UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

FEB 9 1981

MEMORANDUM FOR: James P. Tourtellotte, Esq.
Assistant Chief Hearing Counsel, ELD

FROM: Demetrios L. Basdekas
Reactor Safety Engineer, RSR, RES

SUBJECT: SAFETY IMPLICATIONS OF CONTROL SYSTEMS AND PLANT DYNAMICS, AND THEIR RELEVANCE TO THE TMI-1 RESTART ASLB HEARING

This is in response to Mr. Cutchin's request of February 3, 1981 to provide a written explanation of how my views, presented in documents provided to the Board, have direct application to TMI-1.

I have read pages 11,027-11,030 of the hearing transcript and I believe that the Board desires an explanation specifically focused on my statement that "Even though [the issue of the effects on safety of the control systems] has been treated as a generic issue, it applies directly to the TMI-1...." I believe that the center of the Board's question is the word directly. The following explanatory remarks are intended to answer the Board's question on this point.

The fact that a Failure Modes and Effects Analysis (FMEA) has been performed for the Integrated Control System (ICS) by Babcock & Wilcox does not mean that it has been an effective one in identifying important weaknesses in the TMI-1 (a sister plant of TMI-2) control systems. The recommendations I make in my memorandum to Dr. Ahearne of September 4, 1979 (Reference No. 1 in my memo to you of October 10, 1980) with respect to follow-up effort to complete the FMEA with the objective of accounting plant-unique features, applies directly to the TMI-1, in that the B&W performed FMEA was never extended, as it should have been, to include the TMI-1 plant design features of its control systems and plant dynamics, which are unique to it. An example of lack of such effectiveness is on page 4-32 of B&W-1564 (copy attached). In Item 1-30 it is stated that no effect is expected for the case of steam-generator-level loss of control. This statement is not correct. The implications of this and related failures in the main feedwater control system are discussed in documents No. 12 and 16 on the list of documents I supplied to Mr. Cutchin of your office on October 30, 1980. Control system and other "non-safety" system failures on the secondary side may result in a rapid overcooling of the primary system subjecting the reactor vessel to a pressurized thermal shock that would threaten its very structural integrity. During our meeting in your office with NRR representatives sometime in early September 1980, I mentioned,

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as an example, that my understanding was that TMI-1 was one of two plants in the country that did not have a "safety grade" main feedwater pump trip function on reactor/turbine trip.

Furthermore, I believe that testimony which had been prepared by the staff on this generic issue for the TMI-1 Restart Hearing needed to be challenged. I am addressing this point in the fourth paragraph of my memorandum to you dated October 10, 1980. My use of the word directly was intended to point the direct applicability of my concerns on the subject issue to the TMI-1 in terms of this generic concern, in view of its specific similarities to TMI-2, and the specific points I have discussed earlier.

I hope that this discussion is responsive to the Board's question, and I request that you make a copy of this memorandum available to the Board.

Demetrios L. Basdekas

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Enclosure: As stated

cc: W. S. Farmer
L. H. Sullivan

Table 4-3. (Cont'd)

SHEET & ITEM NO.	INPUT	FAILURE MODE	EFFECTS ON NSS	REACTOR TRIP	REMARKS
1-26 (continued)		0%	No effect if MFWBV is open. If MFWBV is closed, the Loop A S.U. valve goes 80% open, causing the switch from S.U. to Main for feedwater flow indication. Subsequently, the S.U. valve on Loop A will cycle between 50% and 80% open until level reaches the high level limit (FW 17.6).	Possible RC pressure trip	The H/A stations can be used to control level after a trip if necessary.
1-27	Startup Feedwater Flow (Loop B)		Same as Loop A.		
1-28	Temp. Compensated RC Flow, Loop A	100%	This failure could cause an undesired reratioing of feedwater flow and very likely a reactor trip on RC pressure. Control after reactor trip is not changed.	Probable on high RC pressure.	
		0%	Feedwater flow will reratio, with SG-A going on the low level limit, and the SG-B feed flow limited only by BTU limits. For initial load of 100%, there is a net reduction in feedwater flow, and the reactor trips on high pressure. Control after reactor trip is not changed.	Probable on high RC pressure	
1-29	Temp. Compensated RC Flow, Loop B		Same as for Loop A.		
1-30	SG-A, Operate Level	252." (High)	Loop A feed flow is reduced until SG-A reaches the low level limit. The net loss of feedwater flow causes heatup of the primary and reactor trip on high pressure. Control after reactor trip is not changed.	Yes	
		0."	No effect, except that SG-A loses the protection of having a high level limit.	Not expected.	