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Babcock & Wilcox

Power Generation Group

P.O. Box 1260, Lynchburg, Va. 245

Telephone: (804) 384-5111

April 25, 1979

Dr. Roger J. Mattson, Director
Division of Systems Safety
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Dr. Mattson:

In the NRC Staff meeting with B&W and its 177-Fuel Assembly plant owners on Tuesday, April 24, 1979, B&W agreed to describe the scope and status of certain additional analytical work to support the continued safe operation of these plants. The attachment documents this information.

We intend to provide this information to you on or before Friday, May 4, 1979, and suggest that a detailed technical presentation of the information be made to the appropriate Staff engineers at that time with follow-up meetings as deemed necessary.

Very truly yours,

James H. Taylor
James H. Taylor
Manager, Licensing

JHT:dsf

Attach.

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<u>ATTACHMENT</u>

ANALYTICAL WORK TO BE SUBMITTED TO THE STAFF IN SUPPORT OF 177-FA PLANTS BY MAY 4, 1979.

Analyses in this area concern themselves mainly with the prediction of primary system behavior following a loss of main feedwater or other events that result in a loss of feedwater and a delayed establishment of auxiliary feedwater. The analyses can be divided into two general types:

(a) Initial system response which is capable of predicting primary system reactions until such time as two-phase effects become significant. Work in this area will primarily utilize the CADDS computer code. This code is capable of modeling the initial 8 to 10 minutes of such transients.

(b) Long term system response which can deal with primary system evaluation following the establishment of two-phase conditions in the reactor coolant system. These latter analyses will primarily utilize CRAFT code and can be carried out until final resolution of the incident is established.

From our review of anticipated transients, we will provide analyses of those transients which place the greatest constraint on the actions of the auxiliary feedwater system. For these analyses, we will then provide calculations detailing the scenarios which may result from delayed auxiliary feedwater actuation. The delays will be considered in a range from normal actuation to no actuation at all. The specific analyses shown in the following pages will be documented and submitted to the Staff.

CRAFT Analyses (Small Break or Extended Time, 0 Break, Analyses)

Status	Description of Analyses	Result/Expected Result	Length of Evaluation
Done	1. Stuck open PORV, RC pumps on, normal auxiliary feed-water and 2 HPI System trains actuated.	Some two-phase conditions at selected areas within the RCS will be encountered. Natural circulation could be established at any time provided RC pumps were terminated. Reactor core covered at all times with no temperature excursion.	Approx. 1000 s
Done	2. PORV stuck open, RC pumps on, normal auxiliary feedwater, 1 HPIS train	Very similar for Case 1. The degree of two-phase accumulation in the RCS is significantly higher. The return to a subcooled system is delayed but the natural circulation would be established at any time the RC pumps could be terminated.	Approx. 1000 s
Done	3. PORV stuck open, RC pumps on, normal auxiliary feed-water, HPI limited to 200 gpm.	Significant steam accumulation in the RC system such that if RC pumps were terminated after an extended time the collapse of the two-phase fluid and the separation of water and steam would be expected to uncover the core causing severe core damage.	Approx. 1000 s
Progress	4. TMI-2 actual transient best estimate prediction.	This evaluation is being made to demonstrate the capability of the B&W evaluation model to predict the events at TMI-2. Significant void fractions	

Status	Description of Analyses	Result/Expected Result	Length of Evaluation
		within the RC system will develop during the first hour and 45 minutes for which RC pumps were on. Termination of the A loop pumps at that time will result in a steam and water separation. A serious amount of core uncovery resulting in severe core damage will result. It is our intention to carry the analyses through core uncovery and to provide approximate core heatup times to the point of core damage.	
Done	5. .07 ft^2, .02 ft^2 and .01 ft^2 small breaks at the pump discharge without auxiliary feedwater actuation for 20 minutes, no RC pumps and 2 HPI systems	For the .07 ft^2 and .02 ft^2 small breaks significant RCS voids, exist however, the core will never uncover and no cladding temperature excursion will result. These .07 ft^2 .02 ft^2 evaluations were performed for 30 minutes with no auxiliary feedwater and no auxiliary feedwater would be required at any time after a successful resolution of the transient. The .01 ft^2 small break is the most limiting of these transients and requires actuation of either auxiliary feedwater or HPI by [illegible] minutes to prevent core uncovery and the potential for core damage.	Approx. 2000 s

Status	Description of Analyses	Result/Expected Result	Length of Evaluation
In progress	6. A loss of offsite power and loss of main feedwater evaluation with manual initiation of HPI at 20 minutes, no RC pumps, no auxiliary feedwater.	Significant void formation in the primary system but re-establishment of natural circulation would be automatic with the establishment of auxiliary feedwater. This evaluation will also be performed considering further delays in actuation of HPI in order to identify the longest delay possible to prevent core uncovery.	Approx. 3000 s
Done	7. A small break in the steam space of the pressurizer of 1.05^2 in.	This evaluation models the results of a stuck open PORV. Evaluation differs from those shown in items 1, 2, and 3 in that the initiating event is a small break with no RC pumps available, normal auxiliary feedwater and one HPI. Small degree of steam formation in the RCS. Natural circulation can established, no core uncovery or temperature excursion. Incident can be terminated at almost anytime through securing the block valve.	Appeox. 3000 s
In progress	8. References to completed small break LOCA analyses and models.	A listing of small break LOCA analysis licensing submittals which provide further clarification to small break concerns will be furnished.	

Status	Description of Analyses	Result/Expected Result	Length of Evaluation
		within the RC system will develop during the first hour and 45 minutes for which RC pumps were on. Termination of the A loop pumps at that time will result in a steam and water separation. A serious amount of core uncovery resulting in severe core damage will result. It is our intention to carry the analyses through core uncovery and to provide approximate core heatup times to the point of core damage.	
Done	5. .07 ft^2, .02 ft^2 and .01 ft^2 small breaks at the pump discharge without auxiliary feedwater actuation for 20 minutes, no RC pumps and 2 HPI systems	For the .07 ft^2 and .02 ft^2 small breaks significant RCS voids, exist however, the core will never uncover and no cladding temperature excursion will result. These .07 ft^2 .02 ft^2 evaluations were performed for 30 minutes with no auxiliary feedwater and no auxiliary feedwater would be required at any time after a successful resolution of the transient. The .01 ft^2 small break is the most limiting of these transients and requires actuation of either auxiliary feedwater or HPI by 20 minutes to prevent core uncovery and the potential for core damage.	Approx. 2000 s