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

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Roger S. Boyd, Assistant Director
for Reactor Projects, DRL
THRU: Robert L. Tedesco, Chief, RPB-2, DRL

OYSTER CREEK UNIT NO. 1 ACRS MEETING - DOCKET NO. 50-219

An ACRS meeting was held on November 1, 1968 to discuss Jersey Central's Oyster Creek Unit No. 1. Our Report No. 4, concerning the balance of plant items, was the basis for the meeting. The major topics discussed at the meeting are outlined below.

Discussion

General Electric (GE) made a presentation of the current Oyster Creek ECCS. GE used the results of the new model for evaluating the amount of metal-water reaction that would occur during a postulated loss-of-coolant accident. The depth to which the clad would react was calculated based on these predicted metal-water reactions. The maximum depth of reaction was calculated to be less than 1 mil and compared favorably with the results of the ANL experiments. That is, the results of the ANL experiments show that cladding failure by brittle fracture does not occur if the depth of reaction is less than 5 mils. (If the design basis model were used, about 5 times more metal-water reaction would be predicted, but would not result in reacting the cladding to a depth of 5 mils.) GE also discussed the results of the pipe rupture study; the purpose being to use these results as a basis for establishing a limit on allowable primary system (Technical Specifications) leak rate.

GE summarized the chronology associated with the placement of the emergency condenser steam supply line isolation valves. Apparently GE has on record a trip report, dated September 4, 1964, on a meeting with the staff wherein placement of both valves external to the containment was noted. The report, however, did not include a list of attendees. Records with regard to the fabrication and installation of these valves have been incorporated into the site files. Based on the review of the various records, GE stated that the valves are free of defects.

GE stated that the rod worths used in the rod drop analysis are based on a worst case analysis and would not increase even if a rod remains in a fixed position for extended periods of time. GE has not yet developed a requirement on reactivity anomalies for incorporation into the Technical Specifications.

The program for vibrational testing of the reactor pressure vessel was outlined. Cold and hot operational tests of the vessel are planned. Cold testing of the vessel is directed at evaluating the performance of the bottom end; i.e., the lower support system in the pressure vessel. Hot

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operational testing will demonstrate the performance of 'upper components' such as the shroud and steam separators. However, since the dryers are supported directly from the vessel and since strain gages or accolometers are not to be installed, it is not clear that the adequacy of this component will be demonstrated.

Capability to withstand the effects of hydrogen evolution resulting from metal-water reaction as well as the effects of radiolytic decomposition were discussed. Based on GE's model, an equilibrium condition is predicted to occur when approximately 25 moles of hydrogen (3.5% of the containment volume) are generated by radiolytic decomposition. This amount of hydrogen would not result in a flammable mixture. Preliminary results of tests on the Humboldt Bay reactor indicate that the production rate of hydrogen during the radiolytic process is lower (by about a factor of 10) than the rate used in the analytical model. Results of these tests will be reported in about three months. GE also analyzed the effects of hydrogen burning in the containment. Using what GE considers as a reasonable model for the event, it showed that even if 170 moles of hydrogen were burned the design pressure of the containment would not be exceeded. The value of 170 moles represents the amount of hydrogen that could react with oxygen if the containment were assumed to contain air.

A Dresden 2/3 type main steam line isolation valve has been subjected to simulated accident tests at the Stateline plant. The test series is not complete, but 35 blowdown tests were performed. Steam (up to 1200 #/sec) and water (up to 3900 #/sec) were used in varying mixtures to simulate accident conditions. Operation of the valve was considered satisfactory and measured loads were less than the analytically predicted valves.

At the conclusion of the meeting the Committee informed the applicant that it wanted to review the reactor vessel repair program, Technical Specification, in-service inspection and basis for plant turnover before it completes its review.

Essentially the same topics were discussed during a meeting with the applicant on the day before the ACRS meeting, October 31, 1968. Therefore, a separate report on the minutes of that meeting will not be issued. In order to complete the record, however, the attendees for the meeting are attached.

GE made available for the Committee's review the summary of the separation criteria applicable for Oyster Creek. Discussion of this topic was deferred to a subsequent meeting.

Attachments:
1. List of Attendees
2. Oyster Creek Summary

Victor Stello, Jr.
Division of Reactor Licensing

cc: P. A. Morris — Branch Chiefs,
F. Schroeder — RP
S. Levine — Compliance (2)
R. DeYoung — Attendees

ATTENDEES

JERSEY CENTRAL POWER & LIGHT COMPANY

OCTOBER 31, 1968

Jersey Central

G. H. Ritter
I. R. Finfrock, Jr
D. E. Hetrick
T. J. McCluskey
D. R. Rees
G. J. Trowbridge (Counsel)
J. K. Pickard (Counsel)

General Electric

P. A. Ianni
R. C. Holt
T. E. Bloom
W. L. Fiock
J. Barnard
S. Taggart
H. Brammer

AEC

R. S. Boyd
R. Tedesco
V. Stello, Jr.
D. Thompson
H. J. Richings
W. Lowe
R. J. Mattson
M. A. Taylor
D. Sullivan
C. Parr

Burns & Roe

G. A. Lari