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DCP/NRC1149
NTD-NRC-97-5450
Docket No.: 52-003

November 21, 1997

Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, DC 20555

ATTENTION: T. R. QUAY

SUBJECT: AP600 RESPONSE TO FSER OPEN ITEM 720.429F AND TO REQUEST FOR ADDITIONAL INFORMATION

Dear Mr. Quay:

Enclosure 1 of this letter provides the Westinghouse response to FSER open item 720.429F. This open item pertains to the AP600 PRA in-vessel steam explosion topic. The OITS number associated with this open item is #6144. The Westinghouse status column in the OITS will be changed to "Confirm W" until PRA Revision 11 is issued.

Enclosure 2 provides the Westinghouse response to RAI 492.14. The OITS number associated with this RAI is #2987. This response closes, from the Westinghouse perspective, the RAI. The Westinghouse status column in the OITS will be changed to "Action N."

The NRC should review the enclosed responses and inform Westinghouse of the status to be designated in the "NRC Status" column of OITS.

Please contact Cynthia L. Haag on (412) 374-4277 if you have any questions concerning this transmittal.

Brian A. McIntyre

Brian A. McIntyre, Manager
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Enclosures

cc: W. C. Huffman, NRC (Enclosure 1)
J. M. Sebrosky, NRC (Enclosure 2)
N. J. Liparulo, Westinghouse (w/o Enclosures)

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Enclosure 1 to Westinghouse
Letter DCP/NRC1149

November 21, 1997

Question: 720.429F (OITS #6144)

FSER OI Pertaining to In-Vessel Steam Explosion:

Although the report "Lower Head Integrity Under In-Vessel Steam Explosion Loads," DOE/ID-10541 is referenced in PRA Chapter 39 the following companion reports are not referenced in the PRA: "Pre-mixing of Steam Explosions: PM-ALPHA Verification Studies," DOE/ID-10504, and "Propagation of Steam Explosions: ESPROSE.m Verification Studies," DOE/ID-10503. The staff believes these reports should also be listed as a reference to PRA Chapter 39.

Response:

The two companion reports DOE/ID-10503 and -10504 will be included in the references of Chapter 39 in Revision 11 of the PRA.

SSAR Revision: None

PRA Revision:

The attached two pages illustrate the change that will be made in Chapter 39 for Revision 11 of the PRA.

Enclosure 2 to Westinghouse Letter DCP/NRC1149

November 21, 1997

Question: 492.14 -

Table 54-53 summarizes the MAAP4 analysis results of ADS success criteria for shutdown conditions.

a. For the sequences with manual actuation of various ADS stages (3 stage-2 and 3 valves, or 1 stage-4 valve), the results of the actuation times of 30, 60, and 120 minutes (from the event initiation) show that the cases with 60 minutes actuation time give either the highest or the lowest PCT among the three cases. What are the actual physical explanations of these phenomena?

b. For the manual ADS actuation sequences (for ADS success criteria ADTS, ADLS, and ADNS), no results are shown for actuation times of less than 30 minutes. How do you ascertain that ADS actuation earlier than 30 minutes will not result in higher peak cladding temperature than those analyzed?

c. The results for success criterion ADNS for the RNS line break sequence with manual ADS actuation are from the analysis of one break size (2000 gpm) only. Page 54-44 indicated that an RNS line break may have a maximum break flow of 3500 gpm (see RAI #492.13). Justify why this (2000 gpm) is sufficient to cover other break sizes.

Response:

a. In the manual ADS scenarios, there is a trade-off between the decay heat level and the RCS inventory that has been lost when ADS is opened. It can be a benefit to delay manual ADS since this allows a decrease in the decay heat. However, at some point the decrease in the RCS inventory has a greater impact than the decrease in decay heat. In the cases with operator action times at 30 minutes, 60 minutes and 120 minutes, the 60 minute case can produce either the highest or lowest PCT, depending on how the decay heat level balances with the RCS inventory loss up to that point in time.

b. These sequences are loss of heat removal scenarios, in which the loss of RCS inventory does not occur until the RCS heats up, and the RNS relief valves open. The RNS valve does not open within the first 30 minutes of the accident, so loss of RCS inventory cannot occur prior to 30 minutes. Furthermore, the analyses are for shutdown modes when the reactor has been shutdown for at least 8 hours. Therefore, if ADS were manually opened earlier than 30 minutes, when the decay heat is higher, the plant response would be much less limiting than the spurious opening of ADS from full power.

c. The RNS line break cases with automatic ADS actuation were performed with a spectrum of break flowrates. In the context of determining whether core damage or successful core cooling occurred, no significant sensitivity to the break flowrate was found. Therefore, 2000 gpm was used for the manual ADS cases to limit the number of parameter changes and cases. Also note that the cases examine operator action times much greater than were credited as successful core cooling in the shutdown PRA.

SSAR/PRA Revision: None.

the completion of Reference 39-1, confirm the heat transfer assumptions at Rayleigh numbers beyond 10^{15}.

The full-scale, slice-geometry ULPU testing, shown in Figure 39-3 (Reference 39-1), investigates the critical heat flux on the external surface of the lower head of the reactor vessel. The test provides full-height water elevation capability to investigate the effects of varied water height and subcooling. The test determines critical heat fluxes at the various azimuthal locations on the lower head external surface. Advanced ULPU Configuration III testing provides data for prototypical reactor vessel steel material with surface preparations to Westinghouse specifications and external cooling water flow restrictions to model the effect of reactor vessel reflective insulation. This test is also used to provide oscillatory pressure data for the reactor vessel insulation design.

The ROAAM analysis also investigates transient aspects of core relocation to the lower head and development of the steady-state heat transfer system described above. Investigations of lower head vessel failure due to jet impingement (Reference 39-1) and in-vessel steam explosion (References 39-2, 39-6, 39-7 and 39-8) have been performed and it is concluded that these phenomena will not fail the vessel. Investigations of the transient development of molten pool conditions conclude that the steady-state heat fluxes bound the transient conditions. Therefore, vessel failure prior to the development of the natural circulating pool and external cooling is physically unreasonable.

Add for PRA Rev. 9

The results of in-vessel retention ROAAM analysis have been peer-reviewed by an international panel of 17 experts in the fields of severe accident progression, heat transfer, thermal-hydraulics, and structural mechanics. The conclusion that vessel failure is physically unreasonable under thermal-hydraulic conditions of in-vessel retention is considered to be resolved and is credited in the AP600 PRA, provided that the sequence meets the criteria outlined above.

Based on the results of the ROAAM testing and analysis, vessel failure is concluded to be physically unreasonable in the AP600 PRA provided the following conditions are met:

- The reactor coolant system is depressurized.
- The vessel is submerged above the top of the molten debris pool.
- Reactor vessel reflective insulation allows the ingress of water at the bottom and egress of steam at the top.
- The reactor vessel external surface conditions do not preclude the wetting phenomena identified as the cooling mechanism in the ULPU testing.

Each of these items is discussed below.

For all accident classes except 3C (vessel rupture initiating event), maintaining the debris in the vessel is ensured by vessel integrity (success at nodes IR and DP). In accident class 3C, the vessel is failed below the intact core as a result of the initiating event. Since vessel rupture produces core damage, regardless of system availability, the failure of ADS and gravity injection has negligible frequency in accident class 3C. Core damage is caused by the inability to reflood the core until the reactor cavity is filled. AP600 has the unique cavity flooding feature that, once the cavity is filled up to the break, water can reflood back into the vessel as the containment compartments fill to arrest core damage before full core relocation. Only a limited amount of debris is likely to relocate to the lower head. The most likely failure for the reactor vessel initiating event is a local failure above the top of the lower head hemisphere at the beltline of the vessel. This location has the highest fluence and brittleness from exposure. Debris relocated into the lower head is guaranteed to be water cooled in the vessel. Therefore, for accident class 3C, a scalar failure probability value of 0.1 for debris relocation is assigned to node VF. A sensitivity to this value is investigated and discussed in Chapter 43.

39.8 Summary

The fault trees and scalar values linked for nodes IR and VF are summarized in Tables 39-2 and 39-3, respectively.

39.9 References

39-1 Theofanous, T.G., et al., "In-Vessel Coolability and Retention of a Core Melt," DOE/ID-10460, July 1995.

39-2 Theofanous, T.G., et al., "Lower Head Integrity Under In-Vessel Steam Explosion Loads," DOE/ID-10541, ~~released for peer review, July 1995.~~ June 1996.

39-3 Theofanous, T.G., "On the Proper Formulation of Safety Goals and Assessment of Safety Margins for Rare and High-Consequence Hazards," Reliability Engineering & Systems Safety, Summer 1996.

39-4 Theofanous, T.G., et al., "The Probability of Mark-I Containment Failure by Melt Attack on the Liner," NUREG/CR-6025, November 1993.

39-5 Theofanous, T.G., et al., "The Probability of Containment Failure by Direct Containment Heating in Zion," NUREG/CR-6075, December 1994.

39-6 Theofanous, T.G., et.al., "Premixing of Steam Explosions: PM-ALPHA Verification Studies," DOE/ID-10504, September 1996.

39-7 Theofanous, T.G., et.al., "Propagation of Steam Explosions: ESPROSE.m Verification Studies," DOE/ID-10503, August 1996.

39-8 Theofanous, T.G., Volume 1 - "Appendices E, F, and G to DOE/ID-10541," and Volume 2 - "Addenda to DOE/ID-10541, -10503, -10504," October 1997.

Add to PRA

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