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August 16, 1979

Bob Steitler, Westinghouse

All the following information is required:

1. Responses to all questions in 02/15/79 letter from R. Mattson except combining loads.

2. Responses to all TMI-2 related questions.

3. Responses to questions on the 06/08/79 report.

There is some duplication in items 1, 2, and 3 above; however, that just emphasizes the need for the above information.

Your August 25, 1979 letter should address these items and provide a schedule for submittal of the groups of responses.

I am telecopying these concerns to you to assist you in preparing your 08/25/79 letter. Excuse the informal nature of the enclosure

A C Thadani

Ashok C. Thadani

cc: ATWS Task Force

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TMI-2 Impact

A. Long-Term Shutdown With Stuck Open Pressurizer Safety/Relief Valve

i. The major staff concerns relate to the potential effects of the voids generated in the primary system in preventing natural circulation.

ii. Post-TMI deliberations have shown that the tripping of RCPs immediately following a small LOCA may be the most appropriate action. Provide ATWS analyses consistent with this factor.

iii. List the instruments and equipment relied on to mitigate the consequences of ATWS events and provide assurance that the instrumentation and equipment are qualified for ATWS environment.

B. Operator Actions

Provide justification for credit for operator action 10 minutes after the initiation of the postulated ATWS event. Also address the information displayed and the simplicity of operator actions.

C. Blocked PORV

The staff has learned that some plants operate with the PORV blocked because of leakage through the PORV. Since the PORV is relied on ~~(for Alt. #2 plants)~~ to provide capability to limit the overpressure during ATWS events, bases for continued operation with blocked PORV must be provided.

D. Pressurizer Safety/Relief Valve Qualification

Recent studies by the staff indicate a need for assurance that the safety/relief valves would behave as predicted in the ATWS analyses when exposed to two-phase and subcooled water conditions. Since the valves are not qualified for this environment, the staff requires that a program to verify correct valve behavior be initiated and results obtained early.

E. Address abilit: of the computer codes to correctly evaluate the consequences of voids in the primary system, the effect of changes in the water relief model, and the role of the RCPs. Long-term shutdown considerations should also address boron precipitation.

F. Provide bases for the applicability of analyses to specific plant designs so that the staff can continue with the "Early Verification" approach to resolve ATWS. In particular, address conditions and equipment.

Questions on 06/08/79 <u>W</u> Report

1. Section 3.2.4.1: Provide at power MTC measurement values and show how they conform to the 95% and 99% MTC values.

2. Section 3.2.4.3: What MTC value is used for rod withdrawal event? Why is it not a limiting event?

3. Section 3.3:
   a. Provide assurance that the equipment is qualified for ATWS environment (pressure, moisture, etc.); e.g., CVCS, controls, purification, etc.

   b. Do plants conform to the assumptions in this section and Tables 3-1 and 3.2; e.g., SG inventory, ECCS, etc.?

4. Section 5.0: Why are these events most limiting of those listed in 02/15/79 Mattson letter?

5. Section 5.0: For Alt. #4 plants with valve stuck open:

   a. Justify the applicability of the code.

   b. The role of RCP

   c. Impact of moisture on control

   d. Containment instrumentation

6. Results: Provide data beyond 150 sec.
   See Section VII-B of 02/15/79 letter.

7. Section 5.1.4: What is the impact of unavailability of any PORV.

8. Section 5.3: How is boiling in core modeled--its effect on pressure and cooldown.

9. Section 5.4: Describe the consequences using a code which accurately predicts and models voids in primary. Also provide evaluation of:

   a. PORV stuck open.

   b. If automatic ECCS signal is effective--turbine trip, etc.--discuss HPSI design effects.

   c. Role of RCPs.

   d. Long-term shutdown

   e. If 0.9 multiplier not used with HEM.

10. Section 7.0: Explain fully the effect of the following on containment conditions:

   a. Early auto containment isolation.

   b. If auto SI is unavailable.

11. Section 9.0: Totally inadequate. Answer Section IV and VII-B of 02/15/79 Mattson letter.

Questions on Analytical Methods

12. It is stated that FACTRAN uses the local conditions of the coolant (pressure, flow, temperature). However, FACTRAN inputs are obtained from LOFTRAN which calculates only average conditions. How are the local conditions input to FACTRAN?

13. FACTRAN input table shows that Nominal Hot Spot Heat Flux of 418,000 Btu/hr-Ft^2 has been used. How is this number obtained and how is it used on the code?

14. What is the convergence criteria for LOFTRAN and TRANFLO iteration? In which transients the iteration becomes necessary?

15. The data flow in the chart and input indicated in table are not consistent. The data flow indicates that LOFTRAN supplies power as input to the GHINC code. The table does not. Provide consistency.

16. Provide justification for switching from Dougal-Rohsenan correlation to Schrock-Grossman correlation at vapor fraction of 019 in TRANFLO code.

17. In the TRANFLO code, switching from one heat transfer correlation to another is based on criteria such as local boiling based on walltemperature or vapor fraction of 0.9. TRANFLO nodalization provides only one dimensional representation in the secondary side of the table region. However, the flow is three dimensional. One leg of the U tube has higher heat flux than the other leg. This promotes three dimensional flow in the secondary side. Justify complete mixing in the secondary side and show why the differences between the 3-D flow and ensuing heat transfer rates and the selected 1-D model are small or acceptable. However this affect UA versus time curve used in LOFTRAN?

18. Table 3-2 item (5) says that pump start time is not applicable (NA). Since throughout the ATWS analysis, Westinghouse takes credit for the availability for aux. feedwater system within 6, secs, the discrepancy in this table should be rectified.

19. In Section 5.4.2 on the "ATWS accidental depressurization of the reactor coolant system" normal operation of pressurizer pressure control is assumed. In this regard, Westinghouse should provide design information on these heaters, their power and control circuits and their environmental qualification to assure proper operation of the pressurizer pressure control for this event.

20. In Sections 9.0 and 10.0, Westinghouse has made a vague attempt to discuss how the ATWS mitigating systems conform to the requirements of Appendix C of NUREG 0460, Vol. 3. Since the two mitigating systems, i.e., aux. feedwater system and turbine trip, are totally in BOP scoe, Westinghouse did not provide adequate design information for us to evaluate the conformance of these mitigating systems to the above referred requirements.

    We are seeking a generic resolution to ATWS but the mitigating systems relied upon by Westinghouse in their analyses are in BOP scope. To facilitate completion of our generic review of the ATWS fix,

    a) We require Westinghouse to select a specific plant wherein these mitigating systems are presently available and provide all the design information requested by the staff in items 1X-C of our letter of February 15, 1979 for staff's review and evaluation.

b) We also require Westinghouse to distinctly stipulate all the requirements of Appendix C of NUREG 0460 Volume 3 for these two mitigating systems as interface requirements for other BOP designs as part of the ATWS resolution.

21. In Section 9.2 Westinghouse has stated that "there are no identifiable safe sutdown systems per se".

We need to clarify the above statement with Westinghouse, since standard format does require "description of the systems that are needed for safe shutdown of the plant" in SAR Section 7.4.

22. In table 9-1 item c it is stated that "BOP mitigation equipment is outside containment" and "no extreme environmental conditions will apply to BOP equipment".

Since mitigation equipment has to perform adequately to validate the ATWS analyses assumptions, we require that equipment qualification information relevant to the postulated ATWS conditions should be provided for our review and evaluation.

23. Section 3.2.2 - What is the temperature margin (temperature below saturation temp.) assumed in considering the primary pump cavitation?

24. Section 5.1.2 - What is the primary system pressure at 10 minutes into transients? Is it lower than the CVCS pumps discharge head?

25. Section 5.4.3 - During the accidental depressurization of reactor coolant system, the pressurizer becomes filled with water at 125 sec. even when no safety injection is assumed. Explain this phenomenon.

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26. Westinghouse assumes it would take ten minutes to isolate the containment and further that this isolation would be performed by the control room operator. This implies that insufficient consideration has been given to the use of other containment equipment which could produce the necessary containment isolation signal. In particular, these signals could include the high containment pressure or high radiation level signals. Based on past experience with fuel handling accidents inside containment, it is certain that with a radiation release a containment isolation signal will occur much faster than 10 minutes. These considerations could have the affect of increasing the containment pressure and thus the leak rate used in the consequence analysis. Further, it is our understanding that a containment isolation signal will isolate the primary system from the radioactivity cleanup function of the plant CVCS. How has this been incorporated into the radiological consequence analysis?

While the calculated radiological consequences may be smaller when considering the above effects, the potential capability of more rapid containment isolation from automated isolation signals such as high radiation should be addressed in the report. Operator action should be assumed only after all other pathways have been investigated.

Second, the Westinghouse assumption of leakage from the RCS in the accidental depressurization event is not consistent with the position given in the February 15, 1979 letter (page 39) and NUREG-0460, Volume 2. The staff position states that any RCS leakage to containment should have a decontamination factor of 1 for the reference case. This would not permit the use of a retention factor based on the enthalpy of the released coolant as is currently assumed in the report. The values for the reference cases presented in Table 8-4 of the report should be recalculated to take this change into account. In addition to the reference calculations, Westinghouse may submit a "realistic" calculation using the retention factors to illustrate the degree of conservatism in the staff assumptions.

Third the report did not address how the Westinghouse RCS activity (ANSI-N237) compared to the staff position of a steady state value of 0.5 uCi/gram I-131 equivalent prior to the ATWS event. Additional discussion should be provided to clarify this point.

Fourth, the use of the retention factor values for any primary to secondary leakage is inconsistent with the staff position contained in NUREG-0460, Volume 2, Section VI. This NUREG section states that primary to secondary leakage should assume a decontamination factor of 10 (i.e., 10% of the iodine leaking to the secondary side is assumed released to the environment). This correction should be made for the reference case calculations presented in Table 8-4 of the June 8, 1979 report.

Fifth, was the RHR leakage of 1600 cc/hr (per Table 8-1) considered when calculating the reference case radiological consequences presented in Table 8-4. If not, this item should be addressed and all appropriate assumptions used in that calculation should be presented in the report.

Sixth, information appears to be missing from Table 8-1 of the report. The last two lines of that table do not address RHR leakage to the auxiliary building. Provide the necessary reference.

Lastly, if Westinghouse wishes to continue the retention factor approach for a "realistic" case calculation, additional information must be provided. After reading the definition of retention factor in Section 8.2 and seeing its application throughout Section 8 of the report, I'm totally confused about what a retention factor really describes. Example, if a retention factor of 1 means that all the iodine is released (I assume it does), what does a retention factor less than 1 mean? Another example, as discussed in Section 8.2, a retention factor of 2 was used at full power reactor coditions and 100 after the coolant temperature was brought below 212°F. However, in Section 8.4 a retention factor of 0.10 is used for RHR leakage 6 hours after the accident. It would appear logical to me to expect that if RHR coolant temperatures are approaching 212°F as they would be then the retention factor should be at least 100. This type of inconsistency should be removed from the report. Also the appropriate equations and the necessary parameters used in the calculation of the retention factors should be provided.

27. Discuss impact of high pressure on the CVCS and makeup ~~of~~ and purification systems.

28. ATWS - TAP A-9 - Preliminary MEB Comments - W - Submittal of 6/8/79

Components

Valves - Structural Integrity of bodies and bolts - OK

Discs - need confirmation that Design Specs. did not specify a lower hydro test pressure than the ASME Section Table NB-3131-6 requirement of 3725 psia.

Further Info needed - Confirm that this justification is applicable for all NSSS and BOP supplied ASME Cl.1 Valves.

Safety and Relief Valves - No Info. on Struc. integrity or operability.

Reactor Vessels - Vessels for Alternate 3 plants are furnished to W from at least 5 vessel manufacturers. Confirm that the "review of reactor vessels" stated in 6.2.1 encompassed each size vessel designed by each of the vessel manufacturers.

Pressurizers - Most pressurizers for Alternate 3 plants are manufactured at the W-Tampa facility. Pressurizers for some early Alternate 3 plants were designed by others. Confirm that the allowable pressures reported are applicable to all Alt. 3 pressurizers. Provide confirmation that for the max pressure calculated for the loss of load transient (3021 psi), the pressurizer manway cover gasket and heater to bottom head welds retain their integrity. If the pressurizer heaters must function to insure safe shutdown provide assurance that the heater tubing i.e., sheaths will remain integral under an external overpressure of 3021 psi.

Steam Generators - Provide confirmation 6.2.3 is applicable for all sizes and designs of Alternate 3 plant steam generators. Specifically confirm that those manufactured by other than W - Tampa were reviewed also.

Piping (Exc. Safety & Relief Valve Disch.) - o.k.

Reactor Coolant Pump - If pump must be operable to assure plant shutdown, provide results of detailed analyses or tests that demonstrate pump operability during and after exposure to the 3021 psia, LOL ATWS pressure. Discuss the effect of the ATWS pressure on reactor coolant pump seal integrity.

CRD's - o.k.

Instrumentation - Provide assurance that instrumentation essential to safe shutdown will function after exposure to the 3021 psi pressure.

Safety Relief Valve Disch. Piping - Qualitative description of method of analysis provided in 6.2.8 is applicable for those piping systems which are analyzed by W. For many Alternate 3 plants, such piping is in the BOP scope of supply. In general the description provided is not responsive to the information requested by item VIII.B.1.d(2) of the February 15 questions. Results of analyses performed are to be provided, not just qualitative descriptions of an analytical method. Information provided should include justification for the adequacy of the loads used in the analysis that are imposed on the piping and supports that result from the continuous subcooled liquid discharge through the safety and relief valves. If computer codes are used for performing the analyses, justification shall be provided as to the adequacy of the code.

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Pressurizer Quench Tanks - No information provided.