AUG 9 1985

Ms. Ellyn R. Weiss and Mr. Robert D. Pollard
Union of Concerned Scientists
1346 Connecticut Avenue, N.W. Suite 1101
Washington, DC 20036

Dear Ms. Weiss and Mr. Pollard:

I have been asked to respond to your letter to the Commissioners dated April 12, 1985 regarding Three Mile Island, Unit 1 (TMI-1). Your letter was basically concerned about staff positions on the boiler-condenser decay heat removal process. In the following responses, we do not respond directly to the six specific questions asked in your letter because a major portion of the requested information is already contained in the staff's testimony and the hearing record. We do address, however, what we understand to be your underlying concern, namely the apparent discrepancy between the previous staff testimony on TMI-1 restart and more recent statements made by the Office of Nuclear Regulatory Research (RES) in response to Congressional inquiries.

Your April 12, 1985 letter identifies an apparent discrepancy between testimony filed by the staff with the TMI-1 Appeal Board and a statement made in response to Congressional inquiries concerning NRC's research budget. Specifically, the NRC staff, in 1983 testimony filed with the TMI-1 appeal board, stated that experimental testing was not needed prior to restart in order to confirm the effectiveness of boiler-condenser decay heat removal in the TMI-1 plant during small break loss of coolant accidents (LOCAs). More recently however, in responding to Congressional inquiries regarding the research budget, RES stated that testing to assess the effectiveness of the boiler-condenser process to remove heat from the reactor coolant and maintain natural circulation was research needed in response to TMI-1 regulatory concerns. These were set forth in ALAB 729 which recommended continuation of research on decay heat removal capability. The RES statement was made on March 14, 1985 prior to the NRC decision on TMI-1 restart which was reached on May 29, 1985.

We reaffirm our position that experimental testing is not needed to confirm the effectiveness of boiler-condenser decay heat removal for TMI-1. Since we recognize that the statement provided to Congressman Udall implies a contradiction with this position, the following discussion is provided to clarify this issue and show that no discrepancy or contradiction actually exists.

During certain small break LOCA's, a possible steam formation at the top of the hot leg U-bends was predicted to interrupt natural circulation. Under these accident conditions, the reactor coolant pumps are not running and natural circulation causes the flow of cooling water through the core. In the absence of natural circulation, decay heat removal capability would be lost and the primary system would repressurize as the primary system temperature increased. This would have a twofold effect of increasing the rate of coolant loss out of the break and reducing the flow of safety injection water into the primary system. If decay heat removal was not restored, the system would continue to lose more water than was being made up, and eventually core uncovery and core damage would occur. In order to stabilize the system, it would be necessary to reestablish decay heat removal. This would lower the primary system pressure, reduce the break flow, and increase the safety injection flow sufficient to prevent core uncovery and to refill the primary system. Under these circumstances, it may be necessary to rely on the boiler-condenser mode of decay heat removal. The effectiveness of the boiler-condenser mode of decay heat removal was in question.

Because of the primary system configuration at TMI-1 (and all other lowered loop B&W-designed reactors), the staff concluded that before the coolant level in the primary system could drop below the top of the core in the accident sequence described in the proceeding paragraph, a condensing surface would be exposed in the once through steam generators (OTSG) that would be sufficient to remove decay heat and depressurize the primary system. This process is what is referred to as the boiler-condenser mode of decay heat removal. Safety injection flow would then be sufficient so that core uncovery would be avoided. The efficacy of the thermal hydraulic and heat transfer processes involved were not deemed to require experimental data since (a) all B&W reactor licensees had modified their emergency procedures to instruct the operators to raise the secondary side water level to 95 percent of the operating range when cooling by natural circulation, thus ensuring an ample condensing surface in the OTSG at an elevation above the top of the core should the boiler-condenser cooling mode be necessary, (b) the heat transfer correlations associated with condensation heat transfer were well established, (c) systems calculations of this process had been performed by independently developed computer codes, including RELAP4, RELAP5, and CRAFT, and despite differences in the predicted detailed thermal hydraulic behavior associated with the boiler-condenser mode of decay heat removal, all of the codes predicted the ultimate establishment of decay heat removal and no core uncovery, (d) decay heat removal by condensation heat transfer had been experimentally confirmed in test facilities with inverted U-tube steam generators, which, while not exactly the same, exhibit many of the thermal hydraulic characteristics of the B&W OTSG design, and (e) sufficient margin existed such that uncertainties in the analytical results would not have influenced our conclusions. Based on the above considerations, we determined that experimental information demonstrating the efficacy of boiler-condenser decay heat removal was not needed to ensure safe operation of TMI-1.

No new information has been established in the intervening period since we made this determination that would cause us to change our conclusions today. Rather, data recently obtained from the GERDA and OTIS facilities (B&W raised loop simulations) and recent TRAC and REBL analyses on the MIST facility have served to confirm our conclusions.

As stated above, we fully recognized that uncertainties existed in the analysis methods available at the time of the appeal board hearing, and we actively supported the need for a thermal hydraulic experimental facility geometrically similar to the B&W nuclear steam supply system (NSSS) design in

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order to study and quantify the uncertainties in the small break LOCA analyses for B&W designed NSSSs as set forth in ALAB 729 which recommended that confirmatory research into decay heat removal be continued in order to "...increase the current knowledge of thermal-hydraulic behavior during small break loss of coolant accidents".

The need for this information was to provide confirmatory data for the purpose of (1) quantifying the ability of thermal hydraulic codes, such as RELAP5 and TRAC, to calculate the best-estimate, or realistically expected thermal hydraulic response of a B&W NSSS to small LOCAs, (2) providing a data base from which code improvements, if needed, could be based and assessed, and (3) using these codes as an audit tool for confirming selected emergency operator guidelines for B&W NSSSs.

The need for and use of this data, and its relationship to the licensing process, is considered identical to our need for and use of data from other thermal hydraulic facilities, notably LOFT and Semiscale for PWRs and TLTA and FIST for BWRs. Data from these facilities has been used primarily to confirm and quantify safety margins in licensing requirements, to provide a basis for reducing excessive margins, and to provide a data base against which best estimate codes can be assessed and improved.

Therefore, our response to Congressman Udall's staff does not mean that experimental data is needed to conclude that TMI-1 meets the Commission's regulations and can safely remove decay heat and prevent unacceptable core uncovery during small break LOCAs. Rather, it is intended to mean that a generic effort is needed to obtain additional experimental data to be used by the Regulatory staff for the purposes of quantifying thermal hydraulic performance uncertainties in codes used for evaluating small break LOCAs in all B&W-designed NSSSs.

With respect to the use of these codes to confirm emergency operating procedures, you point out statements made in a 1981 NRR Research User Need Letter that implies incorrect operator actions could result as a consequence of unpredicted phenomena producing false symptoms of other events. Since the User Need Letter was written in 1981, we have completed an in-depth review of the B&W Abnormal Transient Operator Guidelines (ATOGs), from which the TMI-1 plant emergency procedures were developed. The ATOG guidelines are designed to treat symptoms of accident conditions such as steam generator heat transfer and core cooling problems and are not dependent on operator event diagnosis for safe shutdown. We approved these guidelines and concluded that since any operator error of significance will manifest itself as an abnormal symptom or plant response and would be treated accordingly, operator error is adequately covered.

Sincerely,

"ORIGINAL SIGNED BY:"

Harold R. Denton, Director
Office of Nuclear Reactor Regulation

*See previous white for concurrences.

ORB#4:DL	ORB#4:DL	OELD	AD:OR:DL	D:DL	DD:NRR	D:NRR	D:DSI
JThoma;cf	JStolz*	JGray*	GLainas*	HThompson*	DEisenhut	HDenton*	RBernero*
8/9/85	7/19/85	7/22/85	7/25/85	7/30/85	8/7/85	8/2/85	7/26/85

D:DE	RES
JKnight*	I have verbal concurrence from RES
7/30/85	8/8/85 (S. BASSETT says R. Minogue concurs in this version) JOT

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Sincerely,

Original signed by
Darrell G. Eisenhut

Harold R. Denton, Director
Office of Nuclear Reactor Regulation

RES	for D:DSI	D:DE
	RBernero	JKnight
8/ /85	7/26/85	7/3 /85

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