unable to examine at least 90 percent of the length of all reactor vessel shell welds, it would be required to justify a reduction in scope to the NRC. The staff notes that, in some cases, access to welds is physically impossible. Paragraph (A)(5) states that licensees making a determination that they are unable to completely satisfy the requirements of 10 CFR 50.55a(g)(6)(ii)(A) shall submit information to the Commission to support the determination and shall propose an alternative to the examination

requirements, which would provide an acceptable level of quality and safety. The NRC staff would then review the licensee's request using the regulatory criteria described above.

In Mr. Stankoski's letter of December 18, 1996, he stated that the Millstone Unit 1 reactor pressure vessel welds that were not accessible could be monitored by acoustic emission techniques. The use of acoustic emission techniques would require the monitoring of sensors placed along the reactor pressure vessel welds to detect the sounds made by crack propagation. The ASME Code endorses using acoustic emission techniques in conjunction with other nondestructive testing techniques endorsed by the ASME Code; however, use of acoustic emission techniques as a stand-alone inspection method for ensuring the integrity of these welds does not satisfy ASME Code requirements. As indicated above, the licensee is attempting to satisfy the ASME Code inspection guidelines using new procedures and equipment. Therefore, it is not clear that any compensatory measures such as acoustic monitoring will be required.

I trust this information will help you answer Mr. Stankoski's concerns.

Sincerely,

L. Joseph Callan
Executive Director
for Operations

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Original Signed by
L. J. Callan

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Executive Director
for Operations

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requirements for an augmented examination of reactor vessels. The rule requires licensees to implement, before the time required by normal updating of the ISI program, provisions in the 1989 Edition of the ASME Code, Section XI, to examine "essentially 100%" of the length of all reactor vessel shell welds. "Essentially 100%" examination is defined in paragraph (A)(2) as more than 90 percent of the examination volume of each weld. In the letter to Mr. Stankoski dated November 26, 1996, the staff erroneously stated that the licensee had submitted a request for relief from the ASME Code requirements on accessibility of the reactor pressure vessel welds. In conversations with NRC, the licensee told us of plans to reexamine these welds before the end of the current 10-year ISI cycle with new procedures and equipment. If the licensee is still unable to examine at least 90 percent of the length of all reactor vessel shell welds, it would be required to seek relief. The staff notes that, in some cases, (1) access to welds is physically impossible and (2) to gain access to certain welds, a significant increase in inspection personnel exposure could be necessary. Paragraph (A)(5) states that licensees making a determination that they are unable to completely satisfy the requirements of 10 CFR 50.55a(g)(6)(ii)(A) shall submit information to the Commission to support the determination and shall propose an alternative to the examination requirements, which would provide an acceptable level of quality and safety. The NRC staff would then review the licensee's request using the regulatory criteria described above.

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FOR SIGNATURE OF : ** GRN ** CRC NO:

Executive Director

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