MAY 28 1986

Docket No. STN 50-601

Mr. E. P. Rahe, Jr., Manager
Nuclear Safety Department
Westinghouse Electric Corporation
Water Reactor Divisions, Box 355
Pittsburgh, Pennsylvania 15230

Dear Mr. Rahe:

SUBJECT: WESTINGHOUSE ADVANCED PRESSURIZED WATER REACTOR (WAPWR), RESAR-SP/90, REVIEW OF MODULE 16

We are continuing our review of the RESAR-SP/90 PDA Module 16 and have identified the need for additional information.

Additional information needed for our review of Volumes 1 and 2 is addressed in enclosure 1. These questions have been classified into three areas: modeling, assumptions, and documentation.

Additional information needed for our review of Volumes 3 and 4 is addressed in enclosure 2. These questions have been classified into two types: information needed to support our independent modeling efforts and needs for correction or justification of information already provided.

You are requested to provide your response to the questions identified in the enclosures within 30 days of the date of this letter. If additional information pertaining to other sections of Module 16 is needed, we will notify you promptly.

Sincerely,

Steven M. Long, Project Manager
PWR Project Directorate #5
Division of PWR Licensing-A

Enclosures:
As stated

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
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Sincerely,

A. Vietti-Cook for

Steven M. Long, Project Manager
PWR Project Directorate #5
Division of PWR Licensing-A

Enclosures:
As stated

ENCLOSURE 1

Questions Resulting from Review of
RESAR-SP/90, Module 16, Volumes 1 & 2

A. Modeling

720.1 Some internal initiating events (IEs) that were significant in other similar studies (e.g., Millstone 3 PSS) are not addressed in the SP/90 PSS. Please provide the basis for disregarding the following initiators:

- loss of dc power (partial and total),
- loss of instrument and control power, and
- loss of instrument air.

These events are often described as common-cause initiators because, in addition to causing a reactor trip, they may also impair the operability of systems required to recover from the trip. The inclusion of these events in the fault trees, when using a large event tree approach, is equivalent to accounting for a common event appearing n times as n independent events, leading to an underestimation of the sequence frequencies. This is true of dc power for example, if in fact loss of dc leads to a trip. Similarly, if 120-V ac vital bus power is lost, several major safety systems would be affected in ways which have not been treated explicitly. It is not clear that this initiating event is negligible for APWR either. It is our impression that loss of instrument air will not affect the APWR as strongly as it affects certain other plants, but this needs to be confirmed.

For example, dc bus unavailability appears in the system fault trees. In a fault tree linking approach, the multiple effects of a loss-of-dc initiating event can straightforwardly be assessed. In the large event tree approach used in the submittal, the corresponding calculation would involve development of a support state corresponding to a loss of a given dc bus, and computation of branch point probabilities conditioned on this support state. It is our impression that the submittal has chosen not to do this, in the belief that these sequences are negligible in the context of a plant with extremely low overall core damage frequency.

720.2 Some combinations of events which have been considered significant in other similar studies, like

- sticking open SRV(s) following station blackout, and
- ATWS in conjunction with SGTR,

apparently were not addressed in SP/90 PSS.

A secondary break or stuck-open SRV in conjunction with a blackout is relatively unlikely; on the other hand, this sequence may overcool the primary system leading to a potential of recriticality. Safety injection is required to provide boration but is not available due to station blackout. It is not clear that this sequence is negligible, compared to the other sequences included in the study. Please provide justification for excluding these combinations of events from the SP/90 PRA.

720.3 The probability of an Interfacing Systems LOCA causing a break outside containment was calculated as 5.8×10^{-3}. Provide the analysis developed to justify the low failure probability of the RHR suction lines outside containment when exposed to the nominal RCS pressure (2250 psia).

720.4 How is long term cooling (LTC) quantified for the following initiating events (IEs):

- steam generator tube rupture (SGTR),
- large secondary side break (SSB),
- ATWS, and
- loss of auxiliary cooling?

Also, in the transient event tree where q=0 for LTC (p. 3.12-1) why are success/failure branches of LTC included in sequences where secondary cooling is successful? In the ATWS event tree these branches are not included. Please explain. Furthermore, what are the values of LTC1, LTC2, and LTC3 for these event trees?

720.5 Turbine Trip (TT) or MSIV closure are important in ATWS sequences. Why is the heading Turbine Trip (TT) or MSIV closure not included in the ATWS event tree? Does SP/90 have an anticipatory TT logic for ATWS scenarios?

720.6 For the SSB and SGTR initiators, it is appropriately stated that no credit is taken for operator actions in the success of safety injection (SI1). Therefore, the unavailability of SI1 should be 1.8×10^{-5} (SS2) in these event trees. The sequence #15 in the SSB event tree, which is SSB and failure of SI1, should have a probability of

$$8\times10^{-4} \times 1.8\times10^{-5} = 1.44\times10^{-8}.$$

This would rank this dominant accident sequence (DAS) as #12 and not as #37, as shown in Table 4.3.1. Similarly, the sequence #31 in the SGTR event tree, which is SGRT, failure of safety injection, and operator fails to stabilize, should have a probability of

$$3.1\times10^{-2} \times 1.8\times10^{-5} \times 1\times10^{-2} = 5.58\times10^{-9}$$

This would rank this DAS as #20 and not as #42, as shown in Table 4.3-1.

720.7 What considerations have been included in the new design of the emergency water storage tank (EWST)? Is it possible to have overflow conditions due to extra inventory from the accumulators, core reflood tanks, and RCS (Reactor Coolant System) under large LOCA? If so, are there components inside the containment that may be affected?

720.8 If there is a break outside containment in the RHR suction line, or HHSI suction line the EWST inventory may be depleted if the MOVs are not closed. Can neglect of these sequences be justified?

720.9 Please discuss the significance of inadvertent opening of the valves in the steam generator overfill protection system.

720.10 For a steam generator tube rupture initiating event, it is possible to have stuck open secondary PORVs or safety valves. It appears that these sequences were not included in the anlysis. Can neglect of such sequences be justified?

720.11 Please provide a revised heat load table for the SW-CCW system including component identification (more specific than just the component type). In addition, if module 13 for auxiliary system is available, please provide it. Otherwise, provide P&ID diagrams of the SW-CCW system and other information required to complete the review of the SW-CCW system fault tree.

720.12 Operator action to feed and bleed the reactor coolant system has been given credit (unavailability of 0.01) if the steam generator overfill protection system (SOF) fails. Is this action different from the OST (operator stabilizing the transient) already modeled in the event tree? For example, sequence #11 involves failure of both the OST and SOF. If OST fails, why does the SOF node still include the success of feed and bleed action? How much time is available to the operator between the failure of OST and the challenge of SOF?

720.13 Should core melt sequences involving the sticking open of PORVs or the leakage of RCP seals be categorized into small LOCA core melt instead of transient core melt for the purposes of back-end analysis? For example, transient event tree sequences 4, 6, 8, 10, etc. involve small break and are categorized as TLFC, TLC, TLF, and TL, respectively.

B. <u>Assumptions</u>

720.14 What is the basis for assigning to the consequential LOCA frequency a value equal to 20% of the random small LOCA frequency?

720.15 Some success criteria were assumed which are not included or are a relaxed version of the SAR success criteria, without any specific system analysis justification. Among these are:

- feed and bleed (one of three PORVs),
- ATWS pressure relief (3/3 safeties and 1/3 PORVs),
- ATWS emergency feedwater (2/4 pumps to all 4 SGs),

- safety injection for small LOCAs, including RHR pumps, and
- long term cooling for large and small LOCAs including containment fan cooling system (CFCS) operation.

At this point, it is our impression that relatively little plant-specific analysis has been performed to support these assumptions. As defined, it appears that the ATWS emergency feedwater function would be unavailable in the occurence of SGTR in conjunction with ATWS. This is because one loop would be unavailable, while all four loops are required. Please clarify this point. PORVs at many plants are unavailable on short notice because the block valves are closed due to PORV leakage. If PORVs are really required during ATWS, then blocking them will significantly increase core damage frequency. Are they really required? Please clarify the treatment of this point.

720.16 Please explain the rather high beta factor (0.4) used for containment fan cooler's control and actuation circuits.

720.17 The runout head for HHSI pump is approximately 1000 ft. Please discuss the potential of overspeed and cavitation under large LOCA conditions where rapid depressurization may lead to much lower pressure than the runout head. Similarly, discuss whether RHR pumps would have the similar concern when operating at a lower pressure than the runout head (340 ft), e.g., large LOCA when they are used as RCS injection pumps.

720.18 A three week test interval has been used in the quantification of the backup seal injection system unavailability. Since the backup seal injection system is an important mitigation system for accidents involving seal leakage, its unavailability plays a significant role in the quantification of total core melt frequency. Since it is designated as a control grade system, not a safety grade system, please clarify the basis for assuming three weeks as the test interval.

720.19 Loss of dc power was included in the fault trees of safety systems in SP-90 PRA. However, it is not possible to assess potential system interactions without information regarding components fed by the dc busses. Table 8.3-1 of Module 9 provides some clues to the identification of major interfaces between dc busses and major safety systems. Please clarify what constitutes load groups 1 and 2 and separation groups 3 and 4. In addition, clarify the relationships between components (e.g., pumps) 1, 2, 3, and 4 and trains A, B, C, and D. Does SP/90 PRA assume, in general (i.e., except service water/component cooling water system), that trains A and D share the same ac and dc buses.

C. Documentation

720.20 In Table 4.1-1, the non-recovery probability of one-of-one DG is listed as 0.21. It seems that this value should be 0.7 according to what is stated on p. 4-2. Please explain.

720.21 What was the success criterion used for the accumulator system? In Table 2.1-2 (p. 2-20), 1-out-of-3 is listed while on p. 2-90, 2-out-of-3 is stated. Also, Table 3.4 seems to indicate a 2-out-of-3 success criterion. Which was the success criterion used in the large LOCA analysis presented in Module 1?

720.22 The definition of error factor (p. 118) is actually what's commonly defined as the range factor. Please explain.

720.23 In Module 1, p. 6.3-15, it is stated the RHR heat exchangers would have CCW cooling automatically when EWST water temperature goes above 195°F. It is not clear why operator action to align CCW to the heat exchangers is required for LTC3 as stated on p. 2-21. Please explain.

720.24 Please explain why transients with late core melt are not included in the definition of plant damage states in Table 2.1-1.

720.25 The air operated valves in the steam supply lines to the Turbine-Driven Pumps (TDP) need, according to the FT logic module (Figure 3.10.2-8), an "S" or other ESFAS to open (see p. 3.10-7). Therefore, for any transient initiator in conjunction with IPS failure, the following CM sequence would appear to exist according to the methodology adopted in the study

$$TR4.IPS.REC = 10/yr*5.7x10^{-7}*0.5 = 2.8x10^{-6}/yr$$

where REC is failure probability to recover the IPS signal within 2 or 3 hours into the accident. Basically, failure of IPS renders all safety systems inoperable, including the TDPs of the EFWS. This scenario was, however, omitted from the analysis. Please explain.

Enclosure 2
Questions Resulting from Review of
RESAR-SP/90, Module 16, Volumes 3 & 4

A. INFORMATION NEEDED FOR STCP SIMULATION OF W-APWR MAAP ANALYSIS

720.26 Please provide the heat transfer area, distance measured from the bottom of core, mass, heat capacity, thickness, and initial temperature of core support plate and grid plates in the bottom head as shown in Figure 5.3-1.

720.27 Please provide the flow area, equivalent diameter, heat capacity, initial temperature, and the fraction of airborne fission product decay heat absorbed in the internal structures listed as No. 11, 12, 13, 14, and 15 in the CIRC input data file.

720.28 Please provide the mass of structures falling into the reactor vessel bottom head at core slump.

720.29 Please provide the mass of structures included in core debris for debris/concrete interaction in the reactor cavity compartment.

720.30 Please provide the total mass of zirconium used for in-vessel metal/water reaction.

720.31. The initial inventories of 40 species are required by the CORSOR code. MAAP only has six groups of fission products. Please provide the breakdown of the six MAAP groups, indicating the quantity of each radionuclide included in each group.

B. JUSTIFICATION AND CORRECTION OF W-APWR MAAP ANALYSIS

720.32 On page 5.5-37 of the Probabilistic Safety Study, the lower compartment equipment volume and surface area are given as 273 ft^3 and 51485 ft^2, respectively. Using these values, the equivalent thickness is computed as 0.0636 inches. It appears that the equivalent thickness of the lower compartment equipment is too thin. Please clarify the volume and surface area given and provide a proper value for the equipment equivalent thickness.

720.33 The gap resistance between steel liner and concrete in the upper compartment and annular compartment is given as 0.2 ft^2 -hr-F/BTU. The gap resistance appears too high. Please justify or correct the gap resistance value.

720.34 The MAAP analysis shows no zirconium is reacted within the vessel for the AEF-2 sequence. Why was zirconium reaction not included in the analysis of this sequence.

720.35 There are errors in the release fraction data for the SEFC-2 sequence given on page 5.5-163 (ie, many of the release fractions exceed 1.0). Please provide the corrected data.