Since all employees were separated in October 1976 during the existence of an active certification; a new investigation would serve no purpose. Consequently, the investigation has been terminated.

Signed at Washington, D.C. this 20th day of April 1978.

MARVIN M. FOOKS,
Director, Office of Trade Adjustment Assistance.

[FR Doc. 78-11651 Filed 4-27-78; 8:45 am]

[7590-01]

NUCLEAR REGULATORY COMMISSION

[Docket No. 50-313]

ARKANSAS POWER & LIGHT CO., ARKANSAS NUCLEAR ONE—UNIT NO. 1

Order for Modification of License

I

The Arkansas Power & Light Co. (the licensee), is the holder of Facility Operating License No. DPR-51 which authorizes the operation of the nuclear power reactor known as Arkansas Nuclear One—Unit No. 1 (the facility) at steady reactor power levels not in excess of 2568 megawatts thermal (rated power). The facility consists of a Babcock & Wilcox Co. designed pressurized reactor (PWR) located at the licensee's site in Pope County, Ark.

II

In accordance with the requirements of the Commission's ECCS Acceptance Criteria, 10 CFR 50.46, the licensee submitted on July 9, 1975, an ECCS evaluation for the facility. The ECCS performance submitted by the licensee was based upon an ECCS Evaluation Model developed by the Babcock & Wilcox Co. (B&W), the designer of the Nuclear Steam Supply System for this facility. The B&W ECCS Evaluation Model had been previously found to conform to the requirements of the Commission's ECCS Acceptance Criteria, 10 CFR Part 50.46 and Appendix K. The evaluation indicated that with the limits set forth in the facility's Technical Specifications, the ECCS cooling performance for the facility would conform with the criteria contained in 10 CFR 50.46(b) which govern calculated peak clad temperature, maximum cladding oxidation, maximum hydrogen generation, coolable geometry and long-term cooling.

On April 12, 1978, B&W informed the NRC that it had determined that in the event of a small break LOCA on the discharge side of a reactor coolant pump, high pressure injection (HPI) flow to the core could be reduced somewhat. Subsequent calculations indicated that in such a case the calculated peak clad temperature might exceed 2,200 F.

Previous small break analyses for B&W 177 fuel assembly (FA) lowered loop plants had identified the limiting small break to be in the suction line of the reactor coolant pump. For this assumed break location all high pressure ECCS flow injected into the system would add to the reactor coolant inventory once the coolant level decreased to the height of the reactor vessel nozzles. Recently, B&W concluded that previous analyses did not consider that some of the high pressure injection (HPI) flow would be lost if the break were located in the discharge line. As a result, recent analyses have shown that the discharge line break is more limiting than the suction line break.

The Arkansas Nuclear One—Unit No. 1 plant has an ECCS configuration which consists of two high pressure injection (HPI) trains. Each train has a HPI pump and the train injects into two of the four reactor coolant system (RCS) cold legs on the discharge side of the RCS pump. (There is also a third HPI pump installed.) The two parallel HPI trains are connected but are kept isolated by manual valves (known as the cross-over valves) that are normally closed. Upon receiving a safety injection signal the HPI pumps are started and valves in the four injection lines are opened. Assuming loss of offsite power and the worst single failure (failure of diesel to start) only one HPI pump would be available and two of the four injection valves would fail to open.

If a small break is postulated to occur in the RCS piping between the RCS pump discharge and the reactor vessel, the high pressure injection flow injected into this line (about half of the output of one high pressure pump) could flow out the break. Therefore, for the worst combination of break location and single failure, only one-half of the flow rate of a single high pressure ECCS pump would contribute to maintaining the coolant inventory in the reactor vessel. This situation had not been previously analyzed and B&W had indicated that the limits specified in 10 CFR 50.46 may be exceeded.

B&W has stated that they have analyzed a spectrum of small breaks in the pump discharge line and have determined that to meet the limits of 10 CFR 50.46, operator action is required to open the two manual operated crossover valves and to manually align the two motor driven isolation valves which had failed to open. This would allow the flow from the one HPI pump to feed all four reactor coolant legs. B&W has assumed that 30 percent of the flow would be lost through the break and 70 percent would refill the core. The licensee has committed to provide for the necessary operator actions within the required time frame. That is, in the event of a small break and a limiting single failure, manual action will be taken to begin opening these valves within five minutes and have them fully opened and an adequate flow split obtained within 10 minutes. The analyses performed by B&W assumed that the flow split was established at 650 seconds by operator action. We conclude that the analyses are a reasonable approximation of the operator action that actually will be taken, provided specific procedures are prepared and followed to assure such action.

B&W has stated that a 0.1 ft.^{2} discharge line break, with the aforementioned operator actions, is the most limiting case. To arrive at this conclusion, B&W has performed analyses at break sizes of 0.5, 0.2, 0.1, and 0.04 ft.^{2}, with and without the assumed operator action. The results for a power of 2568 Mwt, which were obtained using an approved Appendix K model for blowdown, indicate core uncovery for about 250 seconds for the 0.1 ft.^{2} limiting break. For this break size B&W has conservatively estimated the peak clad temperature to be approximately 1,200 F; well below the limits of 10 CFR 50.46(b).

B&W has indicated the manner in which the calculational methods have been revised and has indicated that their revised calculations are wholly in conformance with the requirements of 10 CFR 50.46. However, B&W has not yet had the opportunity to fully present the result of its calculations to the licensee for submittal to the NRC staff, and the staff has accordingly not had the opportunity to fully assess the new calculations. Until the licensee and the staff have an opportunity to review the B&W revised calculations, the staff has recommended and the licensee has agreed, that operating conditions be limited to a range in which ECCS performance for small break conditions is less sensitive to specific calculation inputs.

For this facility, with operation up to 2311 Mwt, ECCS performance calculations for the limiting small break does not even result in core uncovering, if appropriate operator action is properly taken (as described above), thus providing a very substantial margin on peak clad temperature below the limits of 10 CFR 50.46(b).

Therefore, until the staff has had the opportunity to fully assess the B&W revised calculations, operation of the facility at the power level specified in this Order, and in accordance with the operating procedures specified in this Order, will assure that the ECCS will conform to the performance requirements of 10 CFR 50.46(b). Accordingly, such limits provide reasonable assurance that the

8004280 830

public health and safety will not be endangered. Upon notification by the NRC staff, the licensee committed to provide the staff with B&W's reevaluation of ECCS performance applicable to the licensee's facility as promptly as possible, to limit operation as specified herein, and to submit a technical specification requiring appropriate operating procedures to assure required operator action as discussed herein. Such procedures were described and the commitments confirmed by the licensee's letter of April 17, 1978, supplemented by letters dated April 21, 1978. The staff believes that the licensee's action, under the circumstances, is appropriate and that this action should be confirmed by NRC Order. Upon satisfactory completion of our assessment of the revised evaluation, we will accordingly modify the authorization to operate the facility.

IV

Copies of the following document are available for inspection at the Commission's Public document Room at 1717 H Street, Washington, D.C. 20555, and are being placed in the Commission's local public document room at the Arkansas Polytechnic College, Russellville, Ark.

(1) Letters from Mr. Donald A. Rueter to Mr. R. W. Reid, Chief Operating Reactors Branch No. 4, dated April 17 and 21, 1978, and from Mr. Daniel H. Williams to Mr. R. W. Reid dated April 21, 1978.

Accordingly, pursuant to the Atomic Energy Act of 1954, as amended, and the Commission's Rules and Regulations in 10 CFR Parts 2 and 50: *It is ordered,* That Facility Operating License No. DPR-51 is hereby amended by adding the following new provisions:

(1) As soon as possible, the licensee shall submit a reevaluation wholly in conformance with 10 CFR 50.46 of ECCS cooling performance calculated in accordance with the B&W Evaluation Model for operation with operating procedures described in its letters of April 17, 1978, and April 21, 1978,

(2) Until further authorization by the Commission, the power level shall not exceed 2311 Mwt, and

(3) Until further authorization by the Commission, the licensee shall operate in accordance with the procedures described in its letter of April 17, 1978, supplemented by letters dated April 21, 1978.

Dated at Bethesda, Md., this 21st day of April 1978.

For the Nuclear Regulatory Commission.

VICTOR STELLO, Jr.,
Director, Division of Operating Reactors, Office of Nuclear Reactor Regulation.

[FR Doc. 78-11586 Filed 4-27-78; 8:45 am]

[7590-01]

[Docket No. 50-265]

COMMONWEALTH EDISON CO. AND IOWA-ILLINOIS GAS AND ELECTRIC CO.

Issuance of Amendment to Facility Operating License

The Nuclear Regulatory Commission (the Commission) has issued Amendment No. 34 to Facility Operating License No. DPR-30, issued to the Commonwealth Edison Co. (acting for itself and on behalf of the Iowa-Illinois Gas and Electric Co.), which revised Technical Specifications for operation of the Quad Cities Station Unit 2 (the facility) located in Rock Island County, Ill. The amendment is effective as of the date of issuance.

This amendment extends the allowable period of reactor operation with Loop 2A of the Containment Cooling Mode of the RHR System inoperable for 7 days beyond April 24, 1978 provided that a visual inspection is performed daily to assure that proper valve alignment and system integrity is maintained in the "B" RHR loop.

The application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment. Prior public notice of this amendment was not required since the amendment does not involve a significant hazards consideration.

The Commission has determined that the issuance of this amendment will not result in any significant environmental impact and that pursuant to 10 CFR § 51.5(d)(4) an environmental impact statement, or negative declaration and environmental impact appraisal need not be prepared in connection with issuance of this amendment.

For further details with respect to this action, see (1) the application for amendment dated April 13, 1978, (2) Amendment No. 34 to License No. DPR-30, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street NW., Washington, D.C. and at the Moline Public Library, 504 17th Street, Moline, Ill. 61265. A copy of items (2) and (3) may be obtained upon request addressed to the U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Director, Division of Operating Reactors.

Dated at Bethesda, Md., this 21st day of April 1978.

For the Nuclear Regulatory Commission.

GEORGE LEAR,
Chief, Operating Reactors Branch No. 3, Division of Operating Reactors.

[FR Doc. 78-11567 Filed 4-27-78; 8:45 am]

[7590-01]

[Docket No. 50-320]

METROPOLITAN EDISON CO. ET AL.

Granting of Relief From ASME Section XI Inservice Inspection (Testing) Requirements

In the matter of Metropolitan Edison Co., Jersey Central Power Light Co. and Pennsylvania Electric Co.

The U.S. Nuclear Regulatory Commission (the Commission) has granted relief from certain requirements of the ASME Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components" to the Metropolitan Edison Co. The relief relates to the inservice testing program for the Three Mile Island Nuclear Station Unit 2 (the facility), located in Dauphin County, Pa. The ASME Code requirements are incorporated by reference into the Commission's rules and regulations in 10 CFR Part 50. The relief is effective as of its date of issuance.

Relief is granted until the start of commercial operation from compliance with certain inservice testing requirements determined to be impractical for the facility because compliance would result in hardships and unusual difficulties without a compensating increase in the level of quality or safety.

The request for relief complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the letter granting relief. Prior public notice of this action was not required since the granting of this relief from ASME Code requirements does not involve a significant hazards consideration.

The Commission has determined that the granting of this relief will not result in any significant environmental impact and that pursuant to 10 CFR § 51.5(d)(4) an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with this action.

For further details with respect to this action, see (1) the letter from the Metropolitan Edison Co. dated January 3, 1978, and (2) the Commission's letter to the Metropolitan Edison Co. dated April 21, 1978.