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DCP/NRC0510
Docket No.: STN-52-003

May 3, 1996

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

ATTENTION: T. R. QUAY

SUBJECT: WESTINGHOUSE RESPONSES TO NRC REQUESTS FOR ADDITIONAL INFORMATION ON THE AP600

Dear Mr. Quay:

Enclosed are the Westinghouse responses to NRC requests for additional information on the AP600 Design Certification program. Enclosure 1 contains responses to 69 follow-on questions pertaining to Level 2 PRA modeling and severe accident issues. These follow-on questions were provided in a NRC letter dated January 20, 1995.

The NRC technical staff should review these responses.

A listing of the NRC requests for additional information responded to in this letter is contained in Attachment A.

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

John C. Butler for
Brian A. McIntyre, Manager
Advanced Plant Safety and Licensing

/nja

Enclosure
Attachment

cc: D. Jackson, NRC (1 copy enclosure/attachment)
J. Sebrosky, NRC (1 copy enclosure/attachment)

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Attachment A to NSD-NRC-96-4715
Enclosed Responses to NRC Requests for Additional Information

Re: Level 2 PRA & Severe Accident Questions

480.136 through 480.189
480.197 through 480.211

Enclosure 1 to Westinghouse
Letter NSD-NRC-96-4715

May 3, 1996

Definition and Quantification of PDSs (480.136)

Provide documentation of the interface between the Level 1 and Level 2 analysis. Include a listing of key Level 1 sequences mapped into the CET for each accident class, and their respective frequency contribution.

Response:

The Level 1 PRA results are input into the Level 2 PRA through the use of accident classes. The end states of the Level 1 event trees are accident classes. The accident classes are the input to the containment event tree. The containment event tree is quantified by fault tree linking to assure that the upstream failures in each core damage sequence are treated consistently between the system analysis and the containment analysis.

The Level 2 PRA is being updated, and revision 7 is due in June 1996. The containment event tree quantification (Chapter 43) revision will include a section which details the Level 1 / Level 2 interface including a description of the accident class definitions, and dominant sequences and frequencies in each of the accident classes.

CET/Decomposition Event Tree (DET) Structure (480.137)

Many of the DET top events and success criteria are structured using single, limiting parameter values to represent each top event rather than the full range of values over which the associated parameter could vary. For example, instead of using a probability distribution function or discretization to represent the range of water mass available in the reactor cavity for ex-vessel FCI (node WA), the analysis only considers the probability that less than a certain amount (1000 kg) of water is available. An alternative and more scrutable approach would be to develop a probability distribution function for each top event/parameter, to propagate these distributions, and to quantify the impact of the various combinations of event outcomes through supporting deterministic calculations performed for each DET end state. (The DET for in-vessel retention of core debris comes the closest to this approach in terms of having a deterministic calculation to support each end state. However, even here, single parameter values are used to define each top event, without sufficient justification for how these values were selected.) If a single value representation for the top events is used, Westinghouse needs to show that the top event and associated success criteria is defined in such a way that it represents the limiting condition. For example, the DET for ex-vessel FCI assumes that if less than 1000 kg of core debris is coarsely mixed in the water, the reactor vessel and containment will not fail regardless of the outcome of subsequent events in the DET. To justify selection of this value, Westinghouse should show that for any debris mass up to 1000 kg, reactor vessel and containment failure would not be expected, even if all subsequent parameter/events are set to upper bound values. If it was determined that a debris mass of 750 kg or more would be sufficient to fail the vessel given upper bound conditions for all subsequent events (e.g, maximum credible water masses and conversion efficiency), then the success criteria should be modified accordingly, and the top event requantified.

Response:

The AP600 PRA is being updated and will be transmitted to the NRC in June 1996. The methodology that will be employed for addressing severe accident phenomena issues is different than for previous PRA versions. Analyses for the AP600 demonstrate a high probability of success of in-vessel retention of molten core debris (IVR), including allowance for the possibility of in-vessel steam explosion phenomena. As a result, the Containment Event Tree (CET) in the PRA update will not display and quantify the ex-vessel severe accident phenomena that may occur should the vessel fail. The CET model will conservatively assume that vessel failure leads to containment failure for release frequency quantification. In order to show the AP600 containment robustness, deterministic analyses of some of these ex-vessel phenomena will be performed.

The IVR thermal hydraulic phenomena have been researched and analyzed by the DOE Advanced Reactor Severe Accident Program (ARSAP) using the Risk Oriented Accident Analysis Methodology (ROAAM) and the results presented in Ref. 480.137. In the PRA revision, a discussion will be provided on how the AP600 design fulfills the conditions and assumptions of the ARSAP work. This will include:

- RCS depressurization reliability
- cavity flooding system reliability
- illustration that the reactor vessel reflective insulation is "IVR friendly" and will allow sufficient ingress of water and the venting of steam from the cavity
- evaluation of the treatment of the lower head outside surface relative to the cooling of the vessel by the cavity water.

Hydrogen combustion phenomena, including diffusion flames at the IRWST vents is being decomposed with a detailed analysis in the PRA update. A hydrogen analysis will be provided to

assess the probability of containment failure due to hydrogen combustion including deflagration, detonation, and diffusion flames at the IRWST vents for in-vessel hydrogen releases. Ex-vessel core concrete interaction production of combustible gas will be assumed to result in containment failure for the CET model.

The other severe accident issues will be addressed through quantification of the reliability of systems that establish conditions which prevent the phenomena from occurring and threatening the containment integrity. Specifically, the ADS provides highly reliable RCS depressurization capability and eliminates the need for a detailed quantification of the high pressure issues of induced steam generator tube failures and high pressure melt ejection. If the RCS depressurization does not occur, then containment failure will be assumed for the CET model. Creep rupture failure of the RCS piping for preventing steam generator tube failure will not be credited.

The passive containment cooling system provides highly reliable heat removal from the containment and eliminates the possibility of long-term overpressure by decay heat steaming even with the failure of the water cooling of the containment outside shell. Analysis will be provided to support this position in the PRA.

The cavity flooding system provides a highly reliable means of preventing vessel failure and eliminates the need for a detailed quantification of ex-vessel fuel coolant interactions, debris quenching, core-concrete interaction and overpressurization by non-condensible gases. If vessel failure occurs, then containment failure will be assumed for the CET model.

In addition to the above, severe accident phenomena related to ex-vessel conditions will be evaluated and analysis will be provided where appropriate. Analysis will be provided for ex-vessel fuel coolant interactions to demonstrate that the AP600 reactor cavity can withstand any realistically evaluated steam explosions or rapid steam generation that might occur if the reactor vessel were to fail. For ex-vessel core coolability (including Core Concrete Interactions) a determination of the degree and rate of concrete penetration by molten core debris for a partially flooded reactor cavity and a dry cavity will be done. This will be used to determine the containment pressurization rate during the core concrete interactions. Analysis will also be performed to determine the amount and rate of combustible gas generation and resulting hydrogen concentrations and containment pressure and temperature conditions from core concrete interactions, direct containment heating and fuel coolant interactions for realistic severe accident scenarios that result from reactor vessel failure for both flooded and dry cavity conditions.

In summary, the AP600 PRA will contain a CET that includes the evaluation of IVR thermal-hydraulic phenomena and hydrogen combustion phenomena. Other severe accident issues (i.e., ex-vessel phenomena) will be addressed on the CET through quantification of the reliability of systems that establish conditions which prevent the phenomena from occurring and threatening the containment integrity. The CET will not display nor quantify the ex-vessel severe accident phenomena, but rather, deterministic analyses of these phenomena will be performed and documented in the PRA.

In-vessel Retention of Core Debris (480.138)
The DET is based on a steady-state energy balance and heat transfer analysis, and does not consider the likelihood of compromising reactor vessel integrity during the transient portion of the event,before steady state conditions are established. Provide a detailed assessment of the conditions that could exist from the time immediately following debris entry into the lower plenum, through the time at which a stable debris bed configuration would be achieved. This should include evaluation of the maximum transient heat fluxes and corresponding vessel failure probabilities associated with (a) the initial pour of molten debris onto the lower head, (b) different debris bed configurations that could exist on a temporary basis (including a homogeneous debris bed and an oxide layer over a metallic pool), and (c) hot spots and non-uniformities that could exist on a short-term and possibly a long-term basis. Modify the DET to address additional parameters of importance, if appropriate.

Response:
The decomposition event tree analysis for in-vessel retention (IVR) of molten core debris (Chapter 36) is being eliminated from the AP600 PRA. The failure probability of the reactor vessel is based on the Advanced Reactor Severe Accident Program Risk Oriented Accident Analysis Methodology analysis (reference 480.138-1) for issue resolution of IVR and in-vessel steam explosion. The ROAAM analysis concludes that vessel failure is physically unreasonable as long as the RCS is depressurized and the vessel is submerged in water to a depth at least to the top elevation of the debris pool. Thus, a failure probability of 0.0 will be assigned for this node on the CET. The IVR ROAAM analysis addresses these issues and has been peer reviewed by 17 experts in the fields of severe accidents, heat transfer, and structural mechanics.

To credit IVR in the AP600 PRA, analyses are presented to demonstrate that the AP600 meets the conditions and assumptions of the IVR ROAAM. These analyses include:

- illustrating the RCS depressurization system is reliable
- illustrating the cavity flooding system is reliable
- illustrating that the reactor vessel reflective insulation is "IVR-friendly" and allows sufficient ingress of water and the venting of steam from the cavity
- evaluating the treatment of the lower head outside surface relative to the cooling of the vessel by the water.

The reflective insulation is presented as a conceptual design to demonstrate feasibility prior to design certification, and the ARSAP IVR report verifies that the surface treatment specification for the AP600 reactor vessel does not interfere with the heat transfer during IVR. In the PRA, any sequences that are not adequately depressurized or flooded are assumed to fail the reactor vessel.

Reference

480.138-1. Theofanous, T.G., et. al., "In-Vessel Coolability and Retention of a Core Melt," DOE/ID-10460, July 1995.

In-vessel Retention of Core Debris (480.139)
The scatter in available data for assessing upward and downward heat flux from a molten pool, as well as critical heat flux (CHF) on the outside of the reactor vessel, is significant and can result in a variation of about ±30 percent in the estimated heat fluxes. Since the upward/downward heat flux ratio, and CHF values are central to the IVR analysis, the full range of uncertainty in the associated correlations should be explicitly represented in the DET analysis. This can be done by adding top events to address the heat flux ratio and the CHF values, and by defining the associated branches such that the full range of experimental data is represented and bounded. Modify the analysis accordingly.

Response:
Please see response to RAI 480.138.

In-vessel Retention of Core Debris (480.140)
Because of the approximate nature of the heat transfer analysis and the lack of a rigorous reactor vessel structural analysis, there is considerable uncertainty that reactor vessel integrity will be maintained when heat fluxes approach (but are still less than) the mean estimated CHF, and when wall thinning occurs. Accordingly, node IVR should be quantified based on a probability function that considers the thermal and structural uncertainties, including potential non-uniformities, at the extreme conditions, instead of characterizing this event as a go/no-go based on comparison with a single CHF value (1100 kW/m2, in this case). Address this issue.

Response:
Please see response to RAI 480.138.

In-vessel Retention of Core Debris (480.141)
Provide the heat flux data and associated interpolation on which the CHF used for assessing node IVR are based. Discuss the extent to which the heat flux values used bound the scatter in the data.

Response:
Please see response to RAI 480.138.

In-vessel Retention of Core Debris (480.142)
The origin of the coordinate system for measuring angular position in Figure R.1.2 of Revision 1 of the probabilistic risk assessment (PRA) is shown to be above the top surface of the metallic layer rather than at the axis of the hemisphere. Confirm that with the coordinate system used, the heat flux to the wall determined from equation (4), when integrated over the full height of the oxide pool, is equivalent to the average heat flux given by equation (1) (i.e., all energy is conserved).

Response:
Please see response to RAI 480.138.

In-vessel Retention of Core Debris (480.143)
Identify and justify the cavity water temperature assumed in the IVR analyses. Provide an assessment/bounding calculation of the impact that saturated coolant conditions would have on the probability of maintaining debris in-vessel.

Response:
Please see response to RAI 480.138.

In-vessel Retention of Core Debris (480.144)
Identify and justify the RCS pressure assumed in the IVR analyses. Confirm that the weight of the core debris on the lower head was considered in assessing the structural integrity of the reactor vessel in the IVR analysis.

Response:
Please see response to RAI 480.138.

In-vessel Retention of Core Debris (480.145)
Provide an assessment of the probability that core relocation occurs before RCS creep rupture, and the impact that this would have on vessel failure probability. This should be based on consideration of the uncertainties in modelling core melt progression, and supported by comparison of material access authorization program (MAAP) predictions with results from similar analyses performed using other codes.

Response:

The conditions and assumptions of the ROAAM IVR analysis dictate that the RCS is at low pressure for vessel failure to be physically unreasonable (reference 480.145-1). Therefore, IVR is not credited for high pressure scenarios required to produce hot leg creep. Additionally, hot leg creep failure is not credited to prevent steam generator tube failure or high pressure vessel failure in revision 7 of the PRA (see response to RAI 480.137).

Please note that MAAP stands for Modular Accident Analysis Program.

Reference

480.145-1. Theofanous, T.G., et. al., "In-Vessel Coolability and Retention of a Core Melt," DOE/ID-10460, July 1995.

In-vessel Retention of Core Debris (480.146)
Explain why the reactor vessel inner wall temperature in case 3BE (Figure L-49 of the PRA) shows no significant increase after it is in contact with molten material.

Response:

MAAP4 case 3BE.base in revision 6 of the PRA has water in the lower head of the vessel which is replenished through the break. Water in the cavity cools the outside surface. In the model, the debris relocates after the water refloods into the vessel. The debris quenches in the water and does not heat up the vessel wall. The MAAP4 cases are being calculated for revision 7 of the PRA. The Westinghouse position on IVR is not based on or supported by MAAP4 analyses, but by reference 480.146-1.

Reference

480.146-1. Theofanous, T.G., et. al., "In-Vessel Coolability and Retention of a Core Melt," DOE/ID-10460, July 1995.

In-vessel Retention of Core Debris (480.147)
Provide a summary of the reactor vessel insulation and reactor cavity seal design specifications and calculations, showing that the as-built configuration will not limit the ingression of coolant or steam and alter the heat transfer from that predicted in the Level 2 analysis.

Response:

A conceptual design to demonstrate the feasibility of the AP600 reactor vessel reflective insulation has been provided and was presented to the NRC at a meeting on August 17, 1995. Drawings are also available in Appendix K of reference 480.147-1. This insulation is not intended to be a final design, but a design to demonstrate the feasibility of IVR-friendly insulation. Additional reactor vessel insulation details, such as loading requirements, will be presented in revision 7 of the PRA.

The reactor cavity seal ring is a permanent steel seal that is welded between the reactor vessel and the refueling canal. Steam egress from the cavity is through the vessel supports and loop holes in the concrete, and this resistance is included in the ULPU testing for the ex-vessel CHF (reference 480.147-1). It is assumed that the seal ring has no impact on the IVR processes.

Reference

480.147-1. Theofanous, T.G., et. al., "In-Vessel Coolability and Retention of a Core Melt," DOE/ID-10460, July 1995.

In-vessel Retention of Core Debris (480.148)
Provide additional bases for the judgement that heat transfer from the vessel lower head cannot be inhibited as a result of post-accident conditions. This should specifically address (a) why hot spots cannot form as a result of shifting of insulation (due to blowdown forces, the hydrostatic head of water, or seismic motion), or (b) why flow paths cannot be blocked by floating or submerged debris, such as unqualified coatings.

Response:

a) The functional specification of the insulation is that it provides a mounting frame to prevent the insulation from contacting the reactor vessel during vessel cooling either by hydrostatic head or by the oscillatory forces associated with the boiling. If insulation shift is postulated, it is not expected to significantly affect the heat transfer. The vessel is round and the insulation panels are straight which results in single point contact with the vessel or contact along a line which would not seriously impair surface wetting. Considering three-dimensional heat transfer and the large margin to failure as reported in reference 480.148-1, hot spots which could threaten IVR are not considered to be reasonable in this circumstance.

(b) There are no screens to plug and the water flow paths in the insulation are large. The entrance pathway into the insulation is elevated and the water level at the time of IVR is several meters above the bottom of the insulation so floating or submerged debris cannot be injested into the insulation flowpath. The other flowpaths are expelling water and steam so debris would be pushed away.

Reference

480.148-1. Theofanous, T.G., et. al., "In-Vessel Coolability and Retention of a Core Melt," DOE/ID-10460, July 1995.

In-vessel Retention of Core Debris (480.149)
In the event that steam cannot be vented from inside the insulation boundary, explain (a) how the vessel could remain cooled if the vapor space is pressurized to the point that steam is vented at the bottom of the insulation (the outer surface of the vessel would be dry under these conditions), and (b) why a pressure of only 9 inches of water would be needed to vent steam at the bottom of the insulation (rather than the head corresponding to the depth to which the vessel is submerged).

Response:

(a) Steam is vented upwards to the loop compartments through an engineered flowpath in the vessel insulation and through the normal cavity ventilation flowpath. If there is molten debris in the lower head and it is postulated that the outside surface of the vessel is dry, the vessel will fail.

(b) Because of the engineered pathway through the vessel insulation, the pressure required to vent steam from the bottom of the insulation is no longer applicable.

Creep Rupture of Steam Generator Tubes (480.150)
Provide schematics depicting:

a. the nodalization of the vessel with emphasis on the upper head, upper plenum, core, and hot leg connections. (The vessel nodalization was not presented in Figure R.2.3 of the PRA), and

b. the hot leg/cold leg flow configurat:on from the core to the steam generators for times before and after uncovering the bottom of the core barrel.

Response:
In revision 7 of the AP600 PRA, creep rupture failure of RCS piping is not credited for the prevention of steam generator tube failures for high pressure and temperature accident sequences. Only operator and system successes are credited to prevent phenomenological failures. MAAP4 analyses are only used to estimate operator action timing to depressurize high pressure and temperature core damage sequences and they are used in a conservative manner. Timing to steam generator tube failure and to HPME are conservatively estimated.

a. The RCS nodalization schematic will be provided for MAAP4 analyses defending the operator action timing for depressurization.

b. The natural circulation flow pattern as modeled for AP600 in the MAAP4 code is conservative with respect to the heatup of the steam generator tubes. Heat is readily transferred from the core to the RCS metal mass (including the tubes) such that the RCS heats up in a uniform manner. The schematic of the flow will be provided in revision 7 of the PRA.

Creep Rupture of Steam Generator Tubes(480.151)
Based on scoping analyses performed by Idaho Nuclear Energy Commission (INEL), moderate variations in stress and steam generator tube temperature (from those considered by Westinghouse) could result in failure of the steam generator tubes prior to the hot leg or surge line for the case with the secondary system pressurized. Accordingly, the MAAP-calculated natural circulation flow rates and mixing analyses need further justification. In this regard, provide the following:

a. a detailed description of any benchmarking activities performed to validate/calibrate the MAAP natural circulation model and assumptions used in the Westinghouse analysis, such as comparisons to scale tests and analyses using other codes,
b. a description of the upper plenum mixing model used in the calculations, and the influence of the reverse flow from loop 2 on the temperature of the steam entering the hot leg, loop 1,
c. further justification for the natural circulation flow patterns that are computed by the MAAP code, particularly the reverse flow in Loop 2 and through the core, in view of the heat generation due to decay heat and oxidation,
d. a description of how transport of fission products and hydrogen is treated in the loop piping, and how these gases affect the natural circulation in the loop,
e. further substantiation of the magnitude of the natural circulation flow rates contained in Figures R.2-4 through R.2-38 of the PRA, and
f. justification of the magnitude of the steam flow rates through the core for Case HP of Section R.2.3.1.1 of the PRA, where it is stated that the heat transfer to the steam (or in other words steam cooling) keeps the core cooled during this event and "prevents significant melting." Also, justify the magnitude of the increased steam flow through the core that occurs at about 15,000 seconds for this case, causing the substantial decrease in core fuel temperature.

Response:
In revision 7 of the AP600 PRA, high pressure and temperature creep rupture failure of RCS piping is not credited for the prevention of steam generator tube failures. Only operator and system successes are credited to prevent phenomenological failures. MAAP4 analyses are only used to estimate operator action timing to depressurize high pressure and temperature core damage sequences and they are used in a conservative manner. Timing to steam generator tube failure and to HPME are conservatively estimated. This conservative treatment bounds the uncertainties discussed in questions a-f above.

The strong natural circulation flow pattern as modeled for AP600 in the MAAP4 code is conservative with respect to the heatup of the steam generator tubes, minimizing the time for operator action. Heat is readily transferred from the core to the RCS metal mass (including the tubes) such that the RCS heats up in a fairly uniform manner. In this case, the high pressure steam and strong natural circulation prevent significant melting of the core prior to the threat to steam generator tubes.

If strong natural circulation does not occur, then the tubes do not heat up rapidly and the time for operator action to prevent tube rupture increases. In this case, heat is maintained in the core and the time to core degradation and relocation decreases, minimizing the time for the operator to prevent HPME by depressurizing the system prior to the relocation of core debris to the lower head. This case is bounded by low pressure accident sequences which have relatively little natural circulation and heat removal from the core.

Creep Rupture of Steam Generator Tubes (480.152)
Provide the heat transfer coefficient for the convective and radiative contributions at the fuel rod surface from Figure R.2-6 of the PRA, and the heat transfer coefficient for the hot leg, loop 1, for Figure R.2-9.

Response:
Please see response to RAI 480.151.

Creep Rupture of Steam Generator Tubes (480.153)
Confirm that the Inconel 600 creep data developed by INEL and International Nickel Company, and reported in Appendix B of NUREG/CR-5642, as well as available creep data for Inconel 660, is conservatively bounded by the Larson-Miller curves used to represent the AP600 steam generator tubes. Provide a graphical display of the data and failure criteria.

Response:
The AP600 steam generator tubes are made of inconel 690 which is known to have significantly higher strength and ultimate strength at high temperature than inconel 600 (reference 480.153-1). In the DET analysis in revision 2 PRA, the inconel 600 data used to predict tube failure was the lower bound data developed by INEL and the International Nickel Company as taken from reference 480.153-2.

In the PRA revision 7, MAAP4 is used to predict the operator action timing for depressurization of high pressure sequences with respect to steam generator tube creep rupture and debris relocation to the lower head. The methodology used to justify the operator action timing considers the uncertainties associated with natural circulation in the reactor coolant system and conservatively assures that the system is depressurized before creep of the tubes is postulated.

References

480.153-1. Harrold, D. L., et. al., "The Temperature Dependence of the Tensile Properties of Thermally Treated Alloy 690 Tubing," Fifth International Symposium on Environmental Degradation of Material in Nuclear Power Systems - Water Reactors, Monterey, CA, August 25-29, 1991.

480.153-2. Harris, B. L., et. al., "Creep Rupture Failure of Three Components of the Reactor Primary Coolant System During the TMLB' Accident," EGG-EA-7431, November 1986.

Creep Rupture of Steam Generator Tubes (480.154)
As part of the Three Mile Island Margin-to-Failure Analysis, the predictive capabilities of different creep rupture failure criteria were assessed. This study indicates the Manson-Haferd parameter to have better predictive capabilities than the Larson-Miller parameter for the data considered. Discuss the implications of using the Manson-Haferd parameter in lieu of the Larson-Miller parameter in assessing the probability of creep rupture of RCS components for the AP600.

Response:

In revision 7 of the AP600 PRA, high pressure and temperature creep rupture failure of RCS piping is not credited for the prevention of steam generator tube failures. Only operator and system successes are credited to prevent phenomenological failures. MAAP4 analyses are only used to estimate operator action timing to depressurize high pressure and temperature core damage sequences and they are used in a conservative manner. Timing to steam generator tube failure is conservatively estimated. Based on the TMI Margin-to-Failure Analysis, the Larson-Miller methodology underpredicts the time to failure. Therefore, Larson-Miller assessment provides a bounding estimate of the time available for operator action.

Creep Rupture of Steam Generator Tubes (480.155)

Certain limiting condition for operations in the technical specification have implications on the DET for thermally-induced failure of the RCS, particularly the specifications concerning: maximum pre-existing SG tube leakage, maximum wall thinning, and maximum SG tube through-wall crack depth. Justify that the maximum allowable limits on these parameters are accommodated in the DET analysis. This should include (a) additional thermal-hydraulic analyses showing the effect of pre-existing primary-to-secondary side leakage (up to the maximum allowable) on the probability of SG tube creep rupture, (b) confirmation that the 3 mil thinning assumed in the analysis will be the technical specification limit, and (c) an assessment of the impact of elevated SG tube temperatures on fracture mechanics and crack propagation given pre-existing cracks. Also, justify that the behavior of plugged and sleeved steam generator tubes is adequately represented in the DET. The validity of the probabilities assigned to nodes LM, SG, and HL should be confirmed for all of the above conditions.

Response:

In revision 7 of the AP600 PRA, high pressure and temperature creep rupture failure of RCS piping is not credited for the prevention of steam generator tube failures. Only operator and system successes are credited to prevent phenomenological failures. The decomposition event tree for thermally-induced failure of the RCS pressure boundary is eliminated. MAAP4 analyses are used to estimate operator action timing to depressurize high pressure and temperature core damage sequences and they are used in a conservative manner. The analyses bound the assumptions of high and low RCS natural circulation such that, even considering uncertainties in (a) through (c) and tube plugging and sleeving, temperatures that challenge steam generator tubes are not reached in sequences credited as success.

Creep Rupture of Steam Generator Tubes (480.156)
Clarify whether operation of the 4 inch line, 10 inch line, or both lines was credited in case FL of the PRA. If just the 10 inch line was credited, discuss the implications of flooding with both lines on the time to submerge the hot leg, and on the probability of SGTR.

Response:
In revision 7 of the AP600 PRA, high pressure and temperature creep rupture failure of RCS piping is not credited for the prevention of steam generator tube failures. Only operator and system successes are credited to prevent phenomenological failures. Therefore, flooding and time to submerge the hot leg has no implications on the probability of SGTR.

Creep Rupture of Steam Generator Tubes (480.157)
Because the useful range of core exit thermocouples is limited to about 1600 F, it is reasonable to expect that actions to flood the cavity might be taken prior to the core exit temperature reaching the 2000 F value assumed in the analysis. If the decision to flood the cavity were made upon the exit temperatures reaching 1200 F (an entry condition for the Westinghouse Severe Accident Management Guidelines), the hot leg would be submerged about 1 hour prior to the estimated time of hot leg creep rupture in Case FL of the PRA. In view of the significant potential for cavity flooding to preclude creep rupture of the hot leg, provide the following: (a) justification as to why the probability of submerging the hot leg prior to creep rupture should not be explicitly represented in the DET, and (b) controls that will be placed on the use of the cavity flooding system to reduce the potential for submerging the hot leg prior to creep rupture.

Response:
Please see response to RAI 480.156.

Creep Rupture of Steam Generator Tubes (480.158)
It is not clear where the safety/relief valve failure rates used for node SP are described in the Level 1 PRA documentation. Provide additional documentation describing the development of the probability value for node SP. Identify the individual probability values used for failure of valves to reclose and for main steamline break upstream of the isolation valve. Discuss the number of times that the valve(s) would be cycled during the relevant Class 1A transient, and how the probability of reclosing was adjusted to account for multiple cycles. Describe the actions that operators would take in response to a stuck open safety or relief valve, and whether these actions are accounted for in the probability estimate.

Response:
In revision 7 of the AP600 PRA, high pressure and temperature creep rupture failure of RCS piping is not credited for the prevention of steam generator tube failures. Only operator and system successes are credited to prevent phenomenological failures. MAAP4 analyses are used to estimate operator action timing to depressurize high pressure and temperature core damage sequences and they are used in a conservative manner. Timing to steam generator tube failure is conservatively estimated to bound uncertainties. This conservatism includes depressurization of the secondary system.

Creep Rupture of Steam Generator Tubes (480.159)
The probability values assigned to NC and subsequent nodes in the PRA need to be further substantiated by reference to and comparison with results of calculations performed using codes other than MAAP, such as RELAP/SCDAP analyses documented in NUREG/CR-5949 and NUREG/CR-6075, and the assessment of lower head failure documented in NUREG/CR-5642. Provide this information.

Response:
In revision 7 of the AP600 PRA, high pressure and temperature creep rupture failure of RCS piping is not credited for the prevention of steam generator tube failures. Only operator and system successes are credited to prevent phenomenological failures. MAAP4 analyses are used to estimate operator action timing to depressurize high pressure and temperature core damage sequences and they are used in a conservative manner to bound uncertainties. Timing to steam generator tube failure and to HPME are conservatively estimated.

The strong natural circulation flow pattern as modeled for AP600 in the MAAP4 code is conservative with respect to the heatup of the steam generator tubes, minimizing the time for operator action. Heat is readily transferred from the core to the RCS metal mass (including the tubes) such that the RCS heats up in a fairly uniform manner. In this case, the high pressure steam and strong natural circulation prevent significant melting of the core prior to the threat to steam generator tubes.

If strong natural circulation does not occur, then the tubes do not heat up rapidly and the time for operator action to prevent tube rupture increases. In this case, heat is maintained in the core and the time to core degradation and relocation decreases, minimizing the time for the operator to prevent HPME by depressurizing the system prior to the relocation of core debris to the lower head. This case is bounded by low pressure accident sequences which have relatively little natural circulation and heat removal from the core.

In-vessel Steam Explosion (480.160)
As noted in WCAP-13388, numerous studies have been performed to assess the potential for containment failure by in-vessel FCI (α-mode failure) for operating reactors. In order to assess the applicability of these studies to the AP600 design, identify the major differences between the AP600 and operating reactors that would impact the α-mode failure probability, and provide an assessment of how these differences would impact the probability of failure (i.e., whether these differences would tend to increase or decrease the probability of α-mode, and what the net effect would be). This should include an assessment of the effect of the following AP600 design and sequence features on the α-mode failure frequencies estimated in the studies: (a) higher frequency of low pressure sequences, (b) flatter power profile, (c) reduced lower plenum structures and lack of lower head penetrations to break up debris stream, (d) increased opportunity for FCI as a consequence of operator actions to reflood a damaged core that has been retained in-vessel, and (e) reduced reactor vessel strength (due to elevated temperature and wall thinning) for core melt sequences with successful in-vessel retention.

Response:
As part of the ARSAP in-vessel retention of molten core debris program, a ROAAM analysis of in-vessel steam explosion is being performed for the AP600. The draft analysis for peer review is to be provided to the NRC, and will be referenced in revision 7 of the PRA. This analysis supersedes the in-vessel steam explosion DET which will be eliminated from the PRA.

In-vessel Steam Explosion (480.161)
Provide an assessment of the probability of SGTR as a result of an in-vessel FCI, given (a) reflood occurs prior to creep rupture of the RCS, and (b) reflood occurs after RCS depressurization.

Response:
Long-term creep damage in high pressure scenarios failing the RCS pressure boundary prior to steam generator tube failure is not credited for preventing containment bypass in revision 7 of the AP600 PRA. Only system and operator success in depressurizing the RCS is credited. The timing available to the operator to depressurized is conservatively estimated based on upper and lower bounds on natural circulation in the RCS. The result is that successful depressurization is only credited if the ADS is actuated prior to a loss of geometry in the core. Therefore:

a) because of the injection capacity of the passive systems and the normal RHR, the core cannot be reflooded prior to depressurization.

b) after depressurization, there is no molten core mass to mix with the water to create a fuel-coolant interaction. The quasi-static repressurization from the damaged core reflood would be less than the system pressure that the tubes survived prior to depressurization.

Therefore, in both cases the probability of steam generator tube rupture is zero.

Ex-vessel Steam Explosion (480.162)
The DET top events and success criteria appear to be defined on a qualitative basis and quantified arbitrarily, without explicit representation of the uncertainty in either the parameters/processes important to FCI energetics or the containment structural capabilities. A quantitative evaluation of potential FCI impulse loads and reactor cavity structural capabilities (i.e., a fragility curve) is not provided. Furthermore, the DET and supporting assessment does not adequately address the effect of several parameters that could have an important influence on FCI energetics, specifically, mass of debris in the lower plenum, melt superheat, water subcooling, time of explosion triggering, and conversion efficiency. Provide a probabilistic assessment of ex-vessel FCI loads and containment failure probability. This should include quantitative analyses of FCI loads for best-estimate conditions, evaluation of the impact of uncertainties in key parameters on FCI loads (through sensitivity analyses considering the full range of parameter values), and inclusion within the DET of parameters shown to be important determinants for FCI.

Response:
Failure of the depressurized reactor vessel into a flooded reactor cavity is considered to be physically unreasonable based on the IVR ROAAM analysis (reference 480.162-1). Any sequence which fails the vessel is assumed to fail the containment on the containment event tree in revision 7 of the PRA.

Deterministic analysis of ex-vessel steam explosion will be provided in PRA revision 7 to meet the requirements of SECY-93-087.

Reference

480.162-1. Theofanous, T.G., et. al., "In-Vessel Coolability and Retention of a Core Melt," DOE/ID-10460, July 1995.

Ex-vessel Steam Explosion (480.163)
Identify the most likely location for failure of the reactor cavity wall/structures, given an FCI in the reactor cavity. This should consider the effect if the cavity were (a) partially flooded (to a height of about 2.5 m), and (b) fully flooded. Provide a fragility curve for the limiting reactor cavity wall/structures. Provide an assessment of the impact of structural failure on containment integrity.

Response:
Please see response to RAI 480.162.

Ex-vessel Steam Explosion (480.164)
The documentation for CET Node CF1 presented during the October 20, 1994 meeting with the staff indicates that a containment failure probability of 0.01 was assumed for ex-vessel steam explosions involving a large debris mass in a flooded cavity (ESX). However, this value is inconsistent with the documentation provided on page R-10 of the PRA (which indicates that the probability of containment failure is 0.1 based on the split fraction assigned to node MX of the ex-vessel steam explosion tree) and the documentation on page R-131 (which indicates that the split fraction of Node MX is 0.001). Clarify this discrepancy.

Response:
Please see response to RAI 480.162.

Core Concrete Interactions (480.165)
The DET and supporting assessment does not adequately address the effect of several parameters that would appear to have an important influence on the likelihood of debris coolability, specifically, the amount of debris superheat, amount of unoxidized zircaloy in the debris, amount of steel in the debris, debris pour rate, likelihood of debris quenching, eventual slumping of the remainder of the core, and upward heat flux (and effect of crust formation on heat flux). Provide further analyses to demonstrate the impact that variations in these parameters would have on debris coolability. This could be done through sensitivity analyses considering the full range of parameter values. Parameters shown to be important determinants for debris coolability should be considered for inclusion in the DET.

Response:
The DET for debris coolability has been eliminated from revision 7 of the AP600 PRA. Failure of the vessel into a flooded reactor cavity is considered to be physically unreasonable based on the IVR ROAAM analysis (reference 480.165-1). Failure to flood the reactor cavity has a very low frequency and is assumed to result in vessel failure and subsequent containment failure in revision 7 of the PRA. If debris is assumed to be coolable, early containment failure is assumed due to ex-vessel steam explosion. If debris is assumed to be noncoolable, early containment failure is assumed due to hydrogen combustion.

Deterministic debris coolability analyses will be provided in PRA revision 7 to meet the requirements of SECY-93-087.

Reference

480.165-1. Theofanous, T.G., et. al., "In-Vessel Coolability and Retention of a Core Melt," DOE/ID-10460, July 1995.

Core Concrete Interactions (480.166)
The DET assumes that in all sequences there will be sufficient water in the cavity at the time of reactor vessel failure to quench the molten core debris. This does not appear to be true for all sequences. Most notably, class 1A sequences, which constitute about 20 percent of the CDF, would have less than 0.5 meters of water in the cavity at the time of vessel failure. Justify why the DET should not treat the failure to quench debris as a top event, to cover such situations.

Response:
Accident classes 1A and 1AP cases in which manual depressurization fails will not relieve water to the containment prior to a high energy event such as a creep rupture failure of the hot leg nozzle or high pressure melt ejection. In PRA revision 7 these cases are assumed to bypass containment through induced steam generator tube failure by virtue of the large uncertainties associated with such events. Because containment integrity is known to be compromised, the question of debris coolability is not addressed for these cases. The DET for debris coolability has been eliminated from revision 7 of the PRA.

Core Concrete Interactions (480.167)
Describe the condensate drainage paths and partitioning that results in water collecting in the reactor cavity prior to reactor vessel failure in class 1A sequences (if the blowdown from the RCS in these sequences is to the IRWST, the mass of water released appears insufficient to overflow the refueling canal into the reactor cavity). Identify any conservatisms in estimating water level in the cavity, such as accounting for holdup on horizontal surfaces.

Response:
Please see response to RAI 480.166.

Core Concrete Interactions (480.168)
The MAAP input described in Appendix K of the PRA appears to assume overflow from the refueling canal into the reactor cavity (see input for junction #15 on page PK-19) at a lower canal water level than indicated in Figure 4-2 of WCAP-13388, Section 2. Also, the input deck uses a cavity floor area of 45.6 m2 whereas the assessment of ex-vessel debris coolability is based on 53 m2. These differences would lead to overestimates of the water level in the reactor cavity. Discuss the implications of these discrepancies on the assessments of in-vessel debris retention and ex-vessel debris coolability.

Response:
There is no implication of these discrepancies on the in-vessel debris retention since the draining of the IRWST to flood the cavity bypasses the refueling canal. The IRWST must be drained into the cavity to take credit for IVR. The flow path for draining is from the IRWST to the sumps in the loop compartments, through the vertical hatch in the tunnel at the 83 foot elevation to the cavity. The venting pathway is from the cavity to the loop compartments through the nozzle holes in the concrete. Condensate from the PCS is collected in the IRWST. Additionally, any condensate that may be assumed to "rain" into the refueling canal up to the overflow is insignificant with respect to the amount of water from the IRWST which is in the cavity and loop compartments.

There is no implication of these discrepancies on ex-vessel debris coolability. The DET for debris coolability has been eliminated from revision 7 of the AP600 PRA. Failure of the vessel into a flooded reactor cavity is considered to be physically unreasonable based on the IVR ROAAM analysis (reference 480.168-1). Failure to flood the reactor cavity has a very low frequency and is assumed to result in vessel failure and subsequence containment failure in revision 7 of the PRA. If debris is assumed to be coolable, early containment failure is assumed due to ex-vessel steam explosion. If debris is assumed to be noncoolable, early containment failure is assumed due to hydrogen combustion.

Reference

480.168-1. Theofanous, T.G., et. al., "In-Vessel Coolability and Retention of a Core Melt," DOE/ID-10460, July 1995.

Core Concrete Interactions (480.169)

For sequences with reactor vessel failure at the bottom of the vessel (at least 10 percent of the failures) and sequences in which the lower head fails globally due to creep rupture, the debris bed could contain a considerable amount of steel from the lower head. Justify why global failure of the lower head or inclusion of at least some portion of the mass associated with the head is not considered in assessing debris coolability.

Response:

Please see response to RAI 480.165.

Core Concrete Interactions (480.170)

The DET neglects the potential for non-uniform distribution of debris within the reactor cavity, and enhanced concrete erosion in the region immediately below the reactor vessel. Because of the relatively short distance between the reactor vessel breach location and the reactor cavity floor in the AP600 design, incomplete corium jet breakup and particle quenching is likely. Accumulation of particulated debris below the vessel would reduce water depth below the vessel and further limit jet breakup and quenching. The net result would be the formation of a non-coolable debris bed and enhanced concrete ablation below the reactor vessel. Even if the debris eventually spreads within the reactor cavity, significant ablation below the vessel could lead to debris bed depths in this area that exceed the criteria for coolability. Non-uniform distribution of debris in the reactor cavity and its effect on debris coolability needs to be further evaluated. Address this concern.

Response:

Please see response to RAI 480.165.

Core Concrete Interactions (480.171)

The application of Kazimi's analysis to the AP600 suggests that, for pour rates less than 3000 kg/s, the leading edge of the debris will cool to immobilization before spreading throughout the cavity, thereby leading to relatively greater concrete ablation in the area below the reactor cavity. This effect would appear even more pronounced if the plant-specific geometry of the AP600 were considered in lieu of the idealized circular geometry assumed in the Westinghouse analysis (e.g., greater distances debris would need to travel in AP600 to reach the far end of the cavity). Recognizing the impact of pour rate on debris spreading, and the uncertainty in estimating debris pour rates, justify why pour rate is not treated as a top event in the DET.

Response:

Please see response to RAI 480.165.

Core Concrete Interactions (480.172)
Credit for ex-vessel steam explosions completely eliminating the potential for core-concrete interactions is not justified since only a small fraction of the core debris would generally be involved in these interactions. A steam explosion occurring at the onset of the debris pour may render the debris involved in the FCI coolable, but would not affect the coolability of debris which subsequently enters the cavity. Conversely, a steam explosion occurring later in the debris pour would not be expected to affect the coolability of the stratified debris bed that had been formed prior to the explosion. Credit for FCI as a coolability mechanism should be removed. Address this issue.

Response:
Please see response to RAI 480.165.

Core Concrete Interactions (480.173)
There do not appear to be any deterministic bases or calculations to support the position that endstates 5 and 6 are more coolable than the corresponding endstates that involve participation of the full core (endstates 2 and 3). Provide quantitative bases for characterizing the coolability of these corresponding endstates differently.

Response:
Please see response to RAI 480.165.

Core Concrete Interactions (480.174)
Because the AP600 design does not impose any special restrictions on the type of concrete that can be used for the containment basemat (see Q720.27), the impact of CCI should be addressed considering the range of concrete compositions that might be used. Provide an assessment of the impact on the end state frequencies and CCFP of using basaltic rather than limestone concrete.

Response:
Please see response to RAI 480.165.

Hydrogen Combustion (480.175)

Confirm that there are no small pathways by which flames can propagate sideways into the in-containment refueling water storage tank (IRWST) (e.g., ventilation ducts or overflow drain pipes). Identify any inspection, test analysis, and acceptance criteria (ITAAC) that will be proposed to assure that such pathways are not inadvertently introduced during construction.

Response:

No claim has been made that flames cannot propagate into the IRWST. The overflows to the refueling canal and the IRWST vents to the upper compartment could permit flame propagation into the IRWST. There are igniters outside the vents to propagate burns into the IRWST to control the hydrogen concentration in the tank. If the igniters fail, the PRA quantification considers the likelihood of deflagration or detonation of the gas mixture in the IRWST if the upper compartment (including the refueling canal) is greater than 6% hydrogen and flammable to account for propagation.

Hydrogen Combustion (480.176)
Clarify whether the detonability of the mixture in the IRWST or lower compartment is assessed based on the conditions in the compartment: (a) at the time at which the success criteria is initially met, (b) at the most limiting condition existing over the time window when the success criteria is met, or (c) on average conditions over the time window when the success criteria is met. If only the first situation is considered, justify why delayed ignition would not lead to higher probabilities of containment failure.

Response:
The detonability of the mixture in the IRWST, and in the rest of the containment, is assessed at the most limiting conditions in the IRWST.

Hydrogen Combustion (480.177)
The weight assigned to the use of MAAP-based, best-estimate hydrogen generation values in node OX (0.975) of the PRA appears to place undue reliance on the baseline MAAP models without consideration of modelling uncertainties and the possibility that the code (and associated user input parameters and default values) systematically under-predicts hydrogen production. In this regard, provide (a) additional information, including the results of code to code comparisons, to justify that the MAAP baseline and bounding case provide a reasonable central estimate and upper bound estimate of hydrogen production, respectively, and (b) an assessment (sensitivity analysis) of the impact that a more uniform distribution for this node (e.g., a 0.5/0.5 split) would have on the probability of hydrogen combustion and containment failure.

Response:
The best-estimate and upper bound in-vessel hydrogen generation rates were not estimated based on MAAP4 analyses alone. They are also based on consideration of the uncertainties in the MAAP4 models, the events at Three Mile Island, Unit 2 accident, and a review of the expert panel elicitation for the in-vessel issues performed for NUREG-1150. The estimates were made by one of the original members of the expert panel for in-vessel issues. An attempt was made to not rely on tne MAAP4 results. However, the assignment of the split fraction does heavily favor the MAAP4 predicted values. In the hydrogen combustion analyses for the PRA update, this issue will be addressed through sensitivity studies and additional justification of in-vessel hydrogen generation.

Hydrogen Combustion (480.178)
Identify the RCS breach area assumed for creep rupture failure of the hot leg in case 1A of the PRA. Provide an assessment of the maximum hydrogen concentration that would result locally and globally shortly after RCS breach if larger break areas were assumed, and an assessment of the probability of deflagrations/DDT in the lower compartment and containment failure under those conditions. Additional hydrogen released as a result of reflood should be considered as part of this assessment. (Case 3BE with reflood may not bound this situation because of the differences in release rate and location).

Response:
In revision 7, operator action to depressurize the RCS is credited with preventing uncontrolled hydrogen releases to the containment from hot leg creep failure. Failure of the operator action results in an assumed containment bypass through induced steam generator tube rupture. The hydrogen releases to the IRWST and the containment through the ADS in these cases are used to address the impact on containment integrity from DDT and diffusion flames at the IRWST vents.

Hydrogen Combustion (480.179)
Justify that class 1A sequences with large creep rupture induced break areas (and igniters inoperable) are adequately represented in the DET analysis (by case 1A in the assessment of early combustion, and by case 5 in the assessment of intermediate and late combustion).

Response:
Please see response to RAI 480.178.

Hydrogen Combustion (480.180)
Based on information provided in Table R.6-7, R.6-14, and R.6-15 of the PRA, the probability of global burns as well as DDT is assumed to be zero for all sequences with igniters available. Justify that class 1A sequences with large creep rupture induced break areas and igniters operable would not have some non-negligible probability of DDT in the lower compartment.

Response:
Please see response to RAI 480.178.

Reflood of a Damaged Core (480.181)
Although events that progress to control rod melting would be counted as core damage (see Q720.212), the challenge to RCS and containment integrity from recriticality events should also be considered. Provide an assessment of the potential for recriticaiity upon reflood of a damaged core, such as in events where the core is reflooded after control rod/relocation but prior to fuel rod relocation/collapse. Estimate the time window over which recriticality could occur, and the extent of the core inventory that could participate as a function of time.

Response
There is virtually no chance of recriticality in the events that are described above. The water that was credited for injection to the core is from the CMTs, accumulators or IRWST, and it is highly borated. Additionally, since the core is overheated by definition of the accident sequence, the water would be boiling and full of voids which would not moderate the nuclear reaction. Finally, except in the first hours of operation, there would be xenon in the fuel rods for additional shutdown. No recriticality is considered.

Reflood of a Damaged Core (480.182)

The October 4, 1994 response to Q720.232 indicates that RCS pressures following reflood of a damaged core could be as high as 1450 psia. Provide an assessment of the probability of reactor vessel failure or SGTR as a result of this pressure increase, given (a) reflood occurs prior to creep rupture of the RCS, and (b) reflood occurs after RCS depressurization.

Response:

The success criteria for reflood cases have changed. See response to RAI 480.161.

Treatment of Containment Failure Modes in the CET (480.183)
Justify why containment over-temperature, particularly due to diffusion flames, is not included as a potential failure mode.

Response:

Containment over-temperature is considered in that the conditional containment failure probability distribution is determined considering the steel shell to be at a temperature of 400°F. The temperature response of the steel shell predicted by the MAAP4 code for each case is presented to show that the shell temperature is below 400°F with significant margin to account for uncertainties in the MAAP4 modeling.

The diffusion flame scenario is the only case in which the shell temperature can be postulated to exceed the 400°F limit. This case is being considered in detail in PRA revision 7.

Treatment of Containment Failure Modes in the CET (480.184)
For the sequences used to represent each accident class, report the amount of hydrogen released to the IRWST via the ADS valves, and the time period over which this hydrogen is released. Provide an assessment of the maximum containment shell temperatures that would result if this hydrogen is burned as a diffusion flame.

Response:

The diffusion flame scenario is being evaluated in detail in PRA revision 7.

Treatment of Containment Failure Modes in the CET (480.185)
Provide additional information/analyses to support the position that containment pressure will remain below Service Level C when the PCCS is not operable. The use of the GOTHIC code to confirm the MAAP predictions should be considered for this purpose. Include an assessment of the impact of modelling uncertainties and assumptions on the estimated peak pressures, including at a minimum, condensation heat transfer coefficients and the impact of non-condensible gas buildup on these coefficients, air velocity over the containment shell, and baffle plate heatup due to thermal radiation.

Response:
A WGOTHIC analysis of the AP600 containment without PCS water cooling for design basis mass and energy steam releases and for high temperature hydrogen releases equivalent to 100 percent zircaloy oxidation will be presented in revision 7 of the PRA.

Treatment of Containment Failure Modes in the CET (480.186)
Provide an assessment of whether the PCCS will remove sufficient heat to prevent containment over-pressure failure if the air inlets or flow path are completely obstructed (e.g., as a result of water failing to drain from the annulus). Confirm that countercurrent flow in the annulus will not adversely impact the effectiveness of PCCS in this case.

Response:

The inlet flow to the annulus could be obstructed by failing to drain any unevaporated water. The functional specifications for the two annulus drains is that each drain is capable of relieving full PCS water flow to the storm sewer system, and that in the event that the drain downstream pipe becomes plugged, the water is drained to the environment. If water is available to plug the flow inlets, then water is still available for evaporating from the PCS dcme. Countercurrent flow is expected to prevent containment failure, however since there are no analyses to justify this assessment, PCS annulus blockage is treated as a loss of containment heat removal in revision 7 of the AP600 PRA.

The PCS drains are inspected every seven days based on technical specifications. A scalar probability for drain plugging is applied to the AP600 containment event tree in revision 7.

Treatment of Containment Failure Modes in the CET (480.187)
Recognizing that steps to vent the containment may be taken in response to Severe Challenge Guideline SCG-2 of the Westinghouse Severe Accident Management Guidance, discuss why manual containment venting is not considered a containment release mode in lieu of, or in addition to, late containment over-pressure failure. Provide a discussion of the options available to permit venting of the AP600 containment if such a course of action was deemed appropriate during an event, and the feasibility of venting in sequences with late containment over-pressure (e.g., operability of valves and pressure capability of associated vent lines). Discuss (a) whether procedures to vent the containment will be developed as part of either the EOPs or accident management strategies, (b) reasons why venting prior to 72 hours would not reasonably be expected (if that is the case), and (c) anticipated constraints that would be placed on containment venting, such as the earliest time, lowest containment pressure, or maximum projected dose for which venting would be permitted.

Response:

The containment pressure only approaches service level C and the ultimate capacity for cases with hydrogen combustion or noncondensable gas generation following long-term core-concrete interaction. Containment venting would not protect the containment in rapid pressurization events such as hydrogen combustion. In revision 7 of the AP600 PRA, cases which fail the reactor vessel are assumed to fail the containment early bounding a filtered containment venting source term. Therefore, containment venting is not seen as a relevant release category for the AP600 PCS containment in the PRA.

Treatment of Containment Failure Modes in the CET (480.188)
Provide documentation to support the recent Westinghouse position that accidents involving failure of containment isolation will not progress to core damage as a result of loss of coolant from the containment.

Response:
The analysis is discussed in Appendix A of the PRA as part of the PRA Level 1 success criteria.

CET/DET Quantification (480.189)

Provide further elaboration and justification for the probabilities or split fractions assigned to the key branches in the CETs and DETs (as identified through importance and sensitivity analyses discussed under Section 1.8). This should include references to and discussion of calculations and experimental data that form the basis for these values. Specific probabilities/issues under question are:
- extent of zircaloy oxidation at relocation (MD)
- volumetric heat rate of debris pool (DK)
- emissivity of metal pool (QUP)
- probability of secondary side depressurization (SP)
- probability that natural circulation occurs (NC)
- location of reactor vessel failure (ME)
- probability of debris spreading in the cavity (DD)
- probability that the best-estimate hydrogen values apply (OX)

Response:

Based on the revision 7 PRA, cases in which the RCS depressurization fails, cavity flooding system fails or the vessel fails into a fully flooded cavity are assumed to fail the containment. Therefore, the following phenomenological nodes no longer appear on the CET or DETs in the PRA update:
- probability of secondary side depressurization (SP)
- probability that natural circulation occurs (NC)
- location of reactor vessel failure (ME)
- probability of debris spreading in the cavity (DD).

The following nodes are treated explicitly in the ARSAP IVR report (reference 480.189-1):
- extent of zircaloy oxidation at relocation (MD)
- volumetric heat rate of debris pool (DK)
- emissivity of metal pool (QUP)

The hydrogen deflagration, detonation and diffusion flame analyses are being re-examined in the PRA update, so in-vessel hydrogen generation is addressed in PRA revision 7.

Reference

480.189-1. Theofanous, T.G., et. al., "In-Vessel Coolability and Retention of a Core Melt," DOE/ID-10460, July 1995.

Uncertainty in Containment Performance Estimates (480.197)
Perform a systematic assessment to identify and rank the CET and DET parameters/issues with greatest impact on Level 2 results, specifically, containment failure probability and frequency of large release. This can be done via Level 2 importance analyses or a structured set of sensitivity analyses, using containment failure probability and frequency of large release as the figures-of-merit.

Response:
A Level 2 importance analysis will be provided in revision 7 of the PRA.

Uncertainty in Containment Performance Estimates (480.198)

For important parameters/issues (identified in Q480.197), provide a quantitative assessment of the impact that modelling uncertainties and assumptions could have on the Level 2 results (CCFP and frequency of large release). This can be determined by assessing the impact of varying the parameters/issues over the full range of credible values or outcomes. At a minimum, the following parameters/issues should be evaluated through sensitivity analyses:

a. the impact of using CHF values that bound the experimental data in determining whether core debris will be retained in-vessel (IVR),
b. the impact of inadvertently depressurizing the secondary side (SP) in high pressure sequences
c. the impact of removing sidewall failure of the reactor vessel (ME) as a RCS failure mode and reassigning these events to other failure modes,
d. the impact of having igniters unavailable at all times,
e. the impact of removing hot leg creep rupture as a RCS failure mode and reassigning these events to other failure modes,
f. the impact of early actuation of the reactor cavity flood valves (and submerging the hot leg prior to creep rupture failure) in high pressure sequences, and
g. the impact of having the cavity flooding system unavailable at all times.

Response:

The responses are based on revision 7 of the AP600 PRA.

a. The CHF values are being defended outside the PRA by the ARSAP program (reference 480.198-1). The variation of the CHF over the credible range of values has no impact on the conclusion that vessel failure into a flooded cavity is physically unreasonable.
b. There is no impact. The available operator action time to depressurize the RCS to prevent induced tube failure is determined with the secondary system depressurized.
c. There is no impact. Vessel failure into a flooded cavity is physically unreasonable and failure to flood the cavity leading to vessel failure is assumed to result in vessel failure and subsequent containment failure.
d. Impact of igniter failure is addressed in the focused PRA and will be addressed explicitly in the PRA update.
e. Hot leg creep rupture failure is not credited for the prevention of induced tube rupture.
f. Hot leg creep rupture failure is not credited for the prevention of induced tube rupture.
g. The cavity flooding system is a safety-related system and assuming a failure probability of 1.0 is considered to be overly conservative and will not be addressed in the PRA update.

Reference

480.198-1. Theofanous, T.G., et. al., "In-Vessel Coolability and Retention of a Core Melt," DOE/ID-10460, July 1995.

Uncertainty in Containment Performance Estimates (480.199)
Provide an estimate of the containment failure frequency, CCFP, and the probability of large release for the case in which all non-safety systems (in both the Level 1 and Level 2 PRA model) are unavailable.

Response:
The quantification of large release frequency which takes no credit for nonsafety-related systems is provided in the focused PRA sensitivity study for RTNSS. RTNSS has a large release frequency goal of 1.0×10^{-6} per year, but no CCFP goal and no attempt will be made to meet one.

Binning Process and Selection of Representative Sequences (480.200)
Justify that the source terms used to represent each release class provides an appropriate statistical representation of the source terms for all sequences binned within that release class (in terms of timing and magnitude of releases).

Response:
An attempt was made to select sequences to maximize the source terms for each release category. This was done by assuming early times for the failures within each time frame to minimize deposition and allowing the containment to rapidly depressurize for failures. Additional information on source term selection will be documented in PRA revision 7.

Binning Process and Selection of Representative Sequences (480.201)
Provide a more detailed chronology of events for each of the sequences described in Appendix L and Appendix R.2 of the PRA. Specifically, include the times of the following events, as applicable: cavity flooding system actuation, hydrogen system actuation, ADS actuation (specify automatic or manual), core support plate uncovered, creep rupture of RCS (specify location), reactor vessel lower head submerged, hot leg submerged, core relocated to lower plenum, lower plenum dryout.

Response:
Event timing for the requested information will be included in revision 7 of the PRA.

Binning Process and Selection of Representative Sequences (480.202)
It is the staff's understanding that the MAAP results for the 3BE sequence reported in Revision 1 of the PRA do not account for the flow restricting venturi that was recently added to the DVI line in the AP600 design. Since this sequence has been selected by the staff for confirmatory calculations to be carried out using the MELCOR and SCDAP codes, an updated MAAP calculation based on the current plant design is needed. Provide the results of this calculation and supporting documentation.

Response:
A MAAP4 analysis of the AP600 DVI line break case with the flow restriction has been provided (Westinghouse letter NSD-NRC-96-4687 to NRC dated April 4, 1996). The flow restriction will be included in all the DVI line break cases in the PRA update.

Determination of Fission Product Releases (480.203)

Preliminary review of the MAAP input file suggests that certain input parameters selected would tend to limit the release of fission products from the fuel and enhance the radionuclide deposition rate in the RCS and containment. With the FPRAT parameter selected, fission product releases are determined based on the IDCOR/EPRI steam oxidation model, and are limited by saturation vapor pressure for non-volatiles. The use of the NUREG-0772 model without the vapor pressure limitation appears more appropriate. With the value selected for FAERDC, the decay correlation would be favored over the steady-state correlation, resulting in higher deposition rates. A value of 3 appears more appropriate. Provide additional justification for the use of these parameter values. If the present parameters are retained, provide the results of sensitivity analyses addressing the effect of these modelling assumptions.

Response:

The issues presented above will be investigated and the PRA revision 7 will reflect the results of that investigation.

Determination of Fission Product Releases (480.204)

Provide additional documentation to demonstrate that MAAP source term modelling deficiencies identified in past MAAP code assessments have been assessed for the AP600, and that supporting calculations to identify the impact of these deficiencies have been performed.

Response:

A benchmarking exercise (reference 480.204-1) compares the source term results of the MAAP4 code, NAUAHYGROS and simplified hand calculations for various AP600 thermal-hydraulic conditions. The MAAP4 code shows conservatively good agreement with the dry calculations and underpredicts deposition for the cases in which hygoscopicity is important. Hygroscopicity has minimal importance in the AP600 source term cases since the relative humidity in the containment must be above 95% for the effect to be observed, and this only occurs for very short periods of time.

Reference

480.204-1. Hammersley, R.J., D. Leaver, et. al., "Aerosol Deposition in Reactor Containments: A Comparison of NAUAHYGROS and MAAP4" (copy attached).

ANS Reprint

AEROSOL DEPOSITION IN REACTOR CONTAINMENTS: A COMPARISON OF NAUAHYGROS AND MAAP4

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ABSTRACT

This paper compares and contrasts aerosol deposition calculations performed by the NAUAHYGROS code, the MAAP4 code, and simplified hand calculations. These calculations originated with an investigation of the impact of conditions unique to a passive advanced PWR on the prediction of natural aerosol removal in containment. The original plant cases were idealized (simplified and standardized), however, to facilitate the comparison. Good agreement (≤ 50%) between the several calculations was obtained for a variety of thermal-hydraulic conditions, combinations of various natural aerosol removal mechanisms, and inert aerosol injection rates. For all the cases considered in this study, the integrated release fractions of the volatile and non-volatile fission products calculated by the hand calculations and MAAP4 exceeded the release fractions calculated by NAUAHYGROS. Significantly different predictions (nearly an order of magnitude) were found for the two codes for those cases where hygroscopicity was important. NAUAHYGROS calculated much higher removal rates and correspondingly smaller environmental release fractions. Simplified hand calculations were also performed for the dry aerosol gravitational sedimentation cases. Very good agreement (≤ 25%) with the code calculated results were demonstrated for the simplified hand calculations. The simplified hand calculations are seen to be a useful tool for engineering estimates that do not require the use of complex calculational methods and solutions.

BACKGROUND

This paper compares and contrasts the aerosol deposition calculations performed by a sectional code (NAUAHYGROS), a code that employs a non-dimensional correlation (MAAP4), and simplified hand calculations based on the non-dimensional correlation method. The integro-differential equation for aerosol behavior in a closed container for well mixed conditions is solved by these two computer codes.

The solution technique employed by NAUAHYGROS[1] represents the particle size spectrum by a set of size classes, or sections. The size distribution of the particles within each section is specified, and by prescribing the uniform distribution function to each section the number of ordinary differential equations required is simply equal to the number of sections. Thus, the integro-differential equation is replaced by a set of coupled first order differential equations.

A correlation based method for solving the integro-differential equation employed in MAAP4[2] uses the solutions of the integro-differential equation of aerosol coagulation and deposition such that the instantaneous rates of particle deposition are expressed by algebraic forms. Rigorously formulated correlations are assembled by numerically solving the integro-differential equation for two specific limiting situations, one in which the aerosol cloud achieves a steady-state in the presence of an aerosol source and another in which the aerosol mass concentration continually decays by deposition in the absence of a source. An approximate interpolation procedure has ben defined in MAAP4 which applies these correlations to the treatment of aerosol behavior that departs from steady-state or pure decay. In addition, approximate combination laws have been developed for MAAP4 to predict removal rates when selected combinations of deposition mechanisms are operating simultaneously.

If the suspended mass concentration of the aerosol is known or if the aerosol generation rate is known, the deposition rate(s) may be readily evaluated by hand calculations with the algebraic relations that represent the correlations employed in MAAP4.

BASE CASE GEOMETRY AND SOURCE TERM

The comparison was performed by defining a single base case model for the containment geometry and the aerosol rates from the reactor coolant system (RCS) to the containment gas space. The base case geometry was a single containment volume with containment leakage active. The constant containment leak0age rate used was 0.12 volume % per day. No active containment systems (coolers or sprays) were included in the model. The single containment node volume was 47,900 cubic meters, and the deposition area was 1,050 square meters.

A common source term definition was used for all the calculations included in the comparisons provided in this paper. The aerosol source terms for each specie are summarized in Table 1. The source term was represented by the release of aerosols (fission product and inert) from the RCS directly to the containment gas space. The initial aerosol particle size was taken to have a 0.1 μm geometric mean radius with a geometric standard deviation of 2 for the NAUAHYGROS calculations and 0.3 μm for the MAAP4 hygroscopic calculations. Leaver[3] determined best estimate values of 0.25 μm for the geometric mean radius and a 1.69 for the geometric standard deviation based on data from several failed fuel experiments. That work also demonstrated essentially no difference in calculated aerosol removal when the source size parameters were initially taken to be 0.1 μm and 2.0. Epstein[4] notes that the aerosol correlations can be safely used as long as substantial coagulation occurs before significant sedimentation takes place. If this is the case, the initial or source particle size distribution does not influence the steady-state size distribution nor the aerosol decay behavior as time proceeds. The correlations implicitly assume that the majority of particles that make up the source distribution are small enough ($< 0.1\ \mu m$) so that particle coagulation is indeed important. Hence, the impact of the different initial particle sizes (0.1 μm and 0.3 μm) used in the two sets of code calculations in this paper is judged to be small enough to not significantly impact their comparison.

Table 1 - Aerosol Release Rates From the Reactor Coolant System

A) Aerosol Release Rates

	Aerosol Specie	Release Interval: Early (0-9.6 hr) (kg/s)	Release Interval: Late (9.6-24 hr) (kg/s)
i)	Fission Products		
	CsOH	1.9×10^{-3}	9.42×10^{-4}
	CsI	3.84×10^{-4}	1.50×10^{-4}
	Te	8.71×10^{-5}	1.64×10^{-5}
	BaO	1.52×10^{-5}	0
	SrO	9.65×10^{-6}	0
	CeO_2	2.95×10^{-7}	0
	La_2O_3	1.3×10^{-7}	0
	Ru	2.18×10^{-6}	0
	Sb	2.63×10^{-9}	0
ii)	Inerts (1)	2.63×10^{-3} (2)	0

B) Integrated Mass of Aerosol Released From the RCS for a 24 Hour Period

Ratio of Fission Product to Inerts Release Rates	Integrated (24 hr) Aerosol Mass: Fission Products (kg)	Integrated (24 hr) Aerosol Mass: Inerts (kg)
1:0	134.6	0
1:1	134.6	81.6
1:2	134.6	163.2
1:5	134.6	408.0
1:10	134.6	816.0

(1) Inerts are non-radioactive aerosols that come from control rod materials (Ag and Cd), structural materials (SnO_2), and fuel (UO_2).

(2) Multiples (2, 5, 10) of this rate are used in the sensitivity cases.

The release rates from the RCS were typical of PWR source terms and resulted in approximately 55% of the iodine, 48% of the cesium, and 11% of the tellurium in the total core inventory being released over the 24 hour calculation. The 24 hour release period was described by two release intervals which would correspond to "early" and "late" releases from the RCS. For each interval a uniform aerosol release rate was used for each specie. Inert aerosol releases were only considered during the first (early) release interval. The inert aerosol release rate was varied as part of the sensitivity studies performed for these comparisons. A case with only fission product aerosols released from the reactor coolant system is designated as 1:0 (see Table 1).

THERMAL HYDRAULIC CONDITIONS

Several sets of thermal hydraulic conditions for the containment node atmosphere were defined to facilitate the comparison of the codes when different aerosol removal mechanisms were predominant or operable. Figure 1 presents the thermal hydraulic conditions for a low (30-50%) relative humidity sequence. Figure 2 provides the thermal hydraulic condition histories for a moderate (80-85%) relative humidity sequence. The pressure and temperature data are used to specify the physical properties of the gas space. The heat loss through the containment shell (free standing steel containment) is used in the calculation of the thermophoretic deposition velocity and the condensation rate on the containment shell is used to calculate the diffusiophoretic velocity. The aerosol removal mechanisms included in the code calculations used in these comparisons were containment leakage, gravitational sedimentation, diffusiophoresis, thermophoresis, and hygroscopic aerosol sedimentation. The hygroscopic effect of the cesium-iodine and cesium-hydroxide aerosols does not become significant until very high relative humidities are present. Therefore, an additional set of conditions were defined in which the relative humidity was set to 99.9%. Lastly, a completely dry (0% relative humidity) case was simulated by using the low relative humidity case with the diffusiophoretic deposition velocity set to zero. The thermophoretic deposition velocity was also set to zero.

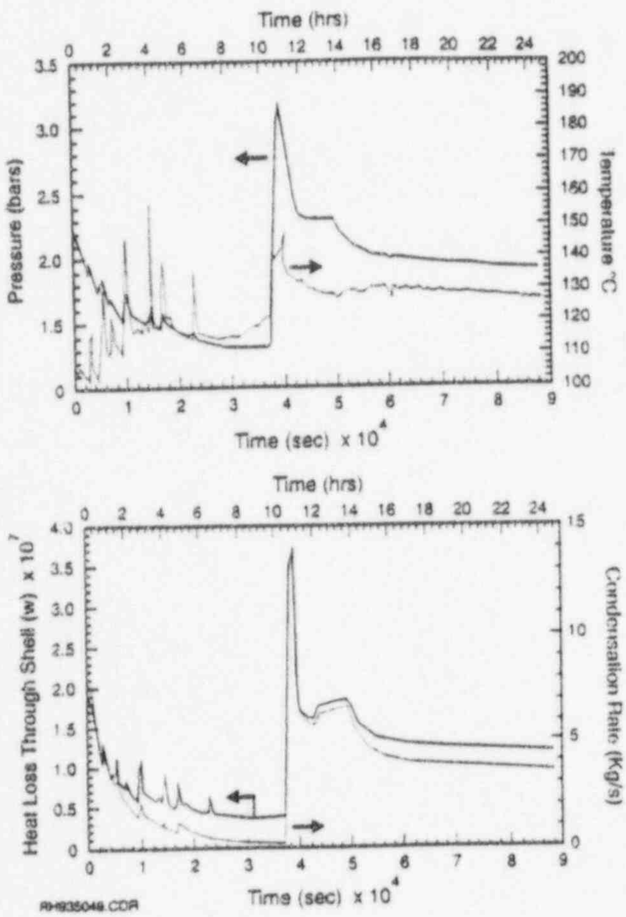

Figure 1 - Thermal-Hydraulic Conditions for Low (30-50%) Relative Humidity Sequence

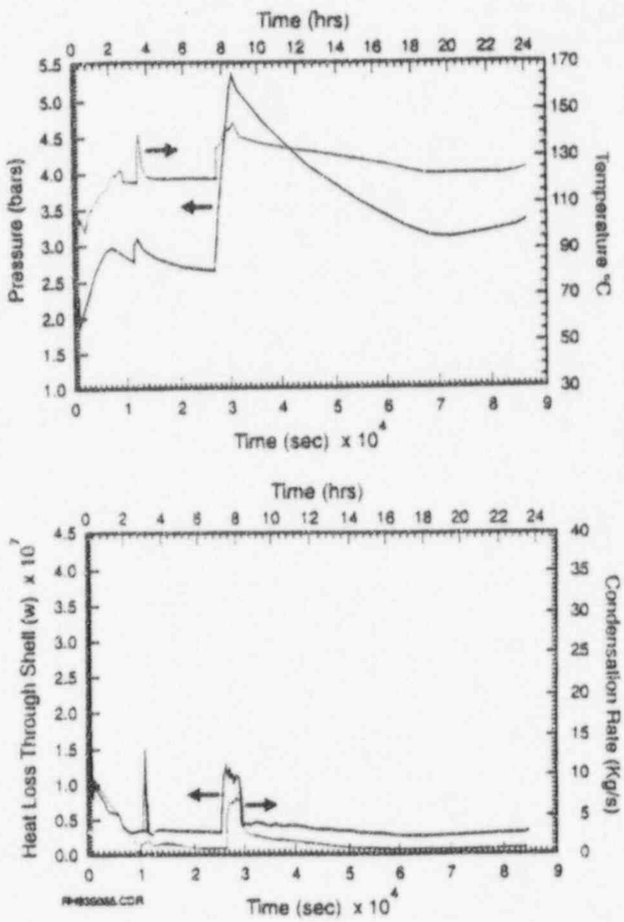

Figure 2 - Thermal-Hydraulic Conditions for Moderate (80-85%) Relative Humidity Sequence

For such a case, only the gravitational sedimentation and containment leakage removal mechanisms would be active. This provided a means of comparing the codes performance and the hand calculational method for a pure coagulation and sedimentation condition.

COMPARISON MATRIX

A series of sensitivity cases that varied the amount of inert aerosols released from the RCS to the containment for each of the four sets of thermal hydraulic conditions was performed. One additional set of sensitivity cases (10 through 14) was performed for the low relative humidity thermal hydraulic condition which set the thermophoretic deposition velocity to zero. In these cases, the importance of the thermophoretic effect due to the temperature gradient near the containment wall was discernable by comparing their release fractions to those for cases 1 through 3. For the purposes of the comparisons provided in this paper, only condensation on the containment shell has been included. In an actual reactor plant, the internal walls, floors, equipment, and other structures provide additional condensation surfaces. The total containment condensation rate could double or triple compared to only the containment shell condensation rate when such additional condensation surfaces are included. Such an increase in condensation rate would also increase the effectiveness of the diffusiophoretic aerosol removal. Table 2 summarizes the set of cases used in the comparisons.

CODE CALCULATION RESULTS

Each of the 18 cases defined in Table 2 was assessed using both the NAUAHYGROS and MAAP4 computer codes. Each code was used to calculate the 24 hour integrated mass of fission products that leaked from the containment to the environment. The fission products were grouped into volatile and non-volatile species. The volatile species were cesium hydroxide, cesium iodide, and tellurium. The non-volatile species were barium oxide, strontium oxide, cerium oxide, lanthanum oxide, ruthenium, and antimony. The environmental releases are presented as release fractions. The release fractions were calculated by dividing the integrated mass of leaked fission products for each of the two groups of species by the total integrated mass of aerosols delivered from the RCS to the containment atmosphere over the 24 hour interval. The results of the calculations for each code are summarized and compared in Tables 3 and 4. These tables include the percentage difference between the two code predictions for each of the cases. The percentage difference was determined by subtracting the NAUAHYGROS calculated release fraction from the MAAP4 calculated release fraction and dividing the difference by the NAUAHYGROS calculated release fraction. This quotient was then cast as a percent and recorded for each case. Positive values of the percentage difference indicate a higher calculated release fraction for MAAP4 compared to NAUAHYGROS.

HAND CALCULATION RESULTS

Correlations for dry aerosol sedimentation based on similitude rules have been developed[4] for making engineering predictions of the instantaneous particle cloud density. The time dependent particle cloud densities are coupled with the knowledge of the containment leakage rate to determine the fission product inventory that escapes containment. The use of correlations in such a methodology provides a basis for performing simple hand calculations. Such calculations have been conducted for the purpose of comparison to the code calculations in this paper. This simplified methodology ignores the benefits of wet containment conditions (thermophoresis and diffusiophoresis) and assumes that dry aerosol sedimentation is the only removal mechanisms other than leakage from the containment. Due to the small containment leakage rate, the airborne aerosol concentration can be assumed to be independent of the leakage. This assumption means that the leakage does not impact the dry aerosol particle size distribution which influences the sedimentation rate.

The hand calculation is performed by considering the 24 hour interval as consisting of a series of phases. During the first phase, the constant influx of aerosols to the containment gas space is assumed to linearly build up the aerosol concentration until either a steady-state condition is reached or the early injection interval ends. If the steady-state airborne aerosol concentration is obtained before the end of the early injection interval, then the steady-state concentration maintains until the end of the early injection interval. Upon the termination of the early injection interval and a switch to the lower late injection rate, a waiting interval is calculated. Per the correlations developed by Epstein, the waiting interval applies while the aerosol size distribution adjusts to the change in the source term delivery rate by transitioning from the first steady-state particle size distribution to a new particle size distribution. This transition involves the coagulation of the particles first due to Brownian motion and then, as a particle mean size increases, by gravitational coagulation. Following the waiting interval, the next phase considered involves the decay of the suspended aerosol mass density to a new steady-state value that corresponds to the second (late) lower aerosol injection rate. The final phase considered begins upon reaching the second steady-state aerosol concentration and continues until the end of the 24 hour interval. By assessing the suspended aerosol mass concentration for this sequence of phases, a time history of the suspended aerosol mass concentration is produced. This time history is integrated with the containment leakage rate to estimate the total 24 hour integrated mass of aerosols that leak from the containment to the environment.

Table 2 - Matrix of Sample Cases Used in NAUAHYGROS/MAAP4 Comparison

Case	Relative[(1)] Humidity	Fission Products To Inerts Release Rates	Active Removal Mechanisms: All[(2)]	Active Removal Mechanisms: No[(3)] Thermophoresis	Active Removal Mechanisms: Only[(4)] Sedimentation
1	low	1:0	X		
2	low	1:1	X		
3	low	1:5	X		
4	medium	1:0	X		
5	medium	1:1	X		
6	medium	1:5	X		
7	high	1:0	X		
8	high	1:1	X		
9	high	1:5	X		
10	low	1:0		X	
11	low	1:1		X	
12	low	1:2		X	
13	low	1:5		X	
14	low	1:10		X	
15	low	1:0			X
16	low	1:1			X
17	low	1:2			X
18	low	1:5			X

(1) Relative humidities: low (30-50%), medium (80-85%), and high (99.9%).

(2) This includes leakage, thermophoresis, diffusiophoresis, hygroscopicity, and gravitational sedimentation. Impaction on internal structures and grating was not considered in these cases.

(3) All the removal mechanisms listed above, except for thermophoresis. The principal effective mechanisms are leakage, diffusiophoresis, and gravitational sedimentation.

(4) Only gravitational sedimentation and leakage are considered.

Table 3 - Comparison of NAUAHYGROS and MAAP4 Results: All Aerosol Removal Mechanisms Active

Case	Relative[1] Humidity	Fission Products to Inerts Release Rates	Release Fraction[2]: NAUAHYGROS non-volatiles ($\times 10^{-4}$)	NAUAHYGROS volatiles ($\times 10^{-6}$)	MAAP4 non-volatiles ($\times 10^{-4}$)	MAAP4 volatiles ($\times 10^{-6}$)	$\%\Delta$[3] non-volatiles	$\%\Delta$[3] volatiles
1	low	1:0	0.93	0.72	1.24	1.02	+ 33	+ 42
2	low	1:1	0.57	0.44	0.77	0.63	+ 35	+ 43
3	low	1:5	0.22	0.16	0.30	0.25	+ 36	+ 56
4	medium	1:0	2.34	1.85	3.19	2.73	+ 36	+ 48
5	medium	1:1	1.33	1.02	1.81	1.51	+ 36	+ 48
6	medium	1:5	0.44	0.31	0.53	0.40	+ 20	+ 29
7	high	1:0	0.31	0.20	1.23	1.0	+ 297	+ 400
8	high	1:1	0.20	0.13	0.76	0.62	+ 280	+ 377
9	high	1:5	0.084	0.058	0.3	0.25	+ 257	+ 331

[1]Relative humidities: low (30-50%), medium (80-85%), and high (99.9%).

[2]The release fraction represents the integrated 24 hour total aerosol mass leaked to the environment as a fraction of the integrated 24 hour fission product mass released to the containment from the RCS.

[3]$\%\Delta$ is the percent difference in the release fraction calculated by each code. A positive value of $\%\Delta$ indicates that the MAAP4 release fraction exceeded the NAUAHYGROS release fraction.

Table 4 - Comparison of NAUAHYGROS and MAAP4 Results: Selected Aerosol Removal Mechanisms Active

A) Active Aerosol Removal Mechanisms Include Leakage, Diffusiophoresis, Hygroscopicity[1] and Gravitational Sedimentation. No Thermophoresis Active.

		Fission Products to Inerts	Release Fraction[3]					
			NAUAHYGROS		MAAP4		%Δ[4]	
Case	Relative[2] Humidity	Release Rates	non-volatiles ($\times 10^{-4}$)	volatiles ($\times 10^{-6}$)	non-volatiles ($\times 10^{-4}$)	volatiles ($\times 10^{-6}$)	non-volatiles	volatiles
10	low	1:0	2.03	1.8	2.54	2.29	+ 25	+ 27
11	low	1:1	1.15	0.99	1.43	1.26	+ 24	+ 27
12	low	1:2	0.78	0.66	0.99	0.86	+ 27	+ 30
13	low	1:5	0.38	0.3	0.49	0.41	+ 29	+ 37
14	low	1:10	0.19	0.14	0.25	0.2	+ 32	+ 43

B) Active Aerosol Removal Mechanisms Include Leakage and Gravitational Sedimentation. No Thermophoresis, Diffusiophoresis, or Hygroscopicity Active.

		Fission Products to Inerts	Release Fraction[3]					
			NAUAHYGROS		MAAP4		%Δ[4]	
Case	Relative[2] Humidity	Release Rates	non-volatiles ($\times 10^{-4}$)	volatiles ($\times 10^{-6}$)	non-volatiles ($\times 10^{-4}$)	volatiles ($\times 10^{-6}$)	non-volatiles	volatiles
15	low	1:0	3.5	2.97	3.89	3.43	+ 11	+ 15
16	low	1:1	1.83	1.47	1.87	1.55	+ 2	+ 5
17	low	1:2	1.19	0.92	1.2	0.97	+ 1	+ 5
18	low	1:5	0.55	0.39	0.55	0.42	0	+ 8

[1]Hygroscopicity active but not a significant removal mechanism in low relative humidity conditions.

[2]Low relative humidity is 30-50%.

[3]The release fraction represents the integrated 24 hour total aerosol mass leaked to the environment as a fraction of the integrated 24 hour fission product mass released to the containment from the RCS.

[4]%Δ is the percent difference in the release fraction calculated by each code. A positive value of %Δ indicates that the MAAP4 release fraction exceeded the NAUAHYGROS release fraction.

The results of applying these correlations to cases 15 through 18 in the matrix of sample cases (see Table 2) are compared in Table 5 with the corresponding results from the two computer codes. Close agreement between the hand calculated values and the code calculated values is demonstrated.

DISCUSSION OF RESULTS

A review of the percent difference shown in Table 3 for cases 1 through 6 demonstrates good agreement between the predicted results from each code. A variation of between 20% and 50% is quite good for aerosol removal calculations and especially if it is realized that the calculated percent differences were performed on very small numbers (release fractions). Good agreement between the predicted release fractions is obtained for those low and medium relative humidity cases with all the aerosol removal mechanisms considered. The relative contribution of these removal mechanisms (thermophoresis, diffusiophoresis, sedimentation, and hygroscopicity) varies throughout the nine cases reported in this table. For example, separate hand calculations can be made to estimate the removal rates (λs) for the thermal-hydraulic profiles provided in Figure 1 at 3.6 and 5.4E4 seconds for the case of no inert aerosols. The resulting estimates are summarized in Table 6 for these two instances. Table 6 shows that the combined removal rate for thermophoresis and diffusiophoresis is approximately an order of magnitude greater than the removal rate due to sedimentation. The thermal-hydraulic profiles (Figures 1 and 2) selected for this paper cover a representative range of conditions within containment for severe accidents. The assessments based on these profiles indicate that the removal mechanisms due to wall heat flux (thermophoresis) and condensation rate (diffusiophoresis) are typically present and dominate over dry sedimentation and leakage. Thus, the thermal-hydraulic conditions are the most influential parameters that affect the several passive aerosol removal mechanisms.

A systematic variation is observed such that the release fractions calculated by MAAP4 were found to be greater than the release fractions calculated by NAUAHYGROS for these cases. The exact reason for this systematic variation has not been exhaustively investigated by the authors. However, it should be noted that the two codes employ different solution techniques (sectional versus non-dimensional correlation). Each code must integrate the wall heat flux and condensation rate profiles and has its own numerics and time step selection algorithms. The relatively small and positive variation between the two solutions to the same governing integro-differential equation implies a consistent application of each code's solution technique.

The degree of agreement is significantly different for cases 7, 8, and 9 which had a very high relative humidity. For these cases hygroscopic effects were important. The observed differences in the release fractions indicate much lower release fraction for NAUAHYGROS. Differences in the release fraction calculated by the two codes for such cases are attributed to the difference in the treatment of hygroscopic aerosol removal incorporated in the codes. The NAUAHYGROS code is a sectional code and simultaneously calculates the growth of aerosol particles due to steam condensation and their agglomeration. The formulation used in MAAP4 assesses sedimentation removal rates (λs) by first considering only dry aerosol agglomeration and sedimentation and then only hygroscopic particle growth and sedimentation. The process with the largest removal rate (λ) is then selected and used in the MAAP4 calculation. MAAP4 does not calculate the combination of hygroscopic particle growth and agglomeration.

Table 4A (cases 10 through 14) compares the code results when the thermophoretic removal mechanism is not included. Good agreement is again demonstrated for a range of inert aerosol release rates. The results summarized in Table 4B exhibit nearly identical results between the predicted release fractions

Table 5 - Comparison of Release Fraction From Hand Calculations and Computer Codes[1]

	NAUAHYGROS		MAAP4		Hand Calculations[2]	
Case	Volatiles ($\times 10^{-4}$)	Non-Volatiles ($\times 10^{-6}$)	Volatiles ($\times 10^{-4}$)	Non-Volatiles ($\times 10^{-6}$)	Volatiles ($\times 10^{-4}$)	Non-Volatiles ($\times 10^{-6}$)
15	3.5	2.97	3.89	3.43	3.97	2.8
16	1.83	1.47	1.87	1.55	2.27	1.61
17	1.19	0.92	1.2	0.97	1.48	1.05
18	0.55	0.39	0.55	0.42	0.63	0.45

(1) Cases 15 through 18 of Table 2 were used for the comparison as these cases only consider leakage and gravitational sedimentation of dry aerosols.

(2) Based on dimensionless correlations developed.[4]

Table 6 - Comparison of Removal Rates (λs) for Figure 1

Event Time (sec)	Removal Rates (sec^{-1}) For: Thermophoresis	Diffusiophoresis	Sedimentation	Leakage
3.6E4	1.2E-4	2.5E-6	1.2E-5	1.4E-8
5.4E4	5.9E-5	5.5E-5	9E-6	1.4E-8

for each code when only gravitational sedimentation and leakage are the active aerosol removal mechanisms. These results are compared to the hand calculations results in Table 5. The high degree of agreement between the results calculated by the computer codes and by hand is encouraging. The hand calculational method provides a useful means of performing engineering estimates without reliance on computer codes, at least for situations with relatively dry conditions. Furthermore, if one chooses to approximate containment aerosol removal by natural removal mechanisms due to only dry aerosol sedimentation, then the use of either the computer codes or the hand calculations could provide satisfactory results regardless of the relative humidity. The approximation of the containment behavior based only on dry aerosol sedimentation provides a bounding estimate for the release fraction compared to the results that would be obtained by detailed code calculations that include additional aerosol removal mechanisms.

The results summarized in Tables 3 and 4 can be rearranged and regrouped to demonstrate the effect of each individual parameter varied in this study. When this is done and the release fraction is redefined as the fraction of each fission product specie's leaked mass divided by that specie's injected mass, then the relative important of the individual parameters can be observed. The thermal hydraulic conditions (see Figures 1 and 2) are found to have the most impact on the release fraction. Varying the thermal hydraulic conditions caused the release fraction to increase by a factor of 2 to 3 (compare case 1 to 4, 2 to 5, and 3 to 6). The relative strength of the thermophoretic and diffusiophoretic removal mechanisms is sequence dependent. The thermophoretic deposition velocity is proportional to the total heat loss into the containment wall and the diffusiophoretic deposition velocity is proportional to the condensation rate. Each set of thermal-hydraulic conditions assessed in this paper is unique and varies with time. When both of these removal mechanisms are present they dominate the aerosol removal provided by gravitational sedimentation. Since thermophoresis and diffusiophoresis generally dominate the removal mechanisms and since they are not particle size dependent, it would be expected that the initial particle size for the airborne aerosols would not significantly impact the calculated release fraction. This was observed in a separate study.[3]

The effect of thermophoresis is demonstrated by comparing case 10 to 1, 11 to 2, and 13 to 3. The release fraction is seen to decrease by a factor of more than one-half when thermophoretic removal is added. Similarly, the effect of diffusiophoresis is demonstrated by comparing case 15 to 10, 16 to 11, 17 to 12, and 18 to 13. The release fraction is seen to decrease by a factor of less than one-half when diffusiophoretic removal is added. The thermophoretic effect is found to be comparable to but slightly larger than the diffusiophoretic effect. The largest release fractions are indicated when neither thermophoresis or diffusiophoresis are considered and only gravitational sedimentation and leakage are considered.

Lastly, the impact of inert aerosols was to decrease the environmental release of the fission product species as the amount of inert aerosol is increased. This effect is evident in Table 3, but is exaggerated because the release fractions in Table 3 are based on the total aerosol into the containment, not on the source of each specie. The decrease in release to the environment is due to the increased removal of fission product species resulting from their increased agglomeration with the inert aerosol, and is therefore most pronounced for the cases in which gravitational sedimentation predominates, since this is the mechanism that is sensitive to aerosol agglomeration. When either thermophoresis or diffusiophoresis or both were active in low or medium relative humidity conditions, they were generally more important than gravitational sedimentation in removing aerosol, and the impact of the inert species was smaller.

CONCLUSION

Good agreement has been obtained between predicted release fractions for the NAUAHYGROS and MAAP4 codes in this study. A range of containment conditions and natural aerosol removal mechanisms were studied. The behavior regarding each code's treatment of hygroscopic aerosol behavior in combination with other aerosol removal mechanisms was identified as the source for the one set of significant departures in their behavior which only occurred for saturation steam conditions.

Hand calculational techniques are found to be suitable engineering tools based on close agreement of

their results and the code calculated results. Scoping or bounding analyses predicted on hand calculations and therefore based on dry aerosol sedimentation could thus be conveniently performed by hand.

The specific results and release fractions presented in this paper are only intended for use in comparing the different calculational methods. It would be difficult to extrapolate the results to an actual reactor case. Rather, each reactor case should be analyzed directly for a number of reasons. The sample case employed in this study included inert aerosol generation only during the early aerosol injection phase. Revaporization effects were not considered in these comparisons. Condensation on internal walls and floors were not included in the condensation rates used for the diffusiophoresis calculated in these comparisons. Hence, the release fractions reported in this study are not intended for direct application in an actual containment assessment. The thermal hydraulic conditions are seen to have the most influence on the several removal mechanisms and these conditions can be sequence specific. However, the relative effects of the parameters varied in this study are valid and suggest that the thermal hydraulic conditions followed by the presence of either thermophoresis or diffusiophoresis and finally followed by the presence of inerts are the most consequential parameters.

ACKNOWLEDGEMENT

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3. D.E. LEAVER, J. LI, R. SHER, Passive ALWR Containment Natural Aerosol Removal, ARSAP Report (1993).

4. M. EPSTEIN, P.G. ELISON, R.E. HENRY, "Correlations of Aerosol Sedimentations", Journal of Colloid and Interface Science, Vol. 113, No. 2 (1986).

Determination of Fission Product Releases (480.205)
For each release class, identify and discuss the phenomena, models, and input assumptions that have the greatest influence on the fission product release estimates.

Response:
Chapter 45 will be updated to include the requested discussion in the PRA revision 7.

Uncertainty in Source Term Estimates (480.206)
To the extent possible, provide comparisons of the MAAP-based source terms for the AP600 with the comparable source term distributions from NUREG-1150, and with revised licensing source terms in NUREG-1465.

Response:
NUREG-1465 does not provide an offsite fission product release source term, but rather a source term from the RCS to the containment. A comparison of releases to the containment could be provided, but would be of limited value, so it will not be provided in the AP600 PRA. The comparison with NUREG-1150 source term will be provided in the PRA update.

Uncertainty in Source Term Estimates (480.207)
Where significant differences exist between the MAAP- and NRC-based source terms, provide a reassessment of risk results using the NRC values.

Response:
If there is a significant difference between MAAP4 and NRC physically based source terms in risk significant release categories, an assessment on the impact to risk will be provided.

Level 3 PRA - Analysis of Consequences (480.208)
Provide additional information concerning the version of the MACCS code used for offsite consequence analysis, and whether the results of the BEIR V studies are reflected in these code calculations. If not, provide an interpretation of the AP600 risk results, considering the differences between the MACCS version and the recent health effects studies.

Response
The version of the MACCS code utilized to perform the current issue of the AP600 level 3 analysis was version 1.5.11. This version does not reflect the results of BIER V. However, for the planned revision of the level 3 analysis, version 1.5.11.1 will be utilized. This version does implement a revised cancer model consistent with recent reports, such as BEIR V, ICRP 60 and LMF-132 (NUREG/CR-6059, SAND92-2146).

Level 3 PRA - Analysis of Consequences (480.209)
Provide a description of how ingestion doses are accounted for in the offsite consequence calculations.

Response
The level 3 PRA modeled only the acute dose and the lifetime dose commitment resulting from exposure during the initial 24 or 72 hours following the onset of core damage. The dominant dose pathways during this period include: cloudshine, cloud inhalation, groundshine, resuspension inhalation and skin deposition. Ingestion is a chronic exposure pathway which is not considered. Evaluation of the chronic dose, and therefore the ingestion pathway, would require detailed information on the land use fractions, crop and dairy utilization, land values, decontamination methods, effectiveness and costs. This data is very dependent upon the site selection. No consistent data is available to evaluate the chronic dose or justify its' applicability to sites other than the referenced site.

Level 3 PRA - Analysis of Consequences (480.210)

Provide additional information regarding the site parameters on which offsite health effects are based, including demographics, meteorology, and evacuation assumptions.

Response

The site data, demographics and meteorology were obtained from the "Advanced Light Water Reactor Document," Volume III, Annex B to Appendix A to Chapter 1, "Utility Requirements for Passive Plants, PRA Key Assumptions and Groundrules," Rev. 2, December 1991. Since this information did not provide topographical information to define the required MACCS site input, the site land use and crop data from the Surry Plant Site (provided with the MACCS model) were used to complete the input and provide an acceptable MACCS input file. Additionally, since the Surry Site is close to the ocean, the ocean sectors were changed to land to avoid assigning the ALWR population to ocean. These changes had no effect on the calculated acute and lifetime committed dose during the initial exposure. In order to maximize the potential dose, no sheltering or evacuation was assumed.

Level 3 PRA - Analysis of Consequences (480.211)
Provide an assessment of offsite consequences (public exposure within a 50-mile radius) assuming no protective measures (evacuation or sheltering) are taken within the first 24 hours of accident initiation.

Response
The basis for the Level 3 analysis was no sheltering or evacuation during the first 24 or 72 hours of the accident. Therefore, Table 49-5 provides the population dose at 80.5 KM (50 miles) for both 24 and 72 hour durations.