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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 19 TO FACILITY LICENSE NO. DPR-2

COMMONWEALTH EDISON COMPANY

DRESDEN NUCLEAR POWER STATION UNIT 1

DOCKET NO. 50-10

INTRODUCTION

By letter dated September 7, 1976, the Commonwealth Edison Company (CECo) requested an amendment to Facility License No. DPR-2 for the Dresden Station - Unit No. 1. The proposed amendment involves revision to the Technical Specifications concerning surveillance tests to verify control rod coupling.

DISCUSSION AND EVALUATION

In 1974, a control rod uncoupling event occurred at Dresden Station - Unit No. 1 following a refueling outage. Four rods were found to be uncoupled as a result of improper control rod installation.

The bottom end of each control blade in Dresden Unit 1 is connected to a control rod drive by a bayonet connection that requires a quarter turn after coupling to lock the rod to the drive. Control rod movement is such that the rods are driven upward into the core to insert neutron absorber material to decrease the core reactivity and downward to remove the neutron absorber and increase the reactivity. When the reactor vessel head is removed the upper ends of the fully inserted control rods are visible and may be engaged by the reactor refueling grapple. When a control rod is coupled to the rod drive at the lower end it may be pulled upward by the refueling grapple, if the rod under-piston pressure is zero. The control rod drive position indicator will move to indicate the rod drive motion. If a control rod is not coupled to the rod drive it can be pulled upward without moving the control rod drive and in this case the rod drive indicator will not move.

8009100 934

Two tests were performed to verify coupling after the 1974 refueling outage. One test (not required by the specifications) was a rod pull test on fully inserted rods using the reactor refueling grapple. This test was performed with the reactor head off. The results of all tests by this method were satisfactory. A second test (required by the specifications) was performed with the reactor head installed by verifying neutronic response of the nuclear instrumentation with individual rod movement. The four uncoupled rods were found by this test method.

Following investigation of the event, the licensee revised their control rod blade pull test to incorporate a more definitive rod coupling test as described in the CECo Control Rod Uncoupling Report - November 12, 1974. The test involves withdrawal of the control rod one notch with the control rod drive, then returning the control rod back to the fully inserted position by the reactor refueling grapple and observing the rod position indicator movement to verify coupling. If the rod drive indicator moves, it confirms that the rod drive and blade are coupled.

An uncoupled control rod may follow the control rod drive downward. However, if the blade sticks in position it may not follow the rod drive downward and remain suspended in the core. If this stuck control blade should subsequently fall after the control rod drive has been withdrawn it would unexpectedly insert reactivity resulting in a control rod drop accident, therefore the NRC requested by letter dated July 20, 1976, that CECo propose a technical specification incorporating the rod coupling verification technique described in their November 12, 1974 report. The licensee's September 7, 1976 request is in response to our letter.

We have reviewed the licensee's proposal and determined that it provides additional assurance that a control rod is not decoupled from its drive. In addition, the test is performed with the reactor vessel head removed (i.e., with the reactor shutdown) which precludes having the reactor critical with an uncoupled rod. The existing coupling test (nuclear instrument check after criticality) would serve as a backup for the proposed test.

On the basis of the above, we conclude that the proposed changes to the specifications provide additional assurance that the health and safety of the public is protected and therefore are acceptable.

ENVIRONMENTAL CONSIDERATIONS

We have determined that the amendment does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendment involves an action which is insignificant from the standpoint of environmental impact and pursuant to 10 CFR §51.5(d)(4), that an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of the amendment.

CONCLUSION

We have concluded, based on the considerations discussed above, that: (1) because the amendment does not involve a significant increase in the probability or consequences of accidents previously considered and does not involve a significant decrease in a safety margin, the amendment does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Date: December 14, 1976