June 24, 1982

Docket No. 50-245
LS05-82-06-093

Mr. W. G. Counsil, Vice President
Nuclear Engineering and Operations
Northeast Nuclear Energy Company
Post Office Box 270
Hartford, Connecticut 06101

Dear Mr. Counsil:

SUBJECT: MILLSTONE NUCLEAR POWER STATION, UNIT 1 - SEP TOPIC III-8.C, IRRADIATION DAMAGE, USE OF SENSITIZED STAINLESS STEEL AND FATIGE RESISTANCE

Reference: Letter, W. G. Counsil to D. L. Ziemann, SEP Topic III-8.C, dated January 31, 1980

Enclosed is a copy of our final evaluation of Systematic Evaluation Program Topic III-8.C, Irradiation Damage, Use of Sensitized Stainless Steel and Fatigue Resistance.

This assessment compares your facility as described in Docket No. 50-245 with the criteria currently used by the regulatory staff for licensing new facilities. The final evaluation differs from the draft in that it incorporates your comment, has been rewritten into a new format and has been reworded for clarity.

This evaluation will be a basic input to the integrated safety assessment for your facility. This topic assessment may be changed in the future if your facility design is changed or if NRC criteria relating to this topic are modified before the integrated assessment is completed.

Sincerely,

James Shea, Project Manager
Operating Reactors Branch No. 5
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Enclosure:
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Mr. W. G. Counsil

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SYSTEMATIC EVALUATION PROGRAM
TOPIC III-8.C

MILLSTONE 1

TOPIC: III-8.C, IRRADIATION DAMAGE, USE OF SENSITIZED STAINLESS STEEL AND FATIGUE RESISTANCE

I. INTRODUCTION

The reactor internals are designed to support and orient the reactor core and control assemblies, provide a flow path for reactor coolant and support in-core instrumentation. The internals are to withstand the forces due to weight, pre-load of fuel assemblies, control rod dynamic loading, vibration, and loss of coolant accident blowdown coincident with earthquake accelerations.

SEP Topic III-8.C is intended to determine if the integrity of the reactor internal structures has been degraded through the use of sensitized steel.

The effect of neutron irradiation and fatigue resistance on material of the internal structures was eliminated from the safety objective of Topic III-8.C in memorandum to D. G. Eisenhut from D. K. Davis and V. S. Noonan dated December 8, 1978. The memorandum concluded that operating experience indicated that no significant degradation of the materials of the reactor internal structures had occurred as a result of either irradiation damage or fatigue resistance. Furthermore, the Standard Review Plan does not address neutron irradiation nor fatigue resistance of the materials of the structures.

II. REVIEW CRITERIA

General Design Criterion 4, "Environmental and Missile Design Bases", Appendix A, 10 CFR Part 50, requires that components be designed to accomodate the effects of and be compatible with the environmental conditions associated with normal operation, maintenance, testing and postulated accident conditions. The use of sensitized stainless steel in the presence of certain environmental conditions can lead to stress corrosion cracking and the eventual loss of structural integrity of the affected component.

III. RELATED SAFETY TOPICS

SEP Topics III-8.A and III.8.B evaluate related items such as control rod drive mechanism integrity and loose parts monitoring, respectively.

IV. REVIEW GUIDELINES

The review of the use of sensitized stainless steel in reactor internals was conducted in accordance with the acceptance criteria of Section 4.5.2, "Reactor Internal and Core Support Materials", of the Standard Review Plan and Regulatory Guides 1.31, "Control of Ferrite Content in Stainless Steel Weld Metal", and 1.44, "Control of the Use of Sensitized Stainless Steel". The materials specifications requirements were those of Sections II and III of the ASME Boiler and Pressure Vessel Code.

V. EVALUATION

The reactor internal components for the Millstone Nuclear Power Station, Unit No. 1, are described and analyzed in Section III-7.3.4 through Section III-7.3.8 of the Final Safety Analysis Report. The internal components were designed to provide support for the fuel and maintain structural clearances during normal and accident conditions. In addition, the internal components provide passageways for the coolant to cool the fuel and means for adequately separating the steam from the coolant water.

Components of the reactor coolant pressure boundary of the Millstone Nuclear Power Station, Unit No. 1, were designed, fabricated, inspected and tested to the requirements of Section III of the ASME Boiler and Pressure Vessel Code, 1965 Edition, including Summer 1965 Addenda plus applicable nuclear code cases. The components of the reactor internal components were analyzed to withstand the combined loadings from pressure, temperature, fluid movement, and seismic acceleration. Calculated stresses were within the criteria required by the ASME Boiler and Pressure Vessel Code.

The primary criteria for material selection for the reactor internal components were the mechanical properties, the material stability and corrosion resistance in the reactor environment. The materials used for fabricating the reactor internals were identified in the FSAR as Type 304 stainless steel, Inconel, and minor quantities of special purpose alloys, such as Stellite. These materials have proven adequate for reactor internal construction as a result of extensive tests, prior usage, and satisfactory performance.

The regulatory position on the use of sensitized stainless steel in reactor internal materials was not addressed in the Millstone Unit No. 1 Final Safety Analysis Report. Experience has shown that at least three elements in combination are necessary to cause cracking in sensitized stainless steel components. These are material susceptibility, an oxygenated water environment, and a threshold total stress. We have not ascertained for this evaluation that the Millstone Unit No. 1 reactor internal components contain sensitized stainless steel in contact with an oxygen saturated coolant water environment. However, the calculated stresses on the reactor internal components do not exceed the threshold stress values associated with intergranular stress corrosion cracking. The threshold stress values are less than the 0.2% off-set yield stress at temperature. Further, in the reactor environment, stress relaxation may occur due to irradiation and temperature effects.

The Licensee Event Reports and the BWR Nuclear Power Experience were reviewed for the Millstone Unit No. 1 in order to correlate reactor internal materials failure to the use of sensitized stainless steel in the components. The events are summarized as follows:

> In September, 1972, during reactor startup, seawater leakage occurred through the main condenser into the hotwell, saturating the condensate demineralizer and allowing excess chloride ion into the primary coolant system. An inservice inspection and testing program was conducted to assess the damage to the reactor internal components. The program showed that intergranular stress corrosion cracking had occurred, causing the failure of the local power range monitors, five neutron source holders and the cladding on the control rod neutron absorber elements. Examination of the other components showed neither intergranular stress corrosion cracking nor evidence of material degradation.
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> Stress corrosion cracking of areas of stainless steel cladding adjacent to reactor vessel head nozzles was identified in 1976. This was evaluated and a report was submitted to the NRC.

We conclude from our review of the Licensee Event Reports and the BWR Nuclear Power Experience that the integrity of the reactor internal components was degraded because of the chloride intrusion and the corrosion attack upon the stainless steel components. The subsequent inservice inspection and testing procedures detected failures in the local power range monitors, neutron source holders and control rod cladding. Failure was attributed to intergranular stress corrosion cracking. After the chloride intrusion and repair and replacement of the failed components, no further incidents of stress corrosion cracking have been reported in the reactor internal components for the Millstone Nuclear Power Station, Unit No. 1.

The inservice inspection program for the reactor internal components is being conducted during the current interval to the requirements of Section XI of the ASME Boiler and Pressure Vessel Code, 1974 Edition, including Summer 1975 Addenda. The program is in compliance with paragraph (g) of Section 50.55a of 10 CFR Part 50. It will assure that the integrity of the included components is maintained during reactor operation.

VI. CONCLUSIONS

We conclude from our review that the stainless steel materials in the reactor internal components are probably not sensitized, that there is an increased potential for cracking due to operation in an oxygen saturated water environment and that the incidents of stress corrosion cracking are expected to be rare because the total stress level in the internal components is relatively low. In the unlikely event that intergranular stress corrosion cracking should occur, operating experience has demonstrated that cracks in the components will be detected by inservice inspection procedures prior to component failures. We conclude that intergranular stress corrosion cracking in the reactor internal components at Millstone Unit No. 1 is not a hazard to the health and safety of the public.