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

TAP A-9

SEP 13 1979

MEMORANDUM FOR: S. H. Hanauer, Director
Unresolved Safety Issues Program

FROM: A. Thadani, Task Manager, A-9

SUBJECT: NRC - CE ATWS MEETING SUMMARY

A meeting was held in Bethesda on August 17 with Combustion Engineering (CE) to discuss plans for resolution of the ATWS issue. At the outset the NRC staff commented that they needed schedule for responses to the 2/15/79 letter from R. Mattson to CE and the TMI-2 related concerns described by the NRC at the 7/25/79 meeting with the industry. NRC staff also expressed its concern that the owners of CE designed plants were not present at the meeting.

CE stated that they had shown the 2/15/79 letter from Mattson to the utilities (staff to investigate why the utilities did not receive copies of this letter from NRC), that CE had been negotiating the scope and the schedule for responses with the utilities, that the ATWS work was suspended in June (because of TMI-2) and that they soon expected to provide a schedule for responses to the 2/15/79 letter, and that the objective of the meeting was to more fully understand the TMI-2 impact on technical aspects of ATWS events.

TMI-2 Related Concerns

a) What assurance do we have that the excessive calculated pressures for some designs modified per Alternative #3 would not result in loss of integrity of reactor coolant pressure boundary. (Note - Some designs may experience peak pressures - 4000 psi).

   CE stated that the stress analyses and functionability of instrumentation and equipment will be addressed in their upcoming report on ATWS.

b) Would increasing the number of safety valves as per Alternative #4 result in insufficient overall risk reduction? Would the primary system integrity be maintained? Would it be better to have larger capacity valves?

   CE believes that they can show that the Alternative 3 fix would provide adequate protection from ATWS events and that increasing the number of safety valves would result in greater potential for LOCA. In response to a staff question, CE stated that they had considered alternative "fixes"

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i) Depressurization events in Alt. 3 and 4, transient + stuck open valve in Alt. 4.

ii) Auto ECCS and no auto ECCS

iii) Turbine trip results

iv) Primary system repressurizes

v) Impact of HPSI design capability

The staff asked CE to develop event trees t decide the sequences to be analyzed.

g) Would single failure cause all PORVs to fail to open? If so, then analyses must be based on all PORVs failing to open. Further, several plants are operating today with PORVs isolated. For these plants credit cannot be taken for relieving capability of these valves.

CE would address the potential for a single failure resulting in failure to open both PORVs under Alt. #4. CE stated that sensitivity studies would cover plants operating with PORV isolated.

h) What assurance do we have that the ATWS events with a stuck open safety valve have been correctly analyzed? What is the potential for core uncovering under this scenario? What is the importance of ECCS actuation, reactor coolant pumps operation, and the pressurizer safety/relief valve discharge model on the potential for uncovering the core? Further, why should more valves not be assumed to stick open following discharge of subcooled water.

CE will address this in conjunction with item f above.

j) For long term shutdown, discuss the following:

i) Available equipment, instrumentation and their qualification. (Must consider the effect of water discharged to the containment via ruptured quench tank).

ii) Impact of loss of offsite power as initiating event - role of control system.

iii) Continued operation of reactor coolant pumps. Also consider tripping of reactor coolant pumps impact on core uncovering.

iv) Describe natural circulation, including effects of non-condensables. Is reflux boiling mode of operation anticipated? If so, justify.

v) Would one anticipate Boron precipitation problem? Also consider TMI-2 type problems with possible letdown line plugging from Boron precipitation.

vi) How are leakage problems from equipment outside containment considered in radiological assessment?

ENCLOSURE 1

Meeting with CE on ATWS - August 17, 1979
Attendance List

A. Thadani	NRC/DSS
S. H. Hanauer	NRC
M. Srinivasan	NRC/DSS
K. I. Parczewski	NRC/DOR/RSB
Francis Akstulewicz	NRC/DSE/AAB
F. Odar	NRC/DSS/AB
Kulin D. Desai	NRC/DSS/MEB
H. Vander Molen	NRC/DOR/RSB
M. R. Fleishman	NRC/SD/EMSB
R. Woods	NRC/DOR/RSB
C. B. Brinkman	CE (Bethesda)
D. Kreps	CE
C. L. Kling	CE