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

November 13, 1979

Docket No. 50-213

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

Dear Mr. Counsil:

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

Enclosed is a copy of our draft evaluation of Systematic Evaluation Program Topic III-8.C. You are requested to examine the facts upon which the staff has based its evaluation and respond either by confirming that the facts are correct, or by identifying any errors. If in error, please supply corrected information for the docket. We encourage you to supply for the docket any other material related to these topics that might affect the staff's evaluation.

Your response within 30 days of the date you receive this letter is requested. If no response is received within that time, we will assume that you have no comments or corrections.

Sincerely,

for Thomas V Wambach

Dennis L. Ziemann, Chief
Operating Reactors Branch #2
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Enclosure:
Topic III-8.C

cc w/enclosure:
See next page

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SYSTEMATIC EVALUATION PROGRAM

PLANT SYSTEMS/MATERIALS

CONNECTICUT YANKEE

Topic III-8.C Irradiation Damage, Use of Sensitized Stainless Steel and Fatigue Resistance

The safety objective of this review is to determine whether the integrity of the internal structures of operating reactors has been degraded through the use of sensitized stainless 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.

Information for this assessment was obtained from the Facility Description and Safety Analysis, Technical Specifications, Safety Evaluation Reports to the ACRS, Licensee Event Reports, and PWR Nuclear Power Experience for the Connecticut Yankee Plant. Our assessment is based on information in topical reports on the behavior of sensitized stainless steel in PWR nuclear steam supply systems and conversations with materials engineers at Combustion Engineering, Westinghouse and General Electric Company.

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The regulatory position is addressed in Section 4.5.2, "Reactor Internals Material", of the Standard Review Plan. The areas currently reviewed in the applicant's SAR are materials specification and the controls imposed on the reactor coolant chemistry, fabrication practices and examination and protection procedures. The materials specification should comply with Section III of the ASME Boiler and Pressure Vessel Code and the fabrication procedures for the components should satisfy the recommendations of Regulatory Guide 1.31, "Control of Ferrite Content in Stainless Steel Weld Metal", and Regulatory Guide 1.44, "Control of the Use of Sensitized Stainless Steel".

The reactor internals and control rod drive mechanisms are described for the Connecticut Yankee Plant in Sections 4.1.7 and 4.1.8, respectively, of the Facility Description and Safety Analyses. The reactor internals were designed to support and orientate the reactor core and control rod assemblies. The internals absorb static and dynamic loads and transmit them to the reactor vessel flange.

Components of the reactor coolant pressure boundary were designed, fabricated, and inspected to the requirements of Section VIII, Division 1, of the ASME Boiler and Pressure Vessel Code, 1962 Edition, including Summer 1963 Addenda plus applicable nuclear code cases. The stress analyses performed (not required by Section VIII) and stress intensity limits applied were in compliance with the rules of Tentative Structural Design Basis for Reactor Pressure Vessels and Directly Associated Components (Pressurized Water Cooled Systems) PB-151987 as modified to account for the stress criteria of Section III of the ASME Boiler and Pressure Vessel Code.

The materials used for constructing the reactor internals were identified in the Facility Description and Safety Analyses as Type 304 stainless steel with minor quantities of special purpose alloys, such as, Inconel 718 and X, Type 410 stainless steel, and cobalt-based alloys. The type of materials used was specified in Westinghouse Equipment Specification, which, in some cases, upgraded or modified the ASME Code requirements. For example, the Equipment Specification required the fabricator to perform Charpy V-notch impact and drop weight tests on the base metal and associated weld metal and side bend tests on the stainless steel cladding for the vessel. The tests are not required by Section VIII of the ASME Code.

A Hazards Analyses report was presented to the ACRS on March 13, 1964. It stated:

> "In our opinion, all components of the primary system of the facility are conventional in design, and the materials and design codes proposed are compatible with the operating conditions expected. Accordingly, we believe the primary system will safely perform its intended functions of cooling the core, transferring heat to the secondary system and containing radioactive materials generated within the primary system."

Insufficient information was included in the Facility Description and Safety Analyses report to ascertain compliance with the recommendations of Regulatory Guide 1.31, "Control of Ferrite Content in Stainless Steel Weld Metal", Regulatory Guide 1.44, "Control of the Use of Sensitized Stainless Steel", and to assure proper control of welding materials and procedures. Therefore, we assume for this assessment that the reactor internal structures contained sensitized stainless steel.

Justification for the use of sensitized stainless steel in PWR quality coolant water was presented in a topical report, WCAP-7477-L, "Sensitized Stainless Steel in Westinghouse PWR Nuclear Steam Supply Systems," written by M. A. Golik, March, 1970. The report reviewed the nature of sensitized Types 304 and 316 stainless steel and the significant factors in the application of sensitized stainless steel in present and future nuclear steam supply systems. In reviewing the PWR operating experience with the Shippingport, BR-3, Saxton, Yankee Rowe, Selni, Connecticut Yankee, San Onofre and Zorita reactors the conclusion was reached that no general problems of intergranular or stress corrosion related to sensitized stainless steel have been encountered in PWR operating reactors. This conclusion was discussed with personnel at Westinghouse and Combustion Engineering who confirmed the conclusion in the report and updated current PWR operating experience.

The operational experience of the Connecticut Yankee Plant was reviewed in the licensees Event Reports and PWR Nuclear Power Experience. None of the events described were traceable to the use of sensitized stainless steel in the fabrication of the reactor internal structures.

The following information was contained in the Safety Evaluation by the Division of Reactor Licensing dated July 1, 1971:

"During the first refueling outage, the applicant performed an inspection of the reactor internals and primary coolant piping that identified the following conditions:

1. Breakage of a structural element in each of two of the control rod clusters.
2. Indications of possible cracks on nozzles of the steam generators.
3. Breakage of the six flexure pieces between the top of the thermal shield and the core support barrel.

The licensee has thoroughly investigated the cause of each deficiency and has taken appropriate action in each case.

The breakage of the control rod cluster assemblies was traced to a manufacturing defect. In each case, the break occurred in a brazed joint which connects a vane, from which control rods are suspended, to a central hub called a spider. The brazed joint in one assembly was examined in detail in a hot cell and found to have no braze material inside the joint, probably due to improper cleaning of the joint prior to brazing. The second failed joint had a similar appearance. It is not feasible to inspect these joints on all control rod assemblies for brazing deficiencies but all joints were visually inspected to verify the integrity of the assemblies before the reactor was restarted. Only two such joints have failed in all the operating reactors and there is no evidence that this condition is prevalent in these assemblies."

"The indications of possible cracks on nozzles of the steam generators were found to be minor surface imperfections resulting from an error in welding procedure. Carbon steel fasteners for shipping covers were welded onto an Inconel weld with 309 stainless steel welding rods. A drawing change had been overlooked, resulting in this improper procedure. The remainder of these fasteners and the stainless steel weld material were removed from the steam generators. There were no indications of stress corrosion cracking on the nozzle safe-ends."

The breakage of the six support flexure pieces was due to a metallurgical problem combined with high-cycle fatigue stress resulting from vibration of the thermal shield. Based on an analysis of the structural supports for the thermal shield, the licensee concluded that these flexure pieces are not necessary and therefore these pieces were not reinstalled. We concluded that removal of the six support flexure pieces did not present significant hazards considerations not described or implicit in the safety analysis report.

The inservice inspection program for the Connecticut Yankee Plant was evaluated in the same Safety Evaluation. The review revealed the following:

"(1) The licensee was not required to comply with the 1970 Edition Section XI of the ASME Inservice Inspection Code, since Connecticut Yankee went into opertion before Section XI was adopted on January 1, 1970. However, the report submitted does satisfy the requirements of IS-622.2 Inservice Inspection Reports of the Code."

"(2) The inspections performed do meet or exceed the requirements specified in the AEC document 'Inservice Inspection Requirements for Nuclear Power Plants Constructed with Limited Accessibility for Inservice Inspection dated January 31, 1969.' High radiation fields prevented the applicant from completing the requirements of Section XI of the ASME Inservice Inspection Code at the reactor vessel flange weld and at the pressurizer surge nozzle.

(3) The techniques and inspection standards employed were in accordance with Appendix IX of Section III of the ASME Code and meet IS-210 of Section XI. The personnel were qualified in accordance with SNT-TC-1A "Recommended Practice for Non-destructive Testing Personnel Qualification and Certification" of the American Society for Nondestructive Testing.

(4) The report states that no significant indications were found.

We conclude that the results of the 1970 Inservice Inspection of Connecticut Yankee are acceptable. This conclusion is based on review and comparison of the results, stated in the report, with the pre-service baseline inspections performed by the licensee in 1966 and 1967."

The inservice inspection program for the reactor internal structure for the current inspection interval for Connecticut Yankee will be conducted to the requirements of Section XI, ASME Boiler and Pressure Vessel Code, 1974 Edition, including Summer 1975 Addenda. The program is in accordance with paragraph (g), Section 50.55a, 10 CFR Part 50.

We conclude from our review of the information submitted by the licensee and the operating information in the Licensee Event Reports together with the PWR _Nuclear Power Experience_ that the integrity of the reactor internal structures for the Connecticut Yankee Plant has not been degraded through the use of sensitized stainless steel. Furthermore, we conclude that the integrity of the internal structures will be assured by an inservice inspection program in accordance with the requirements of paragraph (g), Section 50.55a, 10 CFR Part 50.

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