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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING RELIEF FROM REQUIREMENTS OF SECTION XI ASME CODE

NORTHEAST NUCLEAR ENERGY COMPANY

MILLSTONE NUCLEAR POWER STATION UNIT NO. 1

DOCKET NO. 50-245

1.0 INTRODUCTION

By letter dated September 18, 1980, the Northeast Nuclear Energy Company (the licensee) submitted the Millstone Unit 1 Inservice Inspection and Testing Program for the second ten-year inspection interval. Based on the date of commercial operation, the second ten-year inspection interval will begin on December 28, 1980, and end on December 28, 1990. The Inservice Inspection and Testing Program is based on the current inspection program, which was approved in the Safety Evaluation Report issued September 19, 1979, in support of Amendment No. 64 to Provisional Operating License No. DPR-21, but upgraded to the requirement of Section XI of the ASME Boiler and Pressure Vessel Code, 1977 Edition, including Summer 1978 Addendum.

Paragraph (g)5(iii) of 10 CFR Part 50.55a requires that the licensee notify the Nuclear Regulatory Commission when it is determined that conformance with certain Section XI ASME Code requirements are impractical and that information be submitted in support of this determination. After review and evaluation, relief may be granted and alternative requirements may be imposed pursuant to paragraph (g)6(i) of 10 CFR Part 50.55a by the Nuclear Regulatory Commission.

We have completed our review and evaluation of the request for relief from certain Section XI ASME Code inservice inspection requirements for the Millstone Unit 1 facility. The specific requests for relief and our evaluation follow.

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2.0 EVALUATION OF REQUESTS FOR RELIEF FROM SECTION XI CODE REQUIREMENTS

2.1 CLASS 1 COMPONENTS

A. Reactor Vessel

1. Relief from the volumetric examination of one longitudinal and one circumferential shell weld in the beltline region of the vessel.

(Item No. B 1.10, Examination Category B-A, Table IWB-2500-1)

Code Requirement

The Code requires that one circumferential and one longitudinal beltline region weld be volumetrically examined to essentially 100% of the length during the second inspection interval. The welds should be located at a structural discontinuity, if any. The examinations may be performed at or near the end of the inspection interval.

Basis for the Requested Relief

The reactor vessel is insulated with permanent reflective insulation and surrounded by the concrete biological shield. The annular space between the inside diameter of the biological shield and the outside diameter of the insulation is a nominal 6-1/2 inches. Thus, access for removal of the insulation panels is extremely limited, and this inaccessibility precludes direct examination of the beltline region welds from the outside surface. The interior surface of the reactor vessel is stainless steel clad, and the vessel 's internals, shroud, and jet pumps would make an internal volumetric examination of the beltline region welds impractical.

The Northeast Nuclear Energy Company agreed to perform the augmented program approved by Amendment No. 64 to Provisional Operating License No. DPR-21 when examining these welds. The augmented program consists of:

(a) Examine volumetrically at least 100% of accessible length of each longitudinal weld and at least 100% of the accessible length of each circumferential weld, from either inside or outside the vessel.

(b) Visually inspect to the extent practical, and from the vessel inside surface, the areas of the welds required to be examined.

(c) In the event that Code unacceptable flaw is detected, 100% volumetric examination shall be performed on the welds to meet Section XI ASME Code requirement. This examination will be performed from the inside surface of the reactor vessel.

Evaluation and Conclusion

Imposition of the Section XI ASME Code requirement would subject the licensee to extreme hardship in necessitating removal of the concrete biological shield and the permanently installed insulation to perform the required examination of the welds from the vessel outside surface.

The boiling water reactor vessel is calculated to receive a relatively low fluence. Therefore, the amount of irradiation damage in the beltline region should not be significant during this inspection interval. The reactor vessel is monitored for radiation damage in the beltline region. The surveillance program is in compliance with ASTM E185-66 and has been evaluated with respect to the requirements of ASTM E185-73. We have determined that the program meets the essential requirements of ASTM E185-73 and conforms to the intent of Appendix H of 10 CFR Part 50. The program will provide sufficient data to monitor radiation damage to the vessel beltline materials throughout the service life. In addition, the vessel was designed and fabricated in accordance with the rules of Section III of the 1965 Edition of the ASME Boiler and Pressure Vessel Code. We have evaluated the fracture toughness properties and find that they meet the principle requirements set forth in Appendix G of 10 CFR Part 50. Utilizing the results of the surveillance program to monitor material damage from neutron irradiation and the guidelines in Regulatory Guide 1.99 to establish operating limitations will insure that the reactor vessel will be operated in compliance with the requirements of Appendix G of 10 CFR Part 50. This will provide an acceptable margin of safety to prevent brittle fracture of the vessel during any conditions of normal operation, including anticipated operational occurrences and system hydrostatic tests, to which it may be subjected during the remaining service life.

We conclude that the vessel design, on going surveillance program of the reactor vessel materials from the beltline region, and the augmented examination requirements are adequate to provide an acceptable level of safety and assurance that the vessel structural integrity will not be compromised during the inspection period.

B. Pumps

2. Relief from the scheduling requirement for the visual examination of internal surfaces of at least one pump in each group of pumps performing similar functions in the system. (Item No. B12.20, Examination Category B-L-2, Table IWB-2500-1)

Code Requirement

A visual examination (VT-1) of the internal pressure boundary surfaces of one pump in each of the group of pumps performing similar functions in the system shall be examined during each inspection interval. The examination may be performed at or near the end of the inspection interval.

Basis for the Requested Relief

The Northeast Nuclear Energy Company proposes to postpone the visual examination of the internal surfaces of the recirculation pumps until the need arises to perform major maintenance on the pumps. In the absence of required recirculation pump maintenance, the Section XI ASME Code requirement to disassemble and perform a visual examination of the internal surface of the pump casing is an impractical requirement. Disassembly and inspection of a recirculation pump would result in personnel exposures of about 100 man-rems and require approximately $200,000 in spare parts and other costs. The burden imposed by the Section XI Code examination requirement in the absence of maintenance warrants neither the personnel exposure nor the spare parts and other costs.

The Millstone Unit 1 recirculation pumps have operated satisfactorily during the first inspection interval. Visual examination of the pump casing exteriors during operational leak tests at refueling outages has shown no sign of degradation. Based upon BWR service experience, cast stainless steel pump and valve bodies have performed acceptably with no degradation noted. The integrity of the Millstone Unit 1 recirculation pumps will be assured by performing monthly vibration signature analyses. The analysis is sufficiently sensitive to detect changes occurring as a result of either wear, internal surface erosion, or cavitation damage.

Accelerometers are mounted at right angles on each recirculation pump in areas most sensitive to detect change in internal configuration and coolant flow. Although monthly vibration signature analysis will be made, accelerometers will continuously monitor the pumps operational and structural characteristics.

In addition, the integrity of the recirculation pumps will be assured by a sensitive leak detection system. The Millstone Unit No. 1 Technical Specifications limit unidentified leakage inside containment to 2.5 gpm. In the event a through-wall crack develops in either pump casing, the leak detection system is adequate to provide warning of pump degradation at an early stage of leak occurrence. Northeast Nuclear Energy Company will continue the augmented visual inspections of both pump casing exteriors during periodic leak tests or hydrostatic tests during each refueling outage.

Evaluation and Conclusion

We have reviewed and evaluated the basis of the request for relief from the scheduling requirement of Examination Category B-L-2, pump casings, of Section XI ASME Code. We conclude that the Section XI ASME Code requirement is impractical to conduct, and that relief is necessary. We have reviewed the alternate methods of inspection proposed in lieu of the impractical scheduling requirement, and conclude that they will ensure an adequate margin of component integrity. Pursuant to 10 CFR Section 50.55a(g)6(i), we have granted relief from the specific requirement that the licensee identified to be impractical for the Millstone Unit No. 1 facility, giving due consideration to the burden placed upon the licensee if the Section XI ASME Code requirement was imposed, and which we have determined that by granting such relief will not endanger life or property or comm[illegible] [illegible]ense and security of the public. [(Also see NRC safety Evaluation issued by letter (D. Crutchfield to W. Counsil) dated November 19, 1980.)]

C. Valves

3. Relief from visual examination of the internal surfaces of valves listed below in the Feedwater and Recirculation Water Systems (Item No. 12.40, Examination Category B-M-2, Table IWB-2500-1)

   1-FW-11A
   1-FW-11B
   1-RR-1A
   1-RR-2A
   1-RR-1B
   1-RR-2B
   1-RR-4A
   1-RR-4B

   Code Requirement

   Examination of the internal surfaces of one valve within each group of valves that are of the same constructional design, e.g. globe, gate, or check valve, manufacturing method and that are performing similar functions in the system, e.g. containment isolation, system overpressure protection. The examination may be performed at the end of the inspection interval.

   Basis for the Requested Relief

   The valves are located in piping which penetrate the reactor vessel and cannot be isolated for disassembly in a practical manner for examination of the internal surfaces. Relief is requested from the Code requirement on certain valves which either require draining of the reactor vessel or installing plugs in the piping prior to disassembly of the valve for examination.

   The valves are subjected to the system leakage and hydrostatic tests required by IWB-5221 and IWB-5222 of Section XI of the ASME Code. There are valves of the same constructional design in other piping systems which penetrate the reactor vessel. These valves will experience similar thermal-hydraulic conditions as the listed valves. Northeast Nuclear Energy Company will conduct the required Section XI examination on these valves during this inspection interval. In the event unacceptable conditions are observed, the listed valves will be examined in compliance with Section XI ASME Code requirements; or, in the event the reactor vessel is drained for some other purpose, the internal surfaces of the listed valves will be examined.

Evaluation and Conclusion

We conclude from our review that imposition of the Section XI ASME Code requirement for the examination of the internal surfaces of the listed valves would subject the licensee to extreme hardship. Disassembly of the listed valves would require either draining the reactor vessel or installing plugs in the piping leading to the valves. The examination of valves of the same constructional design and experience similar thermal-hydraulic conditions as the listed valves will provide assurance of acceptable structural integrity. The Section XI ASME Code requirement is impractical for the listed valves, relief is required and granted for this examination requirement.

2.2 CLASS 2 COMPONENTS

A. Pressure Vessels

4. Request to substitute surface examination for volumetric examination of the circumferential shell welds of the LPCI heat exchanger. The surface examination shall cover at least 20% of the weld in proximity to the nozzles. (Item No. C1.30, Examination Category C-A, Table IWC-2500-1)

Code Requirement

Volumetric examination of essentially 100% of the length of the tubesheet-to-shell weld. The examination is to be conducted during each inspection interval.

Basis for the Requested Relief

The geometry of the circumferential tubesheet-to-shell weld on the LPCI heat exchanger is such that volumetric examination of this weld, using either ultrasonic or radiographic techniques, cannot be performed to obtain meaningful results. Northeast Nuclear Energy Company proposes to perform a surface examination of at least 20% of the weld in the nozzle areas.

Evaluation and Conclusion

Volumetric examination of the circumferential tubesheet-to-shell weld in the LPCI heat exchanger is an impractical requirement for the Millstone Unit 1 facility because the design of the heat exchanger prohibits meaningful examination of these welds by either ultrasonic or radiographic techniques. The licensee has proposed to use surface examination in lieu of volumetric examination. We require surface examination of at least 20% of each circumferential tubesheet-to-shell weld be performed during this inspection interval and that the examination be performed on the part of the welds in the vicinity of the nozzles. Due to the design of the heat exchanger and the weld configuration, we conclude that the surface examination of the welds in these areas is adequate to assure the structural integrity of the heat exchanger.

5. Request to substitute surface examination for surface and volumetric examination of nozzle welds to the Shutdown Heat Exchanger. (Item No. C2.20, Examination Category C-B, Table IWC-2500-1)

Code Requirement

Volumetric and surface examinations for pressure retaining nozzle welds (7-1/2 inch nominal thickness) during each inspection interval (Reference Fig. IWC-2520-4, Section XI)

Basis for Requesting Relief

The shutdown heat exchanger nozzle to vessel weld is similar to Fig. IWB-2500-10. The design of the nozzle prevents volumetric examination of the entire length of the weld. Volumetric examination of the weld, using either ultrasonic or radiographic techniques, cannot be performed to obtain meaningful results. Northeast Nuclear Energy Company will perform surface examination of this weld.

Evaluation and Conclusion

Based on drawings and the description of the shutdown heat exchanger nozzle to vessel weld, we conclude that this weld cannot be volumetrically examined over its entire length. Volumetric examination may be practical over part of the length of the weld. We require that volumetric examination be performed to the extent practical, and that surface examination be performed on the nozzle to vessel weld during this inspection interval. We conclude that volumetric examination to the extent practical, combined with surface examination, will assure the integrity of the shutdown heat exchanger nozzle to vessel weld.

2.3 SYSTEM PRESSURE TESTS

A. Class 1 - Class 2 System Boundary

6. Request relief from testing portions of the systems listed at the Code required test pressure:

System	Class 1 Boundary	Class 2 Boundary
Feedwater	1-FW-11A/B	HP Heater Discharge Isolation Valves
Standby Liquid Control	1-SL-8	1-SL-6
Core Spray	1-CS-6A/B	1-CS-5A/B
LPCI	1-LP-11A/B	1-LP-10 A/B
Reactor Cleanup Return	1-CU-29	
CRD Return	Penetration Cap	

Code Requirement

The pressure retaining components shall be subjected to a hydrostatic test at 1.10 times the system operating pressure at least once toward the end of each inspection interval.

Basis for the Requested Relief

Precautions must be taken in view of the differences that exist in Class 1/Class 2 system test pressures to prevent overpressurization of the Class 1 components. The location of check valves in several systems preclude the Class 1 pressure test boundary from extending outward beyond the first of such valves, usually located inside containment even though the class change boundary is outside containment. In these cases, the Class 1 leakage and pressure test boundary would be the inside check valve. Conversely, pressure tests of Class 2 systems would have to be bounded at a stop valve which may or may not be the Class 1/Class 2 boundary.

Evaluation and Conclusion

The design of the listed Class 1 and Class 2 systems does not permit system isolation at the boundary of the systems. In order to prevent overpressurization of the Class 1 portions, the portions of the systems which cannot be tested to ASME Section XI required pressure must be visually inspected to the extent practicable during operation of the facility. In addition, portions of the system piping will be volumetrically examined under Category B-J. We conclude that the examinations and alternate testing procedure are acceptable to assure the systems integrity for safe operation of the facility during this inspection interval.

B. Class 3 Systems

7. Request relief from pressure testing the systems listed at pressures less than the code required test pressure.

Class 3 Systems	System Pressure	Test Pressure
Service Water	55	55
Atmospheric Control	0	43

Code Requirement

The pressure retaining components shall be subjected to a pressure test at 1.10 times the system design pressure at least once toward the end of each inspection interval.

Basis for the Requested Relief

These systems are required for continuous cooling of vital system components and cannot be removed from service for the period of time required for the hydrostatic test. Due to the essential cooling functions of this system, the licensee will test these systems by utilizing the pump head and throttled flow for continuous cooling of the vital system components. Referring to the atmospheric control, the system will be pneumatically tested in compliance to the requirements of the integrated leak test program of Appendix J, 10 CFR Part 50.

Evaluation and Conclusion

The licensee has proposed testing the Service Water System for leakage by utilizing the pump head and throttled flow for continuous cooling of the vital system components and pneumatically testing the Atmospheric Control System in compliance to Appendix J of 10 CFR Part 50. We conclude that the pressures at which the systems will be tested are greater than normal operating pressure and are considered adequate to provide assurance of the structural integrity of the systems.

3.0 SUMMARY

Based on our review, we have found that the Inservice Inspection Program for the ten-year interval from December 28, 1980 to December 28, 1990 is in compliance to the extent practical, within the limitations of design, geometry, and materials of construction, to the requirements of Section XI of the ASME Boiler and Pressure Vessel Code, 1977 Edition, including Summer 1978 Addendum. We, therefore, conclude that the Inservice Inspection Program for the ten-year interval from December 28, 1980 to December 28, 1990 for Millstone Station Unit No. 1 is in compliance with the requirements of 10 CFR Section 50.55a(g).

Pursuant to 10 CFR Section 50.55a(g)(6)(i) we hereby grant relief from the certain requirements, as discussed in this Safety Evaluation, for Millstone Unit 1 ten-year inspection interval from December 28, 1980 to December 28, 1990, and have determined that granting such relief is authorized by law and will not endanger life or property or the common defense and security and is otherwise in the public interest considering the burden imposed on the licensee if the relief is not granted.

4.0 CONCLUSION

We have concluded, based on the considerations discussed above, that: (1) because the relief does not involve a significant increase in the probability of consequences of accidents previously considered and does not involve a significant decrease in a safety margin, the relief does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of this relief will not be inimical to the common defense and security or to the health and safety of the public.

Date: April 10, 1981