APPLICANT: Westinghouse Electric Corporation August 31, 1995

FACILITY: AP600

SUBJECT: SUMMARY OF MEETING TO DISCUSS ISSUES CONCERNING IN-VESSEL RETENTION ON THE AP600

On April 27, 1995, representatives of the Nuclear Regulatory Commission and Westinghouse Electric Corporation met to discuss issues concerning in-vessel retention capability of the AP600 following a severe accident. Attachment 1 is a list of attendees. Both proprietary and non-proprietary versions of the slides and information presented by Westinghouse were submitted by letter dated May 1, 1995.

Westinghouse provided a description of the cavity flooding system for the AP600, followed by a discussion of the timing for flooding. The participants then discussed certain issues concerning in-vessel retention, including the potential for "hot spots" and reactor vessel insulation. The staff indicated that they felt that the AP600 design for retaining corium in-vessel had merit, but that there were concerns that needed to be addressed. Movement of the insulation during and after the course of an accident was a key issue that needed to be addressed. The staff also indicated that they would be reviewing the in-vessel analysis submitted by Westinghouse in draft form after the peer review was completed. Westinghouse agreed to provide the staff access to the peer report.

The participants then discussed Westinghouse's responses to the staff's requests for additional information. Attachment 2 is the status of the followon questions that were discussed at the meeting.

Original signed by
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Docket No. 52-003

Attachments:
As stated

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Westinghouse Electric Corporation Docket No. 52-003

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IN-VESEL RETENTION
ATTENDANCE SHEET
APRIL 27, 1995

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JOHN MONNINGER	NRC/NRR/DSSA

Attachment 1

Status as agreed to at 4/27/95 W/NRC meeting on IVR issues (6B11 White Flint)

FOLLOWON QUESTIONS FOR THE WESTINGHOUSE AP600 DESIGN

EXTERNAL VESSEL COOLING

480.80 Describe the indications that the operators will use to flood the reactor cavity with the in-containment refueling water storage tank (IRWST). WCAP-13388 and Appendix R of the probabilistic risk assessment (PRA) indicate that this has not been finalized; however, the analysis assumes that it is based on a core exit temperature of 2000 °F.

Action W

480.81 Provide the equipment qualification (design-basis accident requirements, if any) and equipment survivability (as discussed in SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs") requirements for the instrumentation and equipment (for example, motor-operated valves) used to initiate reactor cavity flooding along with an analysis of the environmental conditions this instrumentation and equipment must function in resulting from severe accident conditions, along with the time periods.

Action W

480.82 Discuss the incorporation of reactor cavity flooding into the emergency operating procedures and accident management plan. What time span is assumed for the operator to actuate flooding once the core exit temperatures indicate 2000 °F? Was automatic initiation of cavity flooding considered in light of potential hesitancy of operators to initiate flooding?

Action W

480.83 Discuss the power sources for the instrumentation and equipment used for reactor cavity flooding and its availability in all accident sequences. Appendix R of the PRA indicates that the failure of cavity flooding is based on the failure rate of motor-operated valves (0.022 per demand). However, it does not appear that the accident initiator was factored into the credit taken for flooding. Do the motor operated flooding valves fail open on loss of power?

Action W

480.84 Discuss the surveillance and maintenance requirements for all instrumentation and equipment used for reactor cavity flooding.

Action W

480.85 What is the basis for the sizing of the 4-inch and 10-inch lines from the IRWST? Why weren't two 10-inch lines selected? Figure R.1-5 of Appendix R of the PRA indicates that the 4-inch line provides minimal benefit over the 10-inch line when used together and that, when only the 4-inch line is actuated, the time to flooding the reactor cavity to the elevation of the top of the debris pool increases by a factor of 6. What is the success criteria for the level of cavity flooding at the time core debris reaches the reactor vessel lower head?

Action W

Attachment 2

480.86 The effects of delayed cavity flooding need to be further evaluated. Previous experimental studies of melt pool heat transfer have shown that the local wall heat flux is highest near the upper edge of the pool. If delayed cavity flooding is the case, then it is likely that the upper edge of the lower head is not submerged for a prolonged period of time. During this period, local hot spots may develop in the vessel wall near the upper edge region. Localized creep-induced failure or melt-through of the vessel may occur, depending on the local heat flux level and the thermal mass of the wall.

Action W

480.87 Provide an assessment of situations that may arise if the cavity is flooded after some fraction of core material has already relocated to the vessel lower head.

Action W (same as 86 response)

480.88 Appendix R of the PRA states that testing of prototypical reflective insulation leaktightness for accident management purposes has been performed by Fauske and Associates, Incorporated. The staff believes the testing illustrates that water ingression may occur, but this has not been demonstrated sufficiently to make an adequate safety case. In order for credit to be taken for external reactor vessel cooling, controls over the reactor vessel insulation procurement, installation, and maintenance will be necessary to ensure the ingression of water and escape of steam. Provide a discussion of the proposed inspections, tests, analyses, and acceptance criteria; testing; surveillances; procurement specifications; maintenance procedures, and other methods that will ensure adequate control over the insulation. Discuss any proposed additional testing of the insulation to demonstrate that water ingression will occur. Prototypical conditions of insulation configuration should be considered while making this assessment. Westinghouse's responses to Q480.19 and Q720.219 do not sufficiently address these issues.

Active

480.89 Appendix R of the PRA states that there are provisions in the design of the insulation to allow the water in and steam out. Provide a discussion of the specific engineered provisions, and how these provisions will be controlled. The staff does not believe that potential insulation gaps are sufficient to ensure water ingression and steam venting. WCAP-13388 indicated a need for a steam venting pathway from the insulation. Has an engineered pathway been incorporated?

Active

480.90 Provide an analysis of the impact of a dislodged insulation panel resting against the reactor vessel, thereby inhibiting cooling or restricting the steam vent paths. This could result in local dryout and hot spots. Appendix R of the PRA states that forces exerted on the insulation panels and frame due to pressurization by steam generated inside the insulation are estimated to be capable of blowing out panels. Where do the insulation panels go, since the gap between the insulation and the wall of the reactor cavity is very small? Appendix R of the PRA indicates that the typical gap between the insulation and vessel wall is 9 inches; however, Figure 2-4 of

Active

WCAP-13388 appears to indicate substantially less distance in the hemispherical area, and that the 9-inch gap is only applicable to the cylindrical section. If the panels are displaced, it appears that they would block the annulus. What is the minimum distance between the reactor vessel and insulation? Has the potential for dryout and "hot spots" in this area been evaluated?

480.91 Because the inside radius of the insulation is less than the outer, can an argument be made that the external force on the insulation from the flooding of the reactor cavity forces the panels closer together, thereby providing less of a leakage path? As the reactor cavity floods up, the panels may vertically displace sealing joints as they may be less dense than water. Why is this not the case?

Active

480.92 Provide more detailed drawings of the insulation placement around the reactor vessel showing areas of constriction and attachment to the cavity wall. The drawings in WCAP-13388 and Appendix R of the PRA are not sufficient to draw conclusions regarding their acceptability. Describe more thoroughly the mechanism and framework for attaching the insulation to the reactor cavity walls. Appendix R of the PRA indicates that the bottom panels weigh less than 50 pounds and are held in place by their own weight. Provide the prototypical size of the panels. How many does it take to cover the bottom? What is the specific arrangement?

Active

480.93 Appendix R of the PRA states that the insulation panels are not designed for seismic events or to withstand forces other than gravity, and that forces exerted on the insulation panels due to steam pressurization are estimated to be capable of blowing out panels. What assurance is there that the insulation panels won't deform in a manner that hinders external vessel cooling?

Active

480.94 WCAP-13388 discusses the ability of the reactor cavity configuration to release the steam generated. The adequacy of the vapor venting is demonstrated by calculating the steam generation rate and resulting steam velocity through the 9-inch annular gap between the vessel wall and insulation panels. This approach is acceptable as long as the vapor bubbles are small enough to pass through the minimum gap. If vapor bubbles are relatively large, then assessment of the adequacy of the vapor venting capacity cannot be based on the steam velocity alone. The size of the vapor bubbles must also be considered. Experiments have shown relatively large vapor bubbles being released from the bottom center of the reactor vessel. Depending on the conditions of downward facing boiling on the vessel outer surface, it is possible that the steam generated in the water pool may contain relatively large vapor bubbles. In addition, generated vapor bubbles can combine with bubbles generated downstream increasing their size. The WCAP appears to indicate constriction between the lower vessel head and insulation relative to the rest of the annulus.

Active

480.95 How is the counter-current flow of steam and water affected, if the water ingression is from the sides of the insulation rather than the bottom?

Active

480.96 For each severe accident sequence in which credit is taken for reactor cavity flooding, provide an event sequence description (with timing) that indicates transient initiation, scram, containment isolation, key operator actions, flooding levels in the reactor cavity, temperature of IRWST and cavity water, onset of core damage, and core relocation.

Active

480.97 Provide an analysis of the long term effects of reactor cavity flooding, including decreases in the amount of subcooling and an assessment of thermal stratification within the reactor cavity. With time, the reactor vessel walls and water in the reactor cavity will heat up.

Resolved

480.98 The analysis in Appendix R of the PRA is based on a metallic pool layer on top of an oxidic pool layer in the lower reactor vessel. Provide an evaluation of other scenarios, such as an all oxidic pool, homogenous pool, and metallic pool initially present on the bottom.

Action W

480.99 Provide the AP600 reactor vessel structural analyses that demonstrates that virtually complete melt-through of the reactor vessel wall is required to fail the vessel. This analyses was referenced in Appendix R of the PRA. The figures in Appendix R indicate substantial vessel wall melt-through; however, failure has not been identified.

Action W

480.100 In Appendix R of the PRA, the wall heat fluxes from the melt pool to the reactor vessel are calculated by assuming that the pool is in steady-state heat balance. Provide a transient analysis to predict the wall heat fluxes from the melt pool to the reactor vessel. The melt pool heat transfer regime may vary from one region of the melt pool to another and, within the same region, the heat transfer regime may change with time (i.e., varying from conduction to laminar natural convection and then to turbulent natural convection). Thus, the melt pool heat transfer is highly transient. In addition, the melt pool may not be well mixed and, as such, the pool temperature cannot be described by a single value. Depending on the regime of heat transfer, the pool temperature could be highly non-uniform. In light of the fact that the melt pool heat transfer is highly transient and the heat transfer regime may vary both spatially and with time, the correlations of Mayinger may not be applicable. This is especially true when debris crust is present on the boundaries of the melt pool. Because of the growth and remelting of the crust, the no-slip condition at the solid/liquid interface is no longer valid. Rather, a moving boundary condition is imposed on the fluid motions in the melt pool. This moving boundary condition, which was not accounted for in the correlations of Mayinger, could substantially alter the boundary heat fluxes from the melt pool.

Active

480.101 How has the potential for "hot spots" (similar to that which occurred at Three Mile Island) been accounted for in the potential for local dryout and reactor vessel failure in the analysis?

Action w

480.102 Appendix R discusses the heat flux from the core debris through the vessel wall as a function of the angle of the lower head, based on the results of the COPO and UCLA experiments. In addition, the critical heat flux values from the ULPU experiments as a function of the angle of the lower head are discussed. Westinghouse should discuss why the results of the COPO, UCLA, and ULPU experiments are applicable to the AP600 design.

Active

480.103 How is the heat from metal-water reactions within the lower head accounted for in the heat flux from the core debris to the reactor vessel lower head?

Resolved

480.104 Appendix R of the PRA assumes that 90 percent of the time, less than 75 percent of the active cladding has been oxidized. A review of Sequence 3BE in Appendix L of the PRA (which represents 30 percent of the core melt frequency) indicates almost 100 percent metal-water reaction. How is the 90 percent factor justified?

Resolved

480.105 Provide a discussion of the decay power curve for all plausible burnup conditions to support the 1.0 MW/m^3 volumetric heat rate of the oxide pool. Identify the minimum time after scram that external vessel cooling is needed.

Action w
(like 86)

480.106 Appendix R of the PRA indicates that the same conditional probability of reactor vessel failure is applied to all sequences regardless of the initiating event and melt progression. Justify this assumption, or demonstrate that this is a bounding conditional probability value for reactor vessel failure.

Action w
(like 86)

480.107 The conditional probability of reactor vessel failure can easily increase by a factor of 100. If the heat flux values in Table R.1-1 of Appendix R of the PRA are increased by only 8 percent or the critical heat flux decreases by 7 percent, the following sequences result in reactor vessel failure: IVR.4, IVR.7, IVR.8, IVR.12, IVR.15, and IVR.16. This uncertainty is within the extrapolation of the testing results (COPO, UCLA, ULPU) to the AP600 design. Provide further justification for use of the critical heat flux values and internal heat flux values.

Resolved

480.108 Section R.1.5.5 indicates success if the critical heat flux at all points is less than the core debris heat flux. Using this basis, Sequences IVR.8 and IVR.16 were identified as failures. However, looking at Figures R.1-22 and R.1-38, the reactor vessel thickness is only about 3.25 cm for these two cases. Using a criteria of reactor

Action N

vessel thinning, from 15 cm to less than 4 cm would result in significantly more failure cases (IVR.4, 7, 8, 11, 12, 15, and 16). Discuss how the failure of the reactor vessel is defined in Appendix R of the PRA and why.

480.109 Appendix R of the PRA calculates the heat flux across the vessel wall and the pool temperature. Using this, the temperature distribution in the vessel wall should be calculated and, in turn, used to determine the potential for creep rupture failure. Provide these results along with the temperature distribution of the lower head for various sequences analyzed in Appendix R.

Action N

480.110 What is the status of the analysis in WCAP-13388? The assumptions within Appendix R of the PRA and WCAP-13388 are not identical. The heat flux at the edges of the molten pool and bottom of the vessel in the WCAP are substantially less than that in Appendix R.

Resolved

480.111 Appendix R of the PRA states that the crust formed on the reactor vessel wall helps protect the wall from the high temperature debris and limits the heat flux that can be transferred to the wall. The staff disagrees with this statement. The debris crust cannot protect the vessel wall. The melting point of steel is well below the freezing temperature of the molten fuel. Thus, melting of the steel wall could occur underneath the crust. The crust does not play an important role in preventing thinning of the wall. This is shown in the figures in Appendix R, which show substantial thinning of the reactor vessel wall. Justify or clarify this statement.

Resolved

480.112 The pool superheat and heat fluxes in the upward and downward directions are determined by making an overall energy balance under steady-state conditions. This approach is only valid as long as the pool has a crust all around it, and the pool is well mixed. It is possible that a layer of superheated steel will form on the molten pool. Perform an analysis of melt superheats and partitioning of heat fluxes in this configuration.

Closed

480.113 In the event that the reactor vessel fails near the top of the melt pool, there is the potential for an in-vessel steam explosion. Because of the hydraulic head of water in a fully flooded cavity, the water would flow into the reactor vessel. Has an assessment of this scenario been made? Address this issue.

Resolved.

480.114 Provide a discussion on the results of the COPO, UCLA, and ULPU experiments, and how these are applicable to the AP600. Examples of areas to discuss include: dimensions, scaling, prototypicality, shape, incorporation of insulation, temperature of water body, melt composition, melt superheat, time scale, and test matrices. Appendix R of the PRA indicates that the heat fluxes from the ULPU experiments are interpolated between 0 and 60 degrees. Provide the exact locations were the heat fluxes were measured in the experiments. See Q480.102.

Active

480.115 WCAP-13388 references the CECO sponsored experiments performed by Fauske and Associates. However, Appendix R of the PRA does not appear to rely on this information. Are the CECO experiments still used to justify the analysis in any manner?

Resolved.

SEVERE ACCIDENT HYDROGEN GENERATION AND CONTROL

480.116 10 CFR 50.63(a)(2), "Loss of All AC Power," requires that the reactor core and associated coolant, control, and protection systems, including station batteries and any necessary support systems, must provide sufficient capacity and capability to ensure that the core is cooled and appropriate containment integrity is maintained in the event of a station blackout for the specified duration. The PRA for the AP600 shows that station blackout sequences are a significant contributor to overall plant risk. The staff believes that hydrogen igniters are necessary to ensure containment integrity during a station blackout. Therefore, discuss the availability of the igniter system during various sequences, including station blackout.

480.117 In previously reviewed designs, the staff has viewed power diversity and redundancy as important elements to demonstrate availability of the igniters. However, the AP600 igniter system is single train, and, besides the normal onsite and offsite power supplies, the non-safety-related diesel generator is the sole emergency power source. Discuss why this design provides sufficient quality, redundancy, and diversity in accordance with past practice established during the review of the evolutionary designs.

480.118 The hydrogen control system includes recombiners for design-basis events. The staff views equipment needed for design-basis events as safety-related. Discuss why the proposed non-safety-related power supplies are acceptable for a design basis event.

480.119 The staff is concerned about the effect of diffusion flames anchored to the IRWST vents on the containment shell.

480.120 Lumped-parameter codes have limitations when used to predict hydrogen distribution in containments. Lumped-parameter codes tend to over-predict the rate of mixing that can result in under-predicting local hydrogen concentrations. For example, in Test E11.2 performed at the HDR test facility, the actual helium gas concentration in the upper dome region of the containment was 3 times larger than the value the CONTAIN code (a lumped parameter code) predicted at the point of largest discrepancy (25 percent measured versus 8 percent calculated concentration). On what basis does Westinghouse conclude that lumped parameter codes are adequate to predict hydrogen mixing? Also, how is the subnodal physics model capable of sufficiently predicting hydrogen stratification?

480.146
480.142 } Resolved.

480.147, 148, 149, 138, 141
Active

480.143 — Action W — 480.140, 144