Babcock & Wilcox

Power Generation Group

P.O. Box 1260, Lynchburg, Va. 24505
Telephone: (804) 384-5111

June 13, 1979

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

Subject: Commitments Resulting from the TMI-2 Incident

Gentlemen:

Attached is an update of the commitments to the NRC, made by Babcock & Wilcox on behalf of our operating plant customers. This is a total list of the commitments completed and outstanding as I now see them. Note that since my letter to you of April 30, 1979, many commitments have been completed, some have been added, a few have been dropped, and some schedules have been revised; all this has been our attempt to satisfy both our customers and the staff as the required work has become more well defined.

If you have any questions, please call me (Ext. 2817).

Very truly yours,

Edward R. Kane
for James H. Taylor
Manager, Licensing
Nuclear Power Generation Division

JHT:nw

Attachment

cc: Mr. R. B. [illegible]

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References

1) J. H. Taylor to R. J. Mattson, 30 April 79, "Babcock & Wilcox Company Commitments".

2) "Evaluation of Transient Behavior and Small Reactor Coolant System Breaks in the 177 Fuel Assembly Plant" 7 May 79.

3) J. H. Taylor to R. J. Mattson, letter of 18 May 79 transmitting "Report on Analysis Methods for RCS Natural Circulation".

4) J. H. Taylor to Harold Denton, "Independent Auxiliary Feedwater Control System", 22 May 79.

5) J. H. Taylor to D. F. Ross, Jr., "Response to Thermal Shock Concern", June 13, 1979.

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Analytical Commitments from Reference 1

Commitment	Status
I.A Perform calculations, worst case break without AFW for 30 minutes.	Completed in Section 6.2.1 of Ref. 2. NOTE: Operator action was required in 20 minutes to avoid core damage.
I.B Document natural circulation tests conducted at Davis-Besse and Oconee.	Completed in Appendix 1 to Ref. 2.
I.C Document all occurrences of natural circulation which happened inadvertently; include a description of unexpected behavior.	Completed in Appendix 1 to Ref. 2.
I.D Document natural circulation analytical methods.	Completed in Ref. 3.
I.E Summarize and document sensitivity in key parameters identified in the Tedesco report.	Due 6 July 79. B&W believes that much of the work implied by NUREG-0560 has already been submitted. No further submittals are planned aside from those listed in this status report.
I.F (Deleted)	
I.G Define and document thermal shock criteria for operation of low temperature with HPI pumps running and no natural circulation.	Completed in Ref. 5.
I.H Assessment of the safety concerns raised in the report of Dr. Michelson.	Completed in Appendix 5 of Ref. 2.
II.A CRAFT Analyses	
1. Stuck open PORV, RCPs on, normal AFW, 2 HPI pumps	Deleted: item #2 below bounds this case and shows acceptable results.
2. Stuck open PORV, RCPs on, normal AFW, 1 HPI pump.	Completed in Section 6.2.3.1, Case 1 of Ref. 2.

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Commitment		Status
	3. Stuck open PORV, RCPs on, normal AFW, 200 gpm HPI.	Deleted: This case closely parallels item #4 below and results in core damage.
	4. Best estimate of TMI-2 transient.	Completed in Section 3.3 of Ref. 2.
	5. .07/.02/.01 ft^2 breaks.	Completed in Section 6.2.1 of Ref. 2.
	6. LOOP and LOMFW, no RCPs, no AFW, manual HPI at 20 minutes.	Completed in Section 6.2.2 of Ref. 2.
	7. 1.05 in^2 break in the pressurizer steam space	Completed in Section 6.2.3.1, Case 2 of Ref. 2.
	8. References to completed small break LOCA analyses and models.	Clarification is requested of the staff on this item.
II.B	CADDS Analyses	
	1. TMI-2 incident benchmark	Completed in Section 3.2 of Ref. 2.
	2. AFW delay study - best estimate.	Completed in Section 4.3 of Ref. 2.
	3. Reactor trip with LOMF - best estimate.	Completed in Section 4.4 of Ref. 2.
	4. Studies supporting changing PORV and reactor trip setpoints - best estimate.	Completed in Section 4.2 of Ref. 2.
	5. Sensitivity to initial power level.	Deleted.
III.A	Deleted -- This was a repeat of Section II above.	
III.B	Details of B&W's evaluation of the Michelson report.	Completed in Appendix 5 of Ref. 2.
III.C	System response to total loss of heat sink.	Original submittal date - 25 May 79 Revised submittal date - 31 July 79
III.D	Sensitivity study of system response to AFW flow rate.	Original submittal date - 25 May 79 Revised submittal date - 6 July 79

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Commitment		Status
III.E	Effect of anticipatory trip on LOMFW.	Completed in Section 4.4 of Ref. 2.
IV. A.	Benchmark analysis of sequential AFW to OTSG's for LOMFW.	Due 15 June 79.
IV. B.	System response to PORV and Code safety valve actuation.	Completed in Section 6.2.3.2.3 of Ref. 2.
IV. C.	Benchmarking natural circulation cooling in CRAFTII.	Due 2 July 79.
IV. D.	Evaluation of Michelson report and operating criteria for small breaks.	Completed in Appendices 4 and 5 of Ref. 2.
IV. E.	Worst case small break with no AFW and single ECCS failure.	Completed in Ref. 2. Vol. 1, Section 6.0, Supplement 1.
V.	Reliability analysis of the ICS.	Due 27 June 79.
VI. A.	Operating instructions for management of small breaks.	Completed in Appendix 4 of Ref. 2.
VI. B.	ICS FMEA	Included in Item V above.

Non-Analytical Commitments from Ref. 1

I.	Develop means for decoupling AFW control from the ICS.	Completed: As committed in Ref. 4, system descriptions were sent to our customers on 8 June 79.
II. A.	More positive PORV indication.	Several possible methods were sent out for customer review; tests are scheduled for week of 11 June 79 at the Alliance Research Center.
II. B.	Saturated condition indicator for the RCS.	Completed: Proposals for computer alarms have been sent to customers. Proposals for computer-independent meters will be sent soon.

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Commitments Made at the 1 May 79 Meeting in Lynchburg

Commitment	Status
1. CRAFT analysis for .01 ft^2 break to determine if Michelson phenomenon is a problem for 177 FA plants.	Completed in Section 6.2.4.3.1 of Ref. 2.
2. Small break calculation to show HPI provides adequate cooling at 2500 psig (Burp & Slurp)	Completed in Section 6.2.4.3.3 and Vol. 3, Section 6, of Ref. 2.
3. CRAFT analysis for .01 ft^2 break with AFW to only one OTSG.	Completed in Section 6.2.4.3.2 of Ref. 2.
4. Check capability of RC pumps to restart in two-phase flow.	Completed in Appendix 3 of Ref. 2.
5. Discussions on CRAFT steam space break analyses.	Completed in Section 5 of Ref. 2.
6. CRAFT analysis for .01 ft^2 break with AFW to only one OTSG for raised-loop plant.	Completed in Volume 3, Section 6 of Ref. 2.
7. Document CRAFT analyses previously done for Duke Power Company.	Completed in Section 6.2.1 of Ref. 2.

Other Commitments

1. Davis-Besse 1 analysis supporting PORV setpoint = 2365 psig, safety valve setpoint = 2435 psig.	Completed. Transmitted to the customer on 10 May 79.
2. OTSG thermal stress evaluation (no RCS flow, steam inside OTSG tubes, 90° AFW).	Completed in Appendix 2 of Ref. 2.
3. Evaluate the pressurizer safety valve break.	Completed in Section 6.2.3.2.3 of Ref. 2.
4. Probability study for opening the PORV.	Original submittal date - 25 May 79 Revised submittal date - 6 July 79

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