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

November 15, 1979

Docket No. 50-29

Mr. Robert H. Groce
Licensing Engineer
Yankee Atomic Electric Company
20 Turnpike Road
Westboro, Massachusetts 01581

Dear Mr. Groce:

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
Division of Operating Reactors

Enclosure:
Topic III-8.C

cc w/enclosure:
See next page

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November 15, 1979

SYSTEMATIC EVALUATION PROGRAM
PLANT SYSTEMS/MATERIALS
YANKEE ROWE PLANT

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 fatique resistance of the materials of the reactor internal structures.

Information for this assessment was obtained from the Final Safety Analysis Report, Hazards Analysis Reports, Safety Evaluation Reports to the ACRS, Licensee Event Reports and PWR Nuclear Power Experience for the Yankee Rowe plant. Our assessment is based on information in topical reports on the behavior of sensitized stainless steel in PWR nuclear steam supply systems, WAPD-SC-541 "PWR Hazards Summary Report for the Shippingport Reactor," WAPD-PWR-971, "Selection and Application of Materials for the PWR Reactor Plant," and conversations with materials engineers at Combustion Engineering, Westinghouse and General Electric Company.

The regulatory position is addressed in Section 4.5.2, "Reactor Internals Materials" of the Standard Review Plan. The areas currently reviewed in the applicants' 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 Yankee Rowe reactor is generally similar to the Shippingport reactor. The control rods and core support structure are described in Sections 1 and 2 of the Final Safety Analysis Report. The functions of the core support structure are to support and orientate the fuel assemblies, maintain orientation and position of the control rods, and to provide passageway for the reactor coolant. The structure consists of an upper and lower support plate, an upper and lower core support barrel, core barrel, radial support and baffle structure. The internals are supported from the reactor flange.

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Components of the reactor coolant pressure boundary were designed, fabricated and inspected to the requirements of Section VIII of the ASME Boiler and Pressure Vessel Code, 1956 Edition. Stress and deflection analyses were made by the licensee (YAEC-77 and YAEC-105).

The materials used for constructing the reactor internals were identified in the FSAR as Type 304 stainless steel with minor quantities of special purpose alloys, such as Inconel X, type 410 stainless steel, Armco 17-4 PH, and cobalt-base alloys. The type of materials used was specified in the Westinghouse Equipment Specification, which, in some cases, upgraded or modified the ASME Code requirements. Justification for the use and selection of these materials is presented in WAPD-PWR-971, "Selection and Application of Materials for the PWR Reactor Plant."

Insufficient information was included in the FSAR 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 to current PWR operating experience.

The operational experience of the Yankee Rowe Plant was reviewed in the Licensee Event Reports, the PWR Nuclear Power Experience, and the Hazards and Safety Evaluation Reports. The review revealed a number of minor problems with the reactor internals. These problems are described in the Appendix to this report. None of the events described were directly traceable to the use of sensitized stainless steel in the fabrication of the reactor internal structures.

The inservice inspection program for the reactor internal structure for the current inspection interval for the Yankee Rowe plant will be conducted to the requirements of Section XI, ASME Boiler and Pressure Code, 1974 Edition, including Summer 1975 Addendum. 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 Yankee Rowe 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.

APPENDIX

Topic III-8.C Review of Service Experience Yankee Rowe Reactor

The abnormal occurrences described for the Yankee Rowe plant related to the operating experience for the reactor internals are summarized in this Appendix.

1) The Hazards Analysis dated August 31, 1962, described two problems observed during inspection at the first refueling outage.

   (a) Wear occurred on certain parts of the control rod assemblies, and

   (b) Deterioriation of the nickel coating on the Ag-In-Cd control rod absorber section.

   The analysis concluded that these problems have not resulted in an unsafe operation of the reactor..

2) A defective thermal shield was evaluated in a Safety Evaluation dated October 28, 1965.

   "The thermal shield for the Yankee Rowe reactor consists of a hollow cylinder that is larger in diameter than the upper neck of the pressure vessel. It was installed in four sections that were fastened together at four half-lap joints by thirteen bolts. The structure is located in the primary coolant flow annulus between the core barrel and the vessel wall, and is supported by eight support lugs. Recent inspection of the thermal shield has revealed that some of the bolts have failed, and that two joints had separated radially at the upper end and three at the lower end. The maximum measured separation was 3/8-inch. Based on inspection of some of the failed bolts, it is believed that the bolts failed as a result of shear forces."

   "Yankee Rowe proposed to reinforce the bolted joints by the installation of four Joint Clamp Assemblies. Each assembly consists of two vertical clamps that grasp the upper and lower edges of adjacent thermal shield sections. The two sections of each clamp are joined at the top and bottom by bars which span the half-lap joints. The clamp assemblies have been designed to preclude vibration, and to withstand any anticipated static or dynamic stresses."

"Since the thermal shield is located in the primary coolant flow annulus, failure of this structure could cause a potential flow reduction. However, even if one or more joints should become completely separated, there is insufficient clearance between the support lugs and the core support barrel to permit any significant displacement. The maximum possible motion for a thermal shield section would be for it to tilt either against the core barrel or the vessel wall, and would present a flow restriction in only one-quarter of the available flow annulus. This condition would be detected by existing flow and temperature instrumentation, and would not result in any reactor damage. Therefore, we believe that installation of the Joint Clamp Assemblies will improve the integrity of the thermal shield joints and that the safety of reactor operations will not be adversely affected."

It was concluded that the proposed change does not present significant hazards considerations not described or implicit in the hazards summary report, and that there is reasonable assurance that the health and safety of the public will not be endangered.

3) The Hazards Analysis dated March 26, 1962, evaluated the request to install in the reactor up to twelve special tube assemblies containing encapsulated specimens of reactor vessel material. The assemblies were to be placed in eight holes provided in the upper flange of the core baffle and in four guide channels attached to the outer surface of the thermal shield.

   One of the surveillance capsules became loose in the reactor and lodged between the core support and vessel cladding. The cladding was perforated as a result of the event. The event was attributed to fatigue of the fastener. This capsule as well as the remaining attached capsules were removed from the reactor during inspection at the first refueling operation.

4) The Safety Evaluation dated October 25, 1965, evaluated a proposed design change authorizing the installation of four Secondary Core Supports to limit movement of the reactor core in the unlikely event that the primary core support structure should fail.

   "The Yankee Rowe core consists of 76 fuel elements, which weigh about 26 tons, and is contained within the core barrel. The core barrel is suspended by a flange from the top of the pressure vessel, and primary support for this structure is provided by a full penetration weld between the core barrel and the top support flange. This weld has adequate strength to support this structure, but is reinforced by twelve one-inch thick gusset plates spaced radially around the core barrel. The support welds were recently inspected by dye penetrant and ultrasonic testing techniques and found to be in sound condition."

"Yankee Rowe believes that failure of the primary core support is exteemely unlikely, but has proposed to install four Secondary Core Supports to limit the downward motion of the core if a failure should occur. The secondary support consists of four stainless steel straps (3/4" x 7 3/4" in cross section) that are firmly attached to the top and bottom of the thermal shield. These straps provide four points of support under the lower core support plate and would limit the downward motion to 5/8 inch. The secondary supports have been designed to preclude vibration, and to absorb the impact of the core structure. The weight of the core structure would ultimately be supported by the thermal shield support lugs which are an integral part of the pressure vessel wall. We believe that the Secondary Core Supports have been adequately designed, and that their installation would provide a desirable additional safety factor for the Yankee Rowe reactor."

"Since the control rods are inserted from the top of the core, downward movement of the core would result in a reactivity addition. However, the Yankee Rowe reactor is operated as a chemical shim plant with only one group of control rods normally inserted. With this mode of operation, the reactivity addition would be negligible if the core should fall. The maximum potential reactivity addition (control rods at maximum worth) that could occur for this postulated accident, with the Secondary Core Supports in place, would be during shutdown or low power operation. This reactivity addition would be less than 0.005. Yankee Rowe has analyzed this reactivity addition, and we agree that the resulting transient would not result in any damage to the reactor."

It was concluded that the Proposed Change does not present significant hazards considerations not described or implicit in the hazards summary report, and that there is reasonable assurance that the health and safety of the public will not be endangered.

5) The Safety Evaluation dated September 29, 1966, evaluated the proposed change in design to authorize the installation of twenty-four inconel-clad silver-indium-cadmium control rods with zircaloy follower sections that are welded to the control rods. This change was requested to provide a control rod design that would preclude inadvertent disassembly of the follower section under certain reactor shutdown conditions.

"The Yankee Rowe control system contains 24 control rods that are cruciform in shape and contain zircaloy follower sections. The current complement of control rods consists of 20 hafnium rods, two Inconel-clad Ag-In-Cd rods, and two stainless steel clad Ag-In-Cd rods. Yankee Rowe has proposed to replace these rods with 24 new Inconel-clad Ag-In-Cd control rods. The physical dimensions and reactivity worth of these control rods are essentially identical to those currently in service. Two Inconel-clad Ag-In-Cd control rods have been in service during Cores III, IV, and V and inspections during refueling shutdowns have indicated no problems with respect to mechanical wear and corrosion."

"The replacement control rods will have follower sections that are welded to the absorber sections. The current design contains a snap joint that connects the absorber to the follower section by rotation about the vertical axis. Yankee Rowe has reported that inspection of the two control rods removed during the 1965 refueling indicated that some wear of the snap joint has occurred. This wear could allow the follower section to rotate and fall from the absorber during refueling operations when the upper core support plate is removed. However, such separation could not occur during reactor operation since relative rotation sufficient to permit separation of the two sections is precluded by the internal core structure. The use of follower sections that are welded to the absorber sections will prevent inadvertent disassembly of the control rods even during the refueling operation. In this respect we believe that the safety of reactor operations will be improved by the use of the proposed control rods."

The staff concluded that the proposed change does not present significant hazards considerations not described or implicit in the haz rds summary report, and that there is reasonable assurance that the health and safety of the public will not be endangered.

6) A Regulatory Operations inquiry report was filed on November 14, 1972, reporting a loose bolt lying on top of the lower core support plate. The event was described as follows:

"While performing the control rod interchange during the present scheduled outage, it was observed that a bolt was lying on top of the lower core support plate. Observations made with a T.V. camera through 40 feet of water indicated that there were two additional bolts lying loose. An examination of the bolts indicate that they are control rod shroud tie down bolts."

"Preliminary determination identifies two shroud tubes displaced. There are a total of 32 shroud tubes, 8 for shim rods and 24 for moveable control rods. All T.V. monitoring has been recorded on tape."

The licensee summarized the scope and results of the inspection program conducted on the reactor internals following the above described event which occurred after twelve years of reactor operation. The results of the inspection program were as follows:

"Complete visual inspections by the use of binoculars, underwater TV cameras, and boroscopes included the lower internals, significant areas of the upper internals, and areas inside the reactor vessel. Measurements performed by the use of specially designed tools and gaging devices included torque checks on every other lower core support plate to core barrel connecting bolt. These measurements confirmed that these important load carrying bolted connections remained tight. We conclude that this inspection program has identified the extent of the bolting failures in the lower shroud tube assembly and it has confirmed that the other internals components have not experienced significant structural degradation. In addition, during this inspection program an impression was made of the existing cladding defect inside the reactor vessel. A check of measurements of this impression does not indicate measureable changes from measurements made in 1965 and 1968 inspections. All dropped out bolts and locking devices were retrieved from the core support plate and the reactor vessel bottom, except for one bolt. Efforts will be continued to recover this missing bolt. A large foreign object, identified as part of an original low flux specimen holder was also retrieved from the underside of the lower core support plate."

The Hazards Analysis dated March 30, 1973, evaluated the proposed shroud tube design change of the lower internals and the supplemental information presented in support of the change. The event was attributed in part to flow induced vibration acting on the shroud tubes that had loose connecting flange bolts. The Hazards Analysis contained the following:

"We have reviewed your Proposed Change No. 106 and the supplemental items of information, including the results of your inspection program performed on the original shroud tube assembly and other internals; your evaluation of the bolting failures in the original shroud tube assembly; the summary of your mechanical, thermal, and hydraulic evaluation; the guides, codes, standards, and the quality assurance and audit program used in the design and fabrication of the replacement shroud tube assembly; and the preoperational and post-operational inspection and surveillance programs. We have concluded that: (a) the bolting failures were limited to a local area in the original shroud tube assembly and the other internals components have not experienced significant degradation during the period of 12 years of reactor operation, (b) significant design improvements have been incorporated into the new shroud tube assembly, (c) necessary modifications to facilitate installation of the new assembly will not significantly change important performance characteristics, and (d) the specified surveillance program for monitoring the integrity of the new shroud tube assembly is acceptable."

"We have concluded that there are no hazards considerations not described or implicit in the Safety Analysis Report. There is reasonable assurance that the health and safety of the public will not be endangered. Accordingly, pursuant to 50.59 of 10 CFR Part 50, Change No. 106 is hereby authorized as proposed."

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