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

JUL 6 1979

MEMORANDUM FOR: Dr. Max Carbon, Chairman
Advisory Committee on Reactor Safeguards

FROM: S. H. Hanauer, Director
Unresolved Safety Issues Program, NRR

SUBJECT: STATUS AND PROPOSED PLAN FOR RESOLUTION OF "ANTICIPATED TRANSIENTS WITHOUT SCRAM" SAFETY ISSUE

The NRC staff last met with the ACRS to discuss the ATWS issue at the 227th ACRS meeting (March 8-10, 1979). At the conclusion of that meeting the Committee identified several matters to be discussed by the staff at the succeeding or 228th meeting (April, 1979). As a result of the need on the part of both the ACRS and the NRC staff to focus resources on the Three Mile Island 2 accident, these discussions have not taken place.

In order to facilitate the resumption of the dialogue on ATWS between the ACRS and the NRC staff, we are attaching to this letter a discussion of those matters that the Committee has raised relative to BWR's. We are not including any discussion, however, on the PWR related items until, as noted below, we have had a better opportunity to assess the impact of the Three Mile Island 2 accident on the evaluation and resolution of the ATWS issue for such plants.

As you are aware, as a part of the NRR Interim Organization, a technical review group has been formed to work on a "dedicated" basis on each of the Unresolved Safety Issues reported to Congress in the 1978 NRC Annual Report. ATWS is, of course, one of these issues.

As a result of the heavy expenditure of resources on Three Mile Island related activities, essentially no staff effort, and we think greatly reduced vendor effort, has been applied to the ATWS issue for the last 3 months or so. For Boiling Water Reactors, the effect of Three Mile Island 2 on the evaluation of ATWS events is believed to be minimal. General Electric provided a submittal in May with answers to some of the "early verification" questions which were transmitted to all of the vendors by letter of February 15, 1979 from R. J. Mattson. G.E. has committed to providing the balance of the early verification responses for BWRs in two additional submittals, one in early July and the other in the Fall of 1979. We intend to complete the review of these submittals on an expedited basis with the objective of arriving at a proposed rule that will contain general NRC requirements for plant modifications for various classes of BWR plants. It is our intent that for the BWR's, our revised schedule will be made as expeditious as

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possible leading to a proposal to the Commission for rulemaking in the next few months.

For Pressurized Water Reactors, the staff's preliminary assessment is that the Three Mile Island 2 accident scenario raises new questions with regard to the appropriateness and adequacy of the proposed resolution of the ATWS issue for such plants. The reviewers assigned to ATWS within the Unresolved Safety Issues Program are continuing their assessment of the implications of Three Mile Island on our evaluation of ATWS. This assessment includes discussing the accident's possible implications for ATWS with the NRR "Lessons Learned" Task Group in July.

In addition, we plan to meet with representatives of the PWR vendors and utility representatives on July 25 to exchange views on the impact of Three Mile Island on ATWS. After this meeting, we expect to develop a revised schedule for completing our evaluation of ATWS for PWRs.

We will, of course, provide the ACRS with a complete revised schedule for ATWS as soon as one is available.

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Enclosure:
"ATWS - Outstanding ACRS Questions Following 227th Meeting - Applicable to BWR's Only"

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ATWS - Outstanding ACRS Questions Following 227th Meeting - Applicable to BWR's Only

Question:

a) What are the maximum temperature and pressure transients the torus or pressure suppression pools of GE containments can accept without rupture? What are the consequences of torus or suppression pool failure?

Response:

Our interpretation of the first part of this question is: During the extended Safety/Relief Valve blowdown resulting from an ATWS event, what is the maximum combination of pool bulk temperature and oscillating pool pressure that the torus or suppression pools of G.E. containments could accept without rupture?

The specific combination of pressure and temperature at which failure would be expected to occur would be plant specific and has not been determined. We do know, however, from testing that has been performed as a part of the Mark I, II, and III Containment Testing Programs, that continuous safety/relief valve steam discharge can result in a phenomenon that has been termed "steam condensation instability". Condensation oscillations occur as the steam is being discharged to the pool. The amplitudes of these vibrations are relatively small at low pool temperatures. However, continued blowdown into the pool, as would result from an ATWS event, increases the pool temperature. If the temperature rises far enough, a "threshold temeprature" can be reached for some types of discharge devices. When the pool temperature is above the threshold, steam condensation is unstable. The resulting vibration and associated forces are very severe.

In a foreign reactor, an event involving prolonged blowdown led to loss of integrity of the suppression pool via this mechanism. In testing, the condensation load has been observed to be as great as ten times the normal oscillation load at pool temperatures above the threshold.

Current practice for operating BWR's is to restrict the allowable operating temperature of the suppression pool via the Technical Specifications such that the threshold temperature would not be reached in a Loss of Coolant Accident. Typical Technical Specification temperatures used for an operating Mark I plant are shown in Figure 1.

For ATWS events, G.E. has proposed higher maximum allowable temperature limits than those upon which the current Technical Specification values are based. The staff agrees that higher temperature limits are appropriate for plants when quencher type discharge devices are used. Present indications are that temperatures up to approximately 200°F are acceptable. It is our understanding that all BWR plants, both those in operation and those under construction, will be installing quencher discharge devices.

An industry program has already generated a great deal of experimental and theoretical information on quenching phenomena.

An intensive, detailed examination of all relevant data related to suppression pool temperature limits is being conducted by the staff as part of TAP-A39. All three pool configurations and various quencher devices are being reviewed. The pool temperature criteria to be established for ATWS will be based on this work. Completion of criteria is scheduled for December, 1979, for Mark I and Mark II containments, and a few months later for Mark III.

Temperature predictions for various event sequences and various boron injection capacities are given in Parts (b) and (c) of this response.

The second part of question a) above was "What are the consequences of torus or suppression pool failure?"

Figure 2 illustrates, by means of an elementary event sequence diagram, the relative safety significance of maintaining suppression pool integrity. This event sequence, and the related discussion which follows, is based upon analyses performed by the staff for a "typical" BWR/4 plant with a Mark I containment and equipped with Recirculation Pump Trip. However, we believe that the basic conclusions are applicable for all BWR containment designs.

Referring to the bottom half of Figure 2 which summarizes the sequence of events for the case where complete loss of suppression pool integrity occurs, we note the following:

1) Using conservative assumptions the staff estimates that an ATWS event would result in failure of 10 percent to 20 percent of the fuel by clad perforation early in the event i.e., a few seconds. Thus highly contaminated steam would blow down from the safety/relief valves to the suppression pool.

2) A major rupture of the pool would result in a release of 4×10^6 lb. to 8×10^6 lb. of high temeprature (approximately $250^{\circ}F$), highly contaminated water and steam to the auxiliary building. The exact quantity and temperature is plant specific. The water temperature was calculated assuming complete mixing occurs in the pool.

3) The HPCI pump is located in the auxiliary building and thus would be exposed to a steam and radiation environment for which it has not been qualified. If the HPCI pump fails because of exposure to the steam and radiation environment, as was noted above, core melting would probably occur.

4) A loss of suppression pool integrity could (depending on where the failure occurred) result in the loss of a major source of water for reactor coolant system inventory replacement and core cooling. With the loss of the suppression pool (high temperature or loss of integrity) as a water source, the HPCI system can be continued to be supplied from the alternate water source (Condensate Storage Tank). Although the preferred source of water supply for the HPCI system is from the Condensate Storage Tank, manual action is required to replenish the Condensate Storage Tank in about 15 minutes in order to keep the core covered. Additionally, if the initiating event is an OBE, the Co[illegible]rage Tank may not be available on some plants to provide a wa[illegible]y. Even if the HPCI system functions using the Condensate Storage Tank and the core remains covered, it is likely that the steam pressure in the auxiliary building would cause the building blow out panels to open thus permitting a direct path for the contaminated steam to be released to the environment. The staff estimate is that the resulting dose would exceed 10CFR100. General Electric, using different assumptions, has reached a conclusion different from that of the staff.

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5) In summary, suppression pool integrity must be maintained to prevent excessive radioactive release to the environment and to protect the HPCI pump and possibly other safe shutdown equipment in the auxiliary building from a steam and radiation environment for which they have not been qualified.

Question:

b) Provide an evaluation and comparison of the effects of 43, 86, and 400 gpm liquid boron injection rates on the predicted transients in the pressure suppression pool or torus.

Question:

c) Provide a comparison of the effects of various time delays of boron injection (to 10 minutes) on the predicted transients in the pressure suppression pool or torus.

Response:

The staff has summarized in tabular form the results of analyses performed by G.E. for the "worst case transient i.e., highest vessel pressure." The information presented in Fig. 3 demonstrates the effect on the maximum pool temperature of varying both the liquid boron injection rate and the SLCS actuation time.

Note that the 400 gpm SLCS injection rate (Alternate 4 approach per Volume 3 NUREG-0460) provides for the following:

1) Assurance that the suppression pool temperature core limit is not exceeded even if a single active failure is assumed.

2) Assurance that the core is not uncovered even if a single active failure is assumed.

3) Assurance that the suppression pool temperature limit is not exceeded even if the operator action (e.g., RHR in pool cooling mode) is delayed beyond 10 minutes.

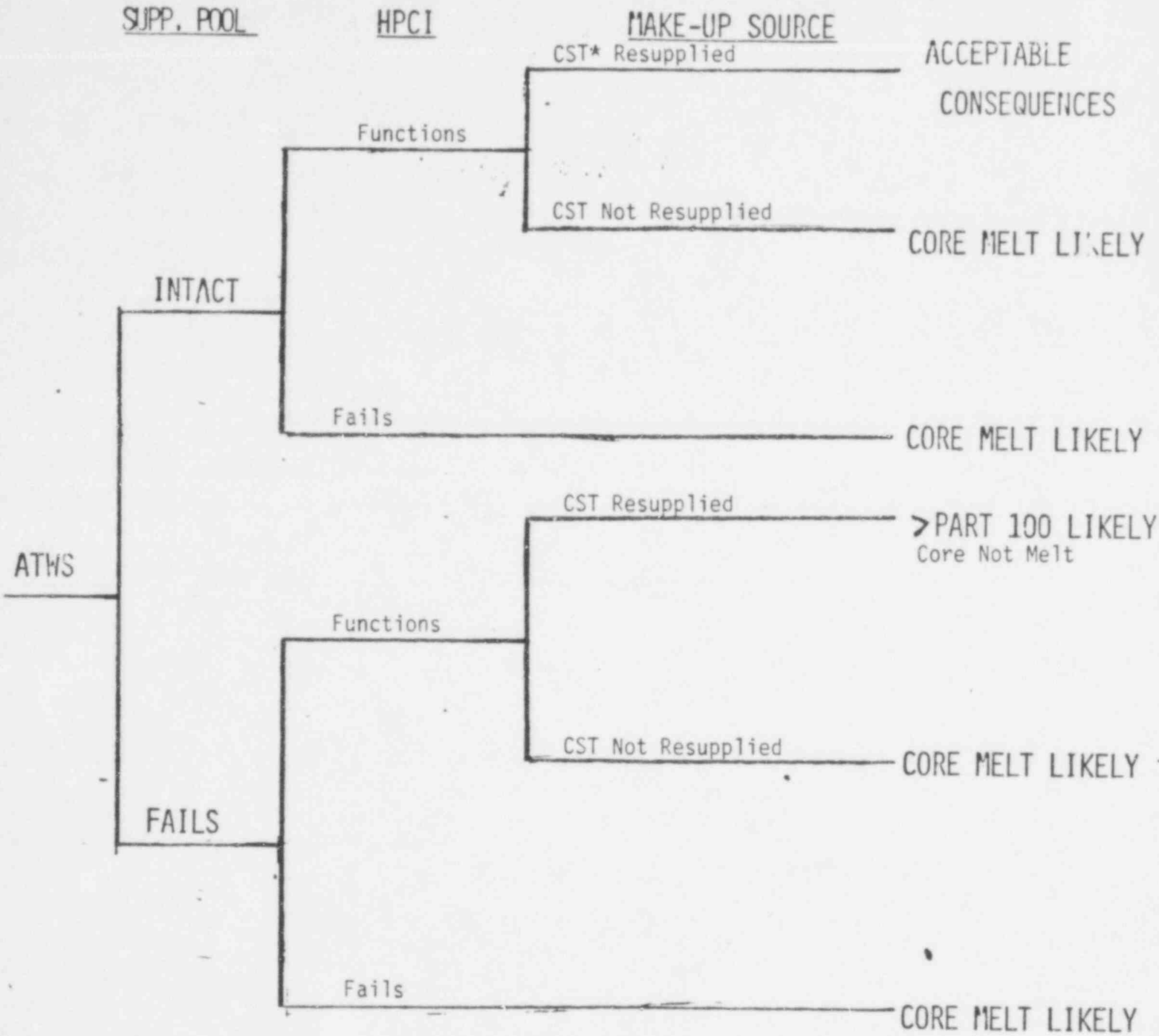
4) Additionally, implementation of Alternate 4 will provide assurance that the suppression pool temperature is low enough to meet the HPCI system suction temperature requirements and thus assure an available HPCI water source even if the Condensate Storage Tank fails. Some plants do not have a Condensate Storage Tank designed for seismic loads, the OBE (an event with a substantial probability of occurrence over the plant lifetime - 10CFR100, Appendix A) could be the initiating event of an ATWS, although of lower frequency than other transients.

BWR TECH SPECS

SUPPRESSION POOL

ACTION	TEMPERATURE (^{o}F)
-- NORMAL OPERATION	≤ 95
-- TESTING LIMIT BUT START COOLING (24 HRS $\rightarrow$ $95^{o}F$)	>105
-- SHUTDOWN	>110
-- SCRAM	>120

Figure 1



SUPPRESSION POOL EVENT SEQUENCE

*Condensate Storage Tank

Figure 2

PRELIMINARY

BWR SUPPRESSION POOL MAXIMUM TEMPERATURE

ATWS EVENT*

PLANT TYPE	EVENT	SLCS (GPM)	ACTUATION TIME (MIN)	PEAK POOL TEMP. (°F)	G.E. REFERENCE
MARK I	MSIV CLOSURE	86	2	200	March 9, 1979 Mtg. with NRC Staff
		86	10	260	March 9, 1979 Mtg. with NRC Staff
		43	2	250	Telephone Conversation w/ G.E.
		43	5	290	Estimated from NEDO-25016 and 11-10-75 Responses
MARK II	MSIV CLOSURE	86	2	180	March 9, 1979 Mtg. with NRC Staff
		86	10	210	March 9, 1979 Mtg. with NRC Staff
		400	1	150	February, 1978 Mtg. w/NRC Staff
MARK III	MSIV CLOSURE	86	2	165	March 9, 1979 Mtg. w/NRC Staff
		86	10	190	March 9, 1979 Mtg. w/NRC Staff
		400	1	150	February, 1978 Mtg. w/NRC Staff

*Considers Effects of Single Active Failures

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Figure 3