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

TERA

October 30, 1980

Docket No. 50-219

Mr. I. R. Finfrock, Jr.
Vice President
Jersey Central Power & Light Company
Post Office Box 388
Forked River, New Jersey 08731

Dear Mr. Finfrock:

RE: SEP TOPIC III-8.C - Irradiation Damage, Use of Sensitized Stainless Steel and Fatigue Resistance.

Enclosed is a copy of our 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. 50-219, with the criteria currently used by the regulatory staff for licensing new facilities. Please inform us if your as-built facility differs from the licensing basis assumed in our assessment.

We have discussed this assessment with your staff and believe the facts concerning your plant are correct. Therefore, our review of this topic is complete and this evaluation will be a basic input to the integrated safety assessment for your facility unless you identify changes needed to reflect the as-built conditions at your facility. This topic assessment may be revised in the future if your facility design is changed or if NRC criteria relating to this topic is modified before the integrated assessment is completed.

Sincerely,

Dennis M. Crutchfield

Dennis M. Crutchfield, Chief
Operating Reactors Branch #5
Division of Licensing

Enclosure:
Completed SEP
Topic III-8.C

cc w/enclosure:
See next page

1

cc

G. F. Trowbridge, Esquire
Shaw, Pittman, Potts and Trowbridge
1800 M Street, N. W.
Washington, D. C. 20036

GPU Service Corporation
ATTN: Mr. E. G. Wallace
Licensing Manager
260 Cherry Hill Road
Parsippany, New Jersey 07054

Anthony Z. Roisman
Natural Resources Defense Council
917 15th Street, N. W.
Washington, D. C. 20006

Steven P. Russo, Esquire
248 Washington Street
P. O. Box 1060
Toms River, New Jersey 08753

Joseph W. Ferraro, Jr., Esquire
Deputy Attorney General
State of New Jersey
Department of Law and Public Safety
1100 Raymond Boulevard
Newark, New Jersey 07012

Ocean County Library
Brick Township Branch
401 Chambers Bridge Road
Brick Town, New Jersey 08723

Mayor
Lacey Township
P. O. Box 475
Forked River, New Jersey 08731

Commissioner
Department of Public Utilities
State of New Jersey
101 Commerce Street
Newark, New Jersey 07102

Gene Fisher
Bureau Chief
Bureau of Radiation Protection
380 Scotts Road
Trenton, New Jersey 08628

Mark L. First
Deputy Attorney General
State of New Jersey
Department of Law and Public Safety
Environmental Protection Section
36 West State Street
Trenton, New Jersey 08625

Joseph T. Carroll, Jr.
Plant Superintendent
Oyster Creek Nuclear Generating
Station
P. O. Box 388
Forked River, New Jersey 08731

Director, Technical Assessment
Division
Office of Radiation Programs
(AW-459)
U. S. Environmental Protection
Agency
Crystal Mall #2
Arlington, Virginia 20460

U. S. Environmental Protection
Agency
Region II Office
ATTN: EIS COORDINATOR
26 Federal Plaza
New York, New York 10007

Robert M. Lazo, Esq., Chairman
Atomic Safety and Licensing Board
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

cc
Dr. Hugh C. Paxton, Member
Los Alamos Scientific Laboratory
P. O. Box 1663
Los Alamos, New Mexico 87544

Dr. Paul W. Purdom, Member
Director, Center for Urban
Research and Environmental
Studies
Drexel University
32nd and Chestnut Streets
Philadelphia, Pennsylvania 19104

SYSTEMATIC EVALUATION PROGRAM
PLANT SYSTEMS/MATERIALS
OYSTER CREEK NUCLEAR POWER 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 materials 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, 1979. 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 or fatigue. Furthermore, the Standard Review Plan (Section 4.5.2) does not address neutron irradiation nor fatigue resistance of the materials of the reactor internal structures.

As a result of incidents of intergranular stress corrosion cracking in piping in the BWR system, special study groups were formed by NRC and industry to evaluate the cause, extent and safety implications of the use of sensitized stainless steel in the nuclear steam supply systems. The study groups identified the incidents with the recirculation system bypass lines, the core spray lines, and the control rod drive return lines. It was concluded that the problem was caused by a combination of high total stresses, sensitization of the austenitic stainless steel in the heat affected zones of welds, and the relatively high oxygen content of the coolant.

The NRC study group recommended an augmented inservice inspection program for stainless steel piping, more stringent monitoring of the leak detection system, modification of plant operating practice, and the use of alternate materials immune to intergranular stress corrosion cracking. The study group concluded in NUREG-0531, "Investigation and Evaluation of Stress-Corrosion Cracking in Piping of Light Water Reactor Plants," that intergranular stress corrosion cracking in piping would be detected prior to unstable crack growth because of the adequacy of the inservice inspection program and the leak detection system. Reactor operating experience has validated the leak-before-break concept of piping integrity, and, it was concluded, that through-wall cracks in the piping systems would be detected before they presented a hazard to the health and safety of the public.

The regulatory position on the use of sensitized stainless steel in reactor internal materials is addressed in the Standard Review Plan Section 4.5.2, "Reactor Internal Materials." 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 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 internal structure for the Oyster Creek Nuclear Power Plant is described in Section III-6 of the Facility Description and Safety Analysis Report. The internal components were designed to provide support for the fuel and maintain the required configuration and 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.

The Oyster Creek reactor was designed and fabricated prior to the adoption of Section III of the ASME Boiler and Pressure Vessel Code, which became effective on May 4, 1964. The reactor vessel was designed, fabricated, inspected and tested in accordance with the requirements of Section I of the ASME Boiler and Pressure Vessel Code, 1962 Edition, including Nuclear Code Cases in effect on December 11, 1963 (the date of the purchase order for the vessel). In the instance that specific details of construction were not covered by Section I requirements, Section VIII requirements of the ASME Code were imposed by the General Electric Company on the vessel manufacturer (Combustion Engineering).

A fatigue analysis for the Oyster Creek reactor vessel was made by the General Electric Company to assure that the reactor vessel and internal components would safely withstand all anticipated normal and emergency operating and transient conditions. The method of analysis was based on the method developed for naval reactors and the industrial experience using it. When Section III became effective, the requirements of the Nuclear Code Cases and the method of analysis were adopted with only minor changes. However, the allowable stresses under Section III exceed those permitted under Sections I and VIII as well as the fatigue curves used in the design specification. Therefore, the Oyster Creek reactor was designed to requirements either more conservative or equivalent to the requirements of Section III of 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 internal components were identified in the Facility Description and Safety Analysis Report as Type 304 stainless steels, Inconel, and minor quantities of special purpose alloys, such as 17-4 PH alloy in the H-1100 heat treated condition and Stellite. These materials have proven adequate for reactor internal construction as a result of extensive tests, prior usage, and satisfactory performance.

During the evaluation of the application for Full Term License in 1974, the licensee discussed compliance with 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." These guides were promulgated after the construction of the Oyster Creek Nuclear Power Plant was completed, and the recommendations of the guides were not explicitly followed. The construction records indicate that welding procedures, qualifications, and acceptance standards meet the requirements of Section VIII and Section IX of the ASME Boiler and Pressure Vessel Code and the USAS B31.1.0 Code for Power Piping.

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 assume for this evaluation that the Oyster Creek Nuclear Power Plant 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 generally associated with intergranular stress corrosion cracking. The threshold stress values are near or greater 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 Oyster Creek Nuclear Power Plant in order to correlate reactor internal materials failure and the use of sensitized stainless steel in the components. The events are summarized as follows:

> During field hydrostatic testing of the pressure vessel in 1967, a leak was observed near one of the control rod drive penetrations. Investigation showed that several welds had been furnace sensitized, causing the failure.
>
> The inservice inspection program conducted in October, 1978, revealed cracks (indications) in the core spray spargers. During the 1980 refueling outage, additional cracks were discovered by a significantly improved procedure for inservice inspection of the spargers.

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 by the use of sensitized stainless steel. The inservice inspection and testing procedures detected failures in the control rod drive penetrations and the core spray spargers. Failures in these components were considered by the NRC Pipe Crack Study Group in NUREG-0531, "Investigation and Evaluation of Stress Corrosion Cracking in Piping of Light Water Reactor Plants."

The inservice inspection program for the reactor internal components for the Oyster Creek Nuclear Power Plant will be conducted in the current inspection interval to the requirements of Section XI of the ASME Boiler and Pressure Vessel Code, 1977 Edition, including Summer 1978 Addenda. This program is in compliance with paragraph (g) of Section 50.55a, 10 CFR Part 50. It will assure that the integrity of the included components is maintained during reactor operation.

We conclude from our review that the stainless steel materials in the reactor internal components are sensitized and that there is an increased susceptibility to intergranular stress corrosion cracking due to operation in an oxygen saturated water environment. However, 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 component will be detected by inservice inspection procedures prior to component failure. We conclude that the use of sensitized stainless steel in the reactor internal components at the Oyster Creek Nuclear Power Plant is not a hazard to the health and safety of the public.