September 23, 2002

MEMORANDUM TO: James Lyons, Director
New Reactor Licensing Project Office
Office of Nuclear Reactor Regulation

FROM: Michael Johnson, Chief **/RA/**
Probabilistic Safety Assessment Branch
Division of Systems Safety and Analysis
Office of Nuclear Reactor Regulation

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION - AP1000 PRA Level2/3, SHUTDOWN RISK, RADIOLOGICAL PROTECTION, AND METEOROLOGY (TAC NO. MB4683)

Attached is a set of request for additional information prepared by the Probabilistic Safety Assessment Branch regarding the AP1000 PRA Level 2/3, Shutdown Risk, Radiological Protection and Meteorology, i.e., Chapters 15 and 19 of the AP1000 DCD, and PRA report. Staff's questions regarding success criteria and Level 1 analysis were provided under separate letter.

Attachment: As stated

CONTACTS: R. Palla, SPSB/DSSA, 415-1095 (720.41 - 720.63)
M. Pohida, SPSB/DSSA, 415-1846 (720.64 - 720.72)
M. Snodderly, SPSB/DSSA, 415-2856 (720.73 -720.81)
A. Drozd, SPSB/DSSA, 415-1308 (RES/720.82-720.97)
M. Hart, SPSB/DSSA, 415-1265 (470.1 - 470.13)
L. Brown, SPSB/DSSA, 415-1232 (451.1 - 451.7)

September 23, 2002

MEMORANDUM TO: James Lyons, Director
New Reactor Licensing Project Office
Office of Nuclear Reactor Regulation

FROM: Michael Johnson, Chief **/RA/**
Probabilistic Safety Assessment Branch
Division of Systems Safety and Analysis
Office of Nuclear Reactor Regulation

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION - AP1000 PRA Level2/3, SHUTDOWN RISK, RADIOLOGICAL PROTECTION, AND METEOROLOGY (TAC NO. MB4683)

Attached is a set of request for additional information prepared by the Probabilistic Safety Assessment Branch regarding the AP1000 PRA Level 2/3, Shutdown Risk, Radiological Protection and Meteorology, i.e., Chapters 15 and 19 of the AP1000 DCD, and PRA report. Staff's questions regarding success criteria and Level 1 analysis were provided under separate letter.

Attachment: As stated

CONTACTS: R. Palla, SPSB/DSSA, 415-1095 (720.41 - 720.63)
M. Pohida, SPSB/DSSA, 415-1846 (720.64 - 720.72)
M. Snodderly, SPSB/DSSA, 415-2856 (720.73 -720.81)
A. Drozd, SPSB/DSSA, 415-1308 (RES/720.82-720.97)
M. Hart, SPSB/DSSA, 415-1265 (470.1 - 470.13)
L. Brown, SPSB/DSSA, 415-1232 (451.1 - 451.7)

DISTRIBUTION:
R.Palla, M. Snodderly M.Pohida M.Hart L.Brown A.Drozd
L.Burkhart M.Rubin M.Reinhart

Accession#ML022660580 NRR-096

G\\:spsb\drozd\Ap1-AMemo-RAI-Sep19.wpd

| OFFICE | *SPSB | *SPSB/DSSA | *SPSB/DSSA | SPSB/DSSA |
|---|---|---|---|---|
| NAME | ADrozd:nyc | MRubin | MReinhart | MJohnson |
| DATE | 09/18/02 | 9/19/02 | 9/19/02 | 9/23/02 |

**OFFICIAL RECORD COPY**

# SPSB REQUEST FOR ADDITIONAL INFORMATION
# REGARDING AP1000 DESIGN

Late Containment Failure (No core damage)

720.41 A late containment failure (LCF) endstate has been added to the Level 1 event trees. Table 4A-1 indicates that these events are assigned to plant damage state (accident class) 2. The footnote to Figure 4A-2 indicates that the LCF endstate is not added to the core damage endstates and is discussed in the Level 2 analysis. However, the Level 2 PRA does not include containment event trees (CETs) for accident class 2 (Chapter 35), and the assessment of events with failure of PCS (Chapter 40) is limited to containment pressure response rather than core cooling. Please clarify whether/how the frequency associated with LCF endstates is addressed in the Level 2 PRA and CETs.

MAAP Analyses

720.42 Westinghouse states that the overall AP1000 plant response to severe accidents as well as the mass, composition, and superheat characteristics of the initial debris relocation is very similar to the AP600, and therefore: (1) the results and insights from the AP600 hydrogen generation and mixing analyses for each accident class are applicable to the AP1000, (2) the release fractions and timing for the AP1000 release categories would be approximately the same as for the AP600, (3) the conclusions of the AP600 analyses regarding the challenge to the lower head integrity from core debris relocation into the lower plenum and the challenge from ex-vessel steam explosions are applicable to AP1000. The premise for this conclusion is questionable given the large differences in core power and mass between the two designs, and the substantial differences in melt progression timing indicated in the results for 1A/1AP sequences in Chapter 36. Also, the dominant accidents within each release category, and their relative contribution, may be different for AP1000 due to differences in the Level 1 PRA and could lead to selection of different representative sequences for some of the release categories. Please provide AP1000-specific analyses of core melt progression and fission product releases for the dominant accident sequences within each release category, and use this information to either define AP1000-specific hydrogen releases and fission product source terms, or to substantiate the applicability of the AP600 hydrogen analyses and fission product releases to AP1000. This should include a comparison of event timing, fraction of core melted, hydrogen generation rates and quantities, mass and superheat characteristics of debris relocating into the lower plenum, and fission product release histories for representative sequences in each accident class.

Operator Actions

720.43 Time windows available for operator actions in AP1000 are shorter than for AP600 (see Table 35-6). For every human action in the Level 2 PRA, please describe the basis for the revised time estimates, and their impact on HEPs and containment performance (i.e., large release frequency).

720.44 Reactor cavity flooding success criteria has been modified to account for higher water depth and earlier flooding times required for AP1000. Operator instructions to flood cavity have been moved from the end of AFR.C-1 in AP600 (before entering the Severe Accident Management Guidelines), to the entry to AFR.C-1 in AP1000. Please confirm that moving this action does not adversely impact other operator actions that might be critical to core damage prevention or mitigation, or conflict with other objectives of AFR.C.

Depressurization

720.45 For AP1000, the thickness of the vessel wall conducting heat at the peak critical heat flux (CHF) is 36 times the minimum required to carry the dead load. Although factor of 36 may appear substantial, 1 psig of internal pressure within the RPV would be roughly equivalent to the dead load for AP1000. Thus, a pressure pulse of 35 psi would be sufficient to eliminate this margin. This is more limiting than the 150 psig criteria used for determining whether the reactor is adequately depressurized to support in-vessel retention. (The analysis of in-vessel retention assumes all sequences with successful depressurization (DP) will have sufficiently low RCS pressure to prevent RPV failure by over-pressure/dead-load. For accident classes 3BE, 3BL, 3BR, and 3D/1D the primary system pressure by definition of the accident class is 150 psig or less, and success at DP is assumed.) Reflood of the damaged core may also produce RCS pressures in excess of the margin. Please address whether some portion of the sequences with success at DP or eventual reactor reflood can result in reactor vessel failure due to over-pressure.

Containment Isolation

720.46 The containment isolation fault tree success criteria tables (Tables 24-5a through-c and 24-8) do not include all of the isolation valves listed in Table 24-1 for the 12 penetrations analyzed (some of which are initially open). Please discuss why only a partial listing is provided.

In-Vessel Retention

720.47 Please discuss the implications of the most recently completed experimental work related to in-vessel retention of molten core debris on the reliability of the in-vessel retention strategy for the AP1000 design, including the work performed as part of the RASPLAV project and any available results from the OECD-sponsored MASCA program at the Russian Research Center, and the SIMECO and FOREVER programs in Sweden. Specifically address the implications of this work on the potential for debris bed stratification and chemical interactions between molten debris and the reactor vessel wall.

720.48 Please provide a quantitative assessment of the uncertainties in the reliability of in-vessel retention for the AP1000 design using the analytical approach and tools developed through the INEEL assessment of in-vessel retention for the AP600 (i.e., J. L. Rempe, et al., "Potential for AP600 In-Vessel Retention Through Ex-Vessel Flooding," INEEL/EXT-97-00779, December 1997). This should include an assessment of the uncertainties in heat transfer, decay heat, and material property assumptions described in Appendix B of the report, and the implications of forming the alternate debris bed configurations described in Section 2.1.2 of the report. Please provide AP1000-specific probability density function results for the final bounding state (comparable to Figures 3-5 through 11 in the report) and for each alternate debris configuration. Justify that the margins to failure are sufficient to support the lower head failure assumptions used in the AP1000 PRA.

720.49 Describe how the water/steam flow path and flow areas specified for the AP1000 in Chapter 5.3.5 were simulated in the ULPU2000 experiments, including scaling effects.

<u>Reactor Vessel Insulation</u>

720.50 Westinghouse claims in Chapter 5.3.5.4 that the forces on the AP1000 reactor vessel insulation following core relocation and cavity flooding can be based on AP600 test results from the ULPU-2000 test program for Configuration III. Although this test data was used to develop the functional requirements for the AP600 reactor vessel insulation and support system, its suitability and applicability for AP1000 has not been established, and is questionable given the substantial differences between the AP600 and AP1000 insulation system designs and accident conditions. The AP1000 design would have higher heat fluxes from the vessel, higher water/steam flow rates and flow velocities through the insulation system, a considerably smaller gap between the insulation and reactor vessel, and closely-fitted hemispherically-shaped insulation panel (versus conically-shaped insulation with a substantial standoff distance from the reactor vessel in AP600). Collectively, these differences could result in substantially different pressure loads and functional requirements for the AP1000 reactor vessel insulation and support system. Westinghouse needs to either: (1) establish the applicability of the ULPU Configuration III test results to AP1000 considering the impact of each of the above factors, or (2) develop AP1000-specific test data based on the prototypical insulation and flow conditions for AP1000, i.e., ULPU-2000 Configuration IV. Note: Westinghouse also states in Chapter 5.3.5.4 that further evaluation of the forces on the reactor vessel insulation and supports is provided in the AP1000 PRA. Such information is <u>not</u> provided in the PRA, e.g., in Chapter 39 “In-Vessel Retention of Molten Core Debris”.

720.51 Details of the insulation design (e.g., specific dimensions or clearances as a function of angle) and the insulation support frame are not provided. The discussion in Chapter 39.10.2 indicates that a (future?) detailed mechanical analysis of the insulation and support frame will verify the specified structural aspects of the support frame and insulation panels. The discussion in Chapter 5.3.5.5 indicates that a (completed?) structural analysis of the AP1000 reactor cavity insulation system demonstrates that it meets the functional requirements discussed in Chapter 5.3.5.4. Please confirm the status of the insulation design. Provide the functional requirements and the structural analysis showing how these functional requirements are met by the AP1000 insulation system.

PCS

720.52 The passive containment cooling system (PCS) was assumed to always be operable in AP600, but is now modeled in AP1000. The success criteria is 1 of 3 PCS water lines open or operator provides an alternate source of water to the containment shell. If PCS operates, other challenges are considered downstream. If PCS fails, paths downstream of PCS failure do not address hydrogen combustion. Westinghouse did not consider operator actions to use the non-safety-related containment spray system, even though such actions would be included within the severe accident management guidelines. Use of the sprays could have both positive (reduce containment pressure and source terms) and negative (de-inert containment atmosphere) impacts on accident progression. Please provide an evaluation and a Level 2 PRA sensitivity case addressing the net impact that spray operation would have on containment release frequency and magnitude.

Intermediate Containment Failure

720.53 The sequence used to quantify the intermediate containment failure probability given failures of PCS and containment venting (Figures 40-5 and 40-6) appears to be a LOCA with full reactor coolant system depressurization and successful core cooling. The resulting failure probability (0.02) is applied to all accident classes. Please justify the applicability of this probability value for each accident class since events with core damage could result in higher containment pressures than the sequence on which the probability value is based.

Hydrogen Combustion

720.54 AP1000 addresses diffusion flames through a defense-in-depth philosophy in the design. As the last level of defense, Westinghouse claims that there is sufficient margin to failure even if design features fail and diffusion flames occur near the containment shell. Westinghouse bases the last statement on previous assessments for AP600, where the quantities of hydrogen produced were lower than for AP1000, and which did not include features (dampers) to preferentially direct hydrogen away from the shell. Failure of a IRWST vent damper could result in local thermal loads greater than if no dampers were present. Please provide additional justification for the application of AP600 insights on creep failure if this is to be considered an additional level of defense.

Direct Containment Heating

720.55 An assessment of direct containment heating (DCH) was performed for AP600 using the methodology developed as part of the DCH issue resolution (i.e., NUREG/CR-6338, 1996). Rather than update this assessment for AP1000, Westinghouse (in Appendix B.3) has reverted to a qualitative argument that the AP1000 design includes reactor cavity design features to decrease the amount of ejected core debris from reaching the upper compartment, as called out in SECY-93-087. This qualitative argument provides an insufficient technical basis for addressing the DCH, given the potential for a greater DCH pressure loading in AP1000 (due to the larger core mass), and the more recent and technically-defensible methodology that is now available. Please provide the results of a deterministic assessment based on the methodology developed as part of DCH issue resolution.

Offsite Consequences

720.56 The offsite consequences for release categories CFI and CI are similar to or less than the corresponding values for AP600. This is unexpected since the fission product inventories for AP1000 are larger than for AP600, and the same release fractions were assumed. Please explain the reasons for this inconsistency.

720.57 Sensitivity and importance analyses for large release frequency and sensitivity analyses for offsite dose risk were provided in Chapter 50 for AP600. Sensitivity analyses and top event importance analyses are provided in Chapter 43 for AP1000. However, component and operator action importance analyses, and sensitivity analyses for offsite dose risk have not been included. Please provide this additional information.

Core-concrete Interactions

720.58 Westinghouse claims that the concrete penetration on the RCDT (sump) side of the cavity is minimal following a hinged failure mode of the reactor vessel, compared to the penetration on the reactor vessel side of the cavity. However, this is predicated on the core debris separating, with the oxide component (about 85-90 percent oxide) remaining on the reactor vessel side of the cavity, and a metallic component (about 75 to 85 percent metal) reaching the RCDT side of the cavity. This debris separation behavior is used by Westinghouse as the basis for concluding that core debris accumulation in the cavity sump would not be controlling for basemat melt-through. It is unclear whether this separation will actually occur given the large uncertainties in the configuration of molten core debris prior to vessel breach (i.e., mixed versus stratified), and the turbulence and mixing that would occur as the debris enters and spreads within the reactor cavity. Westinghouse needs to confirm the robustness of their conclusion and the adequacy of the sump curb design by providing an assessment of the impact on basemat melt-through times and containment pressure (for both limestone and basaltic concretes) assuming that this oxide/metallic separation does not occur following a hinged failure of the reactor vessel, i.e., either a homogeneous melt or an oxide melt reaches the RCDT side of the reactor cavity and enters the sump.

720.59 Please provide a description of: (1) the physical characteristics of the door between the reactor cavity compartment and RCDT room, including its approximate size, construction, bouyancy, hinging arrangement (opening direction and jambs), and pressure retaining capability, (2) the expected response of the door during the flood-up period and following a postulated melt-through of the RV, and (3) the potential for the door to break free and block the inlets to the RV insulation system during flood-up, or remain in place and restrict debris spreading within the reactor cavity following a postulated melt-through of the RV. Also explain why these design details should not be included in the system design description in the DCD.

Severe Accident Mitigation Design Alternatives (SAMDAs)

720.60 In response to 10 CFR 50.34(f)(1)(i), Westinghouse provided an evaluation of potential AP600 design improvements (Severe Accident Mitigation Design Alternatives) in Appendix 1B of the AP600 SSAR. The details of this evaluation, which included a design description and estimated risk reduction and costs for each alternative, and estimated offsite exposure for each of the major release categories, formed the basis for the staff's review. A similar evaluation has not been provided in Appendix 1B of the DCD or in the PRA for AP1000. In order to support the staff's review of potential design improvements, please provide an AP1000-specific evaluation of Severe Accident Mitigation Design Alternatives similar in scope and content to that provided for AP600. Please include the following within the response:

a. a summary of the risk-significant enhancements (i.e., impacting CDF and person-rem doses) incorporated subsequent to the AP600 design, such as the third PCS water injection line and the IRWST vent dampers,

b. the risk (core damage frequency and population dose per year) associated with operation of an AP1000 at the reference site. Include the risk associated with internally- and externally- initiated events, and events at shutdown to the degree that this can be inferred from the associated analyses,

c. the specific site characteristics, including population, meteorology, economic data, and evacuation assumptions on which the population dose estimates are based. Provide these characteristics/interface assumptions in such a way that one can readily determine whether a potential new reactor site is enveloped by the same analysis,

d. the estimated dollar value of completely eliminating all severe accident risk for an AP1000 plant at the reference site, broken down by major cost category (i.e., public exposure, offsite property damage, occupational exposure, onsite cleanup and decontamination, and replacement power)

e. an explanation of how insights from the AP1000-specific PRA and supporting risk analyses for external and shutdown events, including importance analyses and cutset screening, were used to identify potential plant improvements

f. justification that the potential improvements identified through a systematic process as suggested in (e) are included within the set of 15 SAMDAs identified in Appendix 1B of the AP1000 DCD. Provide a supplemental analysis for those risk-significant improvements not included within the list of 15.

ITAAC/COL Action Items

720.61 The list of risk-significant SSCs within the scope of ITAAC (DCD Table 2.3.9-3) and D-RAP (DCD Table 17.4-1) includes the hydrogen igniters but does not include the IRWST louvered vents. The design and testing of these vents should also be included since they are a key element of the defense-in-depth philosophy for hydrogen control and are important for minimizing the potential for creep failure of the containment from diffusion flames.

720.62 The commitment to cover and lock closed access portals to the Passive Core Cooling System (PXS) that may be near the containment wall is important for limiting diffusion flames. Please discuss how this commitment will be addressed in ITAAC or COL action items.

720.63 Westinghouse added a check valve to the AP1000 refueling canal drain to permit the reactor cavity to flood more rapidly (PRA p39-2). This valve should be added to the system description in the DCD (e.g., Chapter 5.3.5.4) and included within the ITAAC related to the PXS (Chapter 2.2, Table 2.2.3-4).

Shutdown Risk Assessment

720.64 The following RAIs pertain to: (1) the frequency of RCS-OD (4.4E-6 per year), the frequency of overdraining the RCS required for midloop operations and (2) the design of the step nozzle for RNS pump suction

a. Human Error Probability (HEP) (RCS-MANODS1) evaluates the probability of failure to observe failure of the hot leg level instruments and failure to close the air-operated CVS valves to preclude overdraining.

(i) This HEP is based on the availability of the pressurizer wide range level instrument which is not safety related and is not in Technical Specifications; and therefore, may not be available during midloop operations. The HEP assessment should not credit the availability of pressurizer wide range level indication or the pressurizer wide range level indication should be referenced in Technical Specifications. Please revise the HEP assessment.

(ii) This HEP should be based on the time to drain the hot leg to the point of the low critical vortexing level based on the current step nozzle configuration, since a comparison is being made between the hot leg level instruments and the pressurizer wide range level instrument. The time window of 3 hours is not applicable for this HEP.

1. In the HEP assessment in the PRA and the DCD, document the time to drain the hot leg to the critical vortexing level given the current step nozzle configuration.

2. Revise the time window in the HEP assessment to reflect the time to drain the hotleg to the low critical vortexing level and revise the HEP assessment

b. HEP (RCS-MANODS2) evaluates the probability of failure to detect the failure of automatic closure of the air-operated valves (CVS-V045 and CVS-V047). The estimated time window for the diagnosis and completion of the action appears to be five minutes. The HEP of 1.3 E-2 seems optimistically low considering the data provided by NUREG/CR-1278 Table 12-4 which suggests that the median joint HEP for diagnosis of an abnormal event annunciated closely time to be around .5. Please revise the HEP or document and justify the method used to arrive at the HEP.

c. In the Shutdown Probabilistic Risk Assessment (Chapter 19E of the AP1000 DCD), on page 19E-5, it states that, “should a vortex occur, the maximum air entrainment into the pump suction as shown experimentally will be no greater than 5%.” This appears to be based on the results reported in “AP600 Vortex Mitigator Development Test for RCS Midloop Operation” dated September 1988. However, in the following reference, “Tennessee Valley Authority, Sequoyah Nuclear Plant, CCP Gas Issue, “ Westinghouse letter to P.G. Trudel, Sequoyah Project Engineer from B. J. Garry, Manager, TVA Sequoyah Plant Domestic Customer Project, TVA-90-1050, September 27, 1990, “ Westinghouse states, “minimizing the gas accumulation does not preclude the possibility of initiating a longer term mechanism such as shaft fatigue, wear ring degradation, bearing wear or seal wear. Therefore, for the long term Westinghouse believes that any accumulation is detrimental to the pump reliability. Westinghouse suggest that a continuous venting system be considered fro long term plant operation.” Based on this reference, please justify the acceptance of the maximum air entrainment into the pump suction to be 5% by volume.

d. Failure of both hot leg level instruments was reported to be 2.9E-4 per demand/draindown on page 54-36 of the AP600 PRA (Revision 8, dated September 30, 1996). Please document in the AP1000 DCD (Chapter 19E) and the AP1000 Shutdown PRA (Chapter 54) the differences between the hot leg level instrumentation used in current Westinghouse plants versus the AP1000 design which supports using this value. The staff has observed problems with incorrectly installed RCS level indication which has resulted at a greater failure rate than 2.9E-4 per RCS draindown.

720.65 The reduced time to boiling during cold shutdown, as compared to the AP600 is given in table 54-3, of the AP1000 shutdown PRA. Please,

a. re-calculate and document the new containment closure failure probability and the revised AP1000 Shutdown LERF frequency in Chapter 54 of the AP1000 Shutdown PRA, and

b. document in Chapter 19 of the AP1000 DCD and Chapter 54 of the AP1000 Shutdown PRA how containment recirculation would function if containment could not be closed during cold shutdown/refueling, considering containment integrity is not required during these modes.

720.66 Please document in Chapter 19 of the AP1000 DCD and Chapter 54 of the AP1000 shutdown PRA, how trash generated during outages in containment will impact containment recirculation.

720.67 In Figure 54-7 of the AP1000 Shutdown PRA, Loss of Offsite Power During RCS Drained Condition Event Tree, failure of the onsite AC power through Diesel Generators top event is missing. Please correct the event tree and re-calculate the sequences adding the diesel generators.

720.68 HEP RHN-MAN05 represents the failure of the operator to initiate gravity injection from IRWST via the RNS suction line. The cues for this HEP include high core exit temperature; however, the core exit thermocouples are not required to be available/operable during modes 5 and 6. Therefore, the HEP assessment should not credit the core exit thermocouples. Please revise the HEP assessment.

720.69 In the AP1000 DCD on page 19E-14, it reads, "In Modes 5 and 6, there is no potential for steam release into the containment immediately following an accident". During AP1000 drained operations with the steam generator channel manway covers removed, an extended loss of RNS during will cause steam to be released in containment until gravity injection from the IRWST is manually initiated or automatically actuated. The statement on page 19E-14 should be revised.

720.70 An assessment of shutdown risk, considering fires, internal floods, and seismic events has not been submitted. Please provide a shutdown assessment of these initiators considering:

a. containment may be open,
b. fire/flood barriers may be breached for maintenance,
c. transient combustibles for a given fire area may be increased to support maintenance.

720.71 The staff noted that common cause failure (CCF) of 4 out of 4 IRWST injection squib valves was estimated as 2.6E-5 in the AP600 Shutdown PRA. Please provide the analysis in the AP1000 Shutdown PRA that documents CCF of 6 high pressure squib valves is also 2.6E-5. Please provide data references.

720.72 Please provide the justification for not including LP squib valves 118A/B in the same common cause group as the six high pressure squib valves. This justification should include differences in operation, maintenance, and design.

Fuel-Coolant Interactions [19.2.3.3.5]

720.73 Why are the companion reports to DOE/ID-10541, “Lower Head Integrity Under In-Vessel Steam Explosion Loads,” no longer referenced in Chapter 34 and 39 of the AP1000 PRA?

720.74 Chapter 34.2.2.1 states that the mass flow rate, superheat and composition of debris is expected to be essentially the same as the AP600 and, therefore, it is reasonable to extend the results of the AP600 in-vessel steam explosion analysis to the AP1000. Please, provide either a sample calculation, an analytical justification or an equivalent basis for this conclusion.

720.75 Chapter 34.2.2.2 states that initial debris mass, superheat and composition are assumed to be the same as the AP600 and, therefore, the results of the AP600 ex-vessel steam explosion analysis are considered appropriate for the AP1000. Please, provide either a sample calculation, an analytical justification or an equivalent basis for this conclusion.

720.76 Shouldn’t Reference B-3 be, “AP600 Probabilistic Risk Assessment,” GW-GL-021, August 1998?

Equipment Survivability [19.2.3.3.7]

720.77 The post-accident sampling system is now the containment atmosphere sampling function. What are the differences between the system and the function.

720.78 In SECY-93-087, the staff recommended that the Commission approve the general criteria that the staff evaluate the ALWR vendor’s review of the various severe accident scenarios analyzed and identify the equipment needed to perform its function during a severe accident and the environmental conditions under which the equipment must function. In its July 21, 1993 SRM, the Commission approved the staff’s position. The staff concluded in the AP600 FSER that applicable environments described in Section 19.2.3.3.7 met the above guidance. Table D.7-2 and Figures D.7-3 thru D.7-69 were an integral part of defining that pressure, temperature and event timing and were the basis for defining the applicable environment described in Section 19.2.3.3.7. These figures do not appear to have been provided for the AP600. How can the staff support a similar finding for the AP1000 without the applicable figures for AP1000?

720.79 The applicable environment for equipment survivability accepted by the staff in Section 19.2.3.3.7 of the AP600 FSER included the late in-vessel and ex-vessel phases of NUREG-1465. The severe accident radiation environment defined in Section D.7.1 of the AP1000 PRA appears to no longer include the late in-vessel and ex-vessel release phases. How can the staff support a similar conclusion for the AP1000 that there is reasonable assurance that the equipment identified will survive late in-vessel and ex-vessel source terms?

720.80 Gamma and Beta Doses in Figures of the AP1000 PRA are less than the corresponding figures for the AP600. Considering the power rating has gone up, one would expect these doses to increase not decrease. Why is this less?

720.81 The AP600 FSER states that the COL applicant referencing the AP600 certified design will perform a thermal lag assessment of the as-built equipment used to mitigate severe accidents to provide additional assurance that this equipment can perform its severe accident functions during environmental conditions resulting from hydrogen burns. This assessment is COL Action Item 19.2.3.3.7-1 and is contained in Section 19.59.10.5 of the AP600 DCD. Where can this COL Action Item be found in the AP1000 DCD? How can the staff make a reasonable assurance finding, such as in Section 19.2.3.3.7, for the AP1000 without this COL Action Item?

Dominant Severe Accident Sequences

720.82 In order to obtain a clearer picture of severe accident progression in the AP1000, MELCOR analyses are contemplated for several sequences that are either dominant overall in terms of CDF or dominant within some set of risk-significant sequences. The AP1000 PRA (Rev. 0, page 59-7) describes in detail the five sequences with highest CDF. Please provide a similarly detailed description of the following additional sequences:

a. Sequence 13 (SGTR initiator, failure of CMT or RCP trip, success of PRHR, failure of full and partial ADS). This is the highest-frequency SGTR-initiated core damage sequence reported.

b. Sequence 20 (Transient, failure of MFW/SFW/PRHR, success of CMT and RCP trip, failure of full and partial ADS). This is the highest-frequency non-bypass sequence expected to be at high RCS pressure at the time that core damage begins.

In-vessel and Ex-vessel Steam Explosion

720.83 It is stated in Chapter 34 of the AP1000 PRA that the FCI cannot produce sufficient energy on a short time scale to produce a missile that would fail the AP1000 containment. This statement is based on direct extension of AP600 assumptions and analysis for AP1000. Please provide the justification that results of AP600 in-vessel steam explosion can be directly extended to containment failure in light of melt progression uncertainties in AP1000.

720.84 The thick core reflector in AP600 is replaced by a core shroud in AP1000 to allow for the additional fuel assemblies. Please justify that this change would not increase the size of the crucible failure at the shroud boundary, and thereby increase the initial pour rate into the lower plenum. In addition, please provide justifications for excluding other in-core crucibles that may be in a more cylindrical forms that would support a failure at some mid-level location, instead of the hemispherical crucible with high heat fluxes near the upper surface of the pool. Under these conditions, the axial heat flux distribution may be somewhat different that would affect the initial pour rate into the lower plenum. Please demonstrate that your analyses consider the impact of melt progression uncertainties.

720.85 The AP600 in-vessel steam explosion analysis neglects the possibility of initially small FCIs (with little energetic potential) being a driver for larger melt crucible failures that would increase the melt pour rate. How were these events considered or bounded for the RPV survival in-vessel? Please elaborate.

720.86 The initial AP600 in-vessel steam explosion analysis for the premixing phase of the FCI was limited to times shorter than 1 second, and the triggering time was much shorter than 1 second. The fuel-coolant mixture will naturally attain an optimum set of conditions that would then cause an optimal set of energetics. Again, Westinghouse should address ways in which this has been factored into the analysis provided for AP600 with applicability to AP1000.

720.87 There is no reference provided in the AP1000 PRA for the assessment of ex-vessel steam explosions except to say that the results of AP600 are applicable to AP1000. The AP600 ex-vessel steam explosion analysis considers the lifting of the RPV (see Chapter 19 in NUREG 1512), and concludes that the steel containment remains intact. Please provide the justification of neglecting the potential for failure of the containment penetrations as a result of violent movement of the reactor pressure vessel subject to steam explosions-impulse loads?

<u>In-vessel Retention</u>

720.88 In order to resolve the issue of in-vessel debris coolability due to lower head cooling for the AP1000, Westinghouse referred to a previous DOE study of the same issue for the AP600 (T. G. Theofanous et al., "In-Vessel Coolability and Retention of a Core Melt," DOE/ID-10460, October 1996), and states that these results are directly applicable to AP1000. The following questions are relevant to the applicability of the DOE AP600 experiments and analyses to AP1000:

a. Preliminary calculations performed for the AP1000 had shown that reactor vessel insulation based on the ULPU Configuration III gave very little margin to dryout, and, as a result, the AP1000 reactor vessel insulation was redesigned to more resemble ULPU Configuration IV (Figure 39-3 in the AP1000 PRA), for which the measured CHF was 20% to 30% higher.

    i. What are the main phenomenological reasons for the substantially higher critical heat flux that is measured for ULPU Configuration IV as compared with ULPU Configuration III?

    ii. In the DCD (e.g., Figure 5.3-7 in the AP1000 DCD), the insulation appears to resemble the Configuration III design. Please confirm the actual configuration (i.e., either Configuration III or Configuration IV) foreseen for AP1000.

iii. Please demonstrate the influence of boiling on the structural integrity of the reactor pressure vessel insulation.

iv. What are the experimental uncertainties associated with the measured critical heat flux for the ULPU Configuration IV design as a function of angle?

b. Debris jet impingement on the lower head was shown in the DOE AP600 study to result in an upper-bound ablation depth of about 12.5 cm (as compared with 15.24 cm minimum total thickness), as per calculations in which ablation depth is directly proportional to pour mass, and in which 1/3 of the AP600 total core mass was used. As this sub-issue was not specifically discussed in the AP1000 supporting documents, please provide justifications for assuming that the lower head failures due to debris impingement in the AP1000 design should be extremely unlikely even with the 26% larger total core mass in the AP1000 as compared with the AP600.

c. What is the impact of the higher fuel enrichment on the melt progression and pour rates into the lower plenum?

d. The analysis performed by Theofanous (T. G. Theofanous et al., “In-Vessel Coolability and Retention of a Core Melt,” DOE/ID-10460, October 1996) and Westinghouse, in portraying the molten pool state inside the lower head, does not consider:

i. The potential focusing effects of a thin metal layer on top of the molten pool (with a high thermal conductivity) on the local heat flux on the vessel wall, especially considering the results of calculations for small molten steel masses documented in the study by INEEL (J. L. Rempe, et al., “Potential for AP600 In-Vessel Retention through Ex-Vessel Flooding – Technical Evaluation Report,” INEEL/EXT-97-00779, Idaho National Engineering Laboratory, December 1997).

ii The potential influence of impurities in the debris bed that could result in stratification of the oxide pool into layers of different composition (as observed in the OECD RASPLAV and MASCA programs).

iii The potential influence of zirconium, steel and uranium partitioning between oxide and metallic phases, on the downward heat transfer (as observed in the OECD MASCA experiments).

720.89 Please demonstrate that considering the above-mentioned phenomenological uncertainties, the AP1000 lower head integrity will not be challenged. In any re-analysis, please consider the uncertainties associated with the measured critical heat flux on the outside surface of the AP1000 lower head (see 720.90.3.a.iv).

<u>Core-Concrete Interaction and Core Coolability</u>

720.90 Since long-term core debris cooling on the cavity floor has not been demonstrated, please provide the implications of extended Core Concrete Interactions (CCI) on combustible gas generation and combustion-induced containment failure.

720.91 As part of the verification of the long-term integrity of the AP1000 containment during CCI, several calculations with MAAP and other supporting codes were carried out and discussed in Appendix B.4 of the PRA. These runs indicated that, absent water cooling from the Passive Containment Cooling Water Storage Tank (PCCWST), basemat melt-through would occur at between 2.8 and 4.5 days, and containment pressure at 24 hours would reach values between 20 psig and 39 psig. There are significant uncertainties regarding the coolabiity of ex-vessel core debris by an overlying water pool. In view of this, please provide basemat erosion and containment pressurization results for MAAP scenarios identical to those examined in Appendix B.4 of the PRA, with the following differences: (1) the cavity is dry (so that credit is not given to debris quenching); and (2) water cooling from the PCCWST is assumed to be unavailable.

<u>Hydrogen Generation, Transport, Mixing and Combustion</u>

720.92 What is the philosophy of igniter placement in the upper compartment? The AP600 containment volume is $1.69\text{x}10^6$ $ft^3$ as compared to $2.07\text{x}10^6$ $ft^3$ for the AP1000. Presumably, the bulk of the increase in volume is in the upper compartment. Given that there is a certain volume of coverage for each igniter in the AP600, how does the AP1000 igniter scheme cover the increase in volume of 380,000 $ft^3$ with the same number of igniters?

720.93 How effective is the AP1000 igniter scheme in removing hydrogen with an increase in hydrogen generation rates over that of the AP600? The larger core of the AP1000 suggests that there may be a 30% increase in hydrogen generation rates. A MAAP4 calculation for a scenario with a conservative oxidization of 100% of reactive cladding would provide assurance that the AP1000 igniter scheme is as effective as that of the AP600.

720.94 What effect does the AP1000 hydrogen generation rate have on the mixture classes used in the Sherman-Berman methodology and how does this affect the probability of containment failure at node DTE in the containment event tree?

720.95 Why is the probability of random ignition assumed to be one during the intermediate time? The basis for this question is that it is not conservative to assume that ignition is guaranteed in the intermediate time when it comes to global detonations. I'm presuming that the steam content in the uniformly mixed gases inside the containment decreases as the PCCS is allowed to cool the containment shell. In the limit (dry mixtures), the concentration of hydrogen is about 14% in the AP1000, assuming 100% active cladding reaction, or about 19% assuming 100% reaction of all core zirconium. This mixture is becoming sufficiently sensitive to undergo a transition to detonation, especially if the entire containment is viewed as one confined compartment with a lot of clutter (individual compartments below the operating deck).

720.96 What would be the safety margin basis for containment performance if the uncertainty in the range of steam inerting concentrations was used? The safety margin is less than 1 psi when hydrogen produced from 100% active cladding reaction is mixed with air saturated with 55% steam. However, there is uncertainty in the steam-inerting limits, as measurements have ranged from 49%-63% (M. G. Zabetakis, "Research on the Combustion and Explosion Hazards of Hydrogen-Water Vapor-Air Mixtures," AECU-3327, U. S. Atomic Energy Commission, September 1956.)

Level-1 Follow-up

720.97 Following RAI 720.24, in the PRA report, Table 6-1, there are two basis for success criteria designated as "M" and "P" which are not defined. Please, provide the missing definitions. Also, there are multiple success criteria basis provided for some of the event cases. Please, provide the logic behind, and the decision making process of using such a multiple criteria basis.

Design Basis Accident Radiological Consequences

470.1 Please provide the following information with regard to the Main Steam Line Break (MSLB) as discussed in Chapter 15.1.5.4 and Table 15.1.5-1:

a. What is the basis for assuming an accident duration of 72 hours for the MSLB? What assumptions lead to the determination of this time?

b. What assumptions were made in the determination of the values for the steam mass releases from both steam generators assumed in the radiological consequences analysis of the MSLB?

470.2 Please provide the following information with regard to the radiological consequences analysis of the design basis Locked Rotor Accident (LRA) as discussed in Chapter 15.3.3 and Table 15.3-3:

a. It is stated that it was determined that as a result of the LRA no fuel is damaged such that the activity in the fuel-cladding gap is released, but that a conservative assumption of 16% of the core fuel rods failed was used in the radiological consequences analysis. How was it determined that no fuel is damaged? What is the basis for the assumption of 16% failed fuel?

b. What is the basis for the assumed accident duration of 1.5 hours for the LRA?

c. What assumptions were made in the determination of the steam mass release from the secondary system assumed in the radiological consequences analysis of the LRA?

d. What is the basis for the leak flashing fraction of 0.04% for the first 60 minutes of the LRA?

e. Table 15.3-3 lists the reactor coolant noble gas activity as equal to the operating limit of 280 milliCi/gm dose equivalent Xe-133. Other accidents list this operating limit as 280 microCi/gm dose equivalent Xe-133. Is this a typographical error?

f. Table 15.3-3 lists a fission product gap fraction of 0.10 for Kr-84. The krypton isotope of concern with respect to gap fractions for non-LOCA design basis accident dose analyses is Kr-85. Is this a typographical error?

470.3 Please provide the following information with regard to the radiological consequences analysis of the design basis Rod Ejection Accident (REA) as discussed in Chapter 15.4.8.3 and Table 15.4-4:

a. A fraction of the fuel rods are assumed to melt in the radiological analysis of the REA. Regulatory Position 3 of RG 1.183, “Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors,” states that, for design basis accident events that do not assume melting of the entire core, radial peaking factors should be applied in determining the inventory of the damaged rods. This does not appear to have been done. Please either update your analysis to include the maximum radial peaking factor in the determination of the source term if it was not included, or provide a basis for why you did not do so.

b. What is the basis for the assumed leak flashing fraction of 4.0% in the radiological consequences analysis of the REA?

c. What assumptions were made in the determination of the steam mass release from the secondary system assumed in the radiological consequences analysis of the REA? What is the basis for the assumed release duration of 1800 seconds?

d. What is the basis for the alkali metal partition coefficient of 0.001 used in the REA radiological consequences analysis? What assumptions were made in the determination of the value?

470.4 With regard to the radiological consequences analysis of the design basis Small Line Break Outside Containment as discussed in Chapter 15.6.2 and Table 15.6.2-1, what is the basis for the assumed leak flashing fraction of 0.41?

470.5 Please provide the following in regard to the radiological consequences analysis of the design basis Steam Generator Tube Rupture (SGTR) as discussed in Chapter 15.6.3.3 and Table 15.6.3-3:

a. What is the basis for the assumed flashing fraction for the break flow, as documented in Figure 15.6.3-10? What assumptions lead to the determination of the time-dependent flashing fraction?

b. What is the basis for the assumed steam release duration of 13.19 hours? What assumptions lead to the determination of this time?

470.6 Please provide the following information in regard to the radiological consequences analysis of the design basis Loss of Coolant Accident (LOCA) as discussed in Chapter 15.6.5.3 and Table 15.6.5-2:

a. How was the main control room activity level (2.0E-6 $Ci/m^3$ of dose equivalent I-131) and time (0.2622 hours) at which the emergency habitability system is actuated determined? What assumptions were made in the determination of these values?

b. What is the basis for the control room unfiltered inleakage assumption of 5.0 cfm?

c. What assumptions and inputs were used to calculate the LOCA doses in the control room due to radiation from adjacent structures and sky-shine?

470.7 All Chapter 15 design basis accident radiological analyses include a discussion of additional radiological consequences of spent fuel pool boiling that may occur coincident with the accident. What assumptions and inputs that were used to calculate the radiological consequences as a result of spent fuel pool boiling?

470.8 Table 15A-2 lists the iodine appearance rates used in design basis accident radiological consequences analyses that include iodine spiking. How were the reactor coolant system iodine appearance rates calculated?

470.9 Appendix 15B: The staff plans to perform an independent Monte Carlo analysis of the uncertainties in prediction of the aerosol removal coefficients for the AP1000 design, to determine if use of the previously determined AP600 values is acceptable. For use in this analysis, please provide:

a. containment geometry inputs for the AP1000 design (volume, upward facing surface area, etc),
b. upper and lower values of the thermal-hydraulic containment parameters relevant to the aerosol removal, i.e., pressure, temperature, relative humidity, steam/air ratio, and steam condensation rate on the heat sinks,
c. ranges of non-safety related containment spray flow rate and spray droplets distribution.

470.10 Appendix 15B, paragraph 15B2.4.2. Please, provide reference for the chosen aerosol size distribution parameters (i.e., radius of 0.22 micrometers, and standard deviation of 1.81). Also, please justify the statement that “sensitivity of aerosol removal coefficient calculations to these values is small” since, in general, the opposite is true.

470.11 Appendix 15B, paragraph 15B.2.6. The paragraph presents a qualitative discussion of the differences between AP600 and AP1000 designs concluding that the use of the AP600 removal coefficients is conservative. Please, provide either a sample calculation, or an analytical justification for this conclusion. Also, one potentially important difference is omitted, i.e., the increased height of the AP1000 containment. It is known that the increased height decreases the rate of aerosol removal, which would be a non-conservative effect. Please, discuss the significance of this issue.

470.12 In the AP1000 PRA (as in the AP600 PRA), a DF of 100 has been credited for SGTRs due to impaction on the SG tubes, based on results of an unpublished paper by Jun Li, David Leaver, James Metcalf, "Aerosol Retention During an Unisolated Steam Generator Tube Rupture Severe Accident Event". Please provide justification for this seemingly high DF estimate in light of more recent analyses and or experiments, if available. In addition, please provide a copy of this paper. In addition, since this release category is also used for ISLOCAs, please justify the use of such a high DF, as a general rule, for all bypass events.

470.13 Please justify why the AP600 release fractions, as calculated by MAAP, should be considered applicable to the AP1000. In addition, please justify ZERO release fraction for tellurium in all release classes (in both chemical forms, $TeO_2$ and $Te_2$)

Meteorology

451.1 What are the references for the meteorological data used in the analyses resulting in selection of the site parameter values presented in Table 2-1?

451.2 Section 2.3.4 discusses calculation of the bounding short-tem relative concentration (X/Q) values for use in the off-site design basis accident dose assessments. Was this description provided for information only and it is expected that the methodology, and all inputs and assumptions selected by the combined operating license (COL) applicant will be evaluated at the time of the COL review? If the methodology, and all inputs and assumptions will be evaluated during the COL review, that requirement should be explicitly stated in Section 2.3.6.4.

If a commitment will not be made that the methodology and all inputs, and assumptions selected by the COL applicant will be evaluated at the time of the COL review, what specific inputs, and assumptions for use in the Regulatory Guide 1.145 methodology are proposed as part of the AP1000 Design Certification? Other than the site meteorological data, what specifically will be provided as part of the COL application?

451.3 What factors contributed to the reduction in the 0 - 2 hour EAB X/Q to 6.0E-4 sec/$m^3$ proposed for the AP1000 from the value of 1.0E-3 sec/$m^3$ for the AP600?

451.4 Section 2.3.5 discusses calculation of the bounding long-term X/Q values for use in the off-site assessment. Was this description provided for information only and it is expected that the methodology and all inputs, and assumptions selected by the COL applicant will be evaluated at the time of the COL review? If the methodology, and all inputs and assumptions will be evaluated during the COL review, that requirement should be explicitly stated in Section 2.3.6.5.

If a commitment will not be made that the methodology and all inputs, and assumptions selected by the COL applicant will be evaluated at the time of the COL review, what specific methodology, inputs and assumptions are proposed as part of the AP1000 Design Certification? Other than the site meteorological data, what specifically will be provided as part of the COL application?

451.5 The first paragraph of 15A.3.3 states that short-term atmospheric dispersion factors are listed in Subsection 2.3.4. This is correct for the off-site values, but not for the control room values. Therefore, either the paragraph should be deleted, the first sentence should be modified to specify that off-site values are provided in Subsection 2.3.4, or the control room values should be inserted into Subsection 2.3.4.

451.6 Section 15A.3.3 discusses calculation of the bounding X/Q values for use in the control room design basis accident dose assessments. Diagrams showing the site plan with release and intake locations are included. Was this description provided for information only and it is expected that the methodology and all inputs, and assumptions selected by the COL applicant will be evaluated at the time of the COL review? If the methodology, and all inputs and assumptions will be evaluated during the COL review, that requirement should be explicitly stated in Section 15A.

If a commitment will not be made that the methodology and all inputs, and assumptions selected by the COL applicant will be evaluated at the time of the COL review, please address the following:

a. What specific methodology, inputs, and assumptions are proposed as part of the AP1000 Design Certification? Other than the site meteorological data, what will be provided as part of the COL application?

b. Do Notes 4 through 7 on page 15A-15 simply list the design basis accidents to which the X/Q values should be applied or are the Notes stating that some other X/Q values have not been listed because they are less than, and therefore bounded by, the listed X/Q values? If only bounding values were provided, provide inputs and assumptions for the values that were not submitted in the AP1000 Design Control Document.

c. Because distances, heights, building dimensions and assumptions have not been provided it is difficult to judge. However, on page 15A-17 the main equipment hatch release location appears to be quite close to the control room HVAC intake, yet the 0-2 hour X/Q value listed is, or is bounded by, 1.2E-3 sec/$m^3$. This value seems low given the short apparent distance relative to other postulated release locations. Further, it is the same value as for the equipment hatch at the staging area which is further away. To what is this attributed?

d. In the Table 15A-5 listing of X/Q values, some of the X/Q values are the same for two release locations and/or consecutive time periods. To what is this attributed? Are these occurrences the result of selecting bounding values or a result of other assumptions? Further, in the case of the ground level containment release points, the proposed values are different than those for the AP600. What factors contribute to the differences?

451.7 Will the environmental impact of heat dissipation systems such as the discharge canal and cooling tower be evaluated as part of the COL application? If so, that requirement should be explicitly stated in the appropriate section of the AP1000 Design Control Document.