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

APRIL 12 1979

MEMORANDUM FOR: E. G. Case, Deputy Director
Office of Nuclear Reactor Regulation

FROM: D. F. Ross, Deputy Director
Division of Project Management

SUBJECT: SUMMARY OF MEETING WITH COMBUSTION ENGINEERING (CE)-CORRECTIVE ACTIONS FOR COMBUSTION ENGINEERING NSSS PLANTS AS A RESULT OF THREE MILE ISLAND UNIT 2 INCIDENT

On April 11 and 12, 1979, the NRC staff met with respresentatives of Combustion Engineering, Incorporated (CE) in Bethesda, Maryland, to discuss short term corrective actions to be implemented at CE pressurized water reactors (PWR) as a result of the incident at Three Mile Island Unit 2. Several CE PWR licensees were in attendance. A list of attendees is attached (Enclosure 1).

April 11, 1979 Meeting

The meeting opened with an overview of the events at Three Mile Island Unit 2 (TMI-2) which require immediate attention by all operating PWRs as these events are perceived by the staff in light of information available at this time. These events are identified as Items 1 thru 12 in the NRC Office of Inspection and Enforcement (OI&E) Bulletin 79-05A of April 5, 1979 (Enclosure 2). The staff specifically noted that the responsibility for development of corrective actions for these items rests with CE and the utilities. The corrective actions that are needed are specific instructions to be issued immediately to licensees of CE PWRs. These corrective measures will be reviewed by the NRC staff and issued by means of an OI&E Bulletin. D. Ross of the NRC staff read through the items in the B&W bulletin (Enclosure 2) and asked for comments and agreement that the items were suitable for CE designed plants.

Items 1 through 3 were agreed to by CE (agreement meaning that the item was appropriate for a CE designed plant).

Item 4 consists of four parts. They concern the overriding of automatic actions of engineered safety features (ESF).

CE was unwilling to speak on this item because they considered this a prerogative of their customers. They stated that they provide guidelines for the development of procedures by the utilities.

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CE stated that SI actuation occurs on the following signals (among others):

1. low primary pressurizer pressure

2. high containment pressure

Containment isolation is actuated by the high containment pressure (with the same set point as for SI actuation).

CE stated that while pressurizer level is the main parameter looked at in the guidelines, other system parameters are also used by some CE customers.

CE has not provided a guideline for turning off the safety injection to its customers.

CE agreed with item 4a (the operator should not override automatic actions of emergency safeguards features).

A representative of Baltimore Gas & Electric Company questioned the reason for a time requirement in item 4b concerning conditions under which the HPI could be turned off. He questioned why a pressure indication was not sufficient. The point was that if the Reactor Coolant System (RCS) were to approach going solid, pressure (subcooling) indication would be sufficient, regardless of the time that the HPI had been in operation. The staff noted his comment.

The question of the HPI causing reactor vessel pressure in excess of the allowable limits was discussed. CE stated that they had performed some fracture mechanics calculations of this type for the steam line break for vessels of different ages and these showed acceptable results. The analyses were previously reported to the staff in the letters listed below.

1. June 4, 1975 letter from W. Corcoran, CE to R. Maccary, NRC.

2. June 24, 1975 letter from W. Corcoran, CE to F. Schroeder, NRC.

CE stated that reactors designed by them have not experienced a stuck open relief valve. For this event, the pressurizer level could increase, although the primary system would be depressurizing.

CE related a case in which the system drained down to a pressurizer level of 1% due to a drain valve that was inadvertently left open during a test (pre-nuclear). There was no flashing or change of direction in pressurizer level. However, this event was not directly applicable to the discussion since the drain was releasing water inventory from the water space rather than steam space.

CE stated that the analysis of the inadvertent opening of a relief valve showed that the two phase level reached the top of the pressurizer with a void fraction in the pressurizer of approxiamtely 25%.

CE stated that under this condition they would expect the pressurizer level instrumentation to give a true indication of level.

The staff asked whether the operator had to take any action based on level. CE responded that he did not and furthermore, that level did not enter into any safety system actions.

The staff then asked CE if the pressurizer could be full while the core was empty. CE replied that the analysis of the inadvertent opening of a relief valve showed that the pressurizer would drain. The staff questioned whether the computer code used for the analysis would have predicted this phenomenon. CE stated that the code was an evaluation model code. The staff will pursue this further with CE.

This completed the discussion of Item 4.

Item 5 concerned verifying that the auxiliary feedwater (AFW) valves are in the proper position. CE agreed with this item and stated further that the AFW systems on all CE operating plants are manually actuated. Analyses of all transients in the FSAR showed that with a delay of 10 minutes before initiating AFW flow, the pressurizer relief valves would not open.

CE agreed that items 6 through 10 of the OI&E Bulletin were appropriate.

CE had stated earlier that they had not come prepared to answer these items in detail. They stated that the answers they gave could be considered those of the Corporation however.

The staff held a caucus and decided that more information was required in order to write the bulletin , and CE was given an additional 24 hours to make comments in the following areas:

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1. ESF reset criteria
2. What criteria are now used for turning off HPI? What temperature and pressure criteria are used?
3. Should NRC advise utilities on CVCS operation?
4. Further discussion of fracture mechanics.
5. Criteria for tripping the reactor coolant pump.
6. List operator actions based on pressurizer level.
7. Provide a CEFLASH calculation for a small break on the pressurizer fluid dynamics.

It was decided to continue the meeting on April 12, 1979, at 1:30 pm.

April 12, 1979 Meeting

The meeting opened with a review of the NRC intentions to issue bulletins to Westinghouse Electric Corporation and Combustion Engineering reactor licensees by tomorrow, April 13, 1979. It was noted that Bulletin 79-06 was issued on April 11, 1979, and contained general guidance for all PWR reactor licensees (except Babcock & Wilcox plant licensees). The bulletins to be issued tomorrow will be based on but provide more specific guidance than Bulletin 79-06.

The meeting then proceeded to the seven point agenda identified at the end of the previous day's meeting. These seven points correspond to the provisions of item 4 of IE Bulletin 79-05A.

The CE representatives discussed these provisions as follows:

4.a. (agenda item 1) CE agreed that this was appropriate for their facilities and suggested additional clarifying instructions to the plant operator. The staff will consider this suggestion.

4.b. (1) and (2) (agenda items 2, 3 and 4) CE stated that they have been investigating these provisions since the TMI-2 incident, and they confirmed the information given to us in the April 11 meeting regarding fracture mechanics. Based on their investigations and the reference information (identified in the April 11, 1979 minutes), they agreed that the provisions of item 4 were appropriate for their facilities.

4.c. (agenda item 5) The CE representatives believe that additional clarification should be added to this provision such that not all reactor coolant pumps should be run, but at least one pump per cooling loop should be run. It was recognized that, for some facilities, the proviso that one pump per loop be run would require all reactor coolant pumps to be run. The staff will consider this addition.

4.d. (agenda item 6 & 7) CE agrees with this provision as it is presently worded. Regarding their emergency core cooling code (CE FLASH) calculation for a small pressurizer break, CE representatives state that some information is available on the CE System 80 docket (Safety Analysis Report Chapter 6). This information provides pressurizer pressure as a function of time. CE will, after a check of proprietary considerations, make available information regarding pressurizer level as a function of time for this small pressurizer steam space break (equivalent to a 4" dia. hole).

Based on the information presented by CE, the staff intends to proceed with issuance of a bulletin with short term corrective actions to CE facility licensees.

Denwood F. Ross

Denwood F. Ross, Deputy Director
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Enclosures:
As stated

cc w/encl:
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