UNITED STATES
NUC EAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

OFFICE OF THE SECRETARY

April 28, 1978

Director
Office of the Federal Register
National Archives and Records Service
Washington, D.C. 20403

Dear Sir:

Enclosed for publication in the Federal Register are an original and two certified copies of a document entitled:

SACRAMENTO MUNICIPAL UTILITY DISTRICT

Docket No. 50-312

ORDER FOR MODIFICATION OF LICENSE

Publication of the above document at the earliest possible date would be appreciated.
This material is to be charged to requisition number D-149.

Sincerely,

Samuel J. Chilk
Secretary of the Commission

Enclosures:
Original and 2 certified copies

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UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

In the Matter of	)	
	)	
SACRAMENTO MUNICIPAL UTILITY DISTRICT	)	Docket No. 50-312
	)	
Rancho Seco Nuclear Station, Unit No. 1	)	

ORDER FOR MODIFICATION OF LICENSE

I.

The Sacramento Municipal Utility District (the licensee), is the holder of Facility Operating License No. DPR-54 which authorizes the operation of the nuclear power reactor known as Rancho Seco Nuclear Station, Unit No. 1, (the facility) at steady reactor power levels not in excess of 2772 megawatts thermal (rated power). The facility consists of a Babcock and Wilcox Company designed pressurized water reactor (PWR) located at the licensee's site in Sacramento County, California.

II.

In accordance with the requirements of the Commission's ECCS Acceptance Criteria, 10 CFR 50.46, the licensee submitted on July 8, 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 Company (B&W), the designer of the Nuclear Steam Supply System

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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 2200F.

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. Recent analyses have shown that the discharge line break is more limiting than the suction line break.

The Rancho Seco Nuclear Station, Unit No. 1, 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% of the flow would be lost through the break and 70% 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. To facilitate this operation the licensee has committed to maintain one of the series-connected, manually operated cross-over valves normally open. 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 .15 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 .3, .2, .15, .1, and .04 ft.^{2}. The results, which were obtained using an approved Appendix K model for blowdown, indicate core uncovery for about 500 seconds for the 0.15 ft.^{2} break. For this break size B&W has conservatively calculated the peak clad temperature to be approximately 1760 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 10CFR50.46(b). For other reasons which are not safety-related, however, the plant is limited to a maximum power of about 2080 megawatts thermal until approximately August, 1978. At this lower power level, the safety margin on peak clad temperature will be even greater.

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 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 submit a technical specification requiring appropriate operating procedures to assure required operator action as discussed herein, and affirmed that plant operation was limited to the maximum power level specified herein. Such procedures were described and the commitments confirmed by the licensee's letter of April 14, 1978, supplemented by letter 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 documents 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 Sacramento City-County Library, Sacramento, California.

(1) Letters from J. J. Mattimoe to Mr. R. W. Reid, Chief Operating Reactors Branch #4, dated April 17 and 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-54 is hereby amended by adding the following new provisions:

(1) As soon as possible, the licensee shall submit a reevaluation wholly in conformance with 10CFR50.46 of ECCS cooling performance calculated in accordance with the B&W Evaluation Model for operation with operating procedures described in its letters of April 14, 1978, and April 21, 1978, except that the time for completion of operator action shall be 10 minutes after initiation of the event.

(2) Until further authorization by the Commission, the power level shall not exceed 2080 Mwt, and

(3) Until further authorization by the Commission, the licensee shall operate in accordance with the procedures described in its letter of April 14, 1978, supplemented by letters dated April 21, 1978, except that the maximum time for completion of operator action shall be 10 minutes after initiation of the event.

FOR THE NUCLEAR REGULATORY COMMISSION

Victor Stello, Jr., Director
Division of Operating Reactors
Office of Nuclear Reactor Regulation

Dated at Bethesda, Maryland,
this 26th day of April 1978.