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U.S. NUCLEAR REGULATORY COMMISSION

June 10, 1981

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Christine N. Kohl, Esquire
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In the Matter of
Sacramento Municipal Utility District
(Rancho Seco Nuclear Generating Station)
Docket No. 50-312

Chief Administrative Judge Rosenthal and Administrative Judges Buck and Kohl:

Please find enclosed, for your information, a letter dated March 25, 1981, from Babcock & Wilcox to Sacramento Municipal Utility District ("Licensee") on the subject "Reactor Coolant Pump Suction Small Break LOCA."

B&W's small-break, loss-of-coolant accident analyses and the resultant operator guidelines were the subject of testimony before the Atomic Safety and Licensing Board in this

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proceeding. See, in this docket, Initial Decision (Permitting Continued Reactor Operation), LBP-81-12, 13 N.R.C. ____, slip op. at paragraphs 88-95, 97-101 (May 15, 1981). Licensee had not completed its evaluation of the B&W letter when the Licensing Board issued its Initial Decision.

Licensee has now completed its review of the B&W letter and determined, pursuant to Licensee's own internal procedures, that the information provided in the letter is not reportable to the NRC. Licensee's witnesses have also reviewed the B&W letter and determined that the information provided in the letter does not warrant any change to their testimony previously given before the Licensing Board. Nevertheless, because the B&W letter might be considered to have some bearing on your review of the Licensing Board's decision, I am serving the letter on the Appeal Board, the Licensing Board and the parties.

Respectfully submitted,

Thomas A. Baxter

Thomas A. Baxter
Counsel for Licen

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Enclosure

cc: per Certificate of Service

Babcock & Wilcox

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March 25 1981
File 177/T1.2
ESC-634
SMUD-81-046

MAR 1 1981

Mr D G. Raasch
Manager Generation Engineering
Sacramento Municipal Utility District
6201 S Street
Sacramento, California 95813

Reference: R. W. Ganthner to D. G. Raasch. letter of October 3. 1980

Subject: Reactor Coolant Pump Suction Small Break LOCA

Dear Mr. Raasch:

Following the TMI-2 accident, the NRC requested several small break accident scenarios be evaluated in order to develop operator guidelines for these events. These analyses included scenarios where auxiliary feedwater (AFW) was assumed not to be available at the start of the event. The assumed worst case small break LOCA (less than 0.01 ft.^{2}) for these analyses is located at the reactor coolant pump discharge. This assumption was made because under normal circumstances a greater degree of HPI penetration into the reactor vessel is achieved during a suction line break. The purpose of this letter is to provide some information regarding this worst case assumption as it is affected in the scenario where HPI is not actuated and AFW is delayed. A brief summary of this situation was provided to you in the referenced letter. Specific details are included in this letter.

For pump suction line breaks, under normal circumstances, 100% of the HPI flow enters the reactor vessel. For pump discharge line breaks only about 70% of the HPI flow enters the vessel. However, for the scenario where break size is such that HPI is not automatically initiated and AFW flow is delayed, the rate of system inventory loss before AFW actuation becomes important. During pump discharge breaks a two phase discharge results due to the effect of the reactor vessel internal vent valves. This reduces the rate of system inventory loss. A pump suction break will result in the loss of lower quality fluid which will deplete system inventory at a higher rate. Thus at the time of AFW actuation the RCS inventory will be less for the pump suction line break than for the pump discharge line break.

Analyses which have been conducted for the pump discharge break condition demonstrate that operator actions to start AFW flow in 20 minutes will result in acceptable conditions. However. for the case where there is a pump suction break of a size such that boil dry of the steam generators occurs prior to an RCS pressure decrease to the HPI actuation setpoint. the 20 minute delay in AFW actuation has not been demonstrated to result in an acceptable RCS inventory condition. That is. the 20 minutes of delay analyzed for in the pump discharge break case has not been analyzed for in the pump suction break case. Thus. although there is certainly a delay in AFW actuation which will result in acceptable conditions. the actual time has not been identified for the pump suction break condition.

While the actual analyses for the pump suction break delayed AFW scenario has not been conducted. there is a significant amount of guidance for the operator regarding actuation of HPI and AFW. Following the TMI-2 accident. small break LOCA operating guidelines were developed. These guidelines instruct the operator to ensure that AFW is being delivered to the steam generators and if it is not, to restore feedwater as soon as possible. Additionally. the guidelines require manual actuation of HPI should the system reach saturated conditions. These actions provide for mitigation of the delayed AFW scenario. Additionally. upgrades of the AFW control system have been implemented which would ensure AFW flow in times on the order of one minute. Thus. although the specific analysis has not been conducted. there is adequate reason to believe that current procedures and system characteristics make the identification of the specific time delay superfluous. However. it is not clear what significance the demonstration of a 20 minute operator response time was to the NRC. The licensing significance associated with the demonstration of a 20 minute operator response time is best determined by each utility. Additionally. the AFW upgrades are plant dependent and B&W cannot assess to what extent the probability of this event has been diminished. At present. the following positions appear to be possible resolution paths on this issue:

1. Review the AFW systems and confirm that the small break LOCA with delayed feedwater is a highly unlikely scenario and need not be considered part of the design basis for the plant. Thus. while the analyses performed may not have considered the worst break location for demonstrating the minimum allowable operator response time. the probability of this event along with the generation of the operator guidelines provide adequate assurance that this transient can be safely mitigated.

2. Use the basic position outlined in Item 1 except report to the Commission the potential change in the previously submitted analyses.

3. Perform detailed evaluations of the pump suction small break LOCA with a delay in the delivery of AFW and determine the time frame available to the operator to restore either feedwater or HPI.

It is B&W's position that Item 3 is not necessary. We believe that either Items 1 or 2 are viable alternatives for the ultimate resolution of this issue.

If you have any questions regarding the nature of this concern, please call me (804-384-5111, extension 2420) or R. W. Ganthner (804-384-5111, extension 2751) at our Lynchburg office.

Very truly yours,

DCHolt

D. C. Holt
Engineering Product Manager

cc: R. A. Dieterich
J. T. Janis
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UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING APPEAL BOARD

In the Matter of	)	
	)	
SACRAMENTO MUNICIPAL UTILITY DISTRICT	)	Docket No. 50-312
	)	
(Rancho Seco Nuclear Generating Station)	)	

CERTIFICATE OF SERVICE

I hereby certify that copies of the foregoing letter to the Atomic Safety and Licensing Appeal Board with attachment were served this 10th day of June, 1981 by deposit in the U.S. mail, first class, postage prepaid, upon the following:

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