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November 8, 1996

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

ATTENTION: T. R. QUAY

SUBJECT: WESTINGHOUSE RESPONSE TO NRC REQUEST FOR ADDITIONAL INFORMATION ON THE AP600

Dear Mr. Quay:

Enclosed are Westinghouse responses to NRC requests for additional information (RAIs) on the AP600 Design Certification program. The enclosure contains responses to 51 RAIs pertaining to the AP600 Probabilistic Risk Assessment Level 2 and severe accident analysis. Also included in the enclosure are two RAI responses pertaining to accident management.

These responses close, from a Westinghouse perspective, the addressed questions. 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.

Brian A. McIntyre, Manager
Advanced Plant Safety and Licensing

/nja

Enclosure

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cc: J. Sebrosky, NRC (enclosure)
J. Kudrick, NRC (w/o enclosure)
N. J. Liparulo, Westinghouse (w/o enclosure)

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Enclosure to Westinghouse
Letter NSD-NRC-96-4881

November 8, 1996

Attachment A to NSD-NRC-96-4881
Enclosed Responses to NRC Requests for Additional Information

Re: Level 2 and Severe Accident Analysis

480.83	480.84	480.85	480.86
480.87	480.102	480.105	480.106

Revision 1 to the following RAIs:

480.120	480.122	480.124	480.126
480.132	480.136	480.137	480.138
480.145	480.146	480.147	480.148
480.150	480.151	480.153	480.154
480.155	480.156	480.158	480.159
480.160	480.161	480.162	480.165
480.166	480.168	480.175	480.177
480.178	480.183	480.184	480.185
480.186	480.187	480.189	480.197
480.198	480.200	480.201	480.202
480.203	480.206	480.207	

Re: Accident Management

480.439
480.212 (Rev. 1)

Question: 480.83 EXTERNAL VESSEL COOLING

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

Response:

The failure probability of 0.022 of cavity flooding valves was the individual valve failure given the success of at least one line of cavity flooding. It was intended to determine the probability of the failure of the 10-inch diameter lines, which means that, by definition, only the 4-inch diameter line was injecting.

Due to a change in the design of the system which is used to flood the reactor cavity, the 10-inch diameter line and 4-inch diameter line have both been changed to 6-inch diameter lines with one motor-operated valve and one explosive valve in each line. This change is reflected in revision 8 of the AP600 PRA. The IVR decomposition event tree has been eliminated from the AP600 PRA and the thermal-hydraulic analysis are provided by the IVR ROAAM as presented in DOE/ID-10460. The 6-inch diameter line characteristics (flow area, length, resistance) are used in PRA Chapter 39 to calculate the IRWST draining time to support the time window available for the operator action after entering the Emergency Response Guideline FR.C-1.

The core-exit thermocouples used to monitor the need for cavity flooding and the valves used to flood the reactor cavity are Class 1E and are powered by Class 1E dc power. The availability of the power sources, availability of the valves, ability of the operator to diagnose the situation, and success of the operator are all considered in the fault tree (IWF) used to quantify the failure probability of the cavity flooding (on the AP600 CET). Since the fault trees are linked to node IR of the CET, the availability of power sources is treated consistently for all sequences on the CET.

Question: 480.84 EXTERNAL VESSEL COOLING

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

Response:

The equipment used to flood the reactor cavity is safety-related equipment including surveillance and maintenance requirements. These requirements are outlined under the In-Service Inspection Program discussed in the response to RAI 210.24.

Question: 480.85 EXTERNAL VESSEL COOLING

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

Response:

The line sizing for the system which is used to flood the reactor cavity is based on the design function of the lines which is to provide suction for the RNS pumps in recirculation mode. The lines are sized based on suction requirements for both pumps running unthrottled.

The design of the lines used for cavity flooding is two 6" diameter lines. The success criteria for the cavity flooding in the AP600 PRA Chapter 39 is a two-tier criterion:

1) the lower head must be submerged (to the 83-foot elevation) prior to the relocation of core debris to the lower plenum, and
2) the vessel must be submerged above the top of the highest in-vessel debris pool (elevation 86-foot elevation) before the slumping of the whole core into the lower plenum.

Question: 480.86 EXTERNAL VESSEL COOLING

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

Response:

In revision 8 of the AP600 PRA Chapter 39, the time window available for the operator to flood the cavity is calculated based on the success criteria outlined in the response to RAI 480.85. Conservatively based on the slowest IRWST draining rate through one 6-inch line and the fastest timing for the core relocation behavior presented in the DOE IVR ROAAM report DOE/ID-10460, there is a 25 minute time window available once the operator enters emergency response guideline FR.C-1. Only 20 minutes are credited. If this 20 minute window is not met, the reactor vessel is assumed to fail. Therefore, the effects of delayed cavity flooding are conservatively accounted for in the PRA.

Question: 480.87 EXTERNAL VESSEL COOLING

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

Response:

Flooding of the reactor cavity after the relocation of core debris to the lower head does not meet the definition of successful flooding based on the two-tier success criterion outlined in the response to RAI 480.85, and the reactor vessel is assumed to fail.

Question: 480.102 EXTERNAL VESSEL COOLING

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

Response:

The scaling and applicability of the COPO, ULPU, mini-ACOPO, ACOPO are discussed in detail in the DOE/ID-10460 report on In-Vessel Retention.

Question: 480.105 EXTERNAL VESSEL COOLING

Provide a discussion of the decay power curve for all plausible burnup conditions to support the 1.0 MW/m3 volumetric heat rate of the oxide pool. Identify the minimum time after scram that external vessel cooling is needed.

Response:

The base case IVR ROAAM analysis presented in DOE/ID-10460 examines decay heat rates from 4 hours, at the earliest, after shutdown. No "physically reasonable" vessel failure is predicted in the ROAAM analysis. Chapter 8 of the In-vessel Steam Explosion ROAAM analysis (DOE/ID-10541) provides the most recent detailed assessment of the core relocation timing which includes the effect of the core reflector which is not modeled in other analyses including MAAP4. The large mass of the reflector slows the core relocation to the lower head with respect to other analyses which do not include it. The first core relocation is predicted to occur 80 minutes (1.33 hours) after rapid oxidation at the earliest. The time delay from accident initiation to rapid oxidation is estimated from the MAAP4 analyses in the PRA as 75 minutes (1.25 hours). Therefore, the earliest first relocation would be at 155 minutes (2.6 hours) after shutdown. An additional 60 minutes is required to establish the naturally circulating molten pool behavior (called Final Bounding State or FIBS in ROAAM reports) which provides the bounding heat transfer to the vessel wall. Therefore, the conservatively earliest time after shutdown for IVR is 215 minutes (3.6 hours). The decay power density at 3.6 hours is less than 5% higher than estimated at 4 hours and is covered by the decay heat sensitivity analyses presented in Chapter 7 of the IVR report.

Question: 480.106 EXTERNAL VESSEL COOLING

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

Response:

To credit successful IVR on the AP600 containment event tree, the initiating event and the accident progression must result in a configuration with the RCS depressurized and the cavity flooded prior to debris relocation to the lower head. In this configuration, all sequences provide the same boundary conditions with respect to IVR. The two-tier success criterion for cavity flooding timing outlined in the response to RAI 480.85 and the decay power density estimation in the response to RAI 480.105 provide a justification of the bounding nature of the timing with respect to the progression of the accident prior to the formation of the circulating pool configuration.

Question: 480.120

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

Response:

Revision 8 of the AP600 PRA does not rely on code results for predicting the hydrogen mixing behavior. Lumped parameter MAAP4 code results presented in Chapter 41 of the PRA are used to provide insights into the plant behavior with respect to hydrogen generation and mixing during the early time frame of the severe accidents, but the quantification of the containment failure probability from hydrogen combustion is based on probability distributions which describe the uncertainty in the hydrogen and steam concentrations, hydrogen generation and containment pressure. The MAAP4 sub-nodal physics modeling is not used. The lumped parameter modeling is not considered to be adequate to conclusively predict hydrogen mixing. Stratification of steam and hydrogen is considered in the revision 8 analysis.

During the early time frame in which hydrogen is generated and not yet mixed in the containment atmosphere, the possibility of local detonation due to DDT in the compartments below the operating deck and in the IRWST is assessed. The hydrogen release rate and location, IRWST water level, and availability of stage 4 depressurization lines determine the hydrogen and steam concentration probability distributions in each compartment. MAAP4 analyses are used to help predict the appropriate ranges of gas concentrations for the variations in the accident sequences in the early time frame. The hydrogen and steam concentrations determine the detonation cell widths and are used to quantify the probability of local detonation in the early time frame. The containment is assumed to fail from the occurrence of any detonation.

In the intermediate time frame, when the hydrogen is mixed in the containment atmosphere, containment integrity is assessed for global combustion peak pressure and temperature. For the global burn assessment, the hydrogen, air and steam are assumed to be well-mixed in the containment and the adiabatic, isochoric complete combustion (AICC) peak pressure is calculated. Hydrogen generation and pre-burn containment pressure probability distributions are used for calculating the AICC peak pressure probability distribution. Containment failure probability is determined by combining the AICC peak pressure probability distribution and a containment fragility probability distribution.

Stratification of steam and hydrogen in the intermediate time frame are considered for the assessment of DDT. AP600 Large Scale Test (LST) for the PCS system suggest that hydrogen and air are well-mixed in the containment, but steam may stratify in the containment such that the steam concentration below the operating deck is significantly lower than the steam concentration above the operating deck. Therefore, for the DDT assessment in the long term, the mixture in the CMT room, a geometry conducive to flame acceleration without a steaming source, is conservatively assumed to be a mixture of dry-air and hydrogen.

The HDR tests 11.2 and 11.4 suggest that hydrogen released above the operating deck may stratify in the containment dome. Unlike HDR, hydrogen released above the operating deck in the AP600 would be through the IRWST near the containment wall into the PCS downflow with little momentum. This countercurrent flow would promote hydrogen mixing in the containment. So for the base case analyses, all hydrogen-air mixtures are assumed to be well-mixed. A sensitivity case to the containment event tree quantification is presented in Chapter 50 in which all cases with significant hydrogen release through the IRWST (specifically accident classes 1AP and 3D) are assumed to fail the containment due to hydrogen stratification, if the igniters are off, or diffusion flame, if the igniters are on. The other cases slowly release hydrogen from low pressure through stage 4 ADS to the loop compartments, bypassing the IRWST.

Therefore, the AP600 PRA and hydrogen analysis do not rely on lumped parameter code results and do not assume that the gas concentrations in the containment are well-mixed as would be predicted by a lumped parameter code.

~~Section 46 of the PRA describes the containment mixing analysis that was conducted to demonstrate compliance with 10 CFR 50.34(f). One criterion from this regulation is that the combustible concentrations of hydrogen would not collect in areas where unintended combustion or detonation could cause loss of containment integrity or loss of appropriate mitigating features. The hydrogen mixing analyses were performed to confirm that this regulation was satisfied by the AP600 containment configuration. The hydrogen mixing analyses were performed with the Westinghouse_GOTHIC computer code, version 1.0 which is a lumped parameter code. For the details of these analyses and a description of the code Section 46 of the PRA should be consulted.~~

~~The performance of lumped parameter codes in predicting mixing in containment depends upon several factors, such as plant specific parameters (containment geometry), sequence specific parameters (light gas injection point and flow rate (break size); availability of active mixing systems), and the nodalization scheme selected to represent them.~~

~~Good mixing is observed and well predicted in tests with large LOCAs or open containments (small number of compartments and large areas between compartments) or active internal mixing (spray operation). For example a large scale test with a large LOCA, and active internal sprays, was run by NUPEC in Japan and compared to several lumped parameter codes as part of International Standard Problem Number 35. Likewise, a large LOCA test (T31.5) with light gas injection near the containment's mid height and no internal sprays was run in the HDR facility in Germany. These test conditions and configurations showed good mixing in containment. However, a small LOCA test (E11.2) in the HDR facility with no internal sprays and light gas injection near the containment's mid height showed stratification until external sprays on the containment dome were turned on and successfully induced mixing in containment. Application of traditional lumped parameter codes to the HDR E11.2 did not perform well in capturing the thermal stratification. It should be noted that the AP600 containment configuration is significantly different than the HDR E11.2 test. The AP600 containment is open with generally large areas between regions and a small number of subcompartments. Furthermore, the injection points for hydrogen are at or below the operating deck which means they are low in the containment.~~

~~To overcome mixing deficiencies in a lumped parameter code, the effects of thermal stratification must be considered. In the MAAP4 code, thermal plume and "subnodal physics" models are included to account for these effects. The thermal plume model considers light (less than ambient density) gaseous discharges from the reactor coolant system into the containment. The light gas plume is accelerated by buoyant forces and slowed because it~~

~~entrains the ambient heavier gas. As the plume rises from one compartment to another, an equal volume of gas flows down to fill the void left by the rising plume. Thus, an exchange of gas between compartments due to plume behavior of gaseous effluents is modeled. The subnodal physics model determines the condition required for flow stagnation in the node in question (based on pressure equilibrium with its horizontally connected node). The required stagnation condition is that the penetration depth (within the node in question) of the lighter gas entering from the node above through a vertically oriented junction, or of the heavier gas entering from the node below exceeds the distance required to clear the nearest horizontally oriented junction, then the junction is shut off to any buoyancy-driven flow. The model, hence, does not always allow buoyancy-driven flow through the vertically oriented junction. The rate of gas mixing among various nodes can then be properly predicted.~~

~~The results of MAAP4 calculations of the HDR Test E11.2 with and without the subnodal physics model are compared in Figures X-1, X-2, and X-3 [Ref. 1]. Figure X-1 shows an excellent improvement in the calculated gas temperature at 0 m elevation from a maximum error of ~ 36 K to about 6 K. Figure X-2 and X-3 also shows a very good improvement in the calculated helium concentration in the upper dome and at the 10 m elevation. It can be noted that the trend of the test data, which is incorrectly predicted without the plume and subnodal physics models, is correctly predicted by the addition of these models.~~

~~Since the AP600 reactor containment is much more open and less compartmentalized than the HDR containment, the accuracy of the MAAP4 code for the reactor containment can be expected to be as good or better than that of the HDR benchmarking. This fact is demonstrated in the MAAP4 and W GOTHIC code benchmarking based on the NUPEC's large scale hydrogen mixing and distribution test # M-7-1 which is a 1/4 scale 4-loop PWR model containment with internal structures reflecting the real plant layout. This test included internal containment sprays which induced good mixing. Most lumped parameter codes, including W GOTHIC and MAAP4, performed well in predicting the helium concentrations [Ref. 2].~~

~~References:~~

1. ~~MAAP4 User's Manual, Volume 2, Part 2, EPRI, 1994.~~

2. ~~Final Comparison Report on ISP-35 NUPEC's Hydrogen Mixing and Distribution Test - Test M-7-1, ISP35-092 Rev. 5, September 10, 1994.~~

Question: 480.122

The first three stages of the ADS vent into the IRWST. The fourth stage vents into containment. It appears to exhaust into the lower containment either in the core makeup tank room or the steam generator room. The staff needs to have a better understanding of where the fourth stage vents into containment. What igniter locations have been provided for a release through this pathway. What effect do the ele ated temperatures of this type of release have on the possible combustion loads?

Response:

The fourth stage ADS for each reactor coolant loop discharges into each respective loop compartment. The fourth stage ADS is attached to each hot leg by single line ~~with a tee and~~ which branches into two ADS ~~valves~~ flow paths in parallel. The elevation of the ADS valves is above the flooded-up water level in the containment (approximately 108 elevation). Six igniters have been located in each steam generator subcompartment. Igniters are placed on each of the subcompartment's four walls and vary in elevation between 115 feet and 130 feet. These igniter locations will be effective for hydrogen release pathways for the fourth stage ADS valves.

~~The effects of elevated temperatures on burns are accommodated in two principal ways in the MAAP4 calculations. The flammability limits for combustible gases (hydrogen and carbon monoxide) are adjusted as a function of the temperature of the gas mixture. As the temperature increases, the lean flammability limit decreases while the rich flammability limit increases. Thus, there is a general broadening of the flammability limits as the gas mixture temperature increases. Secondly, the laminar flame speed is calculated as a function of the gas mixture temperature. The flame speed affects the burning rate which affects the energy release rate during combustion calculations. The energy release rate influences the temperature and pressure in the various containment regions. Thus, the effects of elevated temperatures on the combustion process and sub compartment pressures are included in the calculations performed with MAAP4.~~

If stage 4 is open, in the vast majority of cases, the reactor coolant system is fully depressurized prior to the hydrogen generation. Much of the RCS stored energy has been absorbed in the IRWST water during the ADS stages 1 through 3. Hydrogen is released slowly through stage 4 as it is produced in the core, and natural circulation in the containment provides cooling air to the loop compartments. The temperature in the loop compartment is not expected to be elevated significantly.

In the MAAP4 analysis presented in Attachment A to Chapter 41, this can be seen in the containment pressure and temperature plots for accident classes 3BE, 3BL, 3C, and 3BR. The loop compartment temperature remains below 260°F (400°K), except during reflooding or core slumping events. The accompanying increase in steam reduces the hydrogen concentration and increases the detonation cell width to provide a compensating effect to the increased temperature. The approach used to estimate the containment loading for hydrogen by assuming AICC peak pressure and temperature, and assumed containment failure for detonation is considered to be appropriate.

Question: 480.124

The staff is concerned about the possibility of detonable conditions when combustible gases are released through the IRWST. Near stoichiometric concentrations of hydrogen are predicted to exist at various times throughout the release of hydrogen based on separate work performed by Sandia [1] and the AP600 PRA. Steam concentrations are generally between 10-20% during times of high hydrogen concentrations. The transition of a deflagration to a detonation is the most likely mode of detonation initiation. Peraldi [2] has proposed a criterion for deflagration to detonation transition (DDT) which relates the detonation cell size of a mixture to a characteristic geometric length scale. Sandia used this criteria to give an estimate of the range of hydrogen concentrations that may detonate in the IRWST. Peraldi's criterion states that if a flame speed is near the sound speed in the combustion products, DDT will occur if the detonation cell size is on the order of, or less than, the minimum transverse dimension of the channel. The distance between the surface of the water and the top of the IRWST was estimated to be 0.5 m based on the input for MAAP calculations. Sandia estimated, according to Peraldi's criterion, what mixtures having detonation cell sizes on the order of 0.5 m or less may undergo a DDT. This corresponded to hydrogen-air-steam mixtures having hydrogen concentrations between approximately 18% and 56% for mixtures with 10% steam and 19% and 42% for mixtures with 20% steam. These conditions can occur for relatively long periods of time as was noted in the containment analysis report by Sandia [1].

Response:

The detonable conditions in the gas space of the IRWST as mentioned by the review staff can be postulated only in the event of igniter failures during certain classes of severe accident scenario. Whether DDT would occur or not depends on the availability of an ignition source. The only potential ignition source in the IRWST is the igniters. Operation of the stage 4 ADS valve would provide a flowpath which bypasses the IRWST and significantly reduces the mass of hydrogen released to the IRWST.

The conditional probability of DDT occurrence within the containment is considered in the quantification of the AP600 CET in the early (during hydrogen releases) and intermediate (after hydrogen is mixed in containment) time frames ~~by means of "early burn" and "late burn."~~. In the quantification of these CET top nodes, the conditional probability of random ignition source(s) was conservatively assumed to be 0.5 during the early time frame and 1.0 in the intermediate time frame. ~~Furthermore, t~~ The ~~maximum~~ conditional probability of DDT given an ignition source ~~and a detonable condition,~~ was ~~assigned 0.5~~ estimated using the method described in NUREG/CR-4803. Hence, in the AP600 PRA, a DDT is recognized with a certain probability in the IRWST and other compartments.

Question: 480.126

Besides DDT hot jet initiation is another mechanism that could initiate a detonation and should be addressed in the AP600 design.

Response:

Direct initiation of a detonation by a hot jet in the AP600 design is not considered to be possible. The initiation of a detonation by a hot jet would require a hot burning jet at sonic velocity to enter a non-inerted region with a high hydrogen concentration.

The AP600 design includes igniters which would burn hydrogen as it released such that its global concentration in containment would not exceed 10%. A hydrogen concentration of less than 10% in dry air is not detonable.

If the igniters were not operable the hydrogen concentration in containment could approach 14% if 100% of the zirconium in the core was oxidized by steam. For many sequences the primary system is depressurized by the ~~first three stages of the~~ ADS valves prior to the hydrogen generation. The release of hydrogen from the primary system ~~through the IRWST sparger and water pool~~ would not produce a hot jet. For LOCA sequences with no depressurization, a high temperature, hydrogen rich sonic jet could be postulated ~~produced~~. However, such a jet would not be burning as it exited the RCS as there would be no oxygen within the RCS to support burning. Such jets would lead to a diffusion flame (not detonation) if the jet encountered oxygen within the receiving containment region. Furthermore, such a blowdown to the containment would not encounter a high concentration of hydrogen in the containment. The potential source of hydrogen originates within the RPV due to zirconium oxidation in the core. The blowdown process could transfer the hydrogen from the RCS to the containment but the RCS would be depressurized in the process such that any subsequent hydrogen release from the RCS into a potentially flammable containment atmosphere would not be as a sonic jet.

Gas flows between the containment regions and sub-compartments in the open AP600 containment design during a blowdown are not expected to produce sonic jets. The large areas between containment regions are expected to minimize large sustained differential pressures and sonic velocities. However, if a burn occurred in a compartment below the operating deck a hot jet could be introduced to the upper containment of the AP600. If such a hot (burning) sonic and hydrogen rich jet were produced and if a burn occurred in the upper containment, a diffusion flame would result. Such a result was observed in an integral effects test (IET-11) conducted by Sandia National Laboratories in the Containment Technology Test Facility (Blanchard, T. K., et al., Quick-Look Report on the Eleventh Integral Effects Test (IET-11) in the Containment Technology Test Facility, September 1995). Thus, hot jet initiation of deflagrations but not detonations are considered for the AP600 design.

Chapter ~~Sections 36 through~~ 41 of the AP600 PRA provides decomposition event trees for ~~key severe accident~~ hydrogen combustion phenomena. ~~Section 41 addresses hydrogen combustion.~~ Section ~~43.6~~ 41.3.6 (Other Ignition Sources) addresses hot jets as a potential mechanism for initiating deflagrations. The possibility of deflagrations and the transition from a deflagration to a detonation are addressed and quantified in the hydrogen decomposition event trees.

Question: 480.132

The geometric class assigned to the IRWST and steam generator subcompartments appears to be non-conservative. It appears that a qualitative method was used to assign these geometric classes. The staff would like to have a better understanding of the process used to assign the geometric classifications for the subcompartments stated above.

Response:

The assignment of the geometric classes to the compartments is discussed in section 41.8 of the AP600 PRA report (revision 8) and the rational for each compartment's value is summarized in Table 41-11.

~~The process used to assign the geometric classifications for the IRWST and the steam generator subcompartments is described starting on page 4-15 on the section of WCAP-13388 that discusses the probability and consequences of deflagration and detonation of hydrogen. The process involved gaining an understanding of the geometry of these subcompartments in the AP600 containment. The next step in the process was to review the descriptions of the geometric classes for flame acceleration and make a qualitative comparison between the plant geometry and the class descriptions.~~

~~For example, the IRWST gas space is approximately 6700 cubic feet, has no obstructions, and is vented along the circumference of the containment with 100 openings (1 ft diameter). The vents represent transverse venting if a flame front were to accelerate within the IRWST across its water surface and beneath the ceiling of the subcompartment. Examples provided for geometric class 4 are large volumes with hardly any obstacles and large amounts of venting transverse to the flame path or small volumes without obstacles. Examples for geometric class 3 are large tubes without obstacles or small tubes (several inch diameter) with obstacles. The IRWST gas space geometry was judged to be most similar to the examples provided for geometric class 4. This is a qualitative assignment based on the available information.~~

~~A similar qualitative comparison and assessment was employed to determine that the geometric class that best described the steam generator compartments was class 3.~~

~~The reported geometric classes for the IRWST (class 4) and for the steam generator subcompartment (class 3) are combined with the mixture class (class 4) to determine the potential for a DDT. The sensitivity of the potential for DDT on the geometric class assignments can be determined by referring to Table C-3 of the section on Hydrogen Deflagration and Detonation in WCAP-13388. If the IRWST geometric class is considered to be 3 instead of 4 the result class changes from 5 to 4. In the case of the steam generator subcompartment a change in the geometric class from 3 to 2 would not cause the result class to change, i.e., it would remain at 4. A result class of 4 (see Table C-4) is classified as DDT is possible but unlikely. Thus, given the conservative assumptions (no steam inerting, total failure of the igniter system and 100% oxidation of the zirconium in the reactor core) made to establish the requisite initial conditions for a deflagration and the assessment of the result class for the IRWST and steam generator subcompartments it is concluded that essentially no potential for DDT exists for the AP600 configuration.~~

Question: 480.136 Definition and Quantification of PDSs

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~~ 8 ~~is due in June~~ was transmitted in September 1996. The containment event tree description (Chapter 35) and quantification (Chapter 43) revision will include a section which details the Level 1 / Level 2 interface including a description of the accident class definitions (Table 35-1) and dominant sequences and frequencies in each of the accident classes (Tables 43-5 through 43-13).

Question: 480.137 CET/Diagnostic Evaluation Team (DET) Structure

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~~ was transmitted to the NRC in ~~June~~ Sept. 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~~ does not display and quantifiy the ex-vessel severe accident phenomena that may occur should the vessel fail. The CET model ~~will~~ conservatively assumes 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. 480.137-1. 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 the sufficient ingress of water and the venting of steam from the cavity
- evaluation of the treatment of the lower head outside surface to 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~~ is 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 (Chapter 41). Ex-vessel core concrete interaction production of combustible gas ~~will be~~ is assumed to result in containment failure for the CET model (Chapter 35).

The other severe accident issues will be addressed through quantification of the reliability c' ystems 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 (Chapter 36).

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. PCS failure due to blockage of the air flow is quantified (Chapter 40).

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 (Chapter 35).

In addition to the above, severe accident phenomena related to ex-vessel conditions will be evaluated and analysis will be provided where appropriate (Appendix B). 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 phenomona 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.

Reference

480.137-1. Theofanous, T.G., et. al., "In-Vessel Coolability and Retention of a Core Melt," DOE/ID-10460, July 1995.

Question: 480.138 In-vessel Retention of Core Debris

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, rev. 8. The failure probability of the reactor vessel is based on the Advanced Reactor Severe Accident Program IVR 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~~ is 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 (Chapter 39). These analyses includes:

- 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 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.

Question: 480.145 In-vessel Retention of Core Debris

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~~ 8 of the PRA (see response to RAI 480.137).

Reference

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

Question: 480.146 In-vessel Retention of Core Debris

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~~ 8 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.

Question: 480.147 In-vessel Retention of Core Debris

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 feasability of IVR-friendly insulation. Additional reactor vessel insulation details, such as loading requirements, ~~will be~~ are presented in Chapter 39 of revision ~~7~~ 8 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 Coolaoility and Retention of a Core Melt," DOE/ID-10460, July 1995.

Question: 480.148 In-vessel Retention of Core Debris

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 ocillatory forces associated with the boiling (Chapter 39). 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.

Question: 480.150 Creep Rupture of Steam Generator Tubes

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 configuration from the core to the steam generators for times before and after uncovering the bottom of the core barrel.

Response:

In revision ~~7~~ 8 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 (Chapter 36).

a. The RCS nodalization schematic ~~will be provided for MAAP4 analyses defending the operator action timing for depressurization.~~ is available in MAAP4 User Manual, Vol. 2, Part 3 Descriptions of the PRYSYS and PSHS-P Subroutines.

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.~~

Question: 480.151 Creep Rupture of Steam Generator Tubes

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~~ 8 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 (Chapter 36). 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 sequnces which have relatively little natural circulation and heat removal from the core.

Question: 480.153 Creep Rupture of Steam Generator Tubes

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 strenght 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~~ 8, 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 (Chapter 36).

Reference

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.

Question: 480.154 Creep Rupture of Steam Generator Tubes

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~~ 8 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 (Chapter 36). 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.

Question: 480.155 Creep Rupture of Steam Generator Tubes

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~~ 8 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 (Chapter 36). 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.

Question: 480.156 Creep Rupture of Steam Generator Tubes

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~~ 8 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.

Question: 480.158 Creep Rupture of Steam Generator Tubes

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~~ 8 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 (Chapter 36). Timing to steam generator tube failure is conservatively estimated to bound uncertainties. This conservatism includes depressurization of the secondary system.

Question: 480.159 Creep Rupture of Steam Generator Tubes

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~~ 8 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 (Chapter 36).

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 sequnces which have relatively little natural circulation and heat removal from the core.

Question: 480.160 In-vessel Steam Explosion

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~~ 8 of the PRA. This analysis supersedes the in-vessel steam explosion DET which ~~will be~~ is eliminated from the PRA.

Question: 480.161 In-vessel Steam Explosion

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~~ 8 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 (Chapter 36). 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.

Question: 480.162 Ex-vessel Steam Explosion

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~~ 8 of the PRA.

Deterministic analysis of ex-vessel steam explosion will be provided in PRA revision ~~7~~ 9. ~~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.

Question: 480.165 Core Concrete Interactions

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~~ 8 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~~ 8 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~~ 9. ~~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.

Question: 480.166 Core Concrete Interactions

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~~ 8 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 know 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~~ 8 of the PRA.

Question: 480.168 Core Concrete Interactions

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~~ are no implications ~~of~~ from 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 the amount of water from the IRWST which is in the cavity and loop compartments.

There ~~is~~ are no implications ~~of~~ from these discrepancies on ex-vessel debris coolability. The DET for debris coolability has been eliminated from revision ~~7~~ 8 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~~ 8 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.

Question: 480.175 Hydrogen Combustion

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 are failed, 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~~.

Question: 480.177 Hydrogen Combustion

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 the 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~~ (rev. 8), this issue will be addressed through ~~sensitivity studies~~ hydrogen generation probability distributions and additional justification of in-vessel hydrogen generation.

Question: 480.178 Hydrogen Combustion

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~~ 8, 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 hydogen releases to the IRWST and the containment through the ADS system in these cases are used to address the impact on containment integrity from DDT and diffusion flames at the IRWST vents.

Question: 480.183 Treatment of Containment Failure Modes in the CET

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~~ 8.

Question: 480.184 Treatment of Containment Failure Modes in the CET

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~~ 8.

Question: 480.185 Treatment of Containment Failure Modes in the CET

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~~ best-estimate mass and energy water and steam releases ~~and for high temperature hydrogen releases equivalent to 100 percent zircaloy oxidation will be~~ is presented in Chapter 40 of revision ~~7~~ 8 of the PRA. The results show good agreement and little probability of containment failure (less than $5x10^{-4}$).

Additionally, a sensitivity analysis of the reliability of the PCS system is summarized in Chapter 50, and presented in Table 480.185-1. The results show that if PCS water failure (probability in range of 0.001 to 0.0001, including uncertainty) were assumed to fail the containment, the large release frequency would not increase significantly. Therefore, no further thermal-hydraulic analysis is required.

TABLE 480.185-1

PCS Failure Probability	Large Release Frequency (LRF)	Containment Effectiveness	LRF / CDF
0.0001 (base case)	1.82E-08	89.2 %	10.8 %
0.001	1.84E-08	89.1 %	10.9 %
0.01	1.97E-08	88.3 %	11.7 %
0.1	3.33E-08	80.3 %	19.7 %
1.0	1.69E-07	0.0	100 %

Question: 480.186 Treatment of Containment Failure Modes in the CET

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 sewage 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 dome. 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~~ 8 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~~ 8.

Question: 480.187 Treatment of Containment Failure Modes in the CET

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~~ 8 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.

Question: 480.189 CET/DET Quantification

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~~ 8 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 revision ~~7~~ 8.

Reference

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

Question: 480.197 Uncertainty in Containment Performance Estimates

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~~ is provided in revision ~~7~~ 8 of the PRA.

Question: 480.198 Uncertainty in Containment Performance Estimates

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~~ 8 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 (reference 480.198-1) and failure to flood the cavity leading to vessel falure 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~~ is 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.

Question: 480.200 Binning Process and Selection of Representative Sequences

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 souce 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~~ is better documented in PRA revision ~~7~~ 8 (Chapter 45).

Question: 480.201 Binning Process and Selection of Representative Sequences

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:

Even timing for the requested information ~~will be~~ is included in revision ~~7~~ 8 of the PRA (Chapter 34).

Question: 480.202 Binning Process and Selection of Representative Sequences

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~~ is included in all the DVI line break cases in the PRA ~~update~~ (rev. 8).

Question: 480.203 Determination of Fission Product Releases

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~~ 8 ~~will~~ reflect the results of that investigation.

Question: 480.206 Uncertainty in Source Term Estimates

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 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~~ is provided in the PRA ~~update~~ (rev. 8).

Question: 480.207 Uncertainty in Source Term Estimates

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.~~

There is no significant difference between MAAP4 and NUREG-1150 source terms. No assessment on risk impact is required.

Question 480.439

Westinghouse responses to RAIs 720.54 and 720.55 (May 1993) indicated that numerous accident management strategies or related EOP changes would be adopted for AP600, and that additional accident management strategies would be evaluated and integrated into the AP600 accident management plan if found to be effective. WCAP-13913 "Framework for AP600 Severe Accident Management Guidance" (Dec 1993) was subsequently submitted, but does not provide a complete or current accounting of the critical PRA insights and accident management strategies that would need to be further evaluated by a COL applicant as part of their development of an accident management program, many of which have been developed or refined subsequent to issuance of the topical report. Examples of the insights or strategies that the COL applicant would need to address as part of their plant-specific implementation of accident management include:

- initiation of reactor vessel cavity flooding
- use of fan coolers for fission product removal
- use of igniters to control hydrogen
- reclosing of the ADS valves to control hydrogen diffusion flames and fission products
- makeup to the containment for long term cooling
- makeup to the passive containment cooling system (PCS)
- strategies for reflooding a damaged core which is retained in-vessel
- use of portable battery chargers to backup batteries
- identification and use of additional supplies of borated water
- strategies to enhance or restore flow through the PCS annulus
- use of a firewater pump for injection into the steam generators
- use of existing penetrations to vent containment

Furthermore, the response to RAI 720.56 (May 1993) indicates that the completion of the development of the severe accident management guidance for AP600 is part of the man-machine interface specification. However, neither this specification nor a COL action item describing the necessary actions on the part of the COL applicant have been submitted to our knowledge. (The March 1996 response to RAI 480.212 indicates that the COL applicant will develop plant-specific severe accident guidance based on WCAP-13913, but WCAP-13913 is incomplete as discussed above, and a clear commitment or COL action item has not been provided to assure that this will be done).

Please provide the following additional information to assure that all severe accident insights/strategies to be addressed by the COL applicant are identified and that a process and commitment for performing the necessary plant-specific actions is established:

a) A complete accounting (e.g., annotated list) of severe accident insights/strategies that the COL applicant will be responsible for addressing as part of their plant-specific implementation of accident management,

b) A description of the scope and objectives of each strategy, including whether the strategy is to be incorporated into the Emergency Operating Procedures (EOPs) or the Severe Accident Management Guidance (SAMG), and where in these documents this information is or will be located, and

c) A description of the process by which the insights/strategies to be addressed by the COL applicant will be communicated to the COL applicant, and a corresponding COL action item addressing this commitment

Response:

The overall severe accident management philosophy and high level strategies applicable to AP600 are described in WCAP-13914, Revision 1, "Framework for AP600 Severe Accident Management Guidance," November 1996. The overall philosophy and high level strategies described in the previous version of WCAP-13913 and WCAP-13914 has been reviewed following the completion of the AP600 PRA. The severe accident management insights identified from the AP600 PRA have been incorporated into WCAP-13914, revision 1. Thus, WCAP-13914, revision 1, is a valid basis upon which a COL applicant can develop Severe Accident Management Guidance.

As discussed in WCAP-13914, revision 1, the AP600 Severe Accident Management Guidance should be similar in content and structure to the generic Westinghouse Owners Group Severe Accident Management Guidance (WOG SAMG) that forms the basis for Severe Accident Management Guidance at existing plants. The COL applicant should use the generic WOG SAMG and the information in WCAP-13914, including the PRA insights described in Appendix A of that WCAP, to develop the AP600 Severe Accident Management Guidance. This process will address the example insights and/or strategies delineated in this RAI. The evaluation of the applicable accident management strategies by the COL applicant will include a determination of the appropriate guidance set (e.g., Emergency Response Guidelines versus Severe Accident Management Guidance) where the strategy will reside.

Chapter 19 of the AP600 SSAR will include a COL item that commits the Combined License applicants to developing a severe accident management program.

Question: 480.212

Identify and discuss actions that would be required to prevent or mitigate uncontrolled fission product releases after 24 hours due to (a) long term non-condensible gas generation, (b) depletion of coolant inventory due to normal leakage and early bypass sequences, (c) late containment bypass (temperature-induced SGTR), and (d) depletion of PCSS water inventory.

Response:

As discussed in the response to RAI 720.55 and RAI 720.56, Westinghouse has developed a framework and a set of high level strategies for severe accident management. This work is documented in "Framework for AP600 Severe Accident Management Guidance", WCAP-~~13913, December 1993~~ 13914, revision 1, November 1996. High level strategies to diagnose potential fission product release pathways and then to prevent, terminate and/or mitigate those fission product releases are identified and discussed in WCAP-~~13913~~ 13914. The high level strategies presented in WCAP-~~13913~~ 13914 are applicable to all of the items outlined in this question.

Westinghouse believes that the development of the framework for a severe accident management program for the AP600 plant design, including the identification of high level strategies provides a sufficient basis for the development of the detailed AP600 Severe Accident Management Guidance by the COL applicant.