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UNITED STATES
ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

December 13, 1973

Docket No. 50-219

Jersey Central Power & Light Company
ATTN: Mr. I. R. Finfrock, Jr.
Vice President - Generation
Madison Avenue at Punch Bowl Road
Morristown, New Jersey 07960

Gentlemen:

The Exxon Nuclear Company, Inc. (Exxon) report XN-174, "Densification Effects on Exxon Nuclear Boiling Water Reactor Fuel," November 1973, has been submitted to provide a basis for further review of the effects of BWR fuel densification.

The Regulatory staff considers that modifications to the "Exxon Nuclear Model for Fuel Densification" transmitted to you in our letter of July 16, 1973, are appropriate. The enclosure represents the staff's current conclusions on Exxon BWR fuel densification.

We have requested, by telephone, that Exxon supplement XN-174 with calculations to determine the consequences of fuel densification, using the guidance provided in the enclosure. Exxon has indicated that they expect to complete these calculations on December 13, 1973. It is requested that you provide the necessary analyses and other relevant data for determining the effects of fuel densification on normal operation, anticipated transients, and accidents, including the postulated loss-of-coolant accident, using the guidance in the enclosure. If the analyses indicate that changes in operating conditions are warranted, you should submit proposed changes to your Technical Specifications with the analyses.

To meet our current schedule for completing our review of the Exxon fuel in the Oyster Creek Cycle 3 core and to allow possible early action for your facility, it is requested that you provide your response by noon EST

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December 14, 1973. The response concerning General Electric fuel requested in our letter dated December 5, 1973, may be included with the submittal concerning Exxon fuel.

Sincerely,

Donald J. Skovholt
Assistant Director for
Operating Reactors
Directorate of Licensing

Enclosure:
Modified Exxon Model for Fuel
Densification (12/13/73)

cc w/enclosure:

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MODIFIED EXXON NUCLEAR MODEL FOR FUEL DENSIFICATION

The Exxon Nuclear fuel densification model is described in References 1 and 2. The Exxon model when modified as described below is considered to be suitably conservative for the evaluation of densification effects in BWR fuel.

Possible effects of fuel densification are: (1) power spikes due to axial gap formation; (2) increase in LHGR because of pellet length shortening; (3) creep collapse of the cladding due to axial gap formation; and (4) changes in stored energy due to increased radial gap size. Similarly, the Exxon model for fuel densification consists of four parts: power spike model, linear heat generation model, clad creep collapse model and stored energy model. The required modifications to each of these models are listed below.

Power Spike Model

The Exxon power spike model is acceptable as it is described in Reference 1 as long as it is used in conjunction with a maximum gap size given by the following equation:

$$\Delta L = \left(\frac{0.965 - \rho i}{2} + 0.004\right)L$$

where ΔL = maximum axial gap length

L = fuel column length

ρi = mean value of measured initial pellet density (geometric)

0.004 = allowance for irradiation induced cladding growth and axial strain caused by fuel-clad mechanical interaction.

Linear Heat Generation Model

The following expression should be used to calculate the decrease in fuel column length in determinations of the linear heat generation rate.

$$\Delta L = \frac{0.965 - \rho i}{2} \; L$$

where: ΔL = decrease in fuel column

L = fuel column length

ρi = mean value of measured initial pellet density (geometric)

Credit can be taken for fuel column length increase due to thermal expansio and for the actual measured fuel column length.

Clad Creep Collapse Model

Examination of exposed BWR fuel rods and Regulatory staff calculations show that the clad collapse will not occur in typical BWR fuel during the first cycle of operation. Consequently, no additional creep collapse calculations are required for the first cycle of typical BWR fuel.

For reactors in subsequent cycles of operation the Exxon creep collapse model, described in Reference 3 should be used with the following assumption

1. A conservative value should be used for the clad temperature. Axial temperature variations in the vicinity of a fuel gap as affected by thermal radiation from the ends of the pellets and by axial heat conduction should be taken into account. Effects from any buildup of oxide and crud on the clad surfaces should also be considered.
2. The calculations should be made for the fuel rod having the worst combination of fast neutron flux and clad temperature.
3. No credit should be taken for fission gas pressure buildup.
4. No credit should be taken for end effects. An infinitely long, unsupported length of cladding should be assumed.

5. Conservative values for clad wall thickness and initial ovality should be used. An acceptable approach is to use the two standard deviation limits of as fabricated dimensions.

Stored Energy Model

The Exxon model for calculating pellet-to-cladding heat transfer coefficients GAPEXX is described in reference 2. This model is acceptable for use in the stored energy calculation with the following modifications:

1) Deletion of interfacial pellet to cladding heat transfer term (Ross and Stoute pellet-to-cladding model) in GAPEXX.

2) Replacement of the parabolic-time-dependent kinetic densification model with a logarithmic-time-dependent model expressed below:

For fuels with nominal fabricated densities 92% T.D. or greater,

$$\Delta\rho = \Delta\rho_{max}[0.2198 \ln(t) - 0.5184], \qquad 20 < t < 1000;$$

$$\Delta\rho = \Delta\rho_{max}[0.007t], \qquad t < 20;$$

$$\Delta\rho = \Delta\rho_{max}, \qquad t > 1000.$$

In the above expression, t is effective full power hours (EFPH), and $\Delta\rho_{max}$ is based on densification to 96.5% T.D. (geometric) using a lower 2σ value for the initial density.

3) The fuel pellet located at the most critical position for normal operation, anticipated transients and postulated accident conditions should be analyzed with the densified pellet size as given in item 2).

4) Since the assembly average stored energy is one of the most important inputs to BWR LOCA evaluation, a Technical Specification limit should be imposed on maximum permitted assembly power.

References

1. Oyster Creek Nuclear Generating Station, Docket No. 50-219, Supplement No. 3 to Facility Change Request No. 4, April, 1973.

2. "Densification Effects on Exxon Nuclear Boiling Water Reactor Fuel," XN-174, November, 1973.

3. Merckx, K. R.: "Cladding Collapse Calculational Procedure," JN-72-23, November 1, 1972.

Jersey Central Power & Light Company

MADISON AVENUE AT PUNCH BOWL ROAD • MORRISTOWN, N. J. 07960 • 201-539-6111

December 13, 1973

Mr. A. Giambusso
Deputy Director for Reactor Projects
Directorate of Licensing
United States Atomic Energy Commission
Washington, D. C. 20545

Dear Mr. Giambusso:

Subject: Our Letter Dated December 7, 1973 re.
Oyster Creek Station
Docket No. 50-219
Main Steam Isolation Valve Inspection and Repair

I am enclosing an original and forty copies of a revised letter dated December 7, 1973.

Would you please destroy the copies previously sent to you.

Very truly yours,

Donald A. Ross
Manager, Nuclear Generating Stations

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cc: Mr. J. P. O'Reilly, Director
Directorate of Regulatory Operations, Region I

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Jersey Central Power & Light Company

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MEMBER OF THE General Public Utilities Corporation GPU ELECTRIC POWER COMPANIES SYSTEM

December 7, 1973
(Revised)

Mr. A. Giambusso
Deputy Director for Reactor Projects
Directorate of Licensing
United States Atomic Energy Commission
Washington, D. C. 20545

RECEIVED DEC 19 1973 U.S. ATOMIC ENERGY COMMISSION

Dear Mr. Giambusso:

Subject: Oyster Creek Station
Docket No. 50-219
Main Steam Isolation Valve Inspection and Repair

In a letter dated September 21, 1973, we reported that main steam isolation valve NS03B failed to meet the acceptable leakage rate criterion specified in Technical Specifications 4.5.F.1.D, in a test conducted during the September 1973 plant outage. The letter also outlined a course of action that would be followed to visually and dimensionally check valve NS03B to determine the failure mechanism. The purpose of this letter is to submit a summary report on the results of the extensive inspection conducted and on the repairs that were made to renovate the valve.

Before proceeding with the inspection of the valve, representatives of General Electric Company, Atwood & Morrill Company and MPR Associates met with Jersey Central Power & Light Company personnel to discuss possible failure mechanisms and methods of repair. Until the latest development, it was believed that the lack of straightness of the valve stem was responsible for the failure of NS03B to pass the air tests subsequent to power operation. However, the installation of a specially manufactured stem in the valve during the June 1973 plant outage diminished the probability that a bowed stem was the cause of the recurring leakage problem. Procedures and tests for the disassembly and reassembly of the valve and for obtaining the desired measurements and dimensional data were developed in a joint effort with the vendor, Atwood & Morrill Company. An Atwood & Morrill representative was present for the entire inspection and repair program.

Prior to disassembling the valve, special instruments were installed to indicate and/or record ambient temperature, valve body temperature, stem travel, and operator air and oil pressures. The valve was opened and closed several times to verify the repeatability of measurements. Data was obtained to enable plots of operator air pressure versus open-close stem travel to be made. During the disassembly of NS03B, dimensional data was obtained on all

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critical parts. The data recorded during the inspection, along with the procedures followed for the disassembly and reassembly of the valve, are on file at the plant. Copies of this information will be made available upon request.

An analysis of the inspection data revealed that there was excess clearance between the main poppet guides and the valve body guides. The diameter of the circle formed by the body guides was measured at three elevations in the casing. The dimensions recorded were as follows: Top, 23.515 inches; Middle, 23.510 inches; and Bottom, 23.504 inches. The diameter of the circle formed by the main poppet guide pads was found to be 23.456 inches. It is believed that the large clearance between poppet and body guides, as much as .048 inches near the valve seat, permitted excessive misalignment of the poppet on the valve seat, a condition that is not favorable for a tight seal. The desired guide clearance is .027 to .030 inches on the diameter.

The repair consisted of building up the main poppet guide pads to give the desired clearance. First, the guide lugs were cleaned by machining and then built up 1/8" with Stellite 21 overlay. The guide pads were then remachined and dye checked. The diameter of the circle formed by the remachined guide pads is 23.4815 inches. The weld overlay was done in accordance with Atwood & Morrill Company Specification No. PTH-3, Revision No. 8, entitled "Procedure Specification for Gas Tungsten Arc Hard Surfacing of Valve Trim". An Atwood & Morrill welder, certified to the procedure, performed the overlay work. The fix was made to the poppet guide pads rather than to the body guides, which indicated wear, because there was no mechanism available for dimensional boring of the body guides.

Additional work performed on valve NS03B to improve sealing performance consisted of machining and lapping the pilot poppet and the main poppet seats to 46° for line contact, and machining 1/4" off of the spring seating surface on the stem spring plate to reduce the possibility of metal-to-metal contact of the spring coils while in compression.

Following the completion of the maintenance and repair work on main steam isolation valve NS03B, tests were initiated to determine the leakage rate of each of the four main steam isolation valves. The two valves, NS04A and NS04B, outside the drywell failed to meet acceptable leakage rate requirements. (Reported to the Directorate of Licensing by letter dated October 12, 1973). The leakage path in both valves was identified as being the stem packing. As a precautionary measure, all four isolation valves were repacked. In subsequent tests, the leakage rate of each of the four valves was found to be nondetectable, i.e., <0.1 SCFH.

Two major problems have now been identified and corrected in main steam isolation valve NS03B. The fixes consisted of (1) replacing the original valve stem with one that was manufactured to new specifications and (2) reducing the poppet guide to body guide clearance. We expect that the performance of this valve will be much improved in the future.

Enclosed are forty copies of this report.

Very truly yours,

Donald A. Ross

Donald A. Ross
Manager, Nuclear Generating Stations

cs

cc: Mr. J. P. O'Reilly, Director
Directorate of Regulatory Operations, Region I