50-320

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 WESTINGHOUSE - CORRECTIVE ACTIONS FOR WESTINGHOUSE NSSS PLANTS AS A RESULT OF THREE MILE ISLAND UNIT 2 INCIDENT

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

The meeting opened with an overview of the events at Three Mile Island Unit 2 (TMI-2) which require immediate attention by all operating PWR's 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 W and the utilities. The corrective actions that are needed are specific instructions to be issued immediately to licensees of W PWR's. These corrective measures will be reviewed by the NRC staff and issued by means of an OI&E Bulletin.

W representatives then presented a summary of the activities which they have initiated since the TMI-2 incident to prevent the occurrence of a similar incident at a W facility.

Since April 1, 1979, W has been working with its customers on this issue, and on April 5, 1979, a meeting was held between W and its customers to discuss the potential for the occurrence of a TMI-2 type incident at W facilities. Since then, W has been conducting additional studies concerning specific plant concerns regarding the TMI-2 incident and has conducted some computer analysis of the incident. W has also asked individual utilities to compile plant specific information which may bear on the probability of occurrence of mitigation of a TMI-2 type incident. W representatives stated that the efforts underway with their customers covers all the items identified in IE Bulletin 79-05A and some additional areas of review.

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W then discussed the response of a typical 4-loop (four reactor coolant system cooling loops) PWR to a loss of feedwater (to the steam generators) transient. The transient response reported in individual plant Safety Analysis Reports (SAR) is more conservative than the actual response experienced at W facilities for loss of feedwater. For actual transients, the large steam generator secondary-side inventory provides a buffer between secondary (steam side) transients and primary (reactor coolant system) response to the transients. W is still investigating, but as of this date, they are not aware of any loss of feedwater leading to a primary system pressure increase that caused a pressurizer power operated relief valve (PORV) to open. Therefore, a stuck open PORV similar to that experienced at TMI-2 should not occur for a loss of feedwater transient under normal plant operating conditions. However, if no credit is taken in the analysis for non-safety grade plant control system, PORV lift will occur; and there exist other transients which can lead to a PORV lift (and to the potential for a stuck open PORV).

Because it is not impossible to preclude PORV lift and the potential for a stuck open PORV, W performed computer analyses using conservative assumptions to determine the response of a typical 4-loop PWR to a stuck open PORV. Using the W-FLASH code and 10 CFR 50, Appendix K assumptions, W analyzed a 2½" dia. Loss of Coolant Accident (LOCA) break in the vapor space of the pressurizer. This break size is similar to the size of LOCA caused by a stuck open PORV. W also assumed the steam generators were isolated (main steam isolation valves are shut) and no charging flow makeup to the reactor coolant systems is in progress. Three cases were analyzed:

Case 1 [with auxiliary feed system (AFS) flow to steam generator and with safety injection (SI)]

Results:

a. The reactor core remains flooded with cooling water throughout the duration of the analysis (Approx. 4000 sec.) and the parameters indicate that no uncovering of the core would occur thereafter.

b. The pressurizer steam-water mixture level increases and stabilizes at about a 2/3.

Case 2 [with no AFS and with SI]

Results:

a. Same as Case 1, a. and b.

b. 2/3 of steam generator level is still present at 4000 sec.

c. Reactor coolant system pressure approaches 1100 psi which corresponds to the temperature in the steam generators with safety valves lifting.

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Case 3 [with AFS and no SI (i.e., no water makeup to the reactor coolant system at all)]

Results: a. The reactor core would start to uncover at about 2100 sec.

Additional analysis is being done by W for Case 3 without the steam generators isolated. And a comparison of Cases 1 and 2 indicate that the results are not very sensitive to AFS initiation for the time periods of the analysis.

W discussed the signals which initiate SI. Analyses which they submitted previously (Zion Station and RESAR-3 dockets) show that a small LOCA in the pressurizer steam space may not result in SI initiation because the pressurizer level may not decrease. A coincident pressurizer low level (Lp) and low pressure (Pp) is needed for SI actuation. But their analysis of containment building pressure following this LOCA shows that SI would be initiated by containment pressure high (no. 1) indication setpoint which is set at about 10% of containment design pressure at about 1600 sec. At 1600 seconds, reactor core fuel surface temperature would be at the same temperature as the reactor coolant system coolant which is saturation temperature for 1100 psi. This is far below the temperature necessary for core damage. To provide additional assurance that SI initiates and prevents the core from becoming uncovered, in addition to considering the high containment pressure setpoint SI actuation signal, W has instructed its customers that SI should be manually initiated if Pp decreases to the low Pp setpoint regardless of Lp reading.

W is still evaluating the question of when to manually shut off SI following its activation. The concerns are (1) that the SI system would fill the reactor coolant system completely and thus increase the chances of an overpressure transient which could overpressurize the reactor coolant system or (2) that the operator would shut off SI based on an erroneous pressurizer level and thus increase the chances of a TMI-2 type incident (core uncovery). W presented a logic "tree" that an operator could use to determine if SI should or should not be shut off following events which lead to SI and low or failing pressurizer pressure and/or level.

W agreed that a bulletin similar to Bulletin 79-05A should be sent to its customers, but additional clarification of the need to shut off SI to prevent overpressure as discussed above should be included. W noted that the bulletin provision regarding containment isolation reset is not applicable to its plants because containment isolation valves do not open following an SI reset (as occurred at TMI-2) unless the operator deliberately opens the isolation valves.

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Following the W presentation, the staff discussed the followup action to be taken in light of the W information. A bulletin will probably be issued to W facilities in the next few days. The bulletin will be essentially the same as OI&E Bulletin 79-05A but additional information will be included to determine plant specific corrective measures dealing with:

1. Manual shutoff of SI,

2. Management checking of safety system operability status,

3. Possible elimination of Lp as an SI initiation signal by placing it in a "tripped" state,

4. Possible requirement for containment isolation on high radiation signal for all plants.

The bulletin will state that our best information shows that, under certain transient and/or accident conditions, a level may be present in the pressurizer simultaneously with a decreasing primary system pressure.

Denwood Ross

Denwood F. Ross, Deputy Director
Division of Project Management

Enclosures:
As stated

cc w/encl:
See next page

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TERA
NRC Participants

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Enclosure 1

LIST OF ATTENDEES

WESTINGHOUSE MEETING 04/11/79

NRC

R. S. Boyd, DPM
T. A. Ippolito, DOR
M. H. Fletcher, DOR
R. Lobel, DOR
E. G. Case, NRR
F. Orr, DSS
A. Ignatonis, DSS
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M. Mendonca, DOR
L. B. Marsh, DOR
D. Neighbors, DOR
J. Wetmore, DOR
T. V. Wambach, DOR
A. Burger, DOR
B. C. Buckley, DPM
A. J. Szukiewicz, DSS
J. Guibert, OCM
A. Schwencer. DOR
G. Zwetzig, DOR
S. H. Hanauer, DSS
F. Schroeder, DSS
L. P. Croker, DPM
D. Vassallo, DPM
A. Thadani, DSS
G. Lainas, DOR
D. F. Ross, DPM
D. G. Eisenhut, DOR

Carolina Power & Light

D. B. Waters
J. J. Sheppard

Shaw, Pittman, Potts & Trowbridge

J. H. O'Neill

American Electric Power Serv. Corp.

J. G. DelPeriro

Public Service Electric & Gas

P. A. Moeller

Westinghouse

R. W. Stutter
K. R. Jordan
V. J. Espusito
W. J. Johnson
T. M. Anderson

Southern California Edison

J. Rainsberry

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF INSPECTION AND ENFORCEMENT
WASHINGTON, DC 20555

Enclosure 2

APRIL 5, 1979

IE Bulletin 79-05A

NUCLEAR INCIDENT AT THREE MILE ISLAND - SUPPLEMENT

Description of Circumstances:

Preliminary information received by the NRC since issuance of IE Bulletin 79-05 on April 1, 1979 has identified six potential human, design and mechanical failures which resulted in the core damage and radiation releases at the Three Mile Island Unit 2 nuclear plant. The information and actions in this supplement clarify and extend the original Bulletin and transmit a preliminary chronology of the TMI accident through the first 16 hours (Enclosure 1).

1. At the time of the initiating event, loss of feedwater, both of the auxiliary feedwater trains were valved out of service.

2. The pressurizer electromatic relief valve, which opened during the initial pressure surge, failed to close when the pressure decreased below the actuation level.

3. Following rapid depressurization of the pressurizer, the pressurizer level indication may have lead to erroneous inferences of high level in the reactor coolant system. The pressurizer level indication apparently led the operators to prematurely terminate high pressure injection flow, even though substantial voids existed in the reactor coolant system.

4. Because the containment does not isolate on high pressure injection (HPI) initiation, the highly radioactive water from the relief valve discharge was pumped out of the containment by the automatic initiation of a transfer pump. This water entered the radioactive waste treatment system in the auxiliary building where some of it overflowed to the floor. Outgassing from this water and discharge through the auxiliary building ventilation system and filters was the principal source of the offsite release of radioactive noble gases.

5. Subsequently, the high pressure injection system was intermittently operated attempting to control primary coolant inventory losses through the electromatic relief valve, apparently based on pressurizer level indication. Due to the presence of steam and/or noncondensible voids elsewhere in the reactor coolant system, this led to a further reduction in primary coolant inventory.

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