MEMORANDUM TO: M. R. Snodderly, Cognizant ACRS Staff Engineer

FROM: G. E. Apostolakis, Chairman
Reliability and Probabilistic Risk Assessment Subcommittee

SUBJECT: CERTIFICATION OF THE SUMMARY/MINUTES OF THE MEETING OF THE ACRS SUBCOMMITTEE ON RELIABILITY & PROBABILISTIC RISK ASSESSMENT, JANUARY 23-24, 2003, ROCKVILLE, MARYLAND

I do hereby certify that, to the best of my knowledge and belief, the minutes of the subject meeting on January 23-24, 2003, are an accurate record of the proceedings for that meeting.

George E. Apostolakis,
Subcommittee Chairman

4/10/03
Date

PRE-DECISIONAL

April 4, 2003

MEMORANDUM TO: G. E. Apostolakis, Chairman
Reliability and Probabilistic Risk Assessment Subcommittee

FROM: M. R. Snodderly, Cognizant ACRS Staff Engineer

SUBJECT: WORKING COPY OF THE MINUTES OF THE ACRS SUBCOMMITTEE ON RELIABILITY & PROBABILISTIC RISK ASSESSMENT, JANUARY 23-24, 2003 - ROCKVILLE, MARYLAND

A working copy of the minutes for the subject meeting is attached for your review. Please review and comment on them at your soonest convenience. If you are satisfied with these minutes please sign, date, and return the attached certification letter in the pre-addressed envelope attached.

Attachment: Minutes (DRAFT)

cc: S. Bahadur
S. Duraiswamy
J. Larkins
H. Larson

Date:04/4/03

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
MEETING OF THE ACRS SUBCOMMITTEE ON RELIABILITY & PROBABILISTIC RISK ASSESSMENT
MEETING MINUTES - JANUARY 23-24, 2003
ROCKVILLE, MARYLAND

INTRODUCTION

The ACRS Subcommittee on Reliability & Probabilistic Risk Assessment met on January 23-24, 2003, at 11545 Rockville Pike, Rockville, MD, in Room T-2B3. The purpose of this meeting was to review the probabilistic risk assessment (PRA) for the AP1000 passive plant design and identify possible issues that may need to be addressed prior to or during a July 2003 Full Committee meeting on the staff's draft safety evaluation report (DSER).

The Subcommittee received no written comments or requests for time to make oral statements from members of the public regarding the meeting. The entire meeting was open to public attendance. Michael Snodderly was the cognizant ACRS staff engineer for this meeting. Med El-Zeftawy was the designated federal official. The meeting was convened at 8:32 a.m. on January 23, 2003 and adjourned at 2:20 p.m on January 24, 2003.

ATTENDEES

ACRS Members

G. Apostolakis, Subcommittee Chairman
M. Bonaca, Member
P. Ford, Member
T. Kress, Member
G. Leitch, Member
V. Ransom, Member
S. Rosen, Member
W. Shack, Member
J. Sieber, Member
M. El-Zeftawy, Staff (Designated Federal Official)
M. Snodderly, Staff

Principal NRC Speakers

L. Burkhardt, NRR
W. Jensen, NRR
R. Lee, RES
R. Palla, NRR
M. Pohida, NRR
N. Saltos, NRR

Principal Industry Speakers:

M. Corletti, Westinghouse
E. Cummins, Westinghouse
S. Sancaktar, Westinghouse
T. Schultz, Westinghouse
J. Scobel, Westinghouse

There were approximately four members of the public in attendance at this meeting. A complete list of attendees is in the ACRS Office File and will be made available upon request.

The presentation slides and handouts used during the meeting are attached to the office copy of these minutes.

OPENING REMARKS BY THE SUBCOMMITTEE CHAIRMAN

George Apostolakis, Chairman of the ACRS Subcommittee on Reliability & Probabilistic Risk Assessment convened the meeting at 8:32 a.m. Chairman Apostolakis stated that the purpose of this meeting was to review the PRA provided by Westinghouse Electric Company in support of its application to the NRC for certification of its AP1000 design. Chairman Apostolakis reminded the Committee that it had reviewed certification of the AP600 design, which included a design-specific PRA. He had no specific comments relative to the meeting agenda.

DISCUSSION OF AGENDA ITEMS

Westinghouse Presentations

Michael Corletti, Terry Schulz, Selim Sancaktar, Terry Schulz, and Jim Scobel were the main presenters for Westinghouse. Terry Schulz began the presentation with an overview of the AP1000 passive plant design. Mr. Schulz emphasized differences between the AP600 and AP1000 and how the PRA was used as a design tool for the AP1000. Mr. Sancaktar then gave an hour and 45 minute presentation on the Level 1 portion of the AP1000 PRA. Mr. Sancaktar's presentation addressed internal events at power, shutdown, fire portions of the Level 1 PRA, including uncertainty. Mr. Schulz then gave an hour and 25 minute presentation on the Level 1 success criteria, including Westinghouse's thermal-hydraulic uncertainty assessment. On January Twenty Fourth, Mr. Scobel gave a presentation on the Level 2 and 3 portion of the PRA and additional testing performed to support Westinghouse's in-vessel retention approach. Significant points raised during the presentations include:

- Mr. Schulz provided a description of the major components of the AP1000 passive plant design. Mr. Schulz pointed out that the canned motor reactor coolant pumps do not have shaft seals or lubricating oil. This design feature eliminates the seal LOCA failure mechanism and the likelihood of a fire in the PRA.

- Mr. Schulz then laid out the AP1000 approach to safety. The designers' approach was to use "passive" processes only, one time alignment of valves and no support systems after initial actuation. This resulted in a reduced dependency on operator actions. Non-safety systems are not required to mitigate design basis accidents or meet the NRC safety goals. Active non-safety systems are primarily in the design to support normal operation or anticipated transients. These systems are typically powered by non-safety diesels. These systems also minimize challenges to the passive systems.

- Mr. Schulz then described the passive core cooling system, passive decay heat removal, passive safety injection, LOCA long term cooling, the passive containment cooling system, normal residual heat removal system, I & C systems, and the control room.

- Mr. Schulz provided examples of how the PRA had been used as a design tool for the AP600 and the AP1000.

- Mr. Sancaktar provided an overview of the AP1000 PRA. He provided the following: the 26 initiating events that contribute the most to CDF, the 11 most dominant CDF sequences, sensitivity results, system importances for internal events.

- Mr. Sancaktar explained that Spurious actuation of the fourth stage automatic depressurization system was separated from large break LOCA sequences because only one accumulator instead of two was needed to mitigate the former.

- Mr. Sancaktar provided the following concerning the quantitative shutdown risk evaluation performed for internal events: the CDF is 1.23E-07 events per year, the four most dominant accident sequences. This is an 18 percent increase compared to the AP600. Mr. Sancaktar said that the increase was due to more frequent outages. The AP1000 has an 18 month refueling cycle as opposed to a 24 month refueling cycle for the AP600.

- Mr. Sancaktar provided the following concerning internal flooding and fire: the CDF for internal flooding is 8.8E-10 events per years, the dominant contributor is large pipe breaks in the turbine building with an initiating event frequency in the range of 1.4-2.0E-03, the CDF for internal fire events is 5.61E-08 per year.

- Mr. Sancaktar acknowledged that there may be a need for an ITAAC requiring plant specific walkdowns of the as-built plant to verify the internal flooding and fire PRAs.

- Mr. Schulz provided an overview of the PRA Level 1 Success Criteria. He provided results for the following four groups of analysis: (1) Automatic ADS with CMT and IRWST gravity injection, (2) Automatic ADS with CMT and RNS pumped injection, (3) Manual ADS with Accumulator and IRWST gravity injection, and (4) Manual ADS with Accumulator and RNS pumped injection.

- Mr. Schulz described Westinghouse's philosophy for addressing T/H uncertainty. Westinghouse used MAAP4 to identify low margin sequences and the PRA to identify high risk sequences. They then reanalyzed these sequences with more conservative design basis codes to see if the core remained covered.

- Mr. Scobel than gave an overview of the Level 2 and 3 PRA. He explained the containment event tree structure as well as important simplifying assumptions. For example, high pressure RCS at core damage results in induced SGTR containment bypass and vessel failure and debris relocation into the containment results in early containment failure. Mr. Scobel gave Level 2 quantification results and the 11 most dominant LRF sequences. LRF is estimated to be 1.95E-08 per reactor year. Overall containment effectiveness is 92% and ATWS sequences had the lowest effectiveness. The containment effectiveness for a particular SGTR sequence was 57%.

- Mr. Scobel than discussed in-vessel retention via external cooling of the reactor vessel. Mr. Scobel described design features that had been incorporated to promote in-vessel retention. These included: (1) post-accident RCS depressurization system, (2) smooth lower head, (3) ability to submerge vessel post-accident, and (4) improved reactor vessel insulation design. Mr. Scobel described testing and gave results from the ULPU test facility.

- Mr. Scobel discussed Westinghouse's assessment of severe accident phenomena including: in-vessel fuel-coolant interaction, high pressure core damage, hydrogen generation, dry containment cooling, ex-vessel steam explosion, core-concrete interaction, and equipment survivability.

<u>NRC Staff Presentations</u>

Nicholas Saltos, Walt Jensen, Marie Pohida, Robert Palla, and Richard Lee were the main presenters for the staff. Nicholas Saltos provided the first presentation on the status of the staff's review of the at-power Level 1 PRA for internal and external events. Walt Jensen gave the status of the staff's review of the thermal-hydraulic success criteria. Marie Pohida gave the status of the staff's review of the shutdown PRA. The three presentations occurred between 3:25 and 5 p.m. on January Twenty Third. On January Twenty Fourth, Mr. Palla gave the status of the staff's review of severe accidents and Levels 2 and 3 of the PRA. Mr. Lee than provided an overview of confirmatory work being performed by RES in support of the AP1000 review. These presentations took place from 1:15 to 2 p.m. Significant points raised during the presentations include:

- Mr. Saltos stated that the major objectives of the AP1000 PRA review were to identify design and/or operational changes to address weaknesses, determine appropriate regulatory treatment of non-safety systems, and determine the risk significance of raised issues. Mr. Saltos said that the staff planned to rely on the similarity of the AP1000 to AP600 to reduce review effort and the staff's review would focus on design differences having an impact on PRA models.

- Mr. Saltos identified the major issues as thermal/hydraulic uncertainty success criteria, fire-induced spurious actuation of ADS squib valves, and identification of "certification requirements such as inspection, testing, and analysis acceptance criteria and regulatory treatment of non-safety systems.

- Mr. Jensen discussed the staff's review of thermal/hydraulic uncertainty success criteria. As summarized earlier, Westinghouse plans to address thermal/hydraulic uncertainty by reanalyzing certain low margin/high risk sequences with more conservative design basis codes to see if the core remains covered. Mr. Jensen explained that the staff has performed audit calculations using RELAP5 of Westinghouse's calculations using NOTRUMP. Mr. Jensen than presented RELAP5 results overlapped against NOTRUMP and MAAP4 results.

- Ms. Pohida discussed issues associated with the shutdown PRA. She mentioned common cause failure of the high pressure gravity injection squib valves, high pressure recirculation squib valves, and the low pressure recirculation squib valves. She pointed out the increase in power has led to shorter response times for operator recovery actions such as containment closure and manual gravity injection.

- Mr. Palla stated that the staff intends to address all the major severe accident phenomena. Westinghouse is relying heavily on AP600 analyses for addressing fuel-coolant interactions based on similar debris mass, superheat and composition. He discussed concerns involving external reactor vessel cooling which included the following: reduced margins to CHF and impact of uncertainties, implications of recent experimental work on in-vessel melt retention, increased dead-load on a thinned reactor pressure vessel, and the design of the thermal insulation.

- Mr. Lee discussed confirmatory calculations being performed by the Office of Research using the MELCOR. Mr. Lee explained that the analysis of lower head integrity and in-vessel retention will not be based on MELCOR calculations. Instead, a more detailed approach will be utilized that will consider a wide range of uncertainties (e.g., melt composition and configurations).

- Dr. Ali Behbahani gave a summary of the RASPLAV and MASCA testing. Dr. Behbahani explained that in the RASPLAV experiment if carbon was added to the mass than two stratified layers of oxidic melt occurred. The upper layer having more metal. In MASCA, it was zirconium containing corium. Separation occurred when iron was added. Thereby, you have heavy metal relocated to the lower part of the mass next to the vessel wall.

SUBCOMMITTEE COMMENTS, CONCERNS, AND RECOMMENDATIONS

Subcommittee members raised the following significant points during its discussion with the Westinghouse representatives.

- Member Rosen inquired about the design and reliability of the Stage 4 Automatic Depressurization System valves. The Subcommittee identified this as a critical item. Mr. Schulz provided a drawing of the 14" squib valves. Westinghouse has not tested these valves and is relying on data for much smaller valves provided to BWRs by the vendor. The assumed reliability is based on a purchase specification. The manufacturer will certify the reliability of the propellant and the igniter.

- Westinghouse claims that the uncertainty associated with the AP1000 CDF is a factor of six. The Members questioned how this could be less than operating plants which have been built and have a considerable operating history. Operating plants typically have a factor of ten uncertainty in CDF. Member Apostolakis concluded that Westinghouse considered parameter uncertainty but not model uncertainty.

- Member Apostolakis asked how can you estimate common cause failure for a plant that has not been built. Mr. Sancaktar responded that they had no choice but to do it generically. Member Apostolakis contemplated that perhaps we needed an ITAAC or design acceptance criteria for not only common cause failure analysis but human error analysis too. Member Apostolakis asked to make a note of this issue for consideration by the Full Committee when they review the DSER. Member Apostolakis remarked what good is a major research effort, such as ATHEANA, if we never intend to use it?

- Member Shack observed that the large break LOCA initiating event frequency was 20 times lower than for the AP600. Mr. Sancaktar said that it was due to two factors. First, they separated out spurious ignition of ADS 4 because only one accumulator is needed to mitigate the event. The remaining Large break LOCAs require both accumulators for successful mitigation. Second, the large break LOCA initiating event frequency was assumed to be 5.04E-06 based on NUREG/CR-5750. Member Apostolakis pointed out that if you used the AP600 large break LOCA initiating event frequency than the AP1000 CDF would go up an order of magnitude.

- Members Apostolakis and Shack asked why the plant monitoring system is so important if you have a diverse actuation system. Mr. Sancaktar explained that the two systems are independent. Mr. Sancaktar elaborated that PMS importance is related to the inverse of the failure probability. It so important because it is high reliability.

- Member Kress asked if either the AP600 or the AP1000 PRA had been subjected to the industry peer review process. Mr. Corletti believed that the AP600 PRA had been subjected to a peer review process. The AP1000 was not but the same model was followed.

- Member Shack asked if the larger steam generators could be susceptible to the Palo Verde dry out problem. Mr. Schulz responded that the AP1000 design team has taken advantage of steam generator expertise since the merger of Westinghouse and Combustion Engineering. Mr. Schulz also mentioned that Combustion Engineering has built bigger steam generators then those proposed for the AP1000.

- Member Kress asked if the non-safety related systems would handle the design basis accidents. Mr. Schulz responded that had been considered to a large extent in the PRA. Mr. Schulz discussed taking credit for start-up feed water to mitigate a loss of feed water or the residual heat removal system to provide low pressure injection. Member Kress pointed out that if the non-safety related systems could also mitigate the design basis accidents then this would help answer questions about passive system reliability. Mr. Schulz said that large break LOCAs can not be mitigated without both passive accumulators. Mr. Cummins than stated that the active systems would mitigate most LOCAs but would require manual action.

- Member Leitch asked about the reliability of the inlet MOV to the PRHR heat exchanger. Member Leitch was concerned about the pressure and temperature differences across the valve. Mr. Schulz explained that if the failed leaked or failed it would be contained in the in-containment refueling water storage tank.

- Member Kress asked if they considered importance measures for active non-safety systems. Mr. Schulz responded that they did not use the importance measures directly. They recalculated the core melt frequency and large release frequencies without crediting the non-safety systems. If they could still meet the NRC safety goals without these systems then they were classified as not safety important. Mr. Schulz said that some DAS manual controls were classified as safety important and, therefore, technical specifications were placed on the DAS manual controls.

- Member Sieber asked how do you model the I&C system if you haven't selected a design. Mr. Schulz responded that the Sequoyah protection system was assumed.

- Member Kress observed that number of control room operators needed was less than operating plants and asked for the basis of the decision. Mr. Cummins responded that the decision was based on the EPRI Requirements Document that gave a goal of one operator and one supervisor. Mr. Cummins went on to say that the utility requirements document states that the control room be capable of holding at least three people.

- Member Rosen pointed out that the risk from shutdown operations is about a third. He went on to say that in his experience plants that perform mid-loop evolutions during shutdown contribute about half the risk. Member Rosen asked why AP1000 was less. Mr. Corletti responded that many passive systems are available during shutdown that contribute to a lower CDF during shutdown.

- Member Shack asked if the emergency operating procedures provided for manual actuation of the automatic depressurization system. Mr. Schulz said they did and those procedures are used to determine response times credited in the PRA.

- Member Rosen observed that the AP1000 containment is not only important as a fission product barrier but also is important for providing backpressure to aid ECCS performance.

- Member Kress asked the staff how did they measure PRA quality and determine that the quality is commensurate with its intended use. Mr. Saltos said by evaluating the models, assumptions, and data. Member Kress than asked about the quality in terms of the ASME Standard and whether it was a Category 1, 2, or 3. Mr. Saltos said that the PRA was performed prior to the standard but that the PRA was compatible with the standard.

- Member Leitch asked the staff if they were reviewing the possibility that hot shorts could cause the actuation of all four Stage Four ADS valves. Mr. Saltos said they were reviewing that issue. Mr. Cummins indicated that it may be possible for one hot short to actuate one pair of Stage Four but not all four.

- Member Apostolakis asked the staff if they looked for possible errors of commission. Mr. Saltos said that they did not look for any additional errors and that he was not aware of any new information that would change the results. Member Apostolakis reminded him of the ATHEANA work performed by RES. Mr. Saltos said that some errors of commission were considered for the AP600. Member Apostolakis asked if Mr. Saltos would agree to do it for the AP1000. Mr. Corletti stated that for AP600, the ACRS raised the issue of adverse system interactions. Westinghouse prepared a topical report in response which included a qualitative assessment of the effects of human errors of commission. The staff issued an RAI asking to repeat the systematic assessment for AP1000 which has been just submitted. Mr. Saltos said he would look at it.

- Member Kress observed that the MAAP4 results appear to be more conservative because reactor pressure is higher which means you are getting less injection. He asked if the same critical flow model is used. Mr. Scobel responded that MAAP4 use the Fauske critical flow model.

- Member Ransom commented that there appeared to be a lot of subjectivity in the selection of the low margin sequences and that a statistical sample should have been used.

- Member Apostolakis said that he would be interested in knowing what effect setting the human error probabilities to one would have on shutdown CDF and LERF. Neither the staff nor Westinghouse committed to providing such an analysis. Ms. Pohida stated that no release frequencies for shutdown were reported for AP1000 because containment closure is maintained via technical specification until the time to boiling from decay heat is greater than the time to restore containment integrity. Member Apostolakis clarified that there is some probability that the operators will restore containment integrity in that time. Mr. Cummins said that was beyond what Westinghouse normally does in the PRA.

- Member Leitch asked if there is an operator action to vent the containment in a severe accident situation. Mr. Scobel responded that there is an action in the severe accident management guidelines but it is not credited in the PRA.

- Member Kress asked what the basis was for the assumption that a high pressure core melt accident leads to induced steam generator tube failure. Mr. Scobel explained that the AP1000 does not have a loop seal because of the canned reactor coolant pumps. This results in more uniform primary system heat up due to full loop natural circulation. Member Kress commented that he thought it was a good assumption because it was conservative.

- Member Rosen asked if the exterior surface of the reactor vessel was important for in-vessel retention. Mr. Scobel replied that the optimum surface, which resulted in the highest critical heat flux, was unpainted and oxidized.

- Member Kress asked how in-vessel melt progression was modeled. Mr. Scobel answered that he looked at a spurious ADS Stage 4 case because you can't reflood the vessel and it progresses very rapidly. He explained that Westinghouse put together a finite difference model of the core and internals which used the uncovery timing from MAAP 4. Westinghouse concluded that the downward progression is blocked and that resulted in a sideways failure similar to TMI. Member Kress than asked Westinghouse to clarify the modeling of a variable oxide crust because of a change in heat flux. Mr. Scobel explained you have an isothermal boundary so the crust adjusts its thickness and you get the heat fluxes from the natural circulation. From those heat fluxes you calculate a crust thickness and that fixes the metal temperature.

- Member Ransom surmised that what is important for in-vessel retention is that your melt progression model is considered to be conservative. He suggested a worst case type situation where you get the highest heat transfer and assume natural circulation exists in the metallic and oxide layers. Mr. Scobel agreed and said that a conservative model maximizes the fission products, which generate decay heat, in the bottom layer. Mr. Scobel went on to say that for the AP1000 it was assumed that the bottom metal layer has 40 weight percent of Uranium and that 100 percent of the decay heat from the fission products that come from an equivalent volume of the oxide needed to create that amount of Uranium. The initial masses of the metal involved in the reaction is 3,000 kilograms of stainless steel and 7,000 kilograms of Zirconium.

- Member Shack asked about passive containment cooling performance if water is unavailable. Mr. Scobel responded that if you assume ANS decay heat plus 2 sigma and an outside temperature of 115 degrees, Westinghouse estimated a failure probability of two percent at 24 hours for an accident sequence with a dry containment.

STAFF AND INDUSTRY COMMITMENTS

Westinghouse committed to providing a more complete basis for the assumed reliability of the Stage 4 ADS valves at an upcoming meeting of the Future Plant Designs Subcommittee.

The staff committed to looking at Westinghouse's Adverse Systems Interaction report to see if it identified any new errors of commission that should be modeled in the PRA.

SUBCOMMITTEE DECISIONS AND ACTIONS

Since this was an information only briefing, no letter was recommended by Chairman Apostolakis. The Subcommittee Chairman asked each of the members for their impression of the presentations. Based on the members responses, Chairman Apostolakis concluded that there was a consensus that the staff had identified the proper issues and that upon resolution the PRA appears sufficient to support design certification of the AP1000. The Subcommittee will make a recommendation to support the Full Committee's review of the staff's Draft Safety Evaluation Report in the third quarter of 2003.

BACKGROUND MATERIALS PROVIDED TO THE SUBCOMMITTEE PRIOR TO THIS MEETING

1. Subcommittee status report, including agenda.
2. Compact Disc containing the AP1000 PRA, the AP600 PRA, requests for additional information that the staff has issued on the AP1000 PRA, and the staff's final safety evaluation report on the AP600 PRA.
3. Letter dated July 23, 1998, from R. L. Seale, Chairman, ACRS, to Shirley Ann Jackson, Chairman, NRC, Subject: Report on the Safety Aspects of the Westinghouse Electric Company Application for Certification of the AP600 Passive Plant Design.
4. Presentation Material dated November 7, 2002, from Westinghouse Electric Company, to ACRS, Subject: AP1000 Design Certification Review.
5. Internal Report dated January 2003, from Hossein P. Nourbaksh, Senior Fellow, ACRS, Subject: Review of the AP1000 PRA Internal Events At-Power.

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Note: Additional details of this meeting can be obtained from a transcript of this meeting available for downloading or viewing on the Internet at "http://www.nrc.gov/ACRSACNW' or can be purchased from Neal R. Gross and Co., Inc., (Court Reporters and Transcribers) 1323 Rhode Island Avenue, NW., Washington, DC 20005 (202) 234-4433.

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
MEETING OF THE SUBCOMMITTEE ON
RELIABILITY AND PROBABILISTIC RISK ASSESSMENT
ROOM T-2B3, 11545 ROCKVILLE PIKE, ROCKVILLE, MD
January 23 and 24, 2003

ACRS Contact: Michael R. Snodderly (301) 415-6927
E-mail: mrs1@nrc.gov

- PROPOSED SCHEDULE -

Thursday, January 23, 2003

	TOPIC	PRESENTER	TIME
1)	**Introduction** ▸ Review goals and objectives for this meeting	George Apostolakis, ACRS Mike Corletti, Westinghouse	8:30-8:35 am
2)	**Overview of AP1000 Design** ▸ Design Changes from AP600 ▸ Key AP1000 Key Design Features ▸ Defense in Depth ▸ PRA as a Design Tool	Terry Schulz, W	8:35-10:05 am
	**** BREAK ****		10:05 - 10:20 am
3)	**AP1000 PRA** ▸ Background / Approach / Overview ▸ Scope ▸ Level 1 PRA Internal Events At-Power, Including Uncertainty ▸ Shutdown / Fire PRA	Selim Sancaktar, W	10:20 - 12:20 pm
	**** LUNCH****		12:20 - 1:30 pm
4)	**PRA Level 1 Success Criteria** ▸ Overview ▸ Thermal-Hydraulic Analysis to Support Level 1 PRA ▸ T&H Uncertainty Assessment	Terry Schulz, W	1:30 - 3:30 pm
	**** BREAK****		3:30 - 3:45 pm
2)	**NRC Staff Presentation** ▸ Staff RAIs on Level 1 PRA and Success Criteria	Nick Saltos, NRR Walt Jensen,NRR Marie Pohida, NRR	3:45 - 5:30 pm

6)	**Westinghouse Summary**	Mike Corletti, W	5:30 - 5:45 pm

Friday, January 24, 2003

	TOPIC	PRESENTER	TIME
1)	**Introduction** ▸ Review goals and objectives for this meeting	George Apostolakis, ACRS Mike Corletti, Westinghouse	8:30-8:35 am
2)	**Level 2 and 3 PRA** ▸ Quantification ▸ Level 2 Phenomenological Studies	Jim Scobel, W	8:35-10:15 am
	****BREAK****		10:15-10:30 am
3)	**ULPU Testing Performed for AP1000** ▸ AP600 Background ▸ Test Program ▸ RV Insulation Design	Jim Scobel, W	10:30-11:30 am
4)	**PRA Importance and Sensitivity Studies**	Selim Sancaktar, W	11:30-12:15 pm
	****LUNCH****		12:15 -1:15 pm
5)	**NRC Staff Presentation** ▸ Staff RAIs on Level 2 & 3 PRA	Bob Palla, NRR Richard Lee, RES	1:15 - 2:15 pm
6)	**Westinghouse Summary**	Mike Corletti, W	2:15 -2:30 pm
7)	**General Discussion and Adjournment** ▸ General discussion and comments by Members of the Subcommittee	George Apostolakis, ACRS	2:30-3:00 pm

Note:

Presentation time should not exceed 50% of the total time allocated for a specific item.

Number of copies of presentation materials to be provided to the ACRS - 35.

December 19, 2002

MEMORANDUM TO: Sher Bahadur, Associate Director
for Technical Support, ACRS/ACNW

FROM: Michael R. Snodderly, Senior Staff Engineer

SUBJECT: FEDERAL REGISTER NOTICE REGARDING THE MEETING OF THE ACRS SUBCOMMITTEE ON RELIABILITY AND PROBABILISTIC RISK ASSESSMENT, JANUARY 23-24, 2003, ROCKVILLE, MARYLAND

Attached is a Federal Register Notice regarding the subject meeting. Please have this Notice transmitted for publication as soon as possible.

Attachment:
FR Notice

cc with Attachment:
G. Apostolakis, ACRS
J. Larkins, ACRS
I. Schoenfeld, OEDO
J. Szabo, OGC
A. Bates, SECY
R. Jasinski, OPA
S. Collins, NRR
G. Holahan, NRR
A. Thadani, RES
J. Lyons, NRR
S. Newberry, RES
M. Cunningham, RES
PMNS
Public Document Room

[7590-01-P]

NUCLEAR REGULATORY COMMISSION

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
MEETING OF THE ACRS SUBCOMMITTEE ON RELIABILITY
AND PROBABILISTIC RISK ASSESSMENT

Notice of Meeting

The ACRS Subcommittee on Reliability and Probabilistic Risk Assessment will hold a meeting on January 23-24, 2003, Room T-2B3, 11545 Rockville Pike, Rockville, Maryland.

The entire meeting will be open to public attendance.

The agenda for the subject meeting shall be as follows:

Thursday, January 23, 2003 - 8:30 a.m. until the conclusion of business

The Subcommittee will meet with representatives of the Westinghouse Electric Company and members of the NRC staff to review the Probabilistic Risk Assessment for the AP1000 passive plant design.

Friday, January 24, 2003 - 8:30 a.m. until the conclusion of business

The Subcommittee will continue its discussion of the AP1000 Probabilistic Risk Assessment, including fire, low power and shutdown, and external event risk assessments.

The purpose of this meeting is to gather information, analyze relevant issues and facts, and formulate proposed positions and actions, as appropriate, for deliberation by the full Committee.

Oral statements may be presented by members of the public with the concurrence of the Subcommittee Chairman; written statements will be accepted and made available to the Committee. Persons desiring to make oral statements should notify one of the staff engineers named below five days prior to the meeting, if possible, so that appropriate arrangements can be made. Electronic recordings will be permitted only during those portions of the meeting that are open to the public.

During the initial portion of the meeting, the Subcommittee, along with any of its consultants who may be present, may exchange preliminary views regarding matters to be considered during the balance of the meeting.

The Subcommittee will then hear presentations by and hold discussions with representatives of the NRC staff, Westinghouse Electric Company, and other interested persons regarding these matters.

Further information regarding topics to be discussed, whether the meeting has been canceled or rescheduled, and the Chairman's ruling on requests for the opportunity to present oral statements and the time allotted therefor can be obtained by contacting the Designated Federal Official, Dr. Medhat M. El-Zeftawy (telephone: 301-415-6889) or Mr. Michael R. Snodderly, Cognizant Staff Engineer, (telephone: 301-415-6927) between 7:30 a.m. and 4:15 p.m. (EST). Persons planning to attend this meeting are urged to contact one of the above named individuals at least two working days prior to the meeting to be advised of any potential changes to the agenda.

Date____________

____
Sher Bahadur, Associate Director
for Technical Support, ACRS/ACNW

ACRS
MRS/bjw
12/ /02

ACRS
SD
12/ /02

FILED: GT-270

Mike S.

between 7:30 a.m. and 4:15 p.m. (EST). Persons planning to attend this meeting are urged to contact the above named individual at least two working days prior to the meeting to be advised of any potential changes in the proposed agenda.

Dated: December 19, 2002.

Sher Bahadur,

Associate Director for Technical Support, ACRS/ACNW.

[FR Doc. 02–32694 Filed 12–26–02; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Advisory Committee on Reactor Safeguards Meeting of the ACRS Subcommittee on Reliability and Probabilistic Risk Assessment; Notice of Meeting

The ACRS Subcommittee on Reliability and Probabilistic Risk Assessment will hold a meeting on January 23–24, 2003, Room T–2B3, 11545 Rockville Pike, Rockville, Maryland.

The entire meeting will be open to public attendance.

The agenda for the subject meeting shall be as follows:

Thursday, January 23, 2003—8:30 a.m. Until the Conclusion of Business

The Subcommittee will meet with representatives of the Westinghouse Electric Company and members of the NRC staff to review the Probabilistic Risk Assessment for the AP1000 passive plant design.

Friday, January 24, 2003—8:30 a.m. Until the Conclusion of Business

The Subcommittee will continue its discussion of the AP1000 Probabilistic Risk Assessment, including fire, low power and shutdown, and external event risk assessments.

The purpose of this meeting is to gather information, analyze relevant issues and facts, and formulate proposed positions and actions, as appropriate, for deliberation by the full Committee.

Oral statements may be presented by members of the public with the concurrence of the Subcommittee Chairman; written statements will be accepted and made available to the Committee. Persons desiring to make oral statements should notify one of the staff engineers named below five days prior to the meeting, if possible, so that appropriate arrangements can be made. Electronic recordings will be permitted only during those portions of the meeting that are open to the public.

During the initial portion of the meeting, the Subcommittee, along with any of its consultants who may be present, may exchange preliminary views regarding matters to be considered during the balance of the meeting.

The Subcommittee will then hear presentations by and hold discussions with representatives of the NRC staff, Westinghouse Electric Company, and other interested persons regarding these matters.

Further information regarding topics to be discussed, whether the meeting has been canceled or rescheduled, and the Chairman's ruling on requests for the opportunity to present oral statements and the time allotted therefor can be obtained by contacting the Designated Federal Official, Dr. Medhat M. El-Zeftawy (telephone: 301–415–6889) or Mr. Michael R. Snodderly, Cognizant Staff Engineer, (telephone: 301–415–6927) between 7:30 a.m. and 4:15 p.m. (EST). Persons planning to attend this meeting are urged to contact one of the above named individuals at least two working days prior to the meeting to be advised of any potential changes to the agenda.

Dated: December 20, 2002.

Sher Bahadur,

Associate Director for Technical Support, ACRS/ACNW.

[FR Doc. 02–32695 Filed 12–26–02; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Announcement of Public Meeting on Proposed Plan to Risk-Inform Post-Fire Safe-Shutdown Circuit Analysis Inspection

AGENCY: Nuclear Regulatory Commission.

ACTION: Notice of meeting.

SUMMARY: The Nuclear Regulatory Commission (NRC) will hold a public meeting, in the form of a facilitated workshop, to discuss and gather stakeholder input on proposed risk-informed post-fire safe-shutdown circuit analysis inspection.

DATES: February 19, 2003, 9 a.m. to 4:45 p.m.

ADDRESSES: The meeting will be held at the Nuclear Regulatory Commission, Two White Flint North, Auditorium, 11545 Rockville Pike, Rockville, Maryland 20852.

Referenced documents are available for review in ADAMS, NRC's online document management system at *http://www.nrc.gov* or from the NRC's Public Document Room (PDR), www.pdr.gov, 1–800–397–4209 or 301–415–4737, located on the first floor of One White Flint North, 11555 Rockville Pike, Rockville, MD 20852. Referenced documents and their Accession Nos. are: NRC Information Notice 99–17, "Problems Associated with Post-Fire Safe-Shutdown Circuit Analyses," (ADAMS Accession No. ML023510114); Enforcement Guidance Memorandum EGM–98–002, Revision 2, (ADAMS Accession No. ML003710123); Memorandum dated, 11/29/2000, "Rationale for Temporarily Halting Certain Associated Circuits Inspection Lines of Inquiry During Fire Protection Baseline Triennial Inspections," (ADAMS Accession No. ML003773142); Draft Revision D of NEI 00–01, "Guidance for Post-Fire Safe Shutdown Analysis, 10–15–2002," (ADAMS Accession No. ML023010376); Draft NUREG/CR, "Guidance for Post Fire Safe Shutdown Analysis," (ADAMS Accession No. ML023430533).

FOR FURTHER INFORMATION CONTACT: Joseph Birmingham, Mail Stop O–11F1, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, Telephone: 1–800–368–5642, extension 2829, or e-mail *jlb4@nrc.gov*.

SUPPLEMENTARY INFORMATION: Beginning in 1997, the NRC noticed a number of Licensee Event Reports (LERs) that identified plant specific problems related to potential fire induced electrical circuit failures that could prevent operation or cause mal-operation of equipment necessary to achieve and maintain post-fire safe shutdown. LERs identified problems involving Associated Circuits, Cable Routing, Redundant Train Separation, Wiring Errors, Fire-Induced Hot Shorts, Evaluations of Spurious Operations, Motor Operated Valve Evaluations, Transfer and Isolation Capability, Fuse/Breaker Coordination, High Impedance Faults, High-Pressure/Low-Pressure Interfaces (See NRC Information Notice 99–17 for more information) (Agencywide Documents Access and Management System (ADAMS) Accession No. ML023510114). On November 29, 2000, a memorandum was written that outlined the rationale for halting certain associated circuits inspections while the industry worked to resolve the issue (ADAMS Accession No. ML003773142).

In response to this issue, the Nuclear Energy Institute (NEI), with support from the Boiling Water Reactor Owners Group (BWROG), formed a circuit

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
SUBCOMMITTEE MEETING ON RELIABILITY AND PROBABILISTIC RISK ASSESSMENT

JANUARY 23, 2003
Today's Date

NRC STAFF PLEASE SIGN IN BELOW

PLEASE PRINT

NAME	NRC ORGANIZATION
LAWRENCE BURKHART	NRR/NRLPO
Alireza Behbahani	RES/DSARE/SMSAB
ANDRE DROZD	NRR/DSSA/SPSB
JOE COLACCINO	NRR
Goutam Bagchi	NRR/DE
Garth Parry	NRR/DSSA
John Segala	NRR/NRLPO
Richard Lee	RES/DSARE/SMSAB
Sud Basu	RES/DSARE/SMSAB
S. M. Bajorek	RES/DSARE/SMSAB
Nick Saltos	NRR/DSSA/SPSB
Y. Gene Hsii	NRR/DSSA/SRXB
Walton Jensen	NRR/DSSA/SRXB
MARIE POHIDA	NRR/DSSA/SPSB
Joe Sebrosky	NRR/NRLPO
BOB PALLA	NRR/DSSA/SPSB
JERRY Wilson	NRR/NRLPO
ANDRE DROZD	NRR/DSSA/SPSB

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
SUBCOMMITTEE MEETING ON RELIABILITY AND PROBABILISTIC RISK ASSESSMENT

JANUARY 23, 2003
Today's Date

ATTENDEES PLEASE SIGN IN BELOW

PLEASE PRINT

NAME	AFFILIATION
SPYROS TRAIFOROS	LINK TECHNOLOGIES
Mike Corletti	Westinghouse
Erin Whiting	Westinghouse
ED CUMMINS	WESTINGHOUSE
RON VIJUK	WESTINGHOUSE
Jim Scobel	WESTINGHOUSE
Jean-Luc FORET	EDF/Westinghouse
Selim Sancaktar	Westinghouse
TERRY SCHULZ	WESTINGHOUSE
Hymie Shapiro	AECL-Technologies
~~[illegible]~~	
Charles Brinkman	Westinghouse
Mohsen Khatib-Rahbar	ERI
Erin Whiting	Westinghouse
MICHAEL ZAVISCA	ERI

January 10, 2003

MEMORANDUM TO:	G. Apostolakis M. Bonaca F. Ford T. Kress G. Leitch	V. Ransom S. Rosen W. Shack J. Sieber

FROM: M. Snodderly, Senior Staff Engineer

SUBJECT: ACRS PRA SUBCOMMITTEE MEETING, JANUARY 23-24, 2003 - ROCKVILLE, MARYLAND

The ACRS Reliability and Probabilistic Risk Assessment (PRA) Subcommittee will hold a meeting on January 23-24, 2003 in our Rockville, Maryland Offices. The purpose of the meeting is to: (1) review the AP1000 PRA and Fire Study, and (2) identify possible issues with the AP1000 PRA that may need to be addressed prior to or during a July 2003 Full Committee meeting on the staff's draft safety evaluation report. **Please note that the meeting will convene at 8:30 a.m. on January 23, 2003.**

Previous Material

You should have received a compact disc from me during the December 5-7, 2002 Full Committee Meeting. The compact disc contains the AP1000 PRA, the AP600 PRA, requests for additional information that the staff has issued on the AP1000 PRA, and the staff's final safety evaluation report on the AP600 PRA.

AP1000 PRA Review

On March 28, 2002, Westinghouse tendered its application to the NRC for certification of the AP1000 design. The Committee is expected to report on those portions of the application which concern safety to fulfill the requirement of 10 CFR 52.53. In accordance with 10 CFR 52.47, Westinghouse has provided a design-specific PRA as part of its application for design certification of the AP1000. Since the configuration of the AP1000 reactor and safety systems is the same as the AP600, the AP600 PRA was used as the basis of the AP1000 PRA with relevant changes implemented in the model to reflect the AP1000 design changes. The Committee has reviewed certification of the AP600 design, which included its design-specific PRA. Attached is the Committee's July 23, 1998 report to the Commission which concluded that the AP600 PRA was done well and had been an integral part of the design process.

The Committee was briefed on the most significant AP1000 design changes by Westinghouse on November 7, 2002. The associated briefing material has been attached to this memorandum and the most significant design changes are covered by Slides 19 through 53. The briefing also included a summary of the AP1000 PRA, The Level 1 results are covered by Slides 68 to 93. During the November 7th presentation, the Committee was especially interested in the importance of the Protection and Safety Monitoring System (Slide 78) and the second peak of the core damage frequency distribution (Slide 80). The Level 2 and 3 results are summarized in Slides 94 through 106. Credit is taken for in-vessel retention of molten core debris. For the AP600, the staff concluded that the AP600 includes several design features that make in-vessel retention more likely but the in-vessel retention approach to severe accident mitigation should be balanced with evaluation of ex-vessel phenomena to address uncertainty. I believe that a similar conclusion may also be applicable for the AP1000 because the level of uncertainty has not changed much. Uncertainty has increased because of an increase in power and the number of fuel assemblies, and changes to the reactor vessel internals. Uncertainty has decreased because of design improvements to the insulation on the exterior of the reactor vessel. So the overall uncertainty in this area has not changed much.

Attached you will also find a compact disc with Westinghouse's responses to the staff's request for additional information on the AP1000 PRA. To access the PRA responses left click on the line that says, "720 - Reliability and Risk Assessment." Then left click on the specific response you would like to see. The requests are included on the other disc under the file, "AP1000 PRA RAIs."

The final attachment is a review of the AP1000 internal events at-power PRA by ACRS Senior Fellow, Hossein Nourbakhsh.

Meeting Agenda

The morning of January 23rd will be devoted to an overview of the AP1000 PRA with emphasis on design changes from the AP600. It is then expected that Westinghouse will summarize how these design changes were considered in the AP1000 PRA. The rest of the morning and the early part of the afternoon will focus on the Level 1 portion of the AP1000 PRA. The Level 2 and 3 portions of the AP1000 PRA will be discussed in the afternoon of the 23rd. Discussion of sensitivity, importance, and uncertainty analyses will take place once all three levels of the PRA have been discussed. The first day will conclude with a presentation from the staff on their requests for additional information on the internal events portion of the PRA. For the second day's discussion, Westinghouse has been asked to provide a presentation on the success criteria, including best-estimate thermal-hydraulic calculations used to develop success criteria. Westinghouse will then go over their fire risk assessment, low power and shutdown risk assessment, and external events. The staff will then present a short summary of their requests for additional information in these areas. A copy of a draft Agenda is attached.

Hotel reservations have been made for the Addressees at the Residence Inn in Bethesda MD (301/718-0200) as noted below. Please contact Ms. Barbara-Jo White (301/415-7130/ "bjw2@nrc.gov") if you need to change or cancel these reservations.

G. Apostolakis - 1/22-23
M. Bonaca - 1/22-23
F. Ford - 1/22-23
T. Kress - 1/22-23
G. Leitch - 1/22-23
V. Ransom - 1/22-23
S. Rosen - 1/22-23
W. Shack - 1/22-23
J. Sieber - 1/22-23

Attachments: As Stated

cc w/o attach (via E-mail):
- J. Larkins
- S. Bahadur
- H. Larson
- R. Savio
- S. Duraiswamy
- ACRS Technical Staff & Fellows

Record Copy

STATUS OF AP1000 SHUTDOWN PRA REVIEW

Marie Pohida
Probabilistic Safety Assessment Branch (SPSB)
Division of Systems Safety and Analysis
Office of Nuclear Reactor Regulation

January 23, 2003

Based on Staff review of the AP600 Shutdown PRA, SPSB issued 9 RAIs on the AP1000 Shutdown PRA.

RAIs focused on changes from AP600 PRA to AP1000 PRA

- ❑ **Common cause failure of the high pressure Gravity Injection squib valves and the high pressure recirculation squib valves**

- ❑ **Common cause failure of the two low pressure recirculation squib valves.**

- ❑ **Shorter response time for operator recovery actions including:**

  - **Containment Closure (required to maintain long term cooling water inventory)**
  - **Manual Gravity Injection**

SPSB asked additional RAIs on:

- ❑ **Trash Control during shutdown**
- ❑ **Shutdown Fire/Flood risk assessment**

ŞPSB has not completed their review of the AP1000 RAI responses.

Record Copy

AP1000

AP1000 Design Certification Review

Westinghouse Electric Company

Presentation to

Advisory Committee on Reactor Safeguards

PRA Sub-Committee

January 23 -24 2003

BNFL Westinghouse

AP1000

Agenda

Thursday January 23, 2003

• **Introduction**	**George Apostolakis, ACRS**	**8:30 am**
– Review goals and meeting objectives		
• **Westinghouse Introduction**	**Mike Corletti, Westinghouse**	**8:35 am**
• **Overview of AP1000 Design**	**Terry Schulz, Westinghouse**	**8:40 am**
– Design Changes from AP600		
– Key AP1000 Design Features		
– Defense-in-Depth		
– PRA as a Design Tool		
• **BREAK**		**10:05 am**
• **AP1000 PRA**	**Selim Sancaktar, Westinghouse**	**10:20 am**
– Background / Approach / Overview		
– Scope		
– Level 1 PRA Internal Events At-Power		
– Sensitivity and Uncertainty Assessments		
– Shutdown / Fire PRA		
• **LUNCH**		**12:20 - 1:30 pm**

AP1000

Agenda

Thursday January 23, 2003

• **PRA Level 1 Success Criteria**	**Terry Schulz, Westinghouse**	**1:30 pm**
– Overview		
– Thermal-Hydraulic Analysis to Support Level 1 PRA		
– T&H Uncertainty Assessment		
• **BREAK**		**3:30 pm**
• **NRC Staff Presentation**	**Nick Saltos - Walt Jensen - Marie Pohida**	**3:45 pm**
– Staff RAIs on Level 1 PRA and Success Criteria		
• **Westinghouse Summary**	**Mike Corletti**	**5:30 pm**

AP1000

Agenda

Friday January 24, 2003

• **Introduction**	**George Apostolakis, ACRS**	**8:30 am**
– Review goals and meeting objectives		
• **Level 2 and 3 PRA**	**Jim Scobel, Westinghouse**	**8:35 am**
– Quantification		
• **Invessel Retention of Molten Core Debris**		
• **BREAK**		**10:05 am**
• **Level 2 Phenomenological Studies**		**10:30 am**
• **Summary of PRA Results and Insights**	**Selim Sancaktar, Westinghouse**	**11:45 am**
• **LUNCH**		**12:15 pm**
• **NRC Staff Presentation**	**Bob Palla, NRR - Richard Lee, RES**	**1:15 pm**
• **Westinghouse Summary**	**Mike Corletti, Westinghouse**	**2:15 pm**
• **General Discussion**	**ACRS Members**	**2:30 pm**
• **Adjourn**		**3:00 pm**

AP1000

Design Certification Schedule

Major Milestones

1. **W Submits DCD Application (DCD / PRA)**	**3/28/02**
2. **Staff Issues RAI**	**9/30/02**
3. **W Provide Responses to All RAI**	**12/2/02**
2. **NRC Identify Potential DSER Open Items**	**2/28/03**
4. **W Addresses Potential DSER Open Items**	**4/15/03**
5. **NRC Issues DSER**	**6/16/03**

W Goal is to Address All Open Items Prior to Issuance of DSER

6. **ACRS Full Committee & Letter**	**7 / 2003**

W OBJECTIVE IS TO PROVIDE THE NRC / ACRS WITH THE NECESSARY INFORMATION SO THAT A FINAL SAFETY DETERMINATION ON AP1000 CAN BE MADE IN 2003

AP1000

W Objectives of the Meeting

- **Provide a Thorough Presentation of AP1000 PRA**
  - Level 1 / 2 / 3
  - Supporting T/H Analyses for Level 1
  - Supporting Phenomenological Studies for Level 2
- **Address All ACRS Issues Related to PRA**

ACRS Meetings

AP1000

- **Overview to Full Committee** — **Nov. 7, 2002**
- **PRA Subcommittee** — **Jan. 23/24 2003**
- **Thermal-Hydraulic Subcommittee** — **March 2003**
  - Safety Analysis / Entrainment Issue
  - Containment cooling
- **AP1000 Subcommittee** — **April 2003**
  - Containment structural design
  - Materials
  - Regulatory Treatment of Non-Safety Systems
  - Shutdown Maintenance
- **ACRS Full Committee Meeting** — **June - July 2003**

AP1000

Overview of AP1000 Design

Terry Schulz
Advisory Engineer
412-374-5120 - schulztl@westinghouse.com

AP600 to AP1000 Design Changes

AP1000

- **Increase Core Length & Number of Assemblies**
- **Increase Size of Key NSSS Components**
  - Increased height of Reactor Vessel
  - Larger Steam Generators (similar to W/CE SGs)
  - Larger canned RCPs (variable speed controller)
  - Larger Pressurizer
- **Increase Containment Height & Design Pressure**
- **Capacity Increases in Passive Safety System Components**
- **Turbine Island Capacity Increased for Power Rating**

Retained Nuclear Island Footprint

Comparison of Selected Parameters

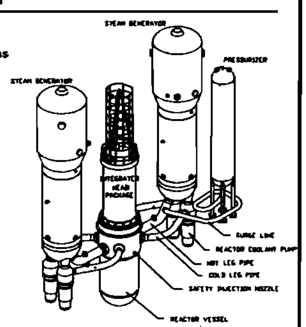
AP1000

Parameter	Doel 4/Tihange 3	AP600	AP1000
Net Electric Output, MWe	985	610	1117
Reactor Power, MWt	2988	1933	3400
Hot Leg Temperature, °F	626	600	610
Number of Fuel Assemblies	157	145	157
Type of Fuel Assembly	17x17	17x17	17x17
Active Fuel Length, ft	14	12	14
Linear Hear Rating, kw.ft	5.02	4.1	5.71
Control Rods / Gray Rods	52 / 0	45 / 16	53 / 16
R/V I.D., inches	157	157	157
Vessel flow (Thermal Design)	295,500	194,200	300,000
Steam Generator Surface Area, ft2	68,000	75,000	125,000
Pressurizer Volume, ft3	1400	1600	2100

AP1000 Major Components

AP1000

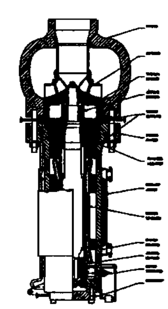
- **Fuel, Internals, Reactor Vessel**
  - Similar to Doel 4, Tihange 3, S. Texas
  - No bottom-mounted instrumentation
  - Use core shroud ala W/CE plants
  - Improved materials - 60 yr life
- **Steam Generators**
  - Features from W SGs in operation
  - Size from W/CE SGs in operation
- **Reactor Coolant Pumps**
  - Canned motor pumps
    - Naval reactors, early commercial reactors, AP600
- **Simplified Main Loop**
  - Same as AP600
  - Reduces welds 50%, supports 80%
- **Pressurizer**
  - 50% larger than operating plants



AP1000 Reactor Coolant Pump

AP1000

- **Based on Field-Proven, Canned Motor Pumps**
  - 1300 units in service
  - 12-year mean time between repair
  - No shaft seals
    - No seal injection / leakoff system
    - No seal leakage / failure
  - Water lubricated bearings
    - No oil lubricating / cooling system
  - Compact, high inertia flywheel
  - AP600 pump tests performed
    - Full size test of compact flywheel
    - Scaled hydraulics tests
    - Air-mixing tests of SG / RCP connection

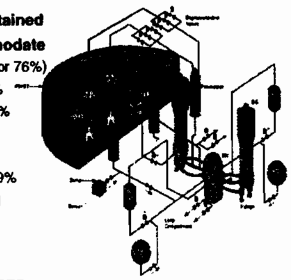


AP1000 Approach to Safety

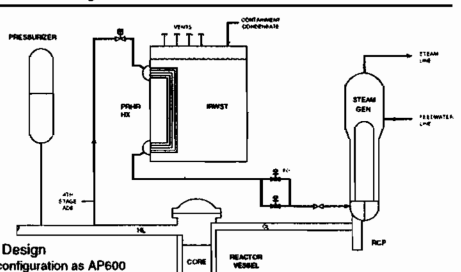
- **Passive Safety-Related Systems**
  - Use "passive" process only, no active pumps, diesels, ...
    - One time alignment of valves
    - No support systems required after actuation
      - No AC power, cooling water, HVAC, I&C
  - Greatly reduced dependency on operator actions
  - Mitigate design basis accidents without nonsafety systems
  - Meet NRC PRA safety goals without use of nonsafety systems
- **Active Nonsafety-Related Systems**
  - Reliably support normal operation
    - Redundant equipment powered by onsite diesels
  - Minimize challenges to passive safety systems
  - Not required to mitigate design basis accidents

AP1000 Passive Core Cooling System

- **AP600 System Configuration Retained**
- **Capacities Increased to Accommodate Higher Power** (1933MW - 3400MW or 76%)
  - PRHR HX Capacity Increased 72%
  - CMT Volume & Flow Increased 25%
  - ADS 4 Flow Increased 93%
  - IRWST Injection Increased 89%
  - Containment Recirc. Increased 139%
- **System Performance Maintained**
  - No core uncovery for SBLOCA
    - ≤ DVI line break
    - Large margin to PCT limit
  - No operator actions required for SGTR



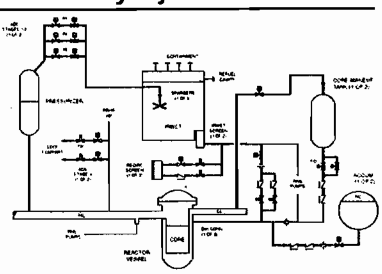
Passive Decay Heat Removal



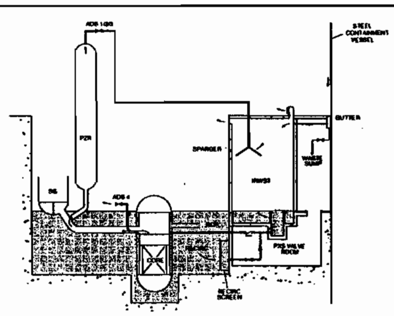
- PRHR HX Design
  - Same configuration as AP600
  - Same elevations as AP600
  - Larger pipe / valve sizes (14" vs 10")
  - Increased HX surface (more tubes / longer horizontal section)

AP1000 Passive Safety Injection

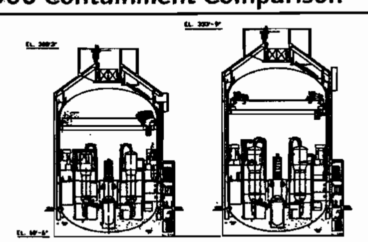
Passive Safety Injection
- Same configuration as AP600
- Same elevations as AP600
- Same Accum capacity
- Increased CMT capacity
  - 25% larger tank
  - 25% more flow
  - Same pipe, larger orifice
- Larger IRWST lines
  - 8" vs 6"
- Larger Recirc lines
  - 8" vs 6"
  - Increased cont. flood level
- Same ADS 1/2/3 lines
- Larger ADS 4 lines
  - 14" vs 10"



LOCA Long Term Cooling

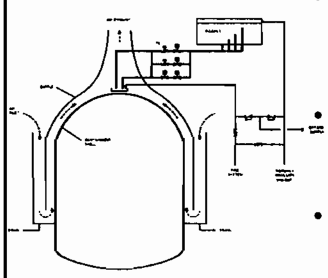


AP1000 Containment Comparison



	AP600	AP1000
Total Free Volume	100%	122%
Design Pressure, psig	45	59
Shell Thickness	1 5/8"	1 3/4"
Material	A537 Class 2	SA738 Grade B

Passive Containment Cooling System



- **PCS Water Storage Tank**
  - Provides 72 hr drain
    - Afterwards use on/offsite water
    - Air only cooling prevents failure
  - Flow decreases with time
    - Uses 4 standpipes
- **PCS Flow Rates**
  - High initial flow
    - Rapidly forms water film
    - Effectively reduces cont pressure
  - Later flows match decay heat
- **Added 3rd Diverse Drain Path**
  - Adds PRA margin
    - T&H uncertainty of cont cooling without water drain

AP1000 Safety Margins

	Typical Plant	AP600	AP1000
- Loss Flow Margin to DNBR Limit	~ 1 - 5%	~16%	~19%
- Feedline Break Subcooling Margin	>0°F	~170°F	~140°F
- SG Tube Rupture	Operator actions required in 10 min	Operator actions NOT required	Operator actions NOT required
- Small LOCA	3" LOCA core uncovers PCT ~1500°F	< 8" LOCA NO core uncovery	< 8" LOCA NO core uncovery
- Large LOCA PCT (with uncertainty)	2000 - 2200°F	1676°F	2124°F

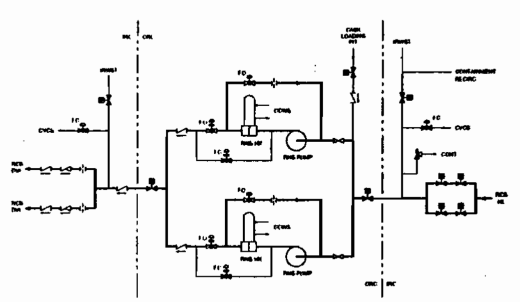
AP1000 Hydrogen Mitigation

- **Design Basis Accidents**
  - Slow long term buildup of H2
  - Uses 2 full size Passive Autocatalytic Recombiners (nonsafety)
    - No power or actuation required
  - Equipment is non-safety based on NRC / industry activities on risk-informed changes to 10 CFR 50.44 (Combustible Gas Control)
- **Severe Accidents**
  - Rapid buildup of H2
  - Uses non-safety igniters distributed in pairs around containment
  - Release paths from RCS ensure standing H2 flames located away from containment walls
    - IRWST vents changed to discharge H2 away from containment wall

AP1000 Active Nonsafety Systems

- **Active Nonsafety System Functions**
  - Reliably support normal operation
  - Minimize challenge to passive safety systems
  - Not required to mitigate design basis accidents
  - Not required to meet NRC safety goals
- **Active Nonsafety System Design Features**
  - Simplified designs (fewer components, separation not required)
  - Redundancy for more probable failures
  - Automatic actuation with power from onsite diesels
- **Active Nonsafety System Equipment Design**
  - Reliable, experienced based, industrial grade equipment
  - Non-ASME, non-seismic, limited fire / flood / wind protection
  - Availability controlled by procedures, no shutdown requirements
  - Reliability controlled by maintenance program

AP1000 Normal RHR System



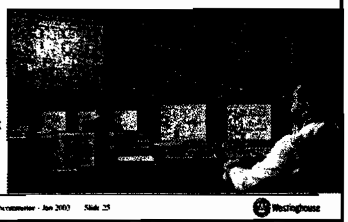
AP1000 I&C Systems

- **Control System (PLS/DDS)**
  - Plant wide non-1E system for all normal displays & controls
  - Microprocessor / software based, multiplexed communications
- **Safety System (PMS)**
  - Plant wide 1E system for all safety displays & controls
  - Microprocessor / software based, multiplexed communications
- **Diverse System (DAS)**
  - Limited scope non-1E system, PRA based displays & controls
    - Backs up PMS where common mode failure is risk important
  - Different hardware & software than PMS, no multiplexing
  - Separate sensors from PMS and PLS

AP1000

AP1000 Advanced Control Room

- **Compact Control Room**
  - Designed for 1 Reactor Operator and 1 Supervisor
- **Displays**
  - Plant status / overview via wall panel (DDS, non 1E)
  - Detail display via workstation video displays (DDS, non 1E)
  - Small number dedicated displays; safety (PMS, 1E) & diverse (DAS, non 1E)
- **Controls**
  - Soft controls (DDS, non 1E) for normal operation
  - Small number dedicated switches; safety (PMS, 1E) & diverse (DAS, non 1E)
- **Advanced Alarm Management**
- **Computer Based Procedures**



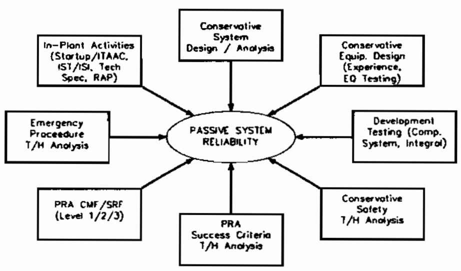
AP1000

PMS Reliability Features

- **Redundant Trains**
  - 4 divisions, physically separated with improved isolation (fiber-optic)
    - Each with own independent battery-backed power supply
  - 2 out of 4 bypass logic, fail safe when appropriate
  - Different plant parameters provide functional diversity
- **Extensive Verification and Validation**
- **Extensive Equipment Qualification**
  - Environmental, seismic, EMC
- **Improved In-Plant Testing**
  - Built-in continuous self-testing and manual periodic testing
- **West. Extensive Experience with Digital I&C Designs**
  - Operating plant upgrades and new plants (Sizewell, Temelin)

AP1000

AP1000 System Reliability



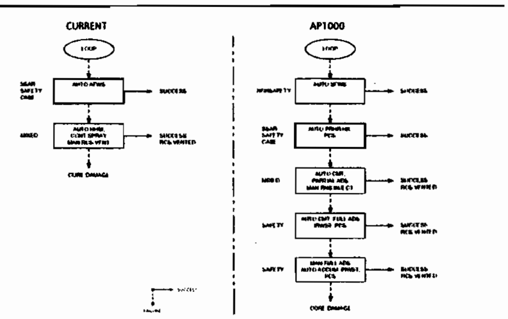
AP1000

System Defense In Depth

- **AP1000 Provides Multiple Levels of Defense**
  - First feature is usually nonsafety active feature
    - High quality industrial grade equipment
  - One feature is safety passive feature
    - Provides safety case for DCD
    - Highest quality nuclear grade equipment
  - Other passive features provide additional defense-in-depth
    - Example; passive feed/bleed backs up PRHR HX
  - Available for all shutdown conditions as well as at power
  - More likely events have more levels of defense

AP1000

Loss of Offsite Power



AP1000

Loss Offsite Power, at Power

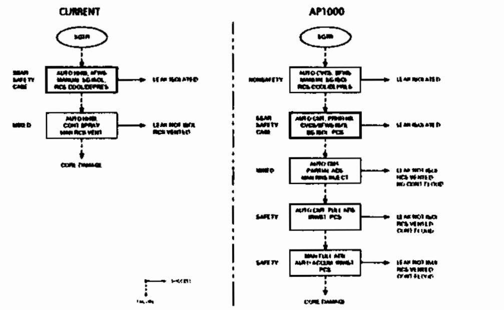
Function / System Use	Notes	Support Systems (I&C, Electrical, Cooling, HVAC) Non-Safety PLS (1)	DC	AC	CW	HVAC	Safety PMS (1)	DC	Diverse DAS (1)
• Reactor Shutdown									
1. Control Rods		-	-	-	-	-	Auto	-	-
2. Control Rods			Yes	-	-	-	-	-	Auto
3. Rods Out	(2)	-	Yes	-	-	-	-	-	Auto
• RCS Inventory Control									
1. CVS		Auto	Yes	Yes	Yes	Yes	-	-	-
2. CMT		-	-	-	-	-	Auto	-	-
3. CMT		-	Yes	-	-	-	-	-	Auto
4. CMT, RNS, part ADS		Man	Yes	Yes	Yes	Yes	Auto	Yes	-
5. CMT, IRWST, full ADS		-	-	-	-	-	Auto	Yes	-
6. CMT, IRWST, full ADS		-	Yes	-	-	-	-	Yes	Man
7. Accum, RNS, part ADS		Man	Yes	Yes	Yes	Yes	Man	Yes	-
8. Accum, IRWST, full ADS		-	-		-	-	Man	Yes	-
9. Accum, IRWST, full ADS		-	Yes	-	-		-	Yes	Man
• RCS Heat Removal									
1. SFW		Auto	Yes	Yes	Yes	Yes	-	-	-
2. PRHR HX		-	-	-	-	-	Auto	-	-
3. PRHR HX		-	Yes	-	-	-	-	-	Auto
4. CMT, RNS, part ADS		Man	Yes	Yes	Yes	Yes	Auto	Yes	-
5. CMT, IRWST, full ADS		-	-	-	-	-	Auto	Yes	-
6. CMT, IRWST, full ADS		-	Yes	-	-	-	-	Yes	Man
7. Accum, RNS, part ADS		Man	Yes	Yes	Yes	Yes	Man	Yes	-
8. Accum, IRWST, full ADS		-	-		-	-	Man	Yes	-
9. Accum, IRWST, full ADS		-	Yes	-	-	-	-	Yes	Man
• Containment Heat Removal									
1. Fan Coolers		Auto	Yes	Yes	Yes	Yes	-	-	-
2. CV external air, water drain		-	-	-	-	-	Auto	-	-
3. CV external air, water drain		-	Yes	-	-	-	-	-	Auto
4. CV external water fire sys only		Man	Yes	-	-	-	-	-	-
5. CV external air only		-	-	-	-	-	-	-	-

Notes:

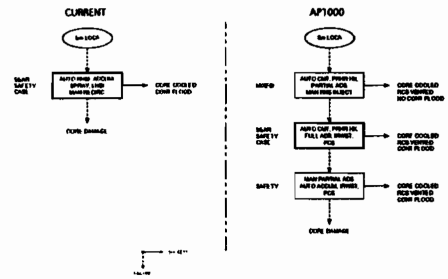
1) PLS provides manual control of all nonsafety equipment by nonsafety MCR soft control switches. PMS provides manual control of all safety equipment by safety MCR system level dedicated controls and safety MCR soft switches. DAS provides manual control of selected safety equipment by nonsafety MCR dedicated controls.

2) Reactor is shut down by negative moderator temperature coefficient as the coolant heats up. Requires automatic RCS pressure relief, turbine trip, PRHR HX or SFW, CMT or CVS actuation.

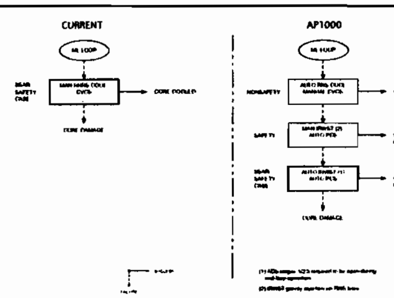
SG Tube Rupture



Small LOCA



Mid-Loop Loss Power



AP1000 PRA

- **Westinghouse Uses PRA as Design & Licensing Tool**
  - 7 PRA major quantifications performed on AP600
    - First in 1987, final in 1997
    - Extensive interaction with plant designers
    - Extensive NRC review / comment
  - AP1000 PRA quantified in 2001
    - Started with AP600 models / analysis
  - Plant designers interact with risk analysis
    - Results reviewed, improvements made (more in AP600)
      - PRA analysis models and supporting T/H analysis
      - Plant operating procedures
      - Plant design

PRA Based Changes (AP600)

- **Analysis Changes**
  - Accum or CMT sufficient for small / medium LOCA
  - One accum sufficient for large LOCA
  - Multiple ADS valve failures acceptable
- **Operation Changes**
  - Manually start RNS after ADS actuation
  - Require containment closure capability during mid-loop
  - Require PXS features to be available during shutdowns

PRA Based Changes (AP600)

- **Design Changes**
  - RNS alignment valves made remote
  - 4th stage ADS valves made diverse from stages 1, 2, 3
  - Added DAS functions
  - Added redundant IRWST injection check valves
  - Added redundant / diverse IRWST recirc valves
  - Made CMT check valves normally open, diverse from accum
  - Provided logic for automatic SGTR protection without ADS

PRA Based Changes (AP1000)

- **AP1000 Analysis Changes**
  - Initiating event frequency changes
    - Larger SGs (more, longer tubes)
    - Increased number SG safety valves
    - Separated spurious ADS stage 4 and large CL LOCA
      - 2 / 2 accum required for CL LOCA, 1/2 accum required for spur ADS 4
  - PRHR HX operation needed for MLOCA without CMTs
    - Provides operators sufficient time for manual ADS
- **AP1000 Operation Changes**
  - Containment recirc MOV normally open (in series with squib valve)
  - Changed IRWST drain proceedure so it occurs earlier in core melt
  - Added Tech Spec on DAS manual controls

PRA Based Changes (AP1000)

- **AP1000 Design Changes**
  - Increased volume and injection rate of CMTs
  - Added 3rd Passive Cont. Cooling drain valve, MOV diverse to AOV
  - Incorporated low boron core, improves ATWT
  - RNS injection water supply changed from IRWST to Cask Load Pit
  - Improved IVR heat transfer via changes to RV insulation gap
  - Improved H2 vents from IRWST to keep H2 flames away from cont.

AP1000 Probabilistic Risk Assessment

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OBJECTIVES

- **The purpose of the AP1000 PRA is to provide inputs to the optimization of the AP1000 design and to verify that the US NRC PRA safety goals have been satisfied**
- **As in the AP600, the PRA is being performed interactively with the design, analysis and operating procedures.**

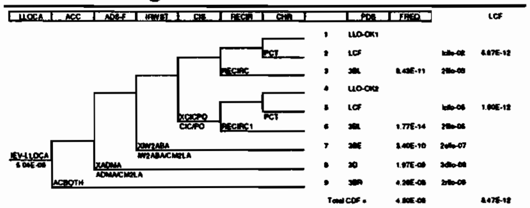
TECHNICAL SCOPE

- **Since the configuration of the AP1000 reactor and safety systems is the same as the AP600, the AP600 PRA is used as the basis of the AP1000 PRA with relevant changes implemented in the model to reflect the AP1000 design changes**

TECHNICAL SCOPE

- **AP1000 plant-specific T&H analyses are performed in order to determine the system success criteria**
- **The CDF and LRF are calculated for internal events at-power. The off-site dose risk analysis is also performed. The external events and shutdown models are also assessed to derive plant insights and plant risk conclusions.**

AP1000 Large LOCA Event Tree



List of Top Events

Event	Description
LLOCA	Large LOCA Event Occurs
ACC	Accumulators Inject
ADS-F	Full RCS Depressurization by ADS occurs
IRWST	RCS Refill from IRWST by Gravity Injection Occurs
CIS	Containment Isolation Occurs
RECIR	Water Recirculation to RPV from the Sump Occurs
CHR	Containment Cooling is Established

AP1000 System Failure Probabilities

- Fault Tree Models are used to calculate system failure probabilities & identify minimal cutsets
- All support systems are modeled in detail
- Component random failures, human errors, tests and maintenance unavailabilities, and common cause are modeled based on standard industry practice

AP1000 PRA System Failure Probabilities

Failure System/Function	Probability	Fault Tree Name
CMT Valve Signal	6.70E-07	CMT-IC11 (one train; auto and manual actuation)
PRHR Valve Signal	1.10E-06	RHR-IC01 (one train; auto and manual actuation)
Passive Cont. Cool.	1.88E-06	PCT
Reactor Trip by PMS	1.20E-06	RTPMS (including operator actions)
Accumulators	6.86E-06	AC2AB
IRWST Inj.	6.90E-06	IW2AB
ADS	9.38E-06	ADS (including operator actions)
Passive RHR	2.08E-04	PRI
Core Makeup Tanks	1.10E-04	CMBSL
125 vdc 1E Bus	3.10E-04	IDADB1 (one bus only)
DC Bus (Non-1E)	3.40E-04	ED1DS1 (one bus only)
RC Pump Trip	6.90E-04	RCT
Chilled Water	1.40E-03	VWH
Containment Isol	1.68E-03	CIC
Reactor Trip by DAS failure))	1.70E-03	DAS (including operator action, excluding MGSET
6900 vac Bus	3.20E-03	ECES1 (one bus only)
CVS	3.40E-03	CVS1
480 vac Bus	5.90E-03	ECEK11 (one bus only)
Service Water	6.20E-03	SWT
Comp. Cooling Water	6.30E-03	CCT
Diesel Generators	1.08E-02	DGEN
Startup Feedwater	1.70E-02	SFWT
Compressed Air	1.30E-02	CAIR
Condenser	2.40E-02	CDS
Main Feedwater	2.80E-02	FWT (including condenser)
RNS	8.10E-02	RNR
Hydrogen Control	1.08E-01	VLH

Contribution of Initiating Events to AP1000 CDF

	IEV CATEGORY	CONTRIBUTION TO CDF	CONTRIBUTION TO CDF (%)	IEV FREQUENCY	CONDITIONAL CDF
1	IEV-SI-LB	9.50E-08	39.43	2.13E-04	4.46E-04
2	IEV-LLOCA	4.50E-08	18.65	5.00E-06	8.99E-03
3	IEV-SPADS	2.96E-08	12.28	5.40E-05	5.48E-04
4	IEV-SLOCA	1.81E-08	7.5	5.00E-04	3.62E-05
5	IEV-MLOCA	1.61E-08	6.69	4.36E-04	3.70E-05
6	IEV-RV-RP	1.00E-08	4.15	1.00E-08	1.00E+00
7	IEV-SGTR	6.79E-09	2.82	3.88E-03	1.75E-06
8	IEV-CMTLB	3.68E-09	1.53	9.31E-05	3.95E-05
9	IEV-ATWS	3.61E-09	1.5	4.81E-01	7.49E-09
10	IEV-TRANS	3.08E-09	1.28	1.40E+00	2.20E-09
11	IEV-RCSLK	1.71E-09	0.71	6.20E-03	2.75E-07
12	IEV-POWEX	1.65E-09	0.69	4.50E-03	3.66E-07
13	IEV-LCOND	1.24E-09	0.52	1.12E-01	1.11E-08
14	IEV-LOSP	9.58E-10	0.4	1.20E-01	7.98E-09
15	IEV-LMFW	8.70E-10	0.36	3.36E-01	2.60E-09
16	IEV-ATW-T	7.12E-10	0.3	1.17E+00	6.08E-10
17	IEV-LCAS	6.72E-10	0.28	3.46E-02	1.95E-08
18	IEV-SLB-V	6.06E-10	0.25	2.36E-03	2.54E-07
19	IEV-PRSTR	5.02E-10	0.21	1.34E-04	3.74E-06
20	IEV-LMFW1	4.52E-10	0.19	1.92E-01	2.36E-09
21	IEV-LCCW	3.23E-10	0.13	1.44E-01	2.24E-09
22	IEV-SLB-U	1.31E-10	0.05	3.72E-04	3.51E-07
23	IEV-ATW-S	1.11E-10	0.05	1.48E-02	7.49E-09
24	IEV-ISLOC	5.00E-11	0.02	5.00E-11	1.00E+00
25	IEV-LRCS	3.52E-11	0.01	1.80E-02	1.96E-09
26	IEV-SLB-D	9.15E-12	0	5.98E-04	1.54E-08
	Totals =	2.41E-07	100.0	2.37*	

* The total initiating event frequency excludes the three ATWS precursor frequencies

Comparison of AP600 and AP1000 CDF

	AP600				AP1000				
IEV CATEGORY	CONTRIBUTION TO CDF	CONTRIBUTION TO CDF (%)	IEV FREQUENCY	CONDITIONAL CDF (P1)	CONTRIBUTION TO CDF	CONTRIBUTION TO CDF (%)	IEV FREQUENCY	CONDITIONAL CDF (P2)	RATIO P2/P1
IEV-LLOCA	5.08E-08	30.67	1.08E-04	4.70E-04	4.50E-08	18.65	5.00E-06	8.99E-03	18.8
IEV-SI-LB	3.82E-08	22.59	1.04E-04	3.67E-04	9.50E-08	39.43	2.13E-04	4.46E-04	1.2
IEV-NLOCA	3.16E-08	18.63	7.70E-04	4.09E-05					
IEV-RV-RP	1.00E-08	5.91	1.00E-08	1.00E+00	1.00E-08	4.15	1.00E-08	1.00E+00	1
IEV-ATWS	8.98E-09	5.31	4.81E-01	1.87E-08	3.61E-09	1.5	4.81E-01	7.49E-09	0.4
IEV-MLOCA	6.23E-09	3.68	1.62E-04	3.85E-05	1.61E-08	6.69	4.36E-04	3.70E-05	1
IEV-SGTR	6.00E-09	3.5	5.20E-03	1.17E-06	6.79E-09	2.82	3.88E-03	1.75E-06	1.5
IEV-SLOCA	4.05E-09	2.4	1.01E-04	4.01E-05	1.81E-08	7.5	5.00E-04	3.62E-05	0.9
IEV-CMTLB	3.54E-09	2.09	8.94E-05	3.96E-05	3.68E-09	1.53	9.31E-05	3.95E-05	1
IEV-RCSLK	2.26E-09	1.34	1.20E-02	1.88E-07	1.71E-09	0.71	6.20E-03	2.75E-07	1.5
IEV-POWEX	1.83E-09	1.08	4.50E-03	4.07E-07	1.65E-09	0.69	4.50E-03	3.66E-07	0.9
IEV-TRANS	1.14E-09	0.67	1.40E+00	8.14E-10	3.08E-09	1.28	1.40E+00	2.20E-09	2.7
IEV-LCOND	1.03E-09	0.61	1.12E-01	9.23E-09	1.24E-09	0.52	1.12E-01	1.11E-08	1.2
IEV-LOSP	1.01E-09	0.6	1.20E-01	8.40E-09	9.58E-10	0.4	1.20E-01	7.98E-09	1
IEV-ATW-T	7.12E-10	0.42	1.17E+00	6.08E-10	7.12E-10	0.3	1.17E+00	6.08E-10	1
IEV-PRSTR	5.58E-10	0.33	2.50E-04	2.23E-06	5.02E-10	0.21	1.34E-04	3.74E-06	1.7
IEV-SLB-V	4.82E-10	0.28	1.21E-03	3.98E-07	6.06E-10	0.25	1.21E-03	5.01E-07	1.3
IEV-ATW-S	3.82E-10	0.23	2.05E-02	1.86E-08	1.11E-10	0.05	1.48E-02	7.49E-09	0.4
IEV-LMFW	3.03E-10	0.18	3.35E-01	9.04E-10	8.70E-10	0.36	3.36E-01	2.60E-09	2.9
IEV-LMFW1	1.76E-10	0.1	1.92E-01	9.18E-10	4.52E-10	0.19	1.92E-01	2.36E-09	2.6
IEV-LCAS	1.73E-10	0.1	3.46E-02	4.99E-09	6.72E-10	0.28	3.46E-02	1.95E-08	3.9
IEV-SLB-U	1.23E-10	0.07	3.72E-04	3.31E-07	1.31E-10	0.05	3.72E-04	3.51E-07	1.1
IEV-LCCW	1.23E-10	0.07	1.44E-01	8.53E-10	3.23E-10	0.13	1.44E-01	2.24E-09	2.6
IEV-ISLOC	5.00E-11	0.03	5.00E-11	1.00E+00	5.00E-11	0.02	5.00E-11	1.00E+00	1
IEV-LRCS	1.27E-11	0.01	1.80E-02	7.06E-10	3.52E-11	0.01	1.80E-02	1.96E-09	2.8
IEV-SLB-D	9.44E-12	0.01	5.98E-04	1.58E-08	9.15E-12	0	5.98E-04	1.54E-08	1
IEV-SPADS					2.96E-08	12.28	5.40E-05	5.48E-04	
Totals	1.69E-07	100	2.30E+00		2.41E-07	100	2.37E+00		

AP1000 PRA Dominant CDF Sequences

	Sequence Frequency	% Contrib	Cum. % Contrib	Sequence ID	Sequence Description	Event Identifier
1	6.86E-08	28.52	28.52	2esil-07	SAFETY INJECTION LINE BREAK INITIATING EVENT OCCURS	IEV-SI-LB
					RCPS TRIP AND CMT INJECTION IS SUCCESSFUL - 1 OF 2 CMT TRAINS	DEL-XCM1A
					SUCCESS OF FULL ADS DEPRESSURIZATION	DEL-ADM
					FAILURE OF ONE OF ONE IRWST INJECTION LINE	SYS-IW1A
2	4.26E-08	17.66	46.18	2llo-09	LARGE LOCA INITIATING EVENT OCCURS	IEV-LLOCA
					ANY ONE OF TWO ACCUMULATOR TRAINS FAIL	SYS-ACBOTH
3	2.13E-08	8.82	55	3dsad-08	SPURIOUS ADS INITIATING EVENT OCCURS	IEV-SPADS
					SUCCESS OF 1/2 OR 2/2 ACCUMULATORS	DEL-AC2AB
					FAILURE OF ADS OR CMT	SYS-XADMA
4	1.98E-08	8.23	63.23	3esil-06	SAFETY INJECTION LINE BREAK INITIATING EVENT OCCURS	IEV-SI-LB
					RCPS TRIP AND CMT INJECTION IS SUCCESSFUL - 1 OF 2 CMT TRAINS	DEL-XCM1A
					FAILURE OF FULL ADS DEPRESSURIZATION	SYS-ADM
5	1.00E-08	4.15	67.38	3crv-02	REACTOR VESSEL RUPTURE INITIATING EVENT OCCURS	IEV-RV-RP
6	8.44E-09	3.5	70.88	2islo-06	SMALL LOCA INITIATING EVENT OCCURS	IEV-SLOCA
					SUCCESS OF CMT & RCP TRIP	DEL-XCM2SL
					SUCCESS OF PASSIVE RHR SYSTEM	DEL-PRL
					SUCCESS OF FULL ADS DEPRESSURIZATION	DEL-ADS
					FAILURE OF NORMAL RHR IN INJECTION MODE	SYS-RNR
					SUCCESS OF TWO OF TWO IRWST INJECTION LINES	DEL-IW2AB
					SUCCESS OF CIS & PRE-EXISTING CONTAINMENT OPENING	DEL-XCICPO
					FAILURE OF RECIRCULATION	SYS-RECIRC

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AP1000 PRA Dominant CDF Sequences

	Sequence Frequency	% Contrib	Cum. % Contrib	Sequence ID	Sequence Description	Event Identifier
7	7.35E-09	3.05	73.89	2mlo-05	MEDIUM LOCA INITIATING EVENT OCCURS	IEV-MLOCA
					SUCCESS OF CMT & RCP TRIP	DEL-XCM2NL
					SUCCESS OF FULL ADS DEPRESSURIZATION	DEL-ADM
					FAILURE OF NORMAL RHR IN INJECTION MODE	SYS-RNR
					SUCCESS OF TWO OF TWO IRWST INJECTION LINES	DEL-IW2AB
					SUCCESS OF CIS & PRE-EXISTING CONTAINMENT OPENING	DEL-XCICPO
					FAILURE OF RECIRCULATION	SYS-RECIRC
8	5.11E-09	2.12	76.05	3dmlo-12	SMALL LOCA INITIATING EVENT OCCURS	IEV-SLOCA
					SUCCESS OF CMT & RCP TRIP	DEL-XCM2SL
					SUCCESS OF PASSIVE RHR SYSTEM	DEL-PRL
					FAILURE OF FULL ADS DEPRESSURIZATION	SYS-ADS
					SUCCESS OF PARTIAL ADS DEPRESSURIZATION	DEL-ADV
					FAILURE OF NORMAL RHR IN INJECTION MODE	SYS-RNR
9	4.46E-09	1.85	77.9	3dmlo-12	MEDIUM LOCA INITIATING EVENT OCCURS	IEV-MLOCA
					SUCCESS OF CMT & RCP TRIP	DEL-XCM2NL
					FAILURE OF FULL ADS DEPRESSURIZATION	SYS-ADM
					SUCCESS OF PARTIAL ADS DEPRESSURIZATION	DEL-ADU
					FAILURE OF NORMAL RHR IN INJECTION MODE	SYS-RNR
10	3.72E-09	1.54	79.44	2sad-09	SPURIOUS ADS INITIATING EVENT OCCURS	IEV-SPADS
					FAILURE OF 2/2 ACCUMULATORS	SYS-AC2AB
11	3.67E-09	1.52	80.96	2ssad-07	SPURIOUS ADS INITIATING EVENT OCCURS	IEV-SPADS
					SUCCESS OF 1/2 OR 2/2 ACCUMULATORS	DEL-AC2AB
					SUCCESS OF ADS & CMT	DEL-XADMA
					FAILURE OF IRW OR CMT	SYS-XIW2ABA

AP1000

Summary of Sensitivity Analysis Results

Case Description	Results
Set LCF sequences to core damage	SI-LB initiating event is the most important contributor (30.84%) to CDF. CDF increases by a factor of 1.3.
Initiating Event Importances	SI-LB (39.43) and LLOCA (18.66) initiating events are the most important contributors to CDF
Accident Sequence Importances	IEV-SI-LB, DEL-XCM1A, DEL-ADM, SYS-IW1A (28.82%) is the most important sequence contributor to CDF.
End State Importances	3BE (33.4%) and 3D+1D (23.9%) are the most important contributors to CDF.
Common Cause Failure Importances	Software CCF of all cards and IRWST sump strainers plugging CCF are the most important contributors to CDF.
Human Error Importances	Operator failure to diagnose SG tube rupture event is the most important contributor to CDF.
Component Importances	IRWST strainer plugged, PRHR H/X plug/leak and IRWST tank failure are most important contributors to CDF.
Set HEPs to 1.0 in core damage output file (no credit for HEPs)	CDF increases by a factor of 57.
Set HEPs to 0.0 in core damage output file (perfect operator)	CDF decreases 8%.
Set HEPs to 0.1 in core damage output file	CDF increases by a factor of 6.5.
Impact of passive system check valve failure probabilities	CDF increases by a factor of 3.7.
Impact of explosive valve failure probabilities	CDF increases by a factor of 2.7.
Impact of reactor trip breaker failure probabilities	CDF has negligible increase.
Impact of RCP breaker failure probabilities	CDF increases by a factor of 1.2.
Sensitivity to standby non-safety systems (CVS,SFW,RNS,DAS,DG)	CDF increases by a factor of 31.

AP1000

AP1000 PRA System Importances

AP1000 SYSTEM IMPORTANCES BY RISK (CDF) INCREASE			
PMS DC-1E IRWST-REC ADS IRWST-INJ	CMT ACC PRHR	DC-Non 1E DAS AC PLS	CAS N-RHR SWS CCS SFW DG MFW SG Overfill Protection
> 300	50-300	2-50	<2

AP1000

AP1000 PRA System Importances

PMS	No credit is taken for PMS in CD sequences
DC-1E	No credit is taken for 1E DC Power in CD sequences
IRWST-REC	No credit is taken for IRWST Recirculation in CD sequences
ADS	No credit is taken for ADS in CD sequences
IRWST-INJ	No credit is taken for IRWST Injection in CD sequences
CMT	No credit is taken for CMT in CD sequences
ACC	No credit is taken for Accumulators in CD sequences
PRHR	No credit is taken for Passive RHR in CD sequences
DC-Non 1E	No credit is taken for Non-1E DC Power in CD sequences
DAS	No credit is taken for DAS in CD sequences
AC	No credit is taken for AC Power in CD sequences
PLS	No credit is taken for PLS in CD sequences
CAS	No credit is taken for CAS in CD sequences
N-RHR	No credit is taken for Normal RHR in CD sequences
SWS	No credit is taken for SWS in CD sequences
CCS	No credit is taken for CCS in CD sequences
SFW	No credit is taken for Startup Feedwater in CD sequences
DG	No credit is taken for Diesel Generators in CD sequences
MFW	No credit is taken for Main Feedwater in CD sequences
SG Overfill Protection	No credit is taken for SG Overfill Protection in CD sequences

AP1000

Importance of PMS and DC-1E Systems

- PMS and DC-1E are the most important systems (by risk increase measure)
- PMS is very reliable and redundant; its reliability is only limited by postulated CCF (such as CCF software).
- In case of a total postulated failure of PMS, the plant relies on DAS (auto or manual) and control systems (only for some transients); in this scenario, the plant CDF goes up by orders of magnitude

AP1000

Sensitivity Analyses Results

- The component, operator action, and system importance analyses provide us input for other AP1000 programs (such as RTNSS, reliability assurance program)
- The sensitivity analyses increase our confidence in the stability of PRA numerical results.

UNCERTAINTY ANALYSIS

AP1000

- The plant CDF uncertainty range is found to be 7.3 E-07 – 2.1 E-08 for the 95% to 05 % interval
- For a lognormal distribution, this would correspond to an error factor of 6, which can be considered as low for rare events

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UNCERTAINTY ANALYSIS

AP1000

- The mean values of the dominant accident sequence frequencies are close to the upper bound (95%) estimates;
- Among the initiating event categories, SI-LB has the highest 95-percentile CDF of 3.2E-07 /year.
- Among the dominant sequences, sequence # 07 of SI-LB event has the highest 95-percentile CDF of 2.1E-07/yr.

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SHUTDOWN EVENTS

AP1000

- A quantitative shutdown risk evaluation is performed for AP1000 for internal events
- The risk profiles of AP1000 and AP600 for events during shutdown conditions are almost identical
- The AP1000 Shutdown PRA has a CDF of 1.23E-07 events per year. This CDF is an 18% increase of the AP600 Level 1 Shutdown CDF of 1.04E-07 events per year

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SHUTDOWN EVENTS

AP1000

- The three events dominating the CDF for each plant are loss of component cooling / service water during drained condition, loss of offsite power during drained condition, and loss of RNS during drained condition
- The initiating event CDF contributions show that the initiating event importance to be similar for the two plants

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SHUTDOWN EVENTS

AP1000

- The twelve dominant accident sequences comprise 77 percent of the level 1 shutdown CDF. They consist of:
  - Loss of component cooling or service water system initiating event during drained condition with a contribution of 64 percent of the CDF

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SHUTDOWN EVENTS

AP1000

- Loss of RNS initiating event during drained condition with a contribution of 6 percent of the CDF
- Loss of offsite power initiating event during drained condition with a contribution of 5 percent of the CDF
- RCS overdraining event during drainage to mid-loop with a contribution of a 2 percent of the CDF.

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INTERNAL FLOODING AND FIRE

- The internal flooding-induced CDF is estimated to be 8.8E-10 events per year for power operations
- The CDF from flooding events at power is not an appreciable contributor to the overall AP1000 plant CDF

INTERNAL FLOODING AND FIRE

- The top five at-power flooding scenarios comprise 91 percent of the at-power flooding-induced core damage frequency
- These scenarios are for large pipe breaks in the turbine building with an initiating event frequency in the range of 1.4 – 2.0 E-03 / year, leading to a loss of CCW/SW event
  - Each scenario has a CDF of 1.2 – 1.8E-10/year.

INTERNAL FLOODING AND FIRE

- Extensive fire hazards analysis review completed for AP600 subsequent to fire AP600 PRA
  - Fire separation improved
  - Fire suppression features incorporated
  - Design features incorporated to address hot-shorts
- AP1000-specific Fire PRA is performed with a resulting CDF of 5.61E-08/yr (for internal events)

INTERNAL FLOODING AND FIRE

- AP600 design features important for fire protection are included in the AP1000
  - Fire separation / fire zones
  - Systems used to achieve safe shutdown
  - Fire suppression features
- AP1000 design is sufficiently robust that internal fires during power operation or shutdown do not represent a significant contribution to plant CDF

SEISMIC MARGINS EVALUATION

- The seismic margin analysis shows the systems, structures, and components required for safe shutdown. HCLPF values are greater than or equal to 0.50g
- This HCLPF is determined by the seismically induced failure of the fuel in the reactor vessel, core assembly failures, IRWST failure, or containment interior failures

SEISMIC MARGINS EVALUATION

- The SMA result assumes no credit for operator actions at the 0.50g review level earthquake, and assumes a loss of offsite power for all sequences
- The SMA shows the plant to be robust against seismic event sequences that contain station blackout coupled with other seismic or random failures
- AP1000 structural design and seismic analysis will be discussed at a future ACRS meeting

Comparison of Low HCLPF SSCs in AP1000 and AP600 Designs

AP1000

Basic Event ID	Description	AP600 HCLPF	AP1000 HCLPF
EQ-CER-INSULATOR	Failure of Ceramic Insulators	0.09g	.09g
EQ-CORE-ASSEMBLY	Core Assembly Failure (not fuel)	0.50g	.50g
EQ-CV-INTER	Interior Containment	0.60g	.50g
EQ-IRWST-TANK	IRWST Failure	0.60g	.50g
EQ-RV-FUEL	Fuel Failure	0.50g	.50g
EQ-AB-EXTWALL	Aux. Building Exterior wall	0.58g	.51g
EQ-AB-FLOOR	Aux. Building Floor	0.58g	.51g
EQ-AB-INTWALL	Aux. Building Interior wall	0.58g	.51g
EQ-PCC-TANK	PCC Tank Failure	0.58g	.51g
EQ-SHDBLD-ROOF	Shield Building Roof	0.58g	.51g
EQ-SHDBLD-WALL	Shield Building Wall	0.58g	.51g
EQ-CABLETRAY	Cable trays - support controlled	0.54g	.54g
EQ-CMT-TANKS	Tank PXS 2A/B (Core Makeup Tank)	0.63g	.54g
EQ-SG-FAILS	Steam Generator Fails	0.65g	.54g
EQ-SGTR	Steam Generator Piping (one or a few)	0.65g	.54g
EQ-ACDISPANEL	120 vac distribution panel	0.51g	.55g
EQ-DC-SWBRD	125 vdc switchboard	0.51g	.55g
EQ-DCDISPANEL	125 vdc distribution panel	0.51g	.55g
EQ-PRZR-FAILS	Pressurizer Fails	0.67g	.55g
EQ-TRSFSWITCH	Transfer switch	0.51g	.55g

Comparison of AP600 and AP1000 PRA Results

AP1000

Scope	AP600	AP1000
Level 1 At-Power Internal Initiating Events	Quantification Performed CDF = 1.7E-07 Several additional cases quantified in response to NRC RAIs	Quantification Performed CDF = 2.4E-07 AP600 additional cases incorporated into the model
Level 2 At-Power Internal Initiating Events	Quantification Performed LRF = 1.8E-08 Containment Effectiveness = 89.5%	Quantification Performed LRF = 2.0E-08 Containment Effectiveness = 91.8%
Level 3 At-Power Internal Initiating Events	Quantification Performed	Quantification Performed
Internal Fire Events	Conservative (via focused PRA) Quantification Performed CDF = 6.5E-07 (internal) CDF = 3.5E-07 (shutdown)	Quantification performed CDF = 5.61E-08
Internal Flooding Events	Quantification Performed CDF = 2.2E-10	Quantification Performed CDF = 8.8E-10
Shutdown Events	Quantification Performed for Level 1 and 2 CDF = 1.0E-07 LRF = 1.5 E-08 Several additional cases quantified in response to NRC RAIs	Quantitative Evaluation Performed CDF = 1.2E-07 LRF = 2.0E-08

AP1000

SUMMARY OF RESULTS

- **The AP1000 PRA results show that**
  - The very low risk of the AP600 has been maintained in the AP1000
  - The AP1000 PRA meets the US NRC safety goals with significant margin

AP1000

PRA Level 1 Success Criteria

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AP1000

Overview

- **Success Criteria Justification**
  - Summary of success criteria (Chapter 6 of PRA)
    - Changes in success criteria vs AP600
  - Success criteria justification
    - Based on analysis - DCD, specific PRA, or other analysis / calculations
    - Summary of PRA analysis
      - Analysis results for small LOCA, large LOCA and ATWS
    - T&H Uncertainty Evaluations
      - Calc of low margin / risk important sequences
      - T&H analysis to bound T&H uncertainty

AP1000

AP1000 Success Criteria

- **Similar to AP600**
  - Similar system design, arrangement, capabilities
  - Several Changes Made to the AP1000 Success Criteria
    - Due to increase in power and other factors
- **Verified Using Same Approach as AP600**
  - Use DCD analysis where applicable
  - Perform special analysis where DCD analysis not applicable
- **AP1000 Success Criteria More Conservative / Robust**
  - Uses same or more equipment for success than AP600
    - For example, uses 3/4 ADS 4 instead of 2/4 ADS 4 (AP600)
      - Even though AP1000 ADS 4 is larger / MW
  - Reduces T&H issues / uncertainty

Success Criteria Basis

- **Provides Critical Functions**
  - Decay heat removal (core cooling)
    - Peak clad temperature < 2200°F
  - RCS inventory control
  - RCS pressure control
    - Less than emergency stress limits, < 3200 psig
  - Containment heat removal and containment isolation
    - Less than emergency stress limits, < ??? psig
  - Reactivity control

AP1000 Full ADS Success Criteria

TABLE A2.3-1 FULL ADS SUCCESS CRITERIA(1)

Event	PRHR HX - on		PRHR HX - off	
	CMT - on	CMT - off	CMT - on	CMT - off
	Accum - off	Accum - on	Accum - off	Accum - on
RCS Transients, Loss of Power, Station Blackout	None (2)	None (2)	Auto 1/2 ADS stage 2/3 and auto 3/4 ADS stage 4	Man 3/4 ADS stage 4
RCS Leak	Auto 3/4 ADS stage 4	Man 3/4 ADS stage 4	Auto 1/4 ADS stage 2,3 and auto 3/4 ADS stage 4	Man 3/4 ADS stage 4
SGTR	None (3)	None (3)	Auto 1/4 ADS stage 2,3 and auto 3/4 ADS stage 4	Man 3/4 ADS stage 4
Small LOCA	Auto 3/4 ADS stage 4	Man 3/4 ADS stage 4	Auto 1/4 ADS stage 2,3 and auto 3/4 ADS stage 4	Man 3/4 ADS stage 4
Medium LOCA	Auto 3/4 ADS stage 4	Man 3/4 ADS stage 4	Auto 3/4 ADS stage 4	(7)
Spurious ADS	(4)		Auto 3/4 ADS stage 4 (5)	
Large LOCA	(4)		Auto 3/4 ADS stage 4 (6)	

Notes:
1. Automatic ADS actuation via PMS. ADS actuation can also be performed manually via PMS or DAS.
2. Successful PRHR HX operation obviates need for ADS.
3. SGTR does not require ADS operation if PRHR HX operates and SGs are isolated.
4. Operation of PRHR HX has no effect on ADS success criteria, use "PRHR HX - off" success criteria.
5. Spurious ADS requires 1/2 accumulators and 1/2 CMT to work.
6. Large LOCA requires 2/2 accumulators and 1/2 CMT to work.
7. No credit is given for success for this case; the time available for operator action is short.

Post ADS Success Criteria

- **Changes Made to Post ADS Success Criteria**
  - Full ADS (IRWST) >> requires 3/4 ADS stage 4
    - AP600 PRA used 2/4 ADS stage 4
    - AP1000 ADS 4 capacity has been increased by more than power
  - Partial ADS (RNS) >> requires 2 of 4 ADS stage 2 or 3
    - AP600 PRA used 1/4 stage 2 or 3
    - ADS stages 1, 2, 3 capacities not increased for AP1000
  - Requires PRHR HX for MLOCAs with only Accum
    - Provides operators more time (> 20 min) to take action
  - Requires 2/4 Cont Recirc if Cont Isol fails
    - 1/4 Cont Recirc if Cont Isol works
  - Full ADS required for large LOCAs to support long term cooling

LOCA Size Definitions

- **Large LOCA (> 9" ID)**
  - Requires 2 of 2 accum
- **Spurious ADS Stage 4 (1 to 4 ADS 4 valves)**
  - Require 1 of 2 accum and 1 CMT
- **Medium LOCA, DVI LOCA, CMT Line LOCA (2-9" ID)**
  - Only requires 1 accum or 1 CMT
  - Depressure RCS below ADS 4 pressure interlock
- **Small LOCA (3/8-2" ID)**
  - Requires PRHR HX or ADS 1/2/3 todepressure RCS below ADS 4 pressure interlock
  - CVS makeup not sufficient
- **RCS Leak (< 3/8" ID)**
  - CVS makeup is sufficient

PRA Success Criteria Analysis

• **Transient (PRHR HX)**	**DCD, LOFTRAN**
• **SGTR (PRHR HX)**	**DCD, LOFTRAN**
• **Non-LOCA Feed-Bleed**	**PRA, MAAP4**
• **LOCA (Small/Med. LOCA)**	**PRA, MAAP4**
• **LOCA (Lg LOCA)**	**PRA, WCOBRA-TRAC**
• **Spurious ADS 4 (Lg LOCA)**	**PRA, WCOBRA-TRAC**
• **ATWS**	**PRA, LOFTRAN**

MAAP4 Code Use

- **Same Approach As AP600**
  - Used for defining success criteria for LOCAs and feed-bleed cooling sequences
    - Provides integrated RCS / containment response
    - Runs fast (hours vs days)
      - Important because of large numbers of runs (hundreds)
        - Break sizes, locations, different sets of multiple failures
    - MAAP4 has been bench marked against NOTRUMP for AP600
      - NOTRUMP has been shown to be applicable to AP1000
    - T&H uncertainty analysis confirms that low margin / risk important sequences will be success
      - Uses detailed DCD codes and methods (NOTRUMP, WCOBRA-TRAC)
- **AP1000 Success Criteria is More Robust**

PRA T&H Analysis

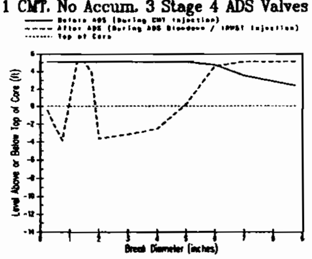
- **LOCAs and Feed-Bleed Cooling Analysis**
  - Considers many different factors
    - Initiating event, LOCA or Feed-Bleed Cooling after non-LOCA
    - LOCA size and location
    - Available mitigating equipment including CMT, Accum, RNS, PRHR HX, ADS, IRWST, Cont Recirc
  - Made use of lessons learned from AP600
    - Test results, DCD analysis, PRA analysis (both success criteria and T&H uncertainty
    - Divided into four groups of analysis
      1. Automatic ADS with CMT and IRWST gravity injection
      2. CMT and RNS pumped injection
      3. Manual ADS with Accum and IRWST gravity injection
      4. Accum and RNS pumped injection

1. Auto ADS with IRWST Gravity Injection

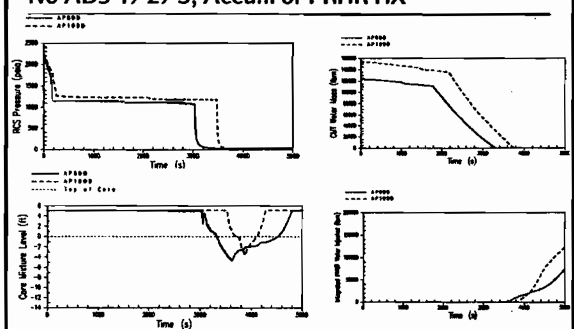
- **Limiting Success Criteria Equipment Assumed**
  - One CMT, no Accum, 1 valve path in one IRWST injection line
    - Same as AP600
  - 3/4 ADS stage 4 , no ADS stage 1/2/3, no PRHR HX
    - AP600 used 2/4 ADS 4
    - For LOCAs < 2" some ADS 1/2/3 or PRHR HX required to reduce RCS pressure to below ADS 4 pressure interlock
  - Containment isolation fails
- **MAAP4 Analysis Was Performed**
  - Break sizes 0.5" up to 8.75"
  - Core uncovery depth and duration is less than AP600
    - Increased capacity PXS, especially ADS 4 & IRWST injection
  - AP1000 success criteria verified

1. Auto ADS with IRWST Gravity Injection

AP1000 Minimum Vessel Mixture Level
Automatic ADS. IRWST Injection
1 CMT. No Accum. 3 Stage 4 ADS Valves



2" HL LOCA, 3/4 ADS4, 1 CMT, 1/1 IRWST No ADS 1/2/3, Accum or PRHR HX

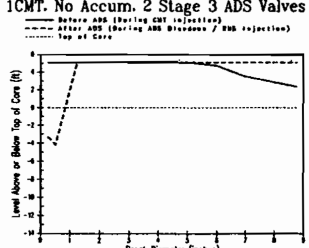


2. Auto ADS with RNS Injection

- **Limiting Success Criteria Equipment Assumed**
  - One CMT, no Accum, 1 RNS pump (SFP Cask Loading Pit)
  - 2/4 ADS stage 2/3, no ADS stage 4, no PRHR HX
    - AP600 used 1/4 ADS 2/3
  - Containment isolation fails
- **MAAP4 Analysis Was Performed**
  - Break sizes 0.5" up to 8.75"
  - Core uncovery depth and duration is less than AP600
  - AP1000 success criteria verified

2. Auto ADS with RNS Injection

AP1000 Minimum Vessel Mixture Level
Automatic ADS. RNS Injection
1CMT. No Accum. 2 Stage 3 ADS Valves



3. Manual ADS w. IRWST Gravity Injection

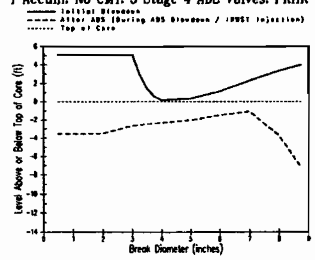
AP1000

- **Limiting Success Criteria Equipment Assumed**
  - One Accum, no CMT, PRHR HX, 1/1 valve / path IRWST injection
    - AP600 does not require PRHR HX, increases time for operator action
  - 3/4 ADS stage 4 , no ADS stage 1/2/3, no PRHR HX
    - ADS 4 manually actuated at 20 min.
    - AP600 uses 2/4 ADS 4
  - Containment isolation fails
- **MAAP4 Analysis Was Performed**
  - Break sizes 0.5" up to 8.75"
  - Core uncovery depth and duration is less than AP600
    - Increased capacity PXS, especially ADS 4 & IRWST injection
  - AP1000 success criteria verified

3. Manual ADS w. IRWST Gravity Injection

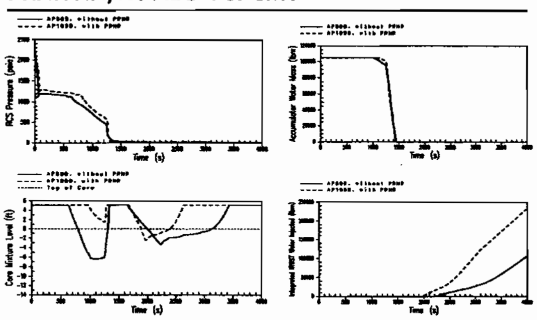
AP1000

AP1000 Minimum Vessel Mixture Level
Manual ADS at 20 Min. IRWST Injection
1 Accum. No CMT. 3 Stage 4 ADS Valves. PRHR



3.5" LOCA, 2/4 ADS 3, 1 Acc, 1/1 IRWST PRHR HX, No ADS 4 or CMT

AP1000



4. Manual ADS with RNS Injection

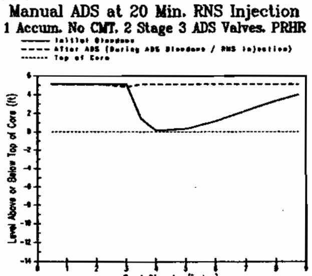
AP1000

- **Limiting Success Criteria Equipment Assumed**
  - One Accum, no CMT, PRHR XH, 1 RNS pump (Cask Loading Pit)
  - 2/4 ADS stage 2/3, no ADS stage 4
    - ADS manually actuated at 20 min.
    - AP600 used 1/4 ADS 2/3
  - Containment isolation fails
- **MAAP4 Analysis Was Performed**
  - Break sizes 0.5" up to 8.75"
  - AP1000 success criteria verified

4. Manual ADS with RNS Injection

AP1000

AP1000 Minimum Vessel Mixture Level
Manual ADS at 20 Min. RNS Injection
1 Accum. No CMT. 2 Stage 3 ADS Valves. PRHR



Large LOCA Success Criteria

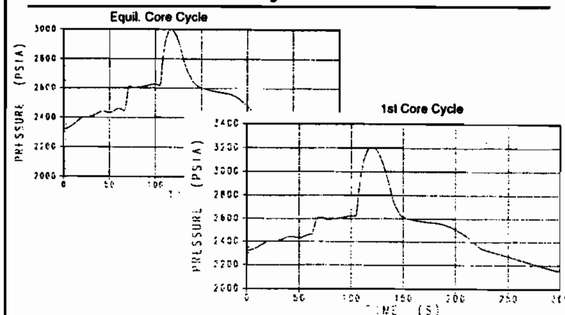
AP1000

- **Large CL LOCAs**
  - Uses 2 of 2 Accum, like DCD analysis
  - Unlike DCD assumes failure of containment isolation and availability of offsite power
  - Was analyzed with WCOBRA-TRAC (RAI 720.012)
    - Calc PCT 1628 F without uncertainty
    - PCT less than DCD case because offsite power was available
- **Spurious ADS 4 Large LOCAs**
  - Limiting case is all four ADS 4 valves opening
  - Uses 1 of 2 Accum, failure cont. isolation, offsite power available
  - Was analyzed with WCOBRA-TRAC (RAI 720.010)
    - Calc PCT 833 F without uncertainty
    - Case analyzed assumed cont isol, because of margin fail cont isol will be OK
- **Both Cases Are Successful**

ATWS Analysis

- **Provides Very Low Unfavorable Exposure Time**
  - AP1000 has low boron core
    - MTC is more negative
    - ATWS "ride out" capability is possible for more than 98.5% of core life
      - Throughout equilibrium core cycles, peak RCS pressure < 3000 psig
      - Through 60% of 1st core cycle, peak RCS pressure < 3200 psig
      - UET < 1.5% over 40 years
- **AP1000 ATWS Analysis**
  - Analyzed with LOFTRAN
  - Equilibrium core has MTC = -12.5 pcm/F at BOL
  - 1st core has MTC = -10.0 pcm/F at 40% life

AP1000 ATWS Analysis Results



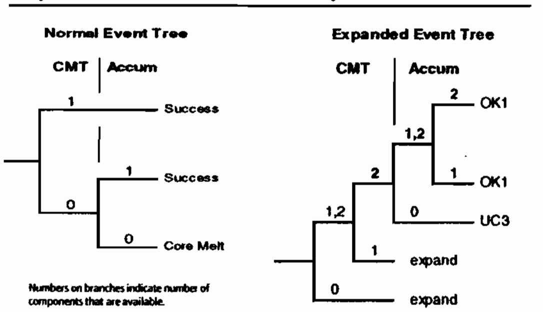
T&H Uncertainty

- **Same Approach As AP600**
  - Detailed evaluation performed (RAI 720.012)
  - Bounds AP1000 T&H uncertainty
    - Determined high risk / low margin cases
      - MAAP4 success criteria analysis used to identify low margin sequences
      - "Expanded" event trees used to identify high risk sequences
      - Bounds more than 98% of LOCA core melt
    - Identified limiting analysis cases
      - 3 small LOCAs, 2 large LOCAs, 2 LTC cases identified
    - Analyzed limiting cases with DCD codes and assumptions
      - Conservative decay heat (Appendix K), line resistances, plant parameters
      - All show successful core cooling

Expand Event Trees

- **Purpose of Expanded Event Trees**
  - Branches with safety equipment are expanded to identify the numbers of safety components that are available
    - The normal event trees only identify the minimum number of safety components that are required
  - Branches with non safety equipment are removed
  - End states changed to differentiate success paths
    - Two general classes, high margin (OK) and low margin (UC)
      - Low margin cases have core uncovery, high margin cases do not
    - More detailed sub-grouping made
      - Based on equipment available / not available
      - Supports selection of T&H uncertainty cases that are analyzed
  - Allows probability of low / high T&H margin cases to be calculated

Expanded Event Tree Example



Numbers on branches indicate number of components that are available.

Expanded Event Tree End States

1.	OK1	More ADS-4 than Design Basis (DB)
2.	OK2	Design Basis
3.	OK3	More ADS-4 / Less ADS-1, 2, 3 than DB
4.	OK4	Less ADS-1, 2, 3 than DB
5.	OK5A	More ADS-4 / CI fails
6.	OK5B	More ADS-4 / CI fails / Less ADS-1, 2, 3
7.	OK6	DB ADS / CI fails
8.	OK7	2 Accumulators / DB for LLOCA
9.	OK8	SI line break with Auto ADS from faulted CMT
0.	OK9	Loss of CMTs for smaller breaks
1.	UC1	No make-up of inventory if RCS pressure greater than 700 psig
2.	UC2A	1 Accumulator depletes prior to operator intervention
3.	UC2B	2 Accumulators deplete prior to operator intervention
4.	UC3	No rapid inventory make-up during blowdown
5.	UC4	Reduced inventory make-up during LLOCA reflood
6.	UC5	No make-up when ADS is actuated
7.	UC6	Less ADS-4 than DBA (ie < 3 of 4 ADS-4)
8.	UC7	Less ADS-4
9.	UC8	No containment isolation / DBA
10.	UC9	No containment isolation / reduced ADS

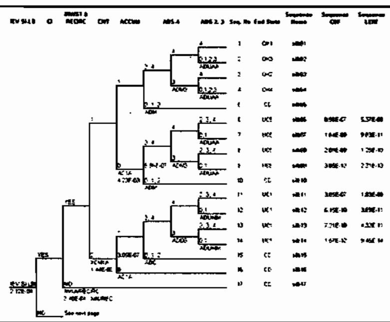
Which Event Trees

- **Selection of Level 1 Event Trees to Expand**
  - AP600 expanded 8 event trees, all with ADS actuation
    - No core uncovery in events / sequences without ADS
  - AP1000 expanded 5 event trees, all with ADS actuation
    - 3 event trees included in AP600 were not expanded for AP1000 since they did not result in limiting T&H analysis cases
      - Small LOCAs, Transients with ADS, SGTR with ADS were not expanded
      - These events did not add any limiting T&H uncertainty analysis cases
        - Some of their end states are not success in AP1000 (for example, 2 / 4 ADS 4 was considered success in AP600 but is not considered success in AP1000)
        - They tend to have more equipment available because they are more probable events
        - ADS occurs later in these events with lower decay heat

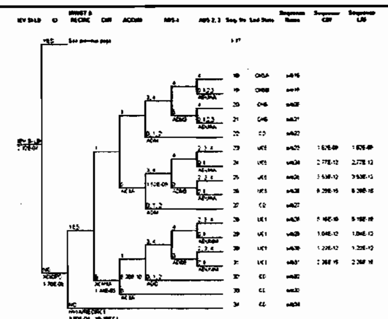
Expanded Event Trees

Initiating Event	AP600	AP1000
Large LOCA	yes	yes
Spurious ADS 4	na	yes
Medium LOCA	yes	yes
CMT Line LOCA	yes	yes
DVI LOCA	yes	yes
Intermediate LOCA	yes	na
Small LOCA	yes	-
SGTR with ADS	yes	-
Transients with ADS	yes	-

Expanded Event Tree - DVI LOCA



Expanded Event Tree - DVI LOCA



Calculation of CDF / LRF

- **Potential CDF**
  - Conservatively assumes low margin sequences (UC) may be core damage
  - System reliabilities based on fault tree calc
    - Base PRA or special fault trees as needed
- **Potential LRF**
  - Based on potential core damage sequences
  - Uses constant ratio 6% for containment isol branches
    - Conservative, same as AP600

Determination of Risk Important Sequences

- **All Low Margin Sequences Are Collected**
  - Includes all UC sequences
  - Sorted by CDF and LRF
  - Criteria for risk importance
    - 1% of baseline CDF or LRF
    - Residue of less important sequences must be small
      - Required to be less than twice the risk important sequences
- **Results**
  - 102 low margin sequences quantified in 5 expanded event trees
  - 13 low margin sequences selected as risk important
    - Covers 99.4% of risk from all low margin sequences
    - Residue of other sequences is < 6% of CDF and LRF

Sorted UC Sequences (Top 25 of 102)

Table 3-1 UC Sequences Sorted by CDF Frequency

	End State	Seq. Name	Sequence CDF	Sequence LRF	% CDF	% LRF	CI	IRWST & RECIRC	CMT	ACC	ADS 4	ADS 2,3	Bounded By Short/Long-Term		
>>	UC5	slb06	8.96E-07	5.37E-08	371.66%	275.86%	YES	YES	1	0	4	2-4	C	FG	
>>	UC	sad06	4.56E-07	2.75E-08	190.05%	140.89%	YES	YES	2	1	4	2-4	E	FG	
>>	UC1	slb11	3.06E-07	1.83E-08	126.76%	94.00%	YES	YES	0	1	4	2-4	A	FG	
>>	UC2B	mlo31	2.89E-07	1.73E-08	119.85%	88.88%	YES	YES	0	2	4	2-4	AB	FG	
>>	UC2B	cmt31	1.34E-07	8.05E-09	55.67%	41.28%	YES	YES	0	2	4	2-4	AB	FG	
>>	UC	sad25	9.12E-08	5.47E-09	37.82%	28.05%	NO	YES	2	2	4	2-4	E	G	(2)
>>	UC3	mlo11	3.01E-08	1.81E-09	12.48%	9.28%	YES	YES	2	0	4	2-4	C	FG	
>>	UC6	llo15	8.51E-09	8.51E-09	3.53%	43.63%	NO	YES	2	2	4	2-4	D	G	
>>	UC3	cmt26	6.42E-09	3.85E-10	2.67%	1.99%	YES	YES	1	0	4	2-4	C	FG	
>>	UC2A	mlo36	2.44E-09	1.47E-10	1.01%	0.76%	YES	YES	0	1	4	2-4	A	FG	
	UC5	slb08	2.08E-09	1.25E-10	0.87%	0.64%	YES	YES	1	0	3	2-4	C	F	
	UC5	slb07	1.64E-09	9.83E-11	0.68%	0.50%	YES	YES	1	0	4	0-1	C	FG	
>>	UC5	slb23	1.52E-09	1.52E-09	0.63%	7.77%	NO	YES	1	0	4	2-4	C	G	
	UC2A	cmt36	1.14E-09	6.88E-11	0.47%	0.35%	YES	YES	0	1	4	2-4	A	FG	
	UC	sad08	1.07E-09	6.42E-11	0.44%	0.33%	YES	YES	2	1	3	2-4		F	
	UC	sad07	8.40E-10	5.04E-11	0.35%	0.26%	YES	YES	2	1	4	0-1	E	FG	
	UC	sad80	7.77E-10	4.66E-11	0.32%	0.24%	NO	YES	2	1	4	2-4	E	G	(2)
	UC1	slb13	7.21E-10	4.32E-11	0.30%	0.22%	YES	YES	0	1	3	2-4		F	
	UC2B	mlo33	6.92E-10	4.15E-11	0.29%	0.21%	YES	YES	0	2	3	2-4		F	
	UC	sad17	6.76E-10	4.05E-11	0.28%	0.21%	YES	YES	1	1	4	2-4	E	FG	
	UC2B	mlo32	6.44E-10	3.86E-11	0.27%	0.20%	YES	YES	0	2	4	0-1	AB	FG	
	UC1	slb12	6.15E-10	3.69E-11	0.26%	0.19%	YES	YES	0	1	4	0-1	A	FG	
>>	UC1	slb28	5.16E-10	5.16E-10	0.21%	2.65%	NO	YES	0	1	4	2-4	A	G	
>>	UC2B	mlo73	4.88E-10	4.88E-10	0.20%	2.50%	NO	YES	0	2	4	2-4	A	G	
	UC2B	cmt33	3.17E-10	1.90E-11	0.13%	0.10%	YES	YES	0	2	3	2-4		F	

Risk Important Sequences

Table 3.4 AP1000 T&H Uncertainty Low Margin / Risk Important Sequences

Case	Sequence	CI	IRWST & RECIRC	CMT	ACC	ADS 4	ADS 2/3	PRHR	Sequence CDF	Sequence LRF	% CDF	% LRF	Short Term	Long Term
1	slb06	YES	YES	1	0	4	2-4	N/A	8.96E-07	5.37E-08	371.7%	275.8%	C	F
2	sad06	YES	YES	2	1	4	2-4	N/A	4.58E-07	2.75E-08	190.1%	140.9%	E	F
3	slb11	YES	YES	0	1	4	2-4	YES	3.06E-07	1.83E-08	126.8%	94.0%	A	F
4	mlo31	YES	YES	0	2	4	2-4	YES	2.89E-07	1.73E-08	119.9%	88.9%	B	F
5	cmt31	YES	YES	0	2	4	2-4	YES	1.34E-07	8.06E-09	55.7%	41.3%	B	F
6	sad25	NO	YES	2	2	4	2-4	N/A	9.12E-08	5.47E-09	37.8%	28.0%	E	G
7	mlo11	YES	YES	2	0	4	2-4	N/A	3.01E-08	1.81E-09	12.5%	9.3%	C	F
8	llo15	NO	YES	2	2	4	2-4	N/A	8.51E-09	8.51E-09	3.5%	43.6%	D	G
9	cmt26	YES	YES	1	0	4	2-4	N/A	6.42E-09	3.86E-10	2.7%	2.0%	C	F
10	mlo36	YES	YES	0	1	4	2-4	YES	2.44E-09	1.47E-10	1.0%	0.8%	A	F
11	slb23	NO	YES	1	0	4	2-4	N/A	1.52E-09	1.52E-09	0.6%	7.8%	C	G
12	slb28	NO	YES	0	1	4	2-4	YES	5.16E-10	5.16E-10	0.2%	2.6%	A	G
13	mlo73	NO	YES	0	2	4	2-4	YES	4.88E-10	4.88E-10	0.2%	2.5%	A	G
								Totals =	2.22E-06	1.44E-07				
	Residue from UC Sequences not selected								1.26E-08	8.62E-10	5.2%	4.4%		
	Residue from sequences with PRHR failure								1.58E-10	9.67E-12	0.1%	0.0%		

Bounding T&H Analysis Cases

- **T&H Uncertainty Cases**
  - 5 short term and 2 long term cooling cases are selected to bound the 13 risk important cases
  - These cases also bound 58 of the 102 low margin cases
    - Covers 99.8% of risk from all low margin sequences

Analysis Case	Initiating Event (1)	Cont. Isol.	IRWST & RECIRC	CMT	ACC	ADS 4	ADS 2/3	PRHR	Bounds Dominant Case
Short-Term Cooling									
A	RCS hot leg (3.0")	no	yes	0	1	4	0	yes	3,10,12,13
B	DE CMT balance line (6.8")	yes	yes	0	2	4	0	yes	4,5
C	DE DVI line (4")	no	yes	1	0	3	0	no	1,7,9,11
D	DE CL LLOCA	no	yes	2	2	4	0	yes	8
E	Spur ADS4 (2)	no	yes	1	1	4	0	yes	2,6
Long-Term Cooling									
F	DE DVI	yes	1/1&1/1	1	0	3	0	no	1-5, 7, 9, 10
G	DE DVI	no	1/1&2/1	1	0	4	0	no	6, 8, 11-13

Notes (1) Break sizes are effective sizes (inside diameter or orifice; not outside pipe diameter).
(2) Spurious ADS assumes all 4 ADS stage 4 valves open at same time as initiating event.

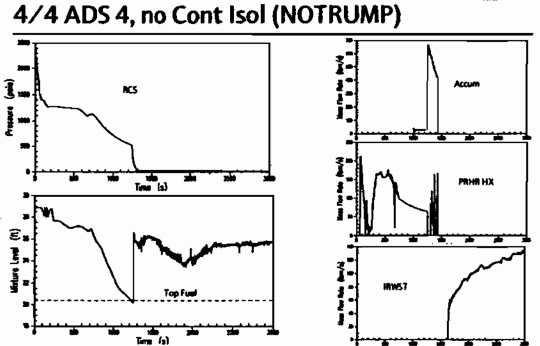
T&H Uncertainty Analysis

- **All of These 7 Cases Have Been Analyzed**
  - Using DCD codes and methods
  - All cases show successful core cooling

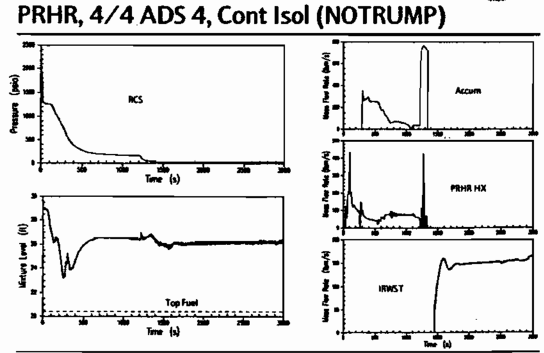
Case	Initiating Event	Cont Isol.	IRWST	REC	CMT	ACC	ADS 4	ADS 2/3	PRHR	CODE	Results	Reference
Short-Term Cooling												
A	RCS hot leg (3.0")	no	yes	yes	0	1	4	0	yes	NOTRUMP	No core uncovery	RAI 720.015
B	DE CMT bal.line (6.8")	yes	yes	yes	0	2	4	0	yes	NOTRUMP	No core uncovery	PRA App A
C	DE DVI line (4")	no	yes	yes	1	0	3	0	no	NOTRUMP	PCT = 1570 F	PRA App A
D	DE CL LLOCA	no	yes	yes	2	2	4	0	yes	WCOBRA-TRAC	PCT = 1856 F (1)	RAI 720.012
E	Spur ADS4	no	yes	yes	1	1	4	0	yes	WCOBRA-TRAC	PCT = 1061 F (1)	RAI 720.010
Long-Term Cooling												
F	DE DVI	yes	1/1	1/1	1	0	3	0	no	WCOBRA-TRAC	No core uncovery	RAI 720.013
G	DE DVI	no	1/1	2/1	1	0	4	0	no	WCOBRA-TRAC	No core uncovery	RAI 720.013

Notes:
(1) Includes DCD Large LOCA uncertainties.

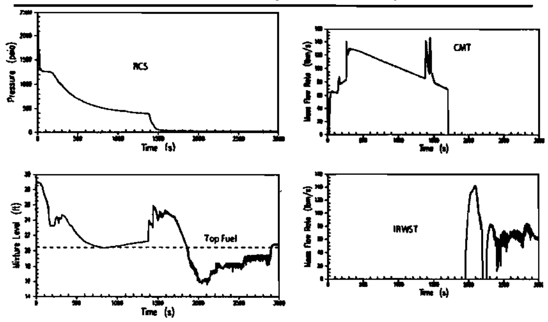
Case A, 3.0" LOCA, 1 Acc, 0 CMT, PRHR 4/4 ADS 4, no Cont Isol (NOTRUMP)



Case B, 6.8" CMT LOCA, 2 Acc, 0 CMT PRHR, 4/4 ADS 4, Cont Isol (NOTRUMP)



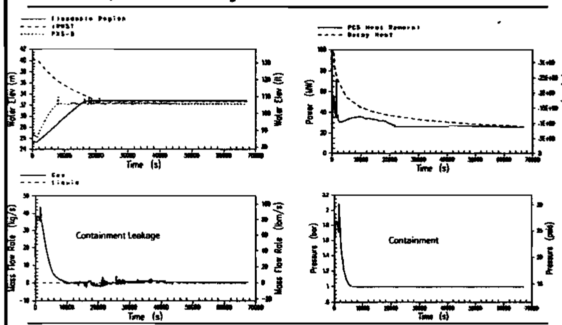
Case C, DVI LOCA, 0 Acc, 1 CMT, no PRHR 3/4 ADS 4, no Cont Isol (NOTRUMP)



T&H Uncertainty Case for Long-Term Cooling with Cont. Isol. Failure

- **Conservative / Limiting Case Analyzed**
  - Largest containment penetration is open (18" HVAC line)
  - DVI LOCA assumed to give lowest initial containment level
    - Causes flooding of PXS valve room where break is located
      - Reduces containment level by ~ x ft
- **LTC Analysis Results**
  - Containment leakage terminated in ~ 2.8 hr (MAAP4)
    - PCS is able to remove decay heat with cont. at atmospheric pressure
    - Leakage of steam/air mix removes air from containment
    - PCS heat transfer improves as partial pres of steam increases
    - Containment recirc level is reduced by ~ 0.3 ft
  - Core remains covered (WCOBRA-TRAC)

Case G, LTC Analysis with Cont. Isol. Failure

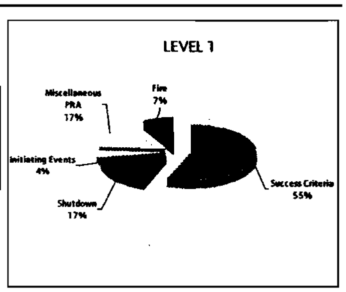


T&H Uncertainty Summary

- **AP1000 T&H Uncertainty Analysis**
  - Has calculated probabilities of low margin sequences
  - Has selected risk important, low margin sequences
  - Has defined 7 bounding T&H uncertainty cases
    - 5 Short and 2 Long-term
  - T&H Analysis has been performed on these cases
    - Using DCD Codes and methods
    - Shows successful core cooling
- **AP1000 T&H Uncertainty is Not Risk Important**
  - ~ 99% of CDF and LRF is bounded by conservative T&H analysis

Summary of RAI on AP1000 Level 1 PRA

RAIs on PRA Level 1	
Success Criteria	30
Shutdown	9
Initiating Events	2
Miscellaneous PRA	9
Fire	4
TOTAL	54



AP1000 PRA Report Updates Included with RAI Responses

- T/H Uncertainties Explicitly Addressed
  - Expanded Event Trees
  - Additional T/H Analyses Performed
    - 99% of Success Sequences Backed -Up with DBA Analysis Models
- Operator Action Times Addressed
- Revision of PRA Chapter 6 and Appendix A
- AP1000-Specific Fire PRA Performed

AP1000 Level 2 / 3 PRA

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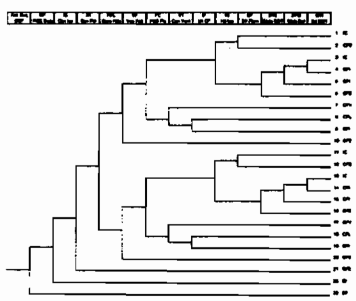
AP1000 Containment Event Tree

- **Used to quantify frequency and magnitude of releases to the environment**
- **Essentially the same structure as AP600 Containment Event Tree**

AP1000 Containment Event Tree Structure



AP1000 Containment Event Tree

- **Phenomena and System Availability**
  - reactor coolant system pressure
  - containment isolation
  - cavity flooding for external reactor vessel cooling
  - in-vessel reflooding
  - vessel failure
  - passive containment cooling water

AP1000 Containment Event Tree (continued)

- **Phenomena and System Availability (continued)**
  - hydrogen control (igniters)
  - containment overtemperature (diffusion flame)
  - hydrogen combustion (deflagration and detonation)
  - containment integrity

AP1000 Containment Event Tree

- **Operator actions**
  - Recovery Actions
    - depressurize RCS
    - isolate containment
    - actuate PCS water
  - Manual Severe Accident Management Actions
    - flood reactor cavity
    - actuate hydrogen control

Containment Event Tree Simplifying Assumptions

AP1000

- **High pressure RCS at core damage results in induced SGTR containment bypass**
- **Vessel failure and debris relocation into the containment results in early containment failure**
  - highly conservative

Interface with Level 1 PRA

AP1000

FUNCTIONAL DEFINITIONS OF LEVEL 1 ACCIDENT CLASSES

Accident Class	Subclass	Definition
1	A	Core damage with RCS at high pressure following transient or RCS leak
	AP	Core damage with no depressurization following small LOCA and RCS leak with passive residual heat removal operating or intermediate LOCA
	D	Core damage with partial depressurization of RCS following transient
3	A	Core damage with RCS at high pressure following anticipated transient without scram or main steam line break inside containment
	BR	Core damage following large LOCA with full RCS depressurization, but accumulator failed
	BE	Core damage following large LOCAs or other event with full depressurization
	BL	Core damage at long term following failure of water recirculation to reactor vessel after successful gravity injection
	C	Core damage following vessel rupture
	D	Core damage following LOCA (except large) with partial depressurization
6	E	Core damage following steam generator tube rupture or interfacing system LOCA. Early core damage (loss of injection)
	L	Core damage following steam generator tube rupture. Late core damage (loss of recirculation)

Interface with Level 1 PRA

AP1000

Accident Class Frequencies
Level 1 PRA Results

Accident Class	Frequency	%	Description
1A	5.0E-9	2.1	High RCS Pressure (Transient or SLOCA)
1AP	1.5E-9	0.6	SLOCA with PRHR Operating
3A	4.4E-9	1.8	High RCS Pressure (ATWS)
3BE	8.1E-8	33.4	RCS Depressurized. Gravity Injection Fails
3BL	2.4E-8	9.9	RCS Depressurized: Gravity Recirculation Fails
3BR	4.6E-8	19.2	RCS Depressurized: LLOCA CMTs and Accum Fails
3C	1.0E-8	4.2	RCS Depressurized. Vessel Failure Initiating Event
3D	6.0E-8	24.8	Partial RCS Depressurization
6	9.5E-9	4.0	SGTR or ISLOCA
Total CDF	2.41E-7	100	

Release Categories

AP1000

IC	Intact Containment	Containment integrity is maintained throughout the accident, and the release of radiation to the environment is due to nominal leakage.	Normal Leakage
BP	Containment Bypass	Fission products are released directly from the RCS to the environment via the secondary system or other interfacing system bypass. Containment failure occurs prior to onset of core damage.	Large Release
CI	Containment Isolation Failure	Fission-product release through a failure of the system or valves that close the penetrations between the containment and the environment. Containment failure occurs prior to onset of core damage.	Large Release
CFE	Early Containment Failure	Fission-product release through a containment failure caused by severe accident phenomena occurring after the onset of core damage but prior to core relocation. Such phenomena include hydrogen combustion phenomena, steam explosions, and vessel failure.	Large Release
CFV	Containment Venting	Fission-product release through a containment vent line during intentional depressurization of the containment	Controlled Release
CFI	Intermediate Containment Failure	Fission-product release through a containment failure caused by severe accident phenomena, such as hydrogen combustion, occurring after core relocation but before 24 hours.	Large Release
CFL	Late Containment Failure	Fission-product release through a containment failure caused by severe accident phenomena, such as a failure of passive containment cooling, occurring after 24 hours.	Large Release

Level 2 PRA Quantification Results

AP1000

Table 43-5

LRF AND CONTAINMENT EFFECTIVENESS BY ACCIDENT CLASS

	3BE	3BL	3BR	1A	1AP	3A	3C	3D	6		TOTAL	%
CDF	8.06E-08	2.40E-08	4.63E-08	5.01E-09	1.48E-09	4.43E-09	1.00E-08	5.97E-08	9.52E-09		2.41E-07	100.0%
CFE	3.22E-09	1.83E-13	1.00E-11	0.00E+00	1.40E-11	0.00E+00	9.99E-10	3.03E-09	1.84E-10		7.47E-09	38.3%
CFI	1.82E-10	4.72E-14	3.55E-12	0.00E+00	1.15E-13	0.00E+00	1.08E-12	1.70E-13	1.00E-14		1.89E-10	1.0%
CFV	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00		0.00E+00	0.0%
CFL	1.34E-13	2.35E-14	5.20E-14	9.71E-15	0.00E+00	2.42E-14	1.04E-14	8.60E-14	5.44E-15		3.45E-13	0.0%
CI	1.32E-10	5.83E-10	7.67E-11	4.60E-12	1.32E-12	6.33E-12	2.66E-11	3.62E-10	1.40E-10		1.33E-09	6.8%
BP	0.00E+00	0.00E+00	0.00E+00	2.04E-09	6.05E-10	4.08E-09	0.00E+00	0.00E+00	3.78E-09		1.05E-08	53.9%
LRF	3.54E-09	5.83E-10	9.23E-11	2.05E-09	6.21E-10	4.08E-09	1.03E-09	3.40E-09	4.11E-09		1.95E-08	100.0%
	18.2%	3.0%	0.5%	10.5%	3.2%	20.9%	5.3%	17.4%	21.1%			
CNTM-EFF	95.6%	97.6%	99.8%	59.1%	57.9%	7.8%	89.7%	94.3%	56.9%		91.9%	

AP1000 Dominant LRF Sequences

AP1000

CONTRIBUTION OF PDS TO LRF

CET SEQ	REL CAT	PDS	FREQ	% CDF	Cumulative % LRF	Sequence Description
23	BP	3A	4.08E-09	1.7	20.9	Containment Bypass. ATWS
23	BP	6	3.78E-09	1.6	40.3	Containment Bypass. SGTR
21	CFE	3BE	2.67E-09	1.1	54.0	Sump Flooding Fails
21	CFE	3D	2.05E-09	0.85	64.5	Sump Flooding Fails
23	BP	1A	2.04E-09	0.85	75.0	Containment Bypass . Induced SGTR
10	CFE	3C	9.97E-10	0.41	80.1	Vessel Failure Initiating Event
12	CFE	3D	9.71E-10	0.40	85.1	Core Reflooding Fails: Diffusion Flame
23	BP	1AP	6.05E-10	0.25	88.2	Containment Bypass
22	CI	3BL	5.83E-10	0.24	91.2	Containment Isolation Fails
6	CFE	3BE	4.75E-10	0.20	93.6	Hydrogen Ignition Fails: Early DDT
22	CI	3D	3.62E-10	0.15	95.5	Containment Isolation Fails

AP1000 LRF Sensitivity Analyses

Sensitivity	Result
No Credit Taken for DP Node for PDS-6	LRF becomes 2.49 E-08/yr, with a CCFP of 10.3 percent
Lesser Reliability for Containment Isolation	LRF becomes 4.05 E-08/yr, with a CCFP of 16.8 percent
Lesser Reliability for Hydrogen Igniters	The LRF becomes 2.31E-08/yr, with CCFP of 9.6 percent
Lesser Reliability for PCS	The LRF becomes 1.97E-08/yr, with CCFP of 8.2 percent
No Credit for Depressurization for High Pressure PDS	The LRF is 2.91E-08/year, with CCFP of 12.1 percent

AP1000 LRF Sensitivity Analysis

Sensitivity	Result
Set PDS-3C Vessel Failure Probability to 1.0	The LRF is 2.85E-08/yr, with a CCFP of 11.8 percent
Set 3D and 1AP Diffusion Flame and Detonation Failure Probability to 1.0	The LRF becomes 7.66E-08/yr, with CCFP of 31.8 percent

AP1000 LRF Importance Analyses

Table 43-10

CET EVENT TREE NODE IMPORTANCES

CET NODE	LRF (per year)	Containment Failure Prob	Containment Effectiveness	Node Failed in Following PDS
BASE LRF	1.95E-08	8.1%	91.9%	N/A
DP RCS Depressurization	2.91E-08	12.1%	87.9%	1A, 1AP, 3A, and 6 set to 1.0.
IS Containment Isolation	2.41E-07	100.0%	0.0%	all PDS set to 1.0
IR Cavity Flooding	1.50E-07	65.6%	34.4%	3BE, 3D, 6 set to 1.0
RFL Core Reflooding	1.91E-08	7.9%	92.1%	3BE set to 1.0
VF Vessel Failure	2.85E-08	11.8%	88.2%	3C set to 1.0

AP1000 LRF Importance Analyses

Table 43-10

CET EVENT TREE NODE IMPORTANCES

CET NODE	LRF (per year)	Containment Failure Prob	Containment Effectiveness	Node Failed in Following PDS
BASE LRF	1.95E-08	8.1%	91.9%	N/A
PC PCS Failure	2.41E-07	100.0%	0.0%	all PDS set to 1.0
IG Hydrogen Igniter Failure	6.28E-08	26.0%	74.0%	all PDS set to 1.0
DF Diffusion Flame	1.41E-07	58.3%	41.7%	3D and 1AP set to 1.0; 3BE set to 0.8535 (85% of 3BE are SI-LB events where DF may exist. Diffusion flame does not apply to other PDS.
DTE Early DDT	2.16E-08	9.0%	91.0%	all PDS set to 1.0
DFG Hydrogen Deflagration	2.17E-08	9.0%	91.0%	all PDS set to 1.0
DTI Intermediate DDT	2.17E-08	9.0%	91.0%	all PDS set to 1.0

AP1000 Level 2 Conclusions and Insights

- **LRF is 1.95×10^{-8} per reactor year.**
  - Goal is LRF less than 1×10^{-6} per reactor year
- **Overall containment effectiveness (CE) is 92%**
- **PDS-3A (ATWS) has lowest CE.**
- **CE for PDS-6 (SGTR) is 57%.**
  - If all SGTR sequences go to bypass overall CE = 89.7%

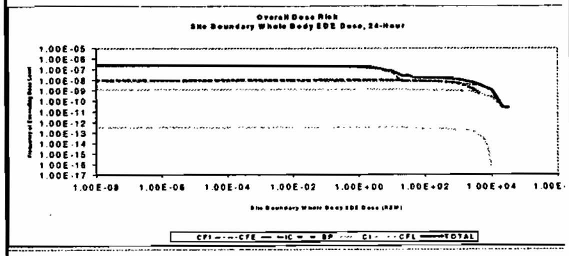
AP1000 Level 2 Conclusions and Insights

- **LRF is not sensitive to the reliability of the hydrogen igniters, but if the igniters are assumed to be failed (probability of 1.0), the CE drops to 74%**
- **If the DF failure probability is 1.0 for all 1AP and 3D sequences, the CE is 84.5%. LRF increase by a factor of 4.**

AP1000 Level 3 PRA

- AP1000 specific source terms calculated with MAAP4
- MACCS2 v. 1.12 used to calculate doses
- Goal
  - Frequency of site boundary whole body dose >25 rem EDE less than 1.0×10^{-6} per reactor-year.

24 Hour Site Boundary Whole Body EDE Dose



AP1000 In-Vessel Retention of Molten Core Debris

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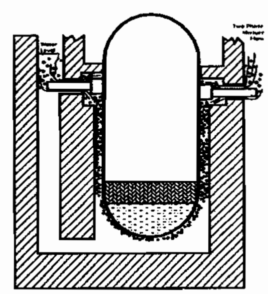
IVR via External Cooling of Reactor Vessel



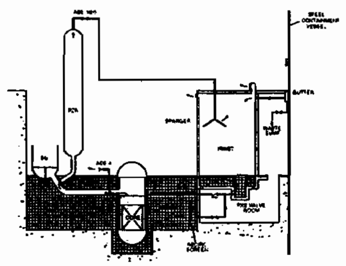
Passive Plant Features Promote IVR

- Reliable post-accident RCS depressurization
  - low stresses on reactor vessel
- No RPV lower head penetrations
  - creep failure of lower head only failure mechanism
- Reactor vessel submerged in water post-accident
  - automatic or manual flooding of cavity with IRWST water

Passive Plant Features Promote IVR

- Core support plate sits low in lower plenum
  - lower plenum debris contacts and melts RPV internals
  - thick metal layer
  - no focusing effect of metal layer
- Reactor vessel insulation designed to promote IVR
  - standoff from reactor vessel
  - provides flowpath for cooling

AP1000 Containment Flooding



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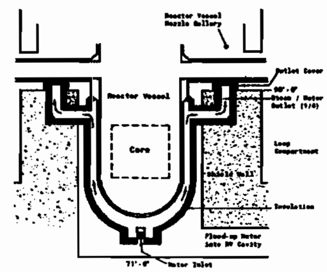
Reactor Vessel Insulation Promotes IVR



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AP600 IVR Assessment

- **Risk Oriented Accident Analysis Methodology**
  - Analysis
  - Test Program
  - Peer Review
- **DOE/ID 10460, "In-Vessel Retention and Coolability of a Core Melt," Theofanous, et. al.**
- **ACOPO test to investigate natural convection heat transfer from debris to vessel at Ra' $\leq 10^{16}$**
- **ULPU test to investigate CHF on external vessel surface**

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AP600 IVR Assessment

- **Exceeding Critical Heat Flux (CHF) is limiting vessel failure criterion**
  - heat flux to vessel wall < CHF is success
- **Steady-state, two-layer debris configuration presents limiting challenge to the reactor vessel**
  - metal over oxide debris bed in lower plenum
- **Large margin to vessel failure**
  - RCS depressurized
  - cavity flooded

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AP1000 vs. AP600

- **Designs are similar**
- **Changes to the AP1000 that potentially impact IVR**
  - power is increased from 1933 to 3400 MWt.
  - 157 14-ft fuel assemblies.
  - core shroud instead of reflector
  - lower core support plate is 1" thicker

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Implement IVR for AP1000

- **Increase critical heat flux (CHF) at vessel surface to maintain margin to failure**
- **Demonstrate thermal failure remains the limiting failure mechanism for increased heat removal**
- **Investigate in-vessel melt progression**
- **Demonstrate that the heat load correlations scale appropriately to the AP1000.**
- **Quantify the margin to failure**

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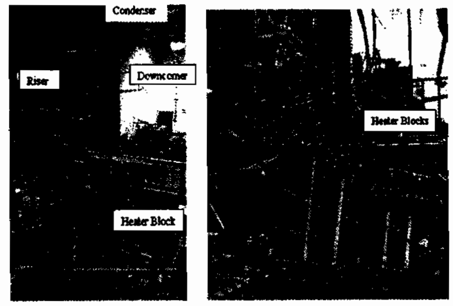
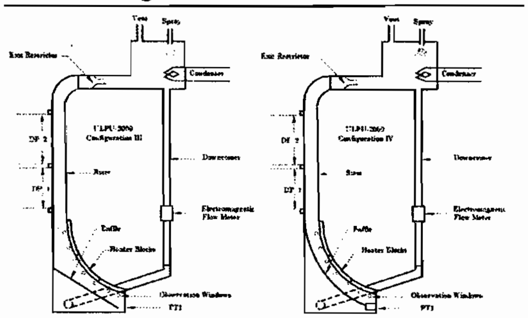
Increase CHF

- **UPLU Configuration IV Test - UC Santa Barbara**
  - Lower Head slice geometry at full scale radius
  - Full scale simulation via power shaping
  - Models AP600 entrance and venting restriction
  - movable baffle, fixed at 90°
- **Tests Completed**
  - examine lower head baffle geometry impact
  - examine water level effects

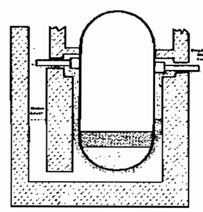
ULPU Facility



ULPU Configurations

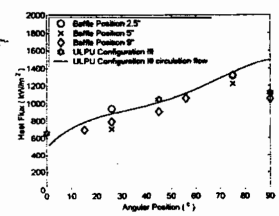


Effect of water level during IVR



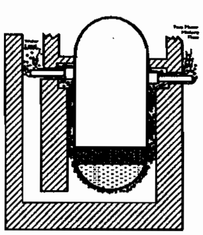
Low Water Level

Pool Boiling



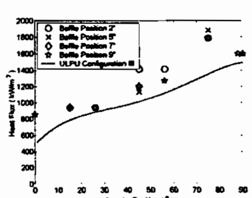
ULPU-IV Data Compared to ULPU-III correlation

Effect of Water Level during IVR



High Water Level

2-Phase Nat Circ



ULPU-IV Data Compared to ULPU-III correlation

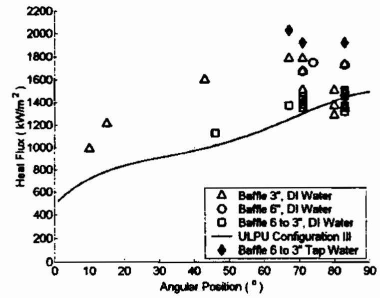
ULPU Configuration IV Conclusions

- **ULPU Configuration IV test report submitted to the NRC**
  - DCP/NRC1510 dated 6/6/2002
- **CHF can be increased sufficiently to provide margin for AP1000**
  - channel flow around lower head
  - high water level for 2-phase natural circulation
- **Adverse exit effect at top of baffle that reduced local CHF**
  - resolved by ULPU Configuration V tests

ULPU Configuration V

- **Funded by DOE International-NERI Program**
- **AP1000 specific inlet/exit modeling**
- **Adjustable baffle design**
- **Additional aspects investigated**
  - surface effects
  - water chemistry
  - exit phenomena
- **Optimization of reactor vessel insulation/water circulation flow path**

ULPU Configuration V



ULPU Configuration V

- **Tests performed show AP1000 CHF can easily be met with margin.**
- **Exit phenomena is negligible**
- **Optimum surface is unpainted and oxidized**

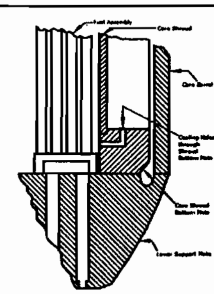
Vessel Structural Failure

- **Confirm that thermal failure criterion is still limiting for increased heat load**
  - large margin to structural failure
- **At a bounding heat flux of 2000 kW/m², vessel thickness is 36 times the thickness required to carry dead load**
- **Thermal failure criterion is still limiting**

In-Vessel Melt Progression

- **AP600 in-vessel melt progression influenced by low power density and radial reflector**
  - downward relocation to lower plenum blocked
  - sideward failure through reflector into dead ended region
  - core barrel failure
  - quickly contacts support plate to mitigate focusing effect
- **AP1000 has higher power density and a core shroud instead of a radial reflector**

AP1000 Core Shroud

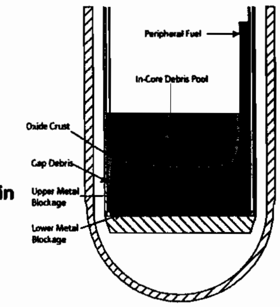


Modeling of Core and Internals Heatup

- **Accident Sequence**
  - fully depressurized
  - earliest core uncovery is conservative (Large LOCA)
  - no vessel reflood
  - conservatively assumed spurious ADS stage 4 opening
- **MAAP4**
- **Finite Difference Model of core and internals**
  - using uncovery timing from MAAP4
- **Hand calculation of core heat up and melting**

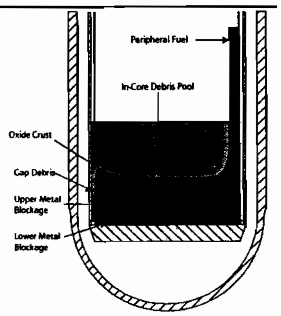
Formation of In-Core Debris Pool

- **Upper core shroud melts prior to fuel melting**
- **Upper core barrel significantly thinned and overheated**
- **Most peripheral fuel assemblies initially remain intact**
- **Oxide blockage at ~1 m above bottom of fuel**



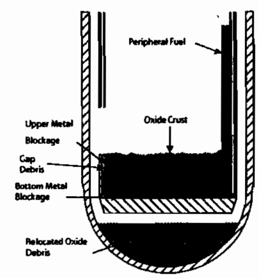
Formation of In-Core Debris Pool

- **Downward relocation pathway blocked by frozen metal and oxide**
- **Gap between shroud and barrel fills with debris**
- **In-core debris pool contact with core barrel**
- **Core barrel fails sidewards near upper surface of pool**

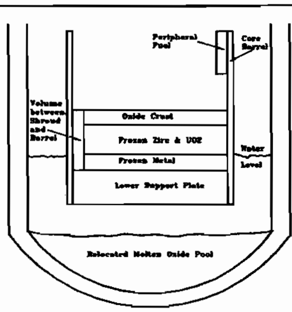


Initial Relocation to Lower Plenum

- **6.2 m^3 of UO_2 and ZrO_2**
- **8 m^3 below lower core support plate**
  - creep of core barrel
- **Occurs at 6000 seconds**
- **Duration of initial relocation is ~500 seconds**
  - ablation of core barrel by relocating debris



Subsequent Relocation of Debris



Subsequent Relocation of Debris

- **Success Criterion**
  - debris contacts lower support plate before dry out
  - mitigates focusing effect
- **Debris contact occurs 6717 seconds**
- **Lower plenum dry out occurs at 6888 seconds**
  - calculated conservatively assuming heat load from 8 m^3 of debris
- **Transient debris configurations are water cooled**
- **Focusing effect is mitigated by inclusion of lower support plate and shroud in metal layer**

RASPLAV and MASCA Tests

- **Addressed in RAI 720.047**
- **In-vessel materials testing**
- **Prototypical materials**
- **Non-prototypic conditions**
  - Rayleigh number too low
  - Heat fluxes too high
  - Ratio of masses not applicable
- **Tests do not contradict position on IVR**

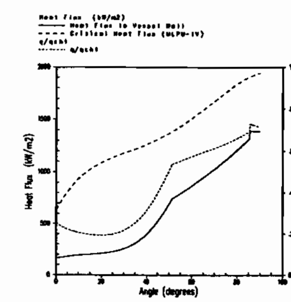
Application of Heat Transfer Correlations

- **Oxide Debris Pool Heat Transfer (Ra′~ 10^{16})**
  - to vessel wall and upward to metal layer
    - Angelini-Theofanous correlations (Ra′≤ 10^{16})
- **Metal Layer Heat Transfer (Ra ~ 10^{10})**
  - to vessel wall
    - Churchill-Chu correlation (Ra < 10^{12})
  - from oxide layer and to top surface
    - Globe-Dropkin correlation ($3x10^5$ < Ra < $7x10^9$)
    - modest extrapolation for thick metal layer

Quantification of Thermal Loads

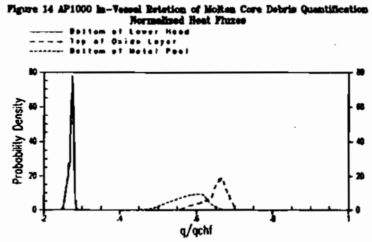
- **Calculate AP1000 thermal loading using DOE/ID 10460 methodology**
- **Use ULPU Configuration IV Critical Heat Flux**
- **Input parameters based on AP1000 power level, geometry of reactor vessel and masses of core materials**
- **AP1000 probability distributions for uncertain input parameters**
  - fraction of cladding oxidized during melt
  - mass of stainless steel in debris
  - time with respect to shutdown (decay heat)

AP1000 Bounding IVR Calculation



Results of Thermal Load Quantification

Figure 14 AP1000 In-Vessel Retention of Molten Core Debris Quantification Normalized Heat Fluxes



Conclusions

- **IVR is successfully demonstrated for AP1000 with margin to failure similar to AP600**
  - CHF is increased
  - ULPU Configuration V has greater margins
- **Insulation geometry and structure are important**
  - forms baffle to direct water smoothly over lower head
- **Two-phase natural circulation is required**
  - deep flooding of the reactor cavity is needed

AP1000

AP1000 Severe Accident Phenomenological Evaluations

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AP1000

AP1000 Severe Accident Studies

- Support Level 2 PRA Quantification
- SECY-93-087 Deterministic Requirements

AP1000

Severe Accident Phenomena

- In-vessel fuel coolant interaction
- High Pressure Core Damage
  - Induced failure of steam generator tubes
  - High pressure melt ejection / direct containment heating
  - Melt attack on the containment pressure boundary
- In-vessel hydrogen generation
- Hydrogen deflagration and detonation
- Diffusion flame overheating containment shell

AP1000

Severe Accident Phenomena (continued)

- Containment overpressure by decay heat
- Reactor vessel integrity
- Ex-vessel fuel coolant interaction
- Core-concrete interaction
- Equipment survivability

AP1000

In-Vessel Fuel-Coolant Interaction

- Lower head integrity under steam explosion loads
- Steam Explosion Assessment for AP600
  - large margin to failure
- AP600 conclusions are extended to AP1000
- AP1000 conditions
  - similar debris relocation pathway
  - similar molten debris mass flow rate
  - same lower plenum geometry

AP1000

High Pressure Core Damage

- Severe Accident Issues
  - Induced failure of steam generator tubes
  - High Pressure Melt Ejection/Direct Containment Heating
  - Melt attack on containment pressure boundary
- Prevention
  - Diverse RCS depressurization capability
    - two train, four stage ADS
    - PRHR Heat Exchanger
  - High pressure core damage frequency < 5% total CDF

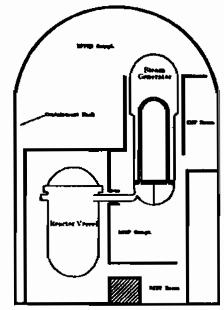
High Pressure Core Damage (continued)

AP1000

- **Mitigation**
  - operator actions to recover ADS, PRHR
  - potential for hot leg or surge line creep rupture
  - torturous pathway from reactor cavity to upper compartment
- **PRA Treatment**
  - assess likelihood of operator actions to depressurize RCS
  - assume induced tube rupture and containment bypass
- **Success Criterion**
  - 2 of 4 ADS stage 4 valves open

HPME Debris Retention

AP1000



Hydrogen Generation

AP1000

- **In-vessel hydrogen generation**
  - cladding oxidation during core uncovery
- **Ex-vessel hydrogen generation**
  - prevented by in-vessel retention of core debris
  - containment pressurization during core-concrete interaction

Hydrogen Combustion

AP1000

- **Threat to containment integrity**
  - locally high temperature (diffusion flame)
  - overpressure (deflagration)
  - dynamic loading (detonation)
- **Prevention**
  - low core damage frequency
- **Mitigation**
  - passive autocatalytic recombiners (PARs)
  - hydrogen igniters

Treatment of Hydrogen in PRA

AP1000

- **In-vessel releases only**
  - vessel failure is conservatively assumed to fail containment early
- **Three scenarios**
  - no reactor vessel reflood
  - early reactor vessel reflood (core relatively intact)
  - late reactor vessel reflood (core geometry lost)

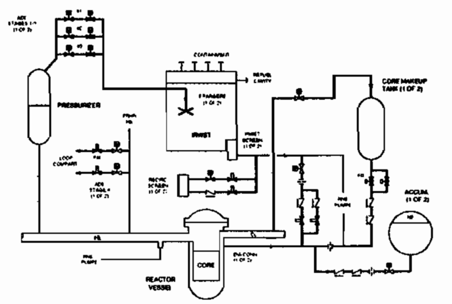
Treatment of Hydrogen in PRA

AP1000

- **Diffusion Flame**
  - postulated at IRWST vents, PXS compt exits
  - mitigated by ADS stage 4
  - preferential release away from containment walls
- **Success Criterion**
  - Hydrogen vented away from containment shell
    - ADS stage 4
    - IRWST pipe vents
    - PXS compartment hatches

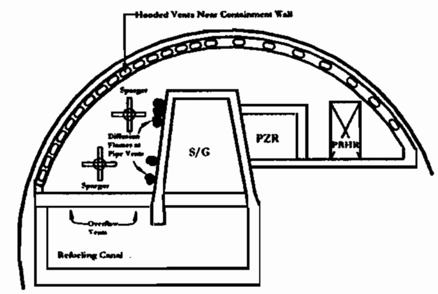
AP1000

Hydrogen Release Through Stage 4



AP1000

Preferential Venting from IRWST



AP1000

Early Detonation

- During hydrogen release from RCS
- Containment not well mixed
  - locally high hydrogen concentrations
- Mitigated by hydrogen igniters
- Deflagration to Detonation Transition (DDT)
  - no source for direct ignition
- Probabilities for early DDT based on AP600
  - RAI showed approach was conservative

AP1000

Sherman - Berman Methodology

- Assign Probability of Deflagration to Detonation Transition
  - flame acceleration
- Function of Gas Mixture and Compartment Geometry
- Detonation cell widths
  - equivalence ratio (measure of mixture with respect to stoichiometry)
  - steam concentration
- Compartment Geometry Classes

AP1000

Global Hydrogen Deflagration

- Intermediate Time Frame (<24 hours)
- Containment well-mixed
- Mitigated by igniters
- Adiabatic peak pressure calculation
- Performed for three general accident scenarios
  - no reflood
  - early reflood
  - late reflood

AP1000

Global Hydrogen Deflagration

- Input probability distributions for each scenario
  - mass of hydrogen generated (cladding oxidation)
  - containment pressure at ignition
- Containment fragility success criterion
  - probability of containment failure vs. pressure
- Probability of containment failure

Global Hydrogen Deflagration

AP1000

- **Safety Margin Basis Calculation**
- **Deterministic Calculation**
  - 100% cladding reaction
  - containment pressure at 55% steam concentration
  - adiabatic peak pressure calculation
- **Peak pressure is 90 psig**
- **Containment Service Level C is 91 psig**

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Intermediate Detonation

AP1000

- **Less than 24 hours after core damage**
- **Containment well mixed**
- **Deflagration to Detonation Transition**
- **Sherman-Berman Mixture Class Probabilities**
  - calculated from hydrogen mass and containment pressure probability distributions
  - air-steam-hydrogen mixture classes
  - dry air-hydrogen mixture classes for CMT room
    - resolves uncertainty with respect to steam stratification
- **Output probability of DDT**

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Containment Overpressure by Decay Heat

AP1000

- **Mitigated by passive containment cooling water**
- **PCS water cooling is more reliable than AP600**
  - added third diverse actuation path
- **Success criterion**
  - at least 1 of 3 PCS actuation paths operates

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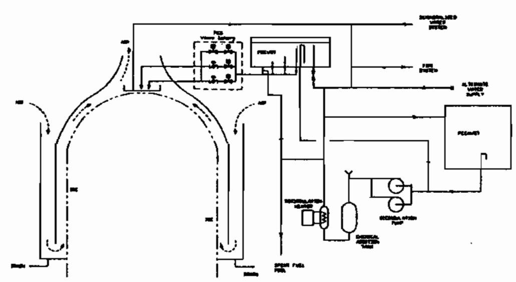
PCS Water Delivery

AP1000



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Dry PCS Cooling

AP1000

- **Dry PCS cooling is sufficient to prevent containment failure for 24 hours.**
- **Success Criterion**
  - containment fragility probability distribution
- **Nominal conditions**
  - 0.0 failure probability in 24 hours
- **Conservative conditions**
  - 0.02 failure probability in 24 hrs
    - ANS 79 decay heat + 2 sigma uncertainty
    - Outside Temperature = 115 F

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Reactor Vessel Integrity

AP1000

- **Vessel integrity maintained via external cooling**
  - cavity fully flooded
- **Vessel Failure Modes**
  - Global failure of lower head (hinged failure)
  - Local failure of lower head
- **Containment conditions**
  - water level at 83' elevation (loop compartment floor)

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Ex-Vessel Steam Explosion

- Prevented by in-vessel retention of core debris
- AP600 assessment
  - hinged failure of the lower head
  - partially flooded cavity
- Similar vessel failure mode for AP1000
- Similar geometry
  - AP1000 vessel is closer to the floor
- AP600 conclusions are extended to the AP1000

Core-Concrete Interaction

- Prevented by in-vessel retention of core debris
- Vessel failure modes
  - hinged failure
  - localized failure
- Concrete Types
  - Limestone
  - Basaltic
- Success Criteria
  - Basemat intact for 24 hours

Core-Concrete Interaction

- MAAP4 calculation of CCI
- Minimum time to basemat failure
  - 2.8 days to melt-through basemat
- Basemat melt-through occurs before containment overpressurization by non-condensable gases

Equipment Survivability

- Identified actions to achieve controlled stable state
- Defined time frames for each action
- Identified equipment and instruments needed for each action
- Determine bounding environments (MAAP4)
- Show reasonable assurance that equipment will perform when needed

Summary of PRA Results and Insights

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Comparison of AP600 and AP1000 PRA Results

Scope	AP600	AP1000
Level 1 At-Power Internal Initiating Events	Quantification Performed CDF = 1.7E-07 Several additional cases quantified in response to NRC RAIs	Quantification Performed CDF = 2.4E-07 AP600 additional cases incorporated into the model
Level 2 At-Power Internal Initiating Events	Quantification Performed LRF = 1.8E-08 Containment Effectiveness = 89.5%	Quantification Performed LRF = 2.0E-08 Containment Effectiveness = 91.8%
Level 3 At-Power Internal Initiating Events	Quantification Performed	Quantification Performed
Internal Fire Events	Conservative (via focused PRA) Quantification Performed CDF = 6.5E-07 (internal) CDF = 3.5E-07 (shutdown)	Quantification performed CDF = 5.61E-08
Internal Flooding Events	Quantification Performed CDF = 2.2E-10	Quantification Performed CDF = 8.8E-10
Shutdown Events	Quantification Performed for Level 1 and 2 CDF = 1.0E-07 LRF = 1.5 E-08 Several additional cases quantified in response to NRC RAIs	Quantitative Evaluation Performed CDF = 1.2E-07 LRF = 2.0E-08

SUMMARY OF RESULTS

- **The AP1000 PRA results show that**
  - The very low risk of the AP600 has been maintained in the AP1000
  - The AP1000 PRA meets the US NRC safety goals with significant margin

Summary of PRA Results

- The total mean core damage frequency is at least two orders of magnitude smaller than those for existing pressurized water reactors
- The total plant severe release frequency is another order of magnitude smaller than that of the core damage frequency; that places such a release frequency in the range of incredible events
- A bounding analysis of the core damage due to internal fire and internal flooding events shows that these two categories of internal events are much lower for AP1000 than are calculated for currently operating plants

Summary of PRA Results

- The severe release frequency is about equal for at-power and shutdown events. The severe release frequency as a percentage of core damage frequency is 8 percent for at-power events and 17 percent for shutdown events
- The results show that the design goals of low core damage frequency and low severe release frequency have been met. The AP1000 frequencies are lower than the NRC and ALWR URD goals set for new plant designs.
- The results show the effectiveness of passive systems in mitigating severe accidents and reflect the reduced dependence of AP1000 on nonsafety systems and human actions

Most Important Level 1 Insights

- The AP1000 design benefits from the high level of redundancy and diversity of the passive safety-related systems; passive safety systems have been shown to be highly reliable, their designs are simple so that a limited number of components are required to function
- AP1000 is less dependent on nonsafety-related systems; the nonsafety-related support systems (ac power, component cooling water, service water, and air) have a limited role in the plant risk profile because the passive safety-related systems do not require cooling water or ac power
- AP1000 is less dependent on human actions; the AP1000 meets the NRC safety goal even when no credit is taken for operator actions

Most Important Level 1 Insights

- **The core damage and large release frequencies are low despite the conservative assumptions made in specifying success criteria for the passive systems. The success criteria have been developed in a more systematic, rigorous manner than typical PRA success criteria. The baseline success criteria are bounding cases for a large number of PRA success sequences. The baseline success sequences, in most cases, have been defined with:**
  - **worst (i.e., the most limiting) break size and location for a given initiating event**
  - **worst automatic depressurization system (ADS) assumption in the success criterion**
  - **worst number of core makeup tanks (CMT) and accumulators**
  - **worst containment conditions for in-containment refueling water storage tank (IRWST) gravity injection.**
  - **Many less-limiting sequences are therefore represented by a baseline success criteria.**

Most Important Level 1 Insights

- Single system or component failures are not overly important due to the redundancy and diversity of safety-related systems in the design. For example, the following lines of defense are available for reactor coolant system (RCS) makeup:
  - chemical and volume control system
  - core makeup tanks
  - partial automatic depressurization system in combination with normal residual heat removal
  - full automatic depressurization system with accumulators and in-containment refueling water storage tank
  - full automatic depressurization system with core makeup tanks and in-containment refueling water storage tank

Most Important Level 1 Insights

- Typical current PRA dominant initiating events are significantly less important for the AP1000 - for example:
  - **Reactor coolant pump (RCP) seal loss-of-coolant accident (LOCA) event has been eliminated as a core damage initiator since AP1000 uses canned motor reactor coolant pumps which do not have seals**
  - **Station blackout and loss of offsite power (LOOP) event is a minor contributor to AP1000 since the passive safety-related systems do not require the support of ac power**
- Passive safety-related systems are available in all shutdown modes
  - **Planned maintenance of passive features is only performed during shutdown modes when that feature is not risk important**
  - **Planned maintenance of nonsafety-related defense-in-depth features used during shutdown is performed at power**

Most Important Level 1 Insights

- The AP1000 passive containment cooling design is highly robust. Air cooling alone can prevent containment failure, although the design has other lines of defense for containment cooling such as fan coolers and alternate sources of passive containment cooling water
- The potential for containment isolation and containment bypass is lessened by having fewer penetrations to allow fission product release; all normally open and risk important penetrations are fail-closed, thus eliminating the dependence on instrumentation and control (I&C) and batteries
- The reactor vessel lower head has no vessel penetrations, thus eliminating penetration failure as a potential vessel failure mode
- The potential for the spreading of fires and floods to safety-related equipment is significantly reduced by the AP1000 layout

Most Important Level 2 Insights

- The containment effectiveness for AP1000 is over 90%, which provides an order of magnitude decrease from CDF to LRF. Since the results already includes CDF sequences that directly bypass the containment, the containment effectiveness for remaining sequences is actually much better. For example, for 5 (3BE,3BL,3BR,3C,3D) of the nine accident classes studied, the containment effectiveness ranges from 89.7 to 99.8%
- Preventing the relocation of molten core debris to the containment eliminates the occurrence of several severe accident phenomena, such as ex-vessel fuel-coolant interactions and core-concrete interaction, which may threaten the containment integrity. Therefore, AP1000, through the prevention of core debris relocation to the containment, significantly reduces the likelihood of containment failure

Most Important Level 2 Insights

- A frequency of 1.0e-08/year has been assigned to the vessel failure initiating event (accident class 3C). In 90% of these events, the vessel is assumed to undergo failures that will be above the beltline: in which case the molten core could be cooled and containment would not be challenged. In the remaining 10% of the cases, the failure is assumed to be below the pressure vessel beltline, whereby the molten core would drop into the containment. In this case, it is conservatively assumed that the containment would fail. A sensitivity analysis is made where by 100% of the failures would be below the beltline. The result shows that the containment effectiveness drops to 88.2%. This change is not significant, and the assumptions behind the case are very conservative.

Most Important Level 2 Insights

- The LRF results are sensitive to failure of hydrogen ignitors. If no credit is taken for hydrogen ignitors, the containment effectiveness drops to 74%.
- However, LRF is not very sensitive to the reliability of hydrogen ignitors; if IG reliability is assumed to be degraded (0.1) across the board for all accident classes, the containment effectiveness becomes 90.5%, which is an insignificant change from the base case.

Most Important Level 2 Insights

- The LRF is dominated (53.9%) by containment failures or bypasses due to SGTR, and unmitigated high-RCS-pressure core damage sequences, classified as BP. The remaining containment failures are dominated by an early containment failure due to reactor cavity flooding failure.
- The LRF is not very sensitive to the reliability of PCS; if PCS reliability is assumed to be 0.001 across the board for all accident classes, the LRF becomes 1.97E-08, which is an insignificant change from the base case.
- The LRF is sensitive to the operator action to flood the reactor cavity in a short time following core damage. This operator action has been moved to the beginning of ERG AFR.C-1 to increase its success likelihood.

Most Important Level 2 Insights

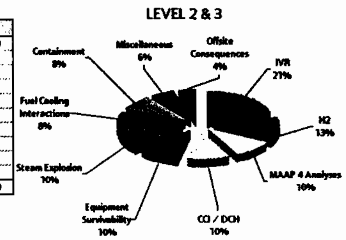
AP1000

- The potential for a release of radioactive materials to the environment is very small. This is largely due to the very small core damage frequency and very small release frequency. The containment design provides enhanced deposition of core materials that could be released in a severe accident, and the passive containment cooling system minimizes the energy available to expel such materials from the containment.
- Deterministic analyses of severe accident phenomena show that AP1000 features are effective in maintaining containment integrity

Summary of RAI on AP1000 Level 2/3 PRA

AP1000

RAIs on PRA Level 2 & 3	
IVR	10
H2 generation, mixing & combustion	6
MAAP 4 Analyses	5
CCI / DCH	5
Equipment Survivability	5
In-vessel & Ex-vessel Steam Explosion	5
Fuel Cooling Interactions	4
Containment (failure, isolation,..)	4
Miscellaneous	3
Offsite Consequences	2
TOTAL	49



AP1000 PRA Report Updates Included with RAI Responses

AP1000

- IVR of Core Melt Debris Analyses
- Revision of PRA Chapter 34 and 39
- Revision of DCD Section 19.39

- Severe Accident Analyses
- Fission-Product Source Term Analyses
- Revision of PRA Chapter 34 and 45
- Revision of DCD Section 19.34

AP1000 PRA Report Updates Included with RAI Responses

AP1000

- H2 generation, mixing and combustion analyses
- Revision of PRA Chapter 41
- Revision of DCD Section 19.41

- MAAP 4 Analyses (Environment)
- Revision of PRA Appendix D

- Offsite dose risk quantification
- Revision of PRA Chapter 49 and 59
- Revision of DCD Section 19.59

AP1000 PRA Report Updates Included with RAI Responses

AP1000

- Revision of PRA Chapter 12 (IRWST CCF)
- Revision of PRA Chapter 29 (IRWST CCF)
- Revision of PRA Chapter 30 (Time window for operator action)
- Revision of PRA Chapter 35 (CET)
- Revision of PRA Chapter 57 (Fire)
- Revision of PRA Chapter 59 (Insights, Fire)
- Revision of DCD Appendix 19E (Shutdown)

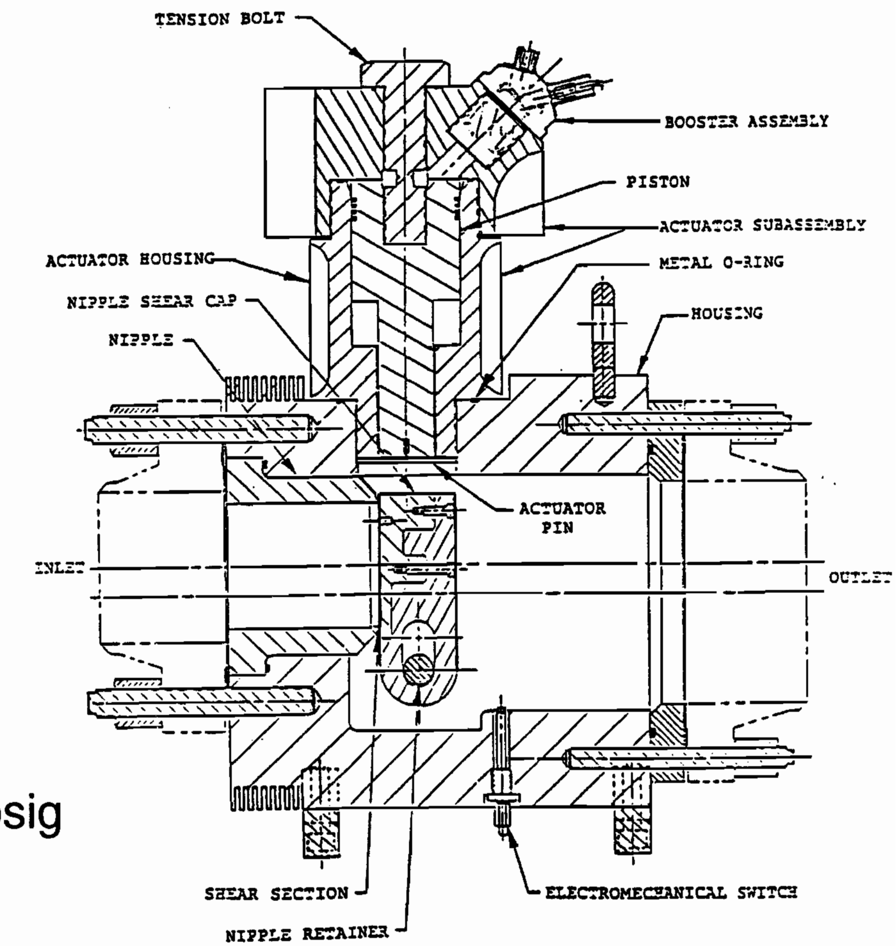
Summary

AP1000

- **AP1000 PRA Report**
  - Complete AP1000-Specific PRA
    - Sufficient for AP1000 Design Certification
  - Demonstrates that the AP1000 meets the US NRC safety goals with significant margin
  - Revision 1 will be issued to include W responses to staff RAI: February 2003

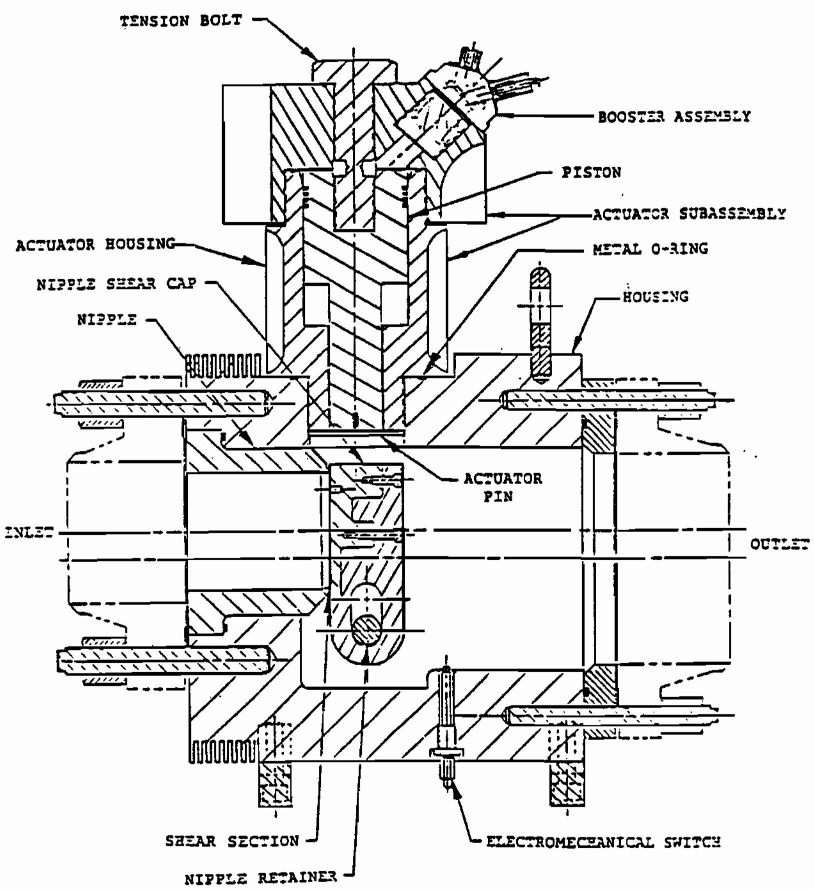
ADS Stage 4 Squib Valve

- **Controls**
  - Two stage “arm” / “fire” circuit prevents spurious opening
  - Three ignitors provided in each valve
    - 2 wired to different PMS divisions
    - 1 wired to DAS
  - Auto opening (PMS) requires
    - SI signal (2/4) and
    - CMT low 1 (2/4) signal and
    - CMT low 2 (2/4 signal and
    - RCS pres < 1300 psig
  - Manual opening requires
    - PLS - 2 step switch & RCS < 1300 psig
    - PMS - 2/2 dedicated switches
    - DAS - 2/2 dedicated switches

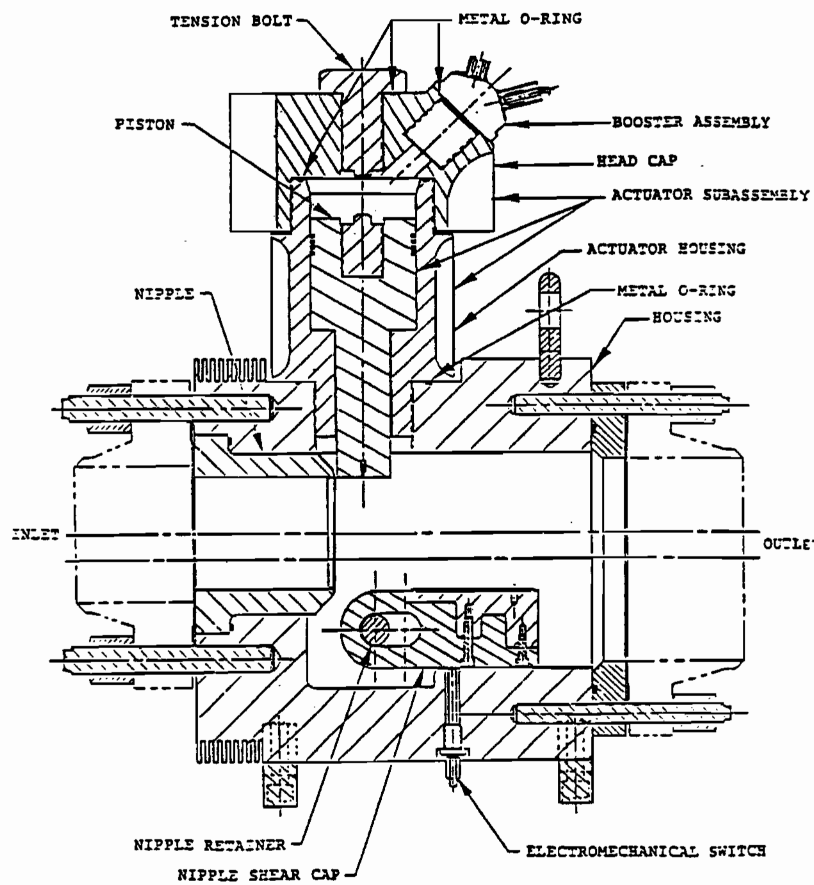


ADS Stage 4 Squib, Closed / Open

Closed



Open



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NRC STAFF REVIEW OF AP1000 LEVEL1 PRA
INTERNAL & EXTERNAL EVENTS AT POWER OPERATION

ACRS Subcommittee on Reliability and Probabilistic Risk Assessment
January 23, 2003

Nicholas Saltos
NRR/DSSA/SPSB

AP1000 PRA REVIEW - - - MAJOR OBJECTIVES

- *ENSURE PRA QUALITY COMMENSURATE WITH ITS INTENDED USE, SUCH AS*
  - *Gain insights about the design*
  - *Support the design and certification processes*
- *ENSURE PROPER INTERPRETATION AND USE OF PRA RESULTS FOR DECISION MAKING IN THE CERTIFICATION PROCESS, SUCH AS*
  - *Identify design and/or operational changes to address weaknesses*
  - *Identify "certification requirements," such as ITAACs*
  - *Determine appropriate regulatory treatment of non-safety systems (RTNSS)*
  - *Determine the risk significance of raised issues*

AP1000 PRA REVIEW - - - APPROACH

- *RELIANCE ON SIMILARITY OF AP1000 TO AP600 CERTIFIED DESIGN TO REDUCE REVIEW EFFORT*
  - *Same system functions, spatial arrangements and capabilities*
  - *The AP1000 PRA uses the AP600 PRA as the starting point*

- *IDENTIFICATION OF DESIGN DIFFERENCES BETWEEN AP1000 AND AP600 HAVING AN IMPACT ON PRA MODELS*
  - *Major differences are due to the power uprate*
  - *Several other minor but potentially significant differences*
  - *Identification of AP1000 PRA areas for review*

- *IDENTIFICATION OF ADDITIONAL DIFFERENCES BETWEEN AP1000 AND AP600 PRAs THAT ARE NOT DUE TO DESIGN DIFFERENCES*

- *FOCUS REVIEW ON IMPACT OF CHANGES ON IMPORTANT ISSUES IDENTIFIED DURING THE AP600 PRA REVIEW*

AP1000 PRA REVIEW -- LEVEL 1 PRA MAJOR ISSUES (OPERATION AT POWER)

- *THERMAL-HYDRAULIC (T/H) UNCERTAINTY/SUCCESS CRITERIA*

- *FIRE-INDUCED SPURIOUS ACTUATION OF ADS SQUIB VALVES*

- *IDENTIFICATION OF "CERTIFICATION REQUIREMENTS," SUCH AS ITAACs AND RTNSS*
    - *Resulting from design differences with respect to AP600*
    - *Could change according to the resolution of outstanding issues*

AP1000 PRA REVIEW - - T/H UNCERTAINTY

ISSUE DESCRIPTION

- *Passive systems rely on small driving forces. The uncertainty in the values of such driving forces can be of comparable magnitude to the predicted values themselves. When T/H uncertainties are considered, "success" accident sequences may actually lead to core damage*

- *This issue was addressed in the AP600 PRA by a risk-based bounding approach which uses conservative assumptions for key T/H parameters:*

  - *Identification of "low T/H margin risk significant" accident scenarios*

  - *Use of DBA computer codes to bound T/H uncertainty*

- *Such an approach relates the impact of T/H uncertainties to changes in success criteria and, thus, to changes in risk*

AP1000 PRA REVIEW - - T/H UNCERTAINTY (continued)

REQUEST FOR ADDITIONAL INFORMATION

- *No sequences beyond those identified for AP600 are classified as "low T/H margin risk significant" on the grounds that the two designs are similar*

- *The staff requested the use of a systematic approach and/or additional analyses, as was done for AP600, to support this argument:*

  - *Differences in T/H parameters (e.g., decay heat and flow rates) can affect plant response for PRA scenarios involving multiple failures and potential system interactions*

  - *Several PRA changes (e.g., IE categories and frequencies, and success criteria) could have changed the risk significance of a sequence*

STATUS

- *Response includes requested systematic approach (under staff review)*

AP1000 PRA REVIEW -- FIRE-INDUCED SPURIOUS ACTUATIONS

ISSUE DESCRIPTION AND RELATED AP600 BACKGROUND

- *AP600 at-power fire CDF is dominated (85% or about 6.5E-7/yr) by fire-induced spurious actuation of ADS explosive valves (EVs) leading to medium LOCA*

- *In AP600 the significant uncertainty in "hot short" probability was addressed by a sensitivity study and design certification requirements*

- *Design features that prevent fire-induced detonation of EVs, such as*

  - *Use controller circuit requiring multiple shorts for actuation*

  - *Routing ADS cables in low voltage cable trays and using redundant series controllers located in separate cabinets*

  - *Provisions for operator action to remove power from the fire zone*

- *Information since AP600 certification indicates that "hot shorts" may not always be independent events and that cable-to-cable interactions cannot be excluded*

AP1000 PRA REVIEW -- FIRE-INDUCED SPURIOUS ACTUATIONS (continued)

REQUEST FOR ADDITIONAL INFORMATION

- *Hot shorts are assumed to be independent events in the AP1000 fire PRA and no cable-to-cable interactions were considered*
- *Studies since AP600 certification (SANDIA, EPRI) indicate that spurious actuations from cable-to-cable interactions (conductors from separate cables could come into close proximity to each other) are credible and likely for some cable types*
- *If ADS cables are routed in same cable tray or a common enclosure:*
  - *Analyze the effect of cable-to-cable interactions*
  - *Assess need for additional design features, beyond AP600, to prevent fire-induced detonation of EVs*

REVIEW STATUS: The staff is interacting with Westinghouse to resolve this issue

AP1000 PRA REVIEW -- CERTIFICATION REQUIREMENTS

ISSUE DESCRIPTION

An important objective of the AP1000 PRA review is to use PRA results and insights to identify "certification requirements"

- *Identify important safety insights, related to design features and assumptions made in the PRA, and use such insights to support "certification requirements," such as ITAACs, TS, D-RAP and COL action items*

- *Support the process used to determine appropriate regulatory treatment of non-safety systems (RTNSS)*

The identification of "certification requirements" requires integrated input from uncertainty, importance and sensitivity studies

AP1000 PRA REVIEW -- CERTIFICATION REQUIREMENTS (continued)

REQUEST FOR ADDITIONAL INFORMATION

- *The staff requested information, similar to what was provided for AP600, showing how PRA results and insights are used to identify "certification requirements" as well as a list of the identified requirements*

- *Differences in "certification requirements" between AP1000 and AP600 result primarily from design differences*

- *Several outstanding issues have the potential, individually or collectively, to affect PRA results and change "certification requirements" with respect to AP600, such as RTNSS. Examples of such issues are:*

  - *Initiating event frequency changes (e.g., LOCAs, SGTR, PRHR-TR)*
  - *Late containment failure modeling issue*
  - *Common cause failure probability of explosive (squib) valves*

REVIEW STATUS: Response under review

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NRC STAFF REVIEW OF THERMAL/HYDRAULIC BASIS FOR AP1000 PRA

ACRS Subcommittee on Reliability and Probabilistic Risk Assessment
Meeting January 23, 2003

Walton Jensen
NRR/DSSA/SRXB

Minimum equipment requirements to prevent CD identified by Westinghouse using MAAP4

- MAAP4 used for scoping analyses to identify the limiting events trees.
- MAAP4 has not been submitted for NRC staff review.
- MAAP4 was benchmarked against Westinghouse licensing codes for AP600.
- MAAP4 results differed from those of the licensing codes because of simplifying assumption in MAAP4.
- Overall conclusions for core cooling were similar for AP600.
- Staff has requested justification that AP600 benchmarks using MAAP4 are valid for AP1000.

Minimum success paths (low margin) identified by MAAP4 are verified by bounding analyses using licensing codes

- WCOBRA/TRAC - LBLOCA and LT Cooling
- NOTRUMP - SBLOCA
- WGOTHIC - Containment

Bounding analyses are performed by Westinghouse in lieu of uncertainty analyses for the T/H parameters. Westinghouse used the same approach for AP600.

NRC staff believes all limiting success paths accepted as the basis for successful core cooling using MAAP4 should be verified using licensing codes.

Westinghouse believes that only success paths with significant risk need to be verified.

Staff is reviewing the risk significance of the unbounded success paths to determine their effect on PRA conclusions.

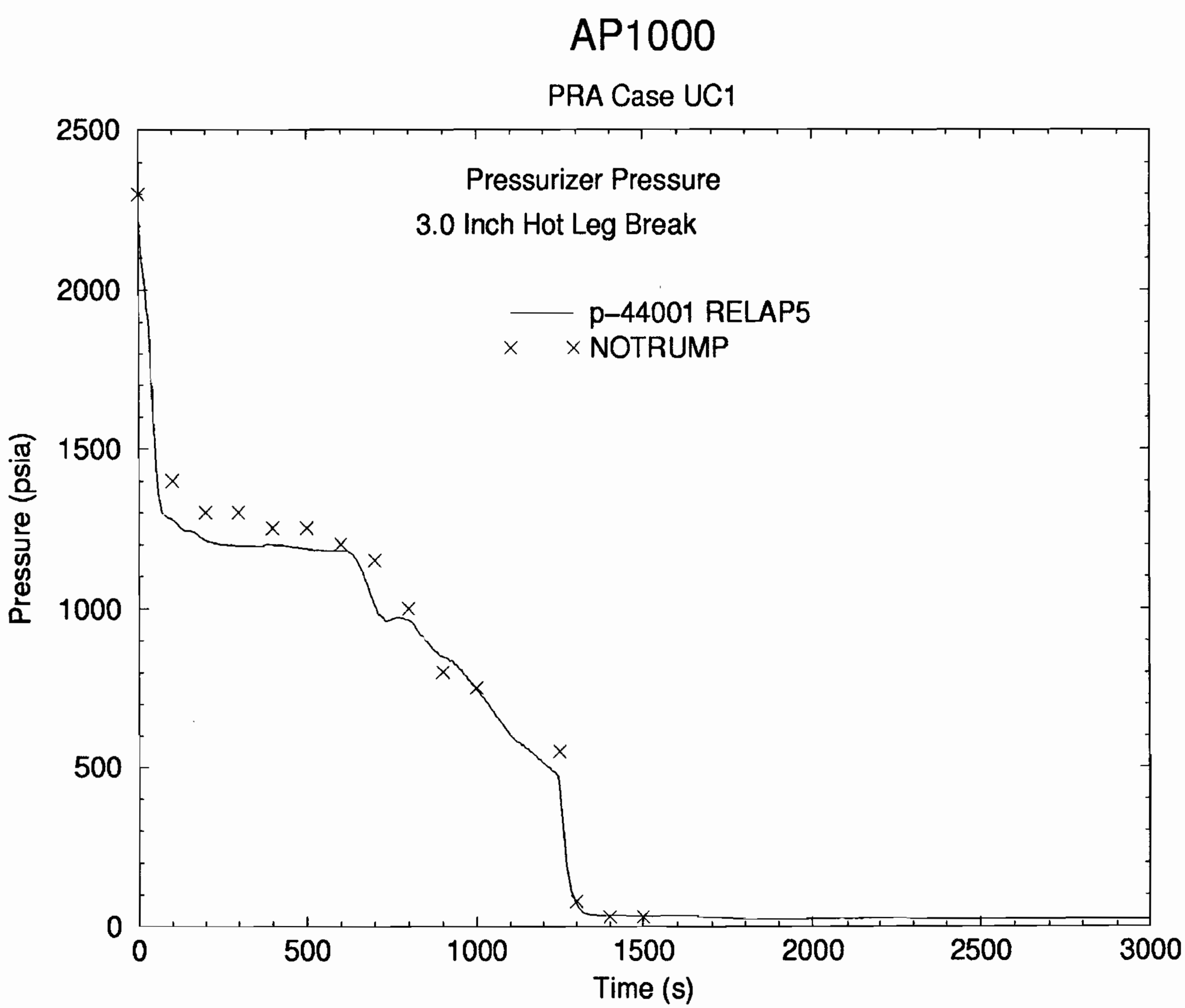
Staff Audit Calculations using RELAP5

Comparisons with NOTRUMP

- Uncertainty Case UC1 3.25 inch HLB assumes the following failures
  - Both CMT > Manual ADS-4 actuation
  - 1 of 2 accumulators
  - All ADS-1,2,3
  - 1 of 2 IRWST line

- Uncertainty Case UC3 DEDVI assumes the following failures
  - 1 of 2 CMTs > Automatic ADS-4 operation
  - Both accumulators
  - All ADS-1,2,3
  - 1 of 4 ADS-4
  - 1 of 2 IRWST Line
  - Containment isolation failed *

* Analysis extended only to initial IRWST injection. Long term cooling was not investigated

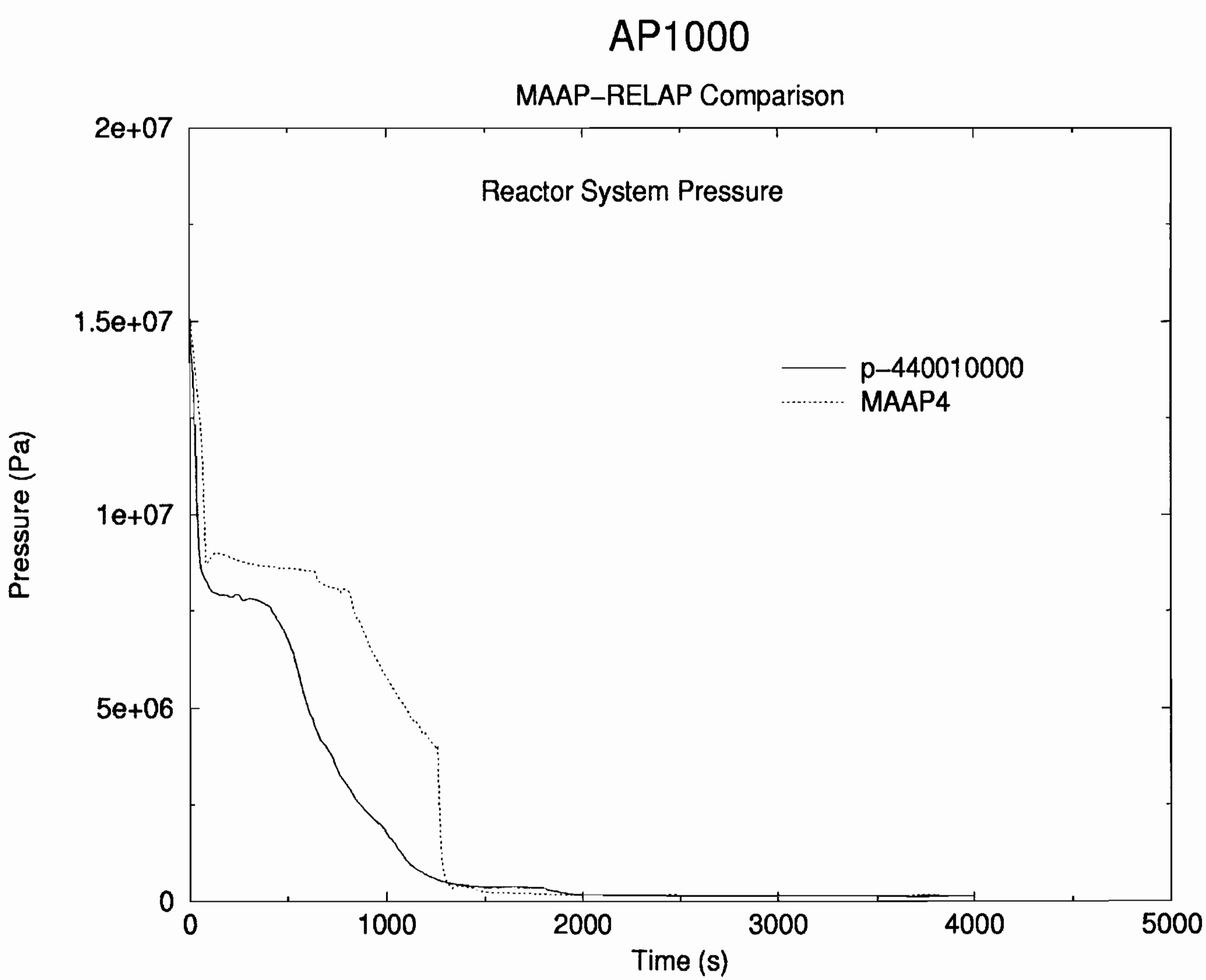


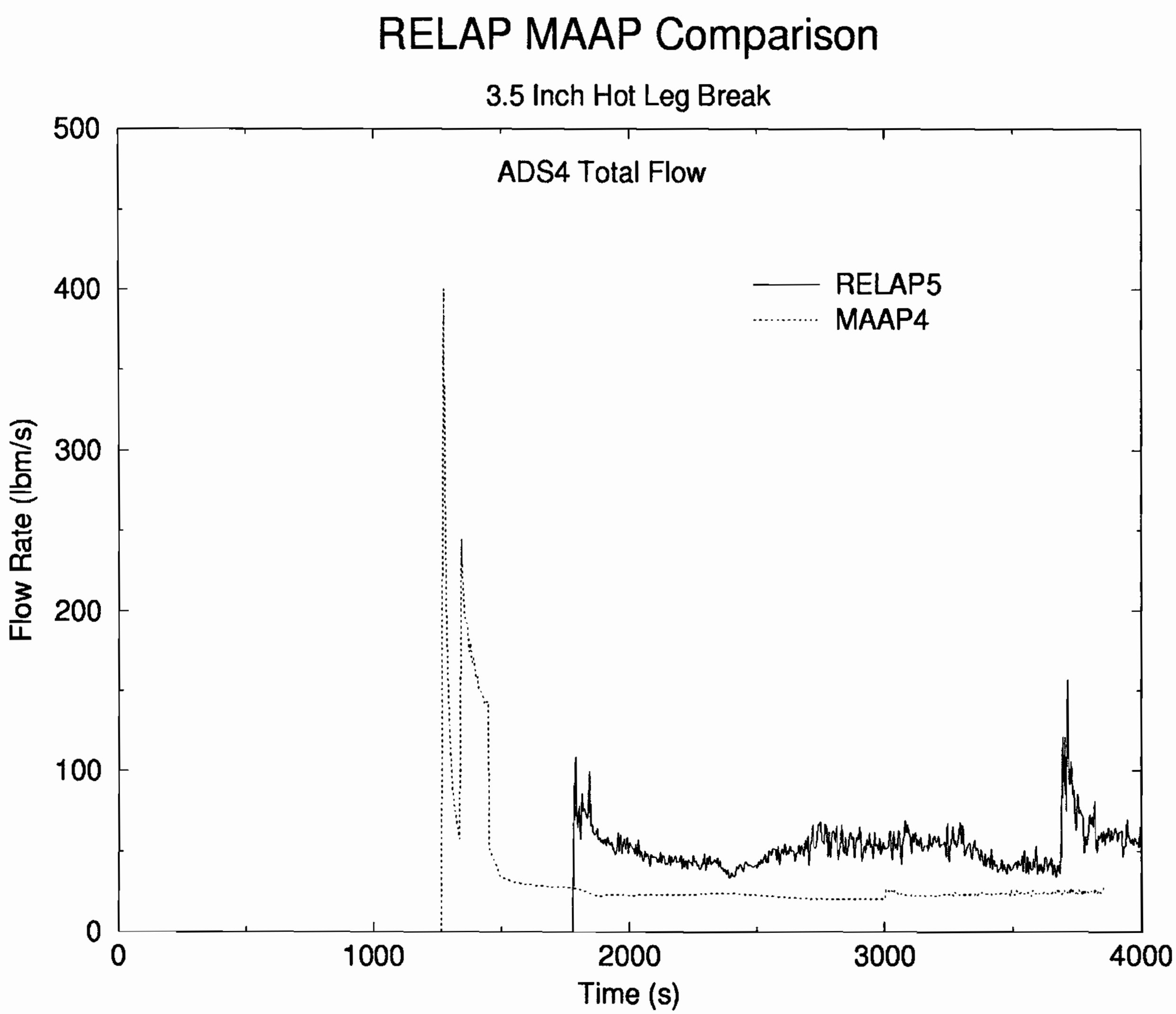
AP1000
PRA Case UC1
Pressurizer Pressure
3.0 Inch Hot Leg Break
p–44001 RELAP5
NOTRUMP
Pressure (psia)
Time (s)
0
500
1000
1500
2000
2500
3000

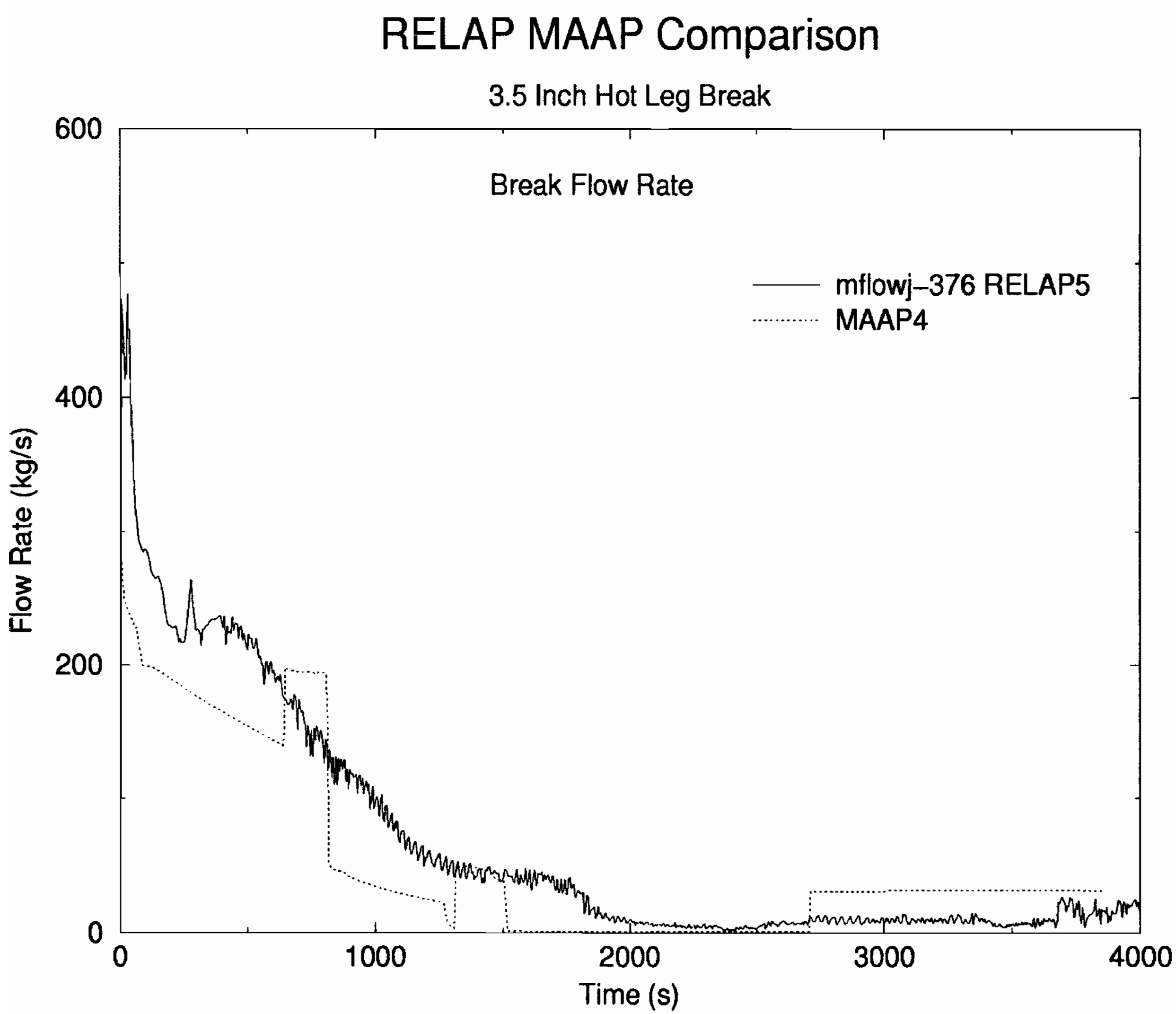
Staff Audit Calculations using RELAP5 (Cont.)

Comparison with MAAP4

- 3.50 inch HLB assuming the following failures
  - 1 of 2 accumulators
  - 1 of 4 ADS-4
  - containment isolation failure


AP1000
MAAP-RELAP Comparison
Reactor System Pressure
p-440010000
MAAP4
Pressure (Pa)
Time (s)
2e+07
1.5e+07
1e+07
5e+06
0
0
1000
2000
3000
4000
5000


RELAP MAAP Comparison
3.5 Inch Hot Leg Break
ADS4 Total Flow
RELAP5
MAAP4
Flow Rate (lbm/s)
Time (s)
0
100
200
300
400
500
0
1000
2000
3000
4000



RELAP MAAP Comparison
3.5 Inch Hot Leg Break
Break Flow Rate
mflowj-376 RELAP5
MAAP4
Flow Rate (kg/s)
Time (s)
600
400
200
0
0
1000
2000
3000
4000

Audit Calculation Conclusions

- NOTRUMP and RELAP5 show the same general trends of reactor system response for the two cases analyzed. Both codes predicted brief periods of core uncovery which were within acceptable limits.

- MAAP4 and RELAP5 predicted different trends of pressure, break flow and ADS4 flow. Although the results were different both codes predicted the core to remain covered and cooled for the case analyzed.

Examples of Unresolved issues

- PRA Appendix A Section A3.3.1indicates success in long term cooling for 3 of 4 ADS4 and Containment Failure. WCOBRA/TRAC analysis were done for 3 of 4 ADS4 without containment failure and for 4 of 4 ADS4 with containment failure.

- PRA Section 6.3.1.5 indicates that sufficient water will be retained within the containment for long term cooling even if containment isolation fails. This conclusion has not been verified.

- AP600 analyses have been utilized to justify many of the success paths for AP1000. These need to be shown to be applicable for AP1000.

- NRC staff is reviewing the risk significance of the minimum success paths which Westinghouse has not bounded by analyses using licensing codes.

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NRC STAFF REVIEW OF LEVEL 2/3 PRA AND SEVERE ACCIDENTS FOR AP1000

ACRS Subcommittee on Reliability and Probabilistic Risk Assessment
January 24, 2003

Robert Palla
NRR/DSSA/SPSB

Issues Related to Level 2/3 PRA and Severe Accidents

Review Objectives and Approach: Same as for Level 1 PRA

Key issues identified in RAIs:

Applicability of AP600 results based on similar debris mass, superheat, and composition
- hydrogen generation and mixing
- thermal loads and pressure loads on RPV
- in-vessel and ex-vessel FCI
- fission product release fractions and timing

External reactor vessel cooling
- reduced margins to CHF and impact of uncertainties
- implications of recent experimental work on in-vessel melt retention
- increased dead-load on thinned RPV
- design of thermal insulation

Hydrogen control
- Diffusion flame mitigation strategy
- Igniter placement philosophy and effectiveness

Issues Related to Level 2/3 PRA and Severe Accidents (cont)

Impact of containment spray on releases

Deterministic assessment of DCH pressure loads

Impact of alternate debris spreading assumptions on basemat melt-thru/containment over-pressure

Additional analysis/documentation (equivalent to that provided for AP600):

- P/T histories to support equipment survivability assessment
- Importance analysis results based on Large Release Frequency
- Evaluation of potential cost-beneficial improvements (SAMDAs)

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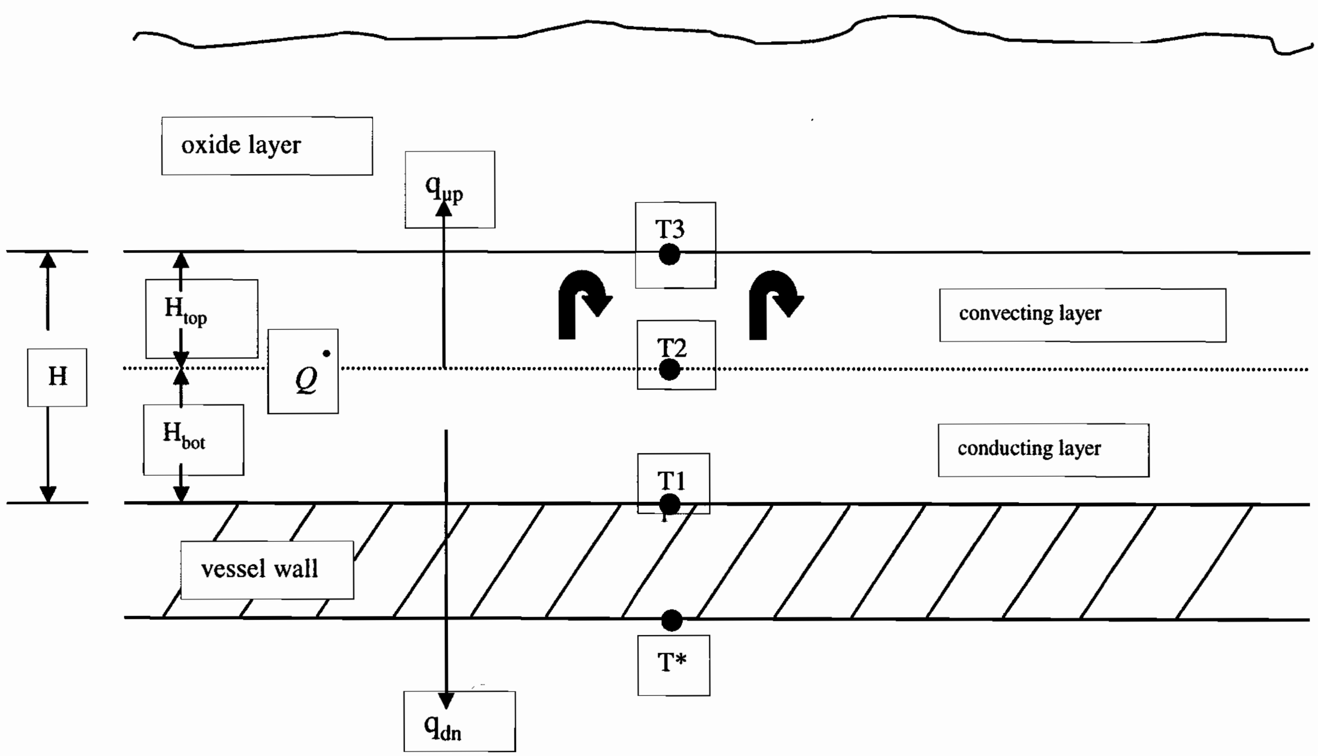
Model for Assessing Bottom Metal Layer



Assumptions

- Infinite slab
  - –conservative at minimum margin point of 0°
- Bottom metal layer has 40 wt% uranium
  - –consistent with INEL assumptions and peer review comments
- 100% of decay heat from the fission products in an equivalent volume of oxide needed to create uranium
- Initial masses of metal involved in the reaction
  - –3000 kg stainless steel
  - –7000 kg zirconium

Bottom Metal Layer Properties

- Masses of metals
  - –5566 kg zirconium
  - –2277 kg stainless steel
  - –5224 kg of uranium
- Volume of layer = 1.53 m^3
  - –layer height in lower plenum = 0.58 m
- Power density of layer = 1.38 MW/m^3

Calculation

- Conduction layer $$T_2 - T_1 = \frac{\dot{Q} H_{bot}{}^2}{2 * K_m}$$

$$q_{m-dn} = \dot{Q} H_{bot} = K_W \frac{T_1 - T^*}{x_W}$$

- Convection layer $$Nu_{up} = 0.297 Ra'^{0.233} \Pr^{0.0645}$$

$$\mathrm{q_{m\text{-}up}} = \mathrm{h_{up}} (\mathrm{T_2} - \mathrm{T_3})$$

- Equations are solved to converge on T_2

Bottom Metal Layer Heat Flux Results

- Peak heat flux to the vessel wall is 415 kW/m^2
  - qchf = 640kW/m^2 based on ULPU-IV
  - q/qchf = 0.65
- Bounding result still has margin to failure

Focusing Effect in Top Metal Layer

- Sinking metal debris to bottom thins top metal layer
  - –7000 kg zirconium
  - –3000 kg stainless steel
- Don't credit reduced heat load from oxide layer
- Bounding metal layer heat flux = 1578 kW/m^2
  - –qchf = 1875 kW/m^2 based on ULPU-IV
  - –q/qchf = 0.84 at 83°, minimum margin to failure
- Bounding result still has margin to failure

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United States Nuclear Regulatory Commission

AP1000 SEVERE ACCIDENT ANALYSIS

PRESENTED TO THE
ACRS SUBCOMMITEE ON
RELIABILITY AND PROBABILISTIC RISK ASSESSMENT

BY

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JANUARY 24, 2003

AP1000 SEVERE ACCIDENT CALCULATIONS

Overall Objective: Gain insights into severe accident evolution and containment challenges in AP1000, and develop the basis for understanding of other issues (e.g., in-vessel and ex-vessel steam explosions, core concrete interactions, hydrogen combustion)

Code Model: A relatively detailed MELCOR 1.8.5 input deck has been developed by Energy Research, Inc. (ERI) utilizing the available AP1000 drawings, design information, and the Westinghouse provided MAAP input deck. Additional design data were also requested and received from Westinghouse through the RAI process.

Due to the parametric nature of the lower head heat transfer model in MELCOR, results of a more detailed calculation (i.e., based on a molten pool convection model) are used to derive certain MELCOR parameters (e.g., debris/molten pool to lower head heat transfer coefficients).

Note: The analysis of lower head integrity and in-vessel retention (IVR) will not be based on the MELCOR calculations. Instead, a more detailed approach will be utilized that will consider a wide range on uncertainties (e.g., melt composition and configurations)

SPECIFIC SCENARIOS BEING ANALYZED

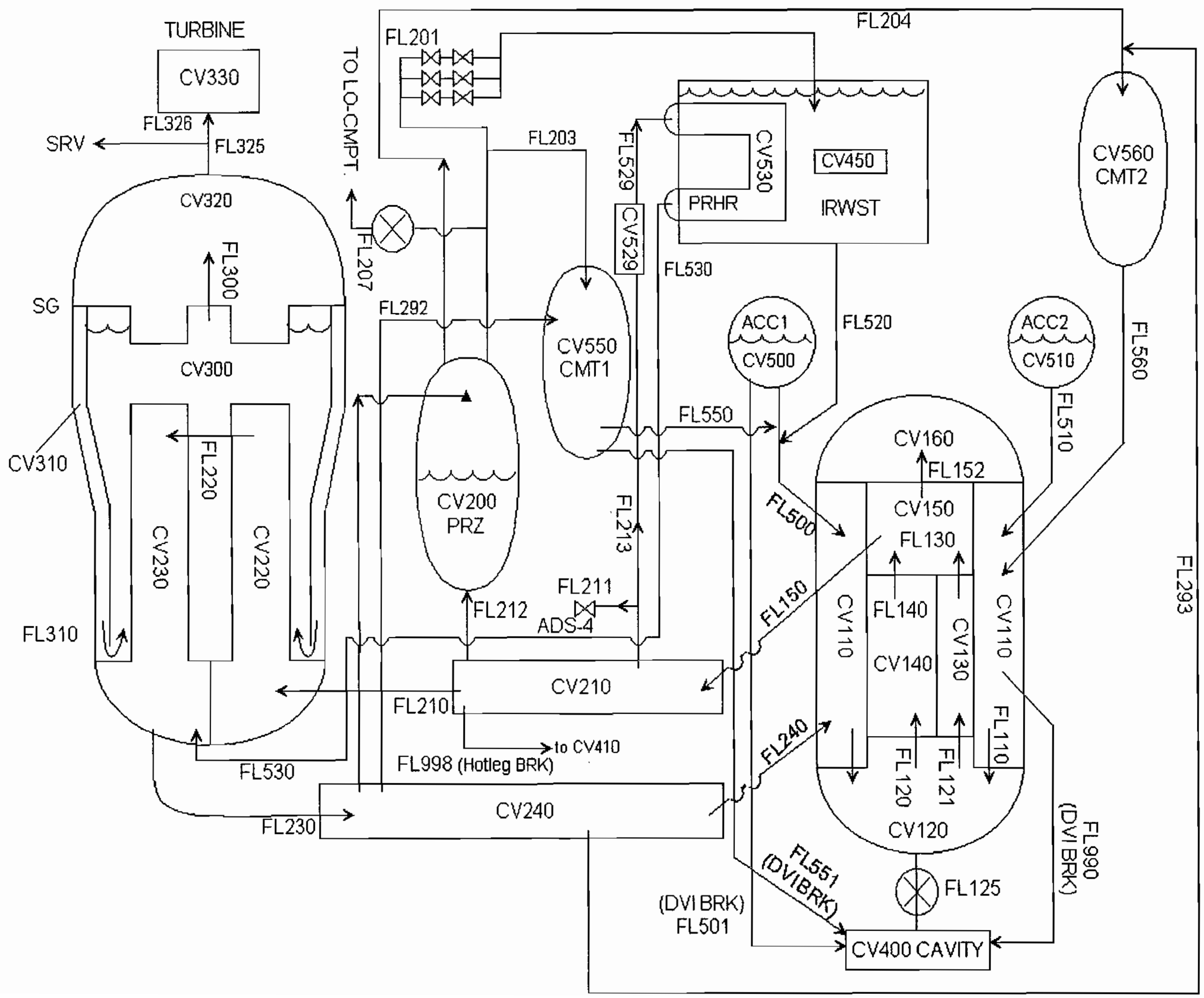
Scenario	Description	Basis for Selection
3BE	• DVI line break • CMTs and accumulators available • ADS successful • PRHR unavailable • IRWST cavity injection successful • Gutters directed to the cavity/sump	• PRA Sequence #1 (29% of CDF) • Fully depressurized RCS • IVR expected to be successful
LLOCA	• LLOCA in the hot leg • CMTs and 1 of 2 accumulators available • PRHR available • Gutters directed to the IRWST • IRWST for core cooling and cavity injection partially successful	• PRA Sequence #2 (18% of total CDF) • IVR unsuccessful due to delay in cavity flooding • MCCI due to shallow water in cavity
3D	• Spurious ADS actuation (stage 1/2/3) • CMTs unavailable • 1 of 2 accumulators not available • Full ADS (stage 4) unsuccessful • IRWST not actuated for core cooling or cavity flooding	• PRA Sequence #3 (9% of CDF) • Highest-frequency sequence with partial RCS depressurization • IVR unsuccessful
1A	• Transient initiated by loss of MFW • CMTs and accumulators available • PRHR and SFW unavailable • ADS unsuccessful • IRWST not actuated for core cooling or cavity flooding • Gutters directed to the cavity/sump	• PRA Sequence #20 (0.6% of CDF) • Highest-frequency sequence with no RCS depressurization • IVR unsuccessful • Potential for T-SGTR (to be studied later, possibly)

PLANNED SENSITIVITY STUDIES

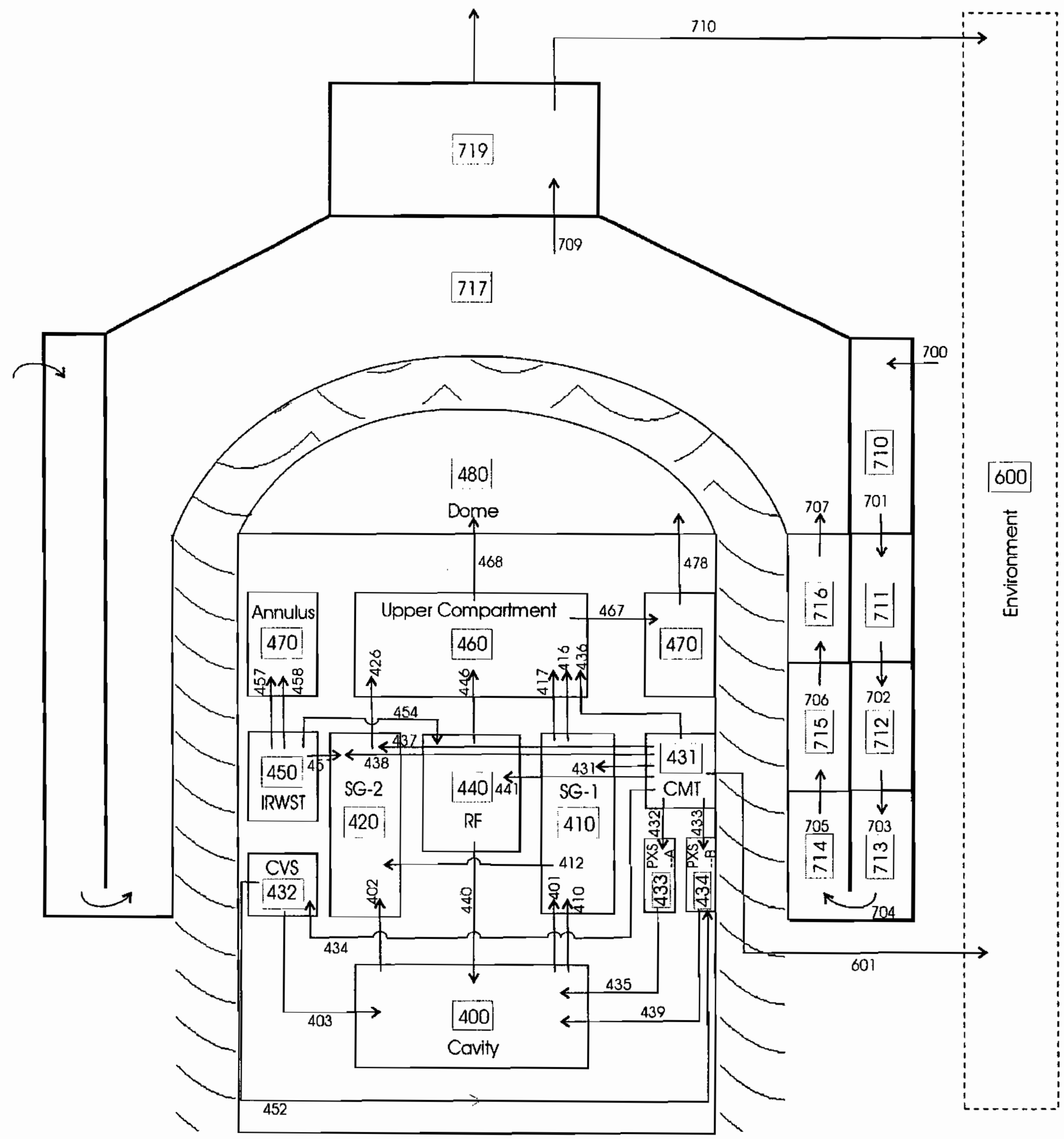
- MCCI Sensitivity Studies - Intended to provide insights into the impact of concrete aggregate, the extent of cavity flooding, and the quantity of debris relocating into the cavity, on cavity erosion, and noncondensible gas generation on containment loads and fission product release behavior.

- Impact of Containment Spray Operation - Intended to provide insights into the impact of containment sprays (to be activated manually, as designed, and to operate continuously), on containment loads and fission product release behavior.

MELCOR NODALIZATION OF AP1000 REACTOR COOLANT SYSTEM



MELCOR NODALIZATION OF AP1000 CONTAINMENT



Record Copy

Discussion of results

- Redox and exchange processes between molten corium and liquid steel taking place during the tests can be described:

MA-1

$(U_{0,59}Zr_{0,41})O_{1,45}+Fe_{0,69}Ni_{0,10}Cr_{0,21} \leftrightarrow (U_{0,55}Zr_{0,38}Fe_{0,05}Ni_{0,004}Cr_{0,01})O_{1,57}+(U_{0,16}Zr_{0,10}Fe_{0,50}Ni_{0,08}Cr_{0,17})O_{0,04}$

1434.2 g 185.4 g 1265.3 g 354.3 g

(C=33.13, U/Zr=1.45) (U/Zr=1.50)

MA-2

$(U_{0,56}Zr_{0,44})O_{1,77}+Fe_{0,68}Ni_{0,10}Cr_{0,22} \leftrightarrow (U_{0,52}Zr_{0,41}Fe_{0,04}Ni_{0,01}Cr_{0,01})O_{1,72}+(U_{0,06}Zr_{0,04}Fe_{0,62}Ni_{0,08}Cr_{0,2})O_{0,01}$

1560.2 g 189.0 g 1535.5 g 213.7 g

(C=74.63, U/Zr=1.27) (U/Zr=1.28)

MA-3 (FP simulants are not considered)

$(U_{0,53}Zr_{0,47})O_{1,41}+Fe_{0,76}Ni_{0,09}Cr_{0,15} \leftrightarrow (U_{0,54}Zr_{0,45}Fe_{0,013}Ni_{0,002}Cr_{0,0041})O_{1,76}+(U_{0,17}Zr_{0,18}Fe_{0,49}Ni_{0,06}Cr_{0,10})O_{0.01}$

1459.9 g 176.2 g 1186.8 g 449.3 g

C=36.52, U/Zr=1.14) (U/Zr=1.21)

MA-4 (FP simulants are not considered)

$(U_{0,54}Zr_{0,46})O_{1,44}+Fe_{0,72}Ni_{0,09}Cr_{0,19} \leftrightarrow (U_{0,53}Zr_{0,38}Fe_{0,08}Ni_{0,002}Cr_{0,01})O_{1,80}+(U_{0,11}Zr_{0,14}Fe_{0,53}Ni_{0,07}Cr_{0,14})O_{0,03}$

1446.0 g 365.3 g 1158.3 g 653.0 g

(C=39.67, U/Zr=1.17) (U/Zr=1.37)

Michael Snodderly

AP1000 Probabilistic Risk Assessment

Presentation to NRC Staff

May 1, 2002

Agenda for Today's Meeting

• 9:00	Introduction / Objectives	Mike Corletti
• 9:20	Overview of AP1000 Plant	Terry Schulz
• 10:20	Overview of AP1000 PRA	Selim Sancaktar
• 10:40	Level 1 PRA	Selim Sancaktar
• Lunch Break		
• 2:00	Level 2 and 3 PRA	Jim Scobel
• 3:00	Success Criteria	Terry Schulz
• 4:00	Feedback from Staff	
• 4:30	Public Comment	

AP1000 Design Certification

Mike Corletti
AP1000 Project
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Phased Approach to AP1000 Licensing

- **Phase 1**
  - –Establish goals and estimate for Prelicensing Review
  - –Westinghouse prepare submittals to support goals
- **Phase 2**
  - –NRC perform Prelicensing Review
  - –NRC estimate Cost and Schedule for AP1000 Design Certification
  - –Westinghouse develop Safety Analysis Report
- **Phase 3**
  - –NRC perform Design Certification Review

Results from Pre-Certification Review (Phase 2)

Pre-Certification Review Complete

- SECY-02-0059
  - –Design Acceptance Criteria can be used for AP1000
    - –Main Control Room and I&C - Same approach as AP600
    - –Piping DAC approach is acceptable
      - –Detailed review will be performed as part of Design Certification
- March 25th Letter to Westinghouse on Remaining Issues
  - –AP600 tests are applicable to AP1000
  - –AP600 analysis codes validated to these tests can also be used for AP1000
    - –Treatment of entrainment phenomenon in the upper plenum / hot leg in SBLOCA analysis will be addressed in Design Certification review
- ACRS Letter Endorsing AP1000 Conclusions

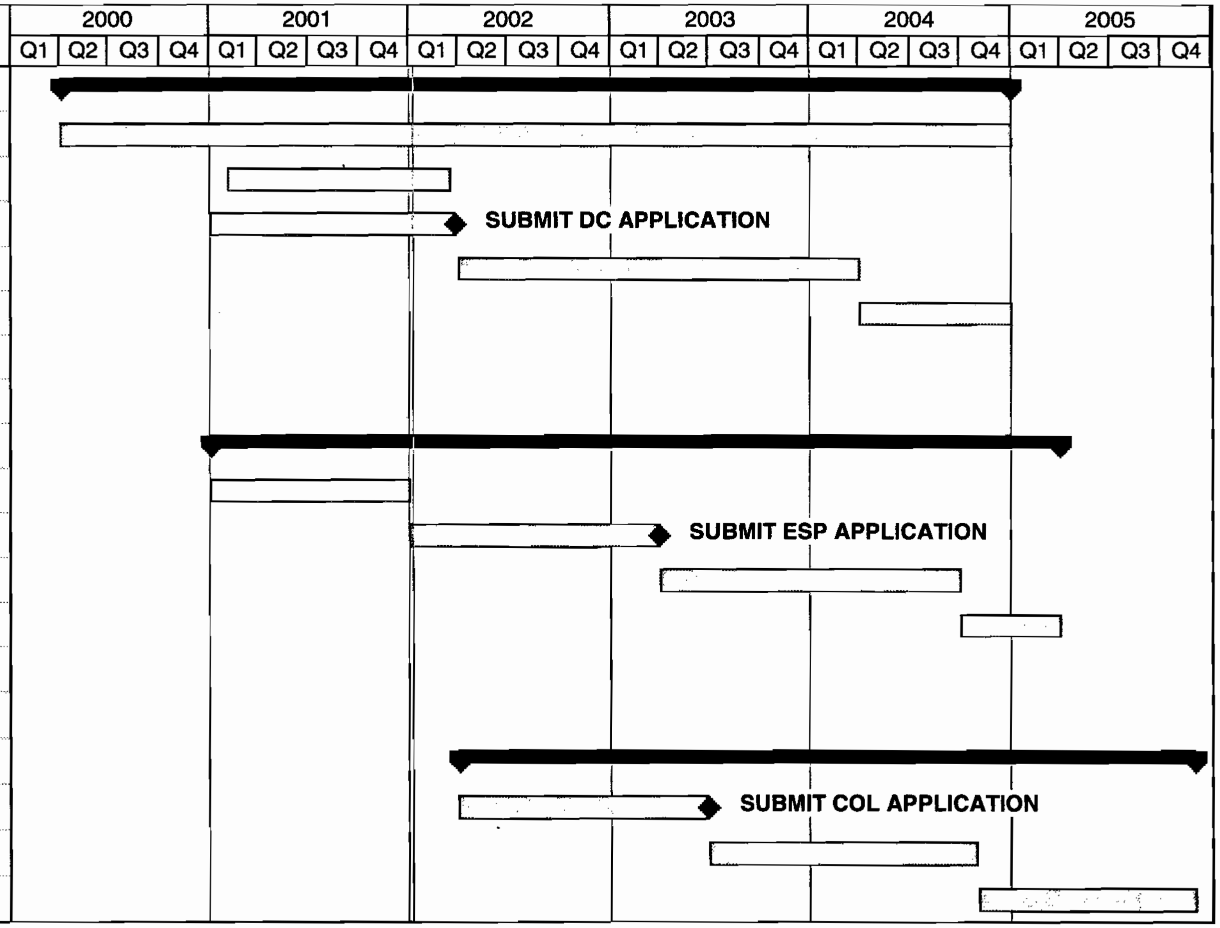
AP1000 Design Certification Application

Submitted March 28, 2002

- AP1000 Design Control Document (DCD)
  - –Tier 1 Information
    - –Inspections, Tests, Analysis and Acceptance Criteria (ITAAC)
  - –Tier 2 - Information
    - –Standard Safety Analysis Report
    - –Technical Specifications
    - –PRA Insights
- AP1000 PRA Report submitted with application

Schedule for AP1000 Design Certification

ID	Task Name
1	**AP1000 Design Certification**
2	Engineering Design
3	NRC Pre-Certification Review
4	Prepare Safety Analysis Report
5	NRC Review (Issue FDA)
6	Hearings / Rule (Issue DC)
7	
8	
9	**U.S. Utilities Early Site Permit**
10	Decide on Plan and Select Site
11	Prepare Application
12	NRC Review
13	Hearings (Issue ESP)
14	
15	
16	**U.S. Utility Combined Op License**
17	Prepare Application
18	NRC Review (Issue SER)
19	Hearings (Issue COL)



Overview of AP1000 Plant

Terry Schulz
Advisory Engineer
412-374-5120 - schultzl@westinghouse.com

AP1000 Design Objectives

- **Reduce Cost by Increasing Plant Power Rating**
  - –Obtain a capital cost that can compete in U.S. market $1000/KWe for n^{th} twin plant
- **Retain AP600 Objectives and Design Detail**
  - –Increase the capability/capacity within "space constraints" of AP600
  - –Retain credibility of "proven components"
  - –Retain the basis for the cost estimate, construction schedule and modularization scheme
- **Retain AP600 Licensing Basis**
  - –Meet regulatory requirements for Advanced Passive Plants
  - –Accept AP600 policy issues

AP1000 Design Features

- **Integrated Power Plant Design**
- **Proven Power Producing Components (Reactor, Fuel, ...)**
- **Simplified RCS Loops with Canned Motor Pumps**
- **Simplified Passive Safety Systems**
  - Increase safety margins and address severe accidents
- **Simplified Nonsafety DID Systems**
- **Microprocessor, Digital Technology Based I&C**
- **Compact Control Room, Electronic Operator Interface**
- **Optimized Plant Arrangement**
  - Construction, Operation, Maintenance, Safety, Cost
- **Extensive Use of Modular Construction**

AP600 to AP1000 Design Changes

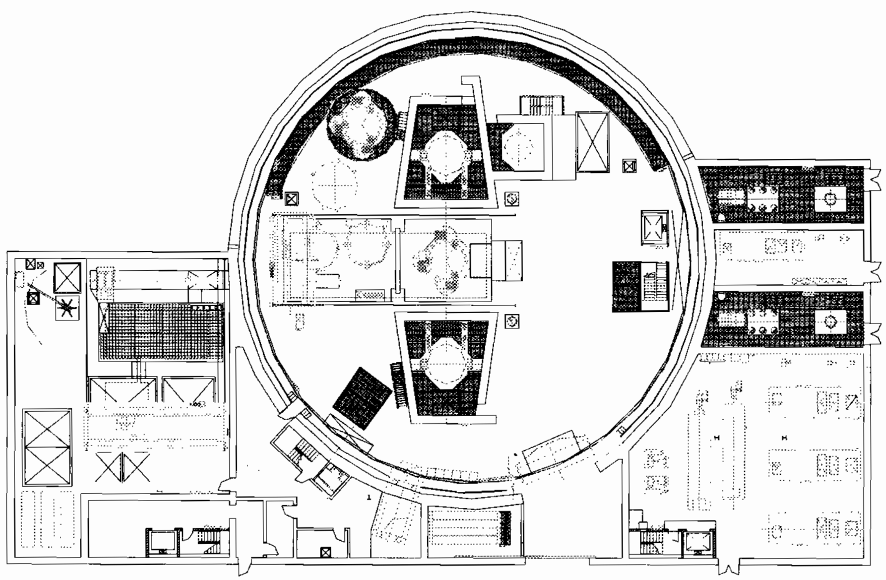
- **Increase Core Length & Number of Assemblies**
- **Increase Size of Key NSSS Components**
  - –Increased height of Reactor Vessel
  - –Steam Generators (Δ125, similar to ANO replacement)
  - –Larger canned RCPs (variable speed controller)
  - –Larger Pressurizer
- **Increase Containment Height**
- **Some Capacity Increases in Passive Safety System Components**
- **Turbine Island Capacity Increased for Power Rating**

Retained Nuclear Island Footprint

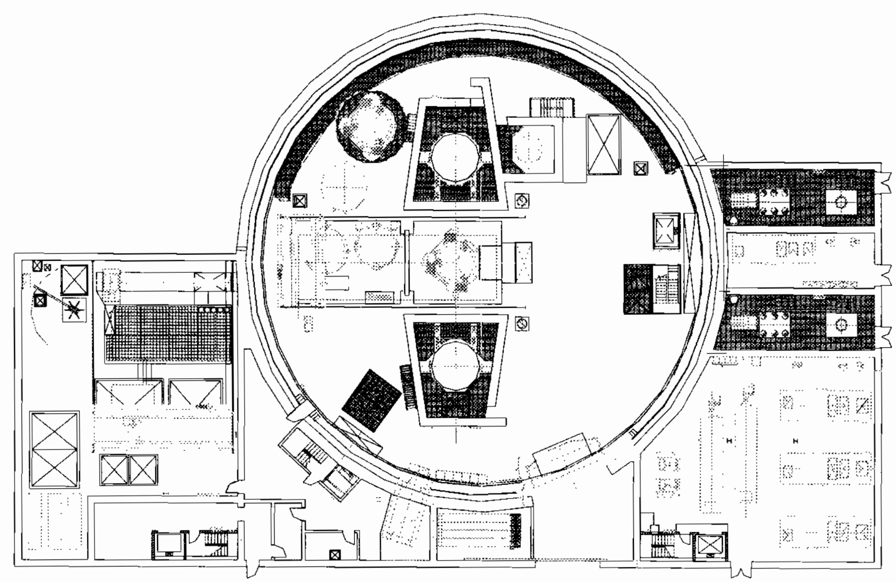
AP1000 General Arrangement

Plan at Elevation 135'

AP600

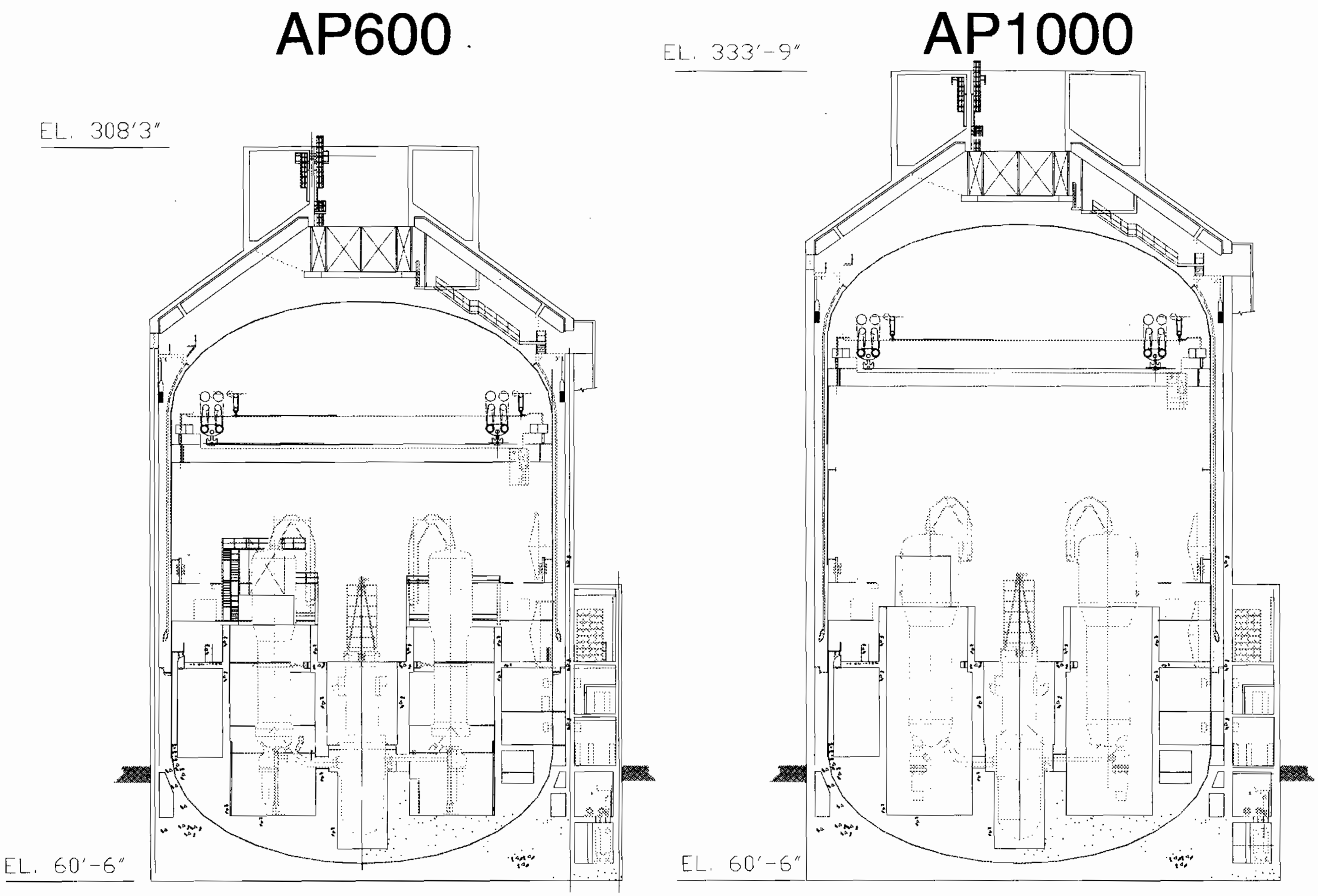


AP1000



AP1000 General Arrangement

Containment Section View

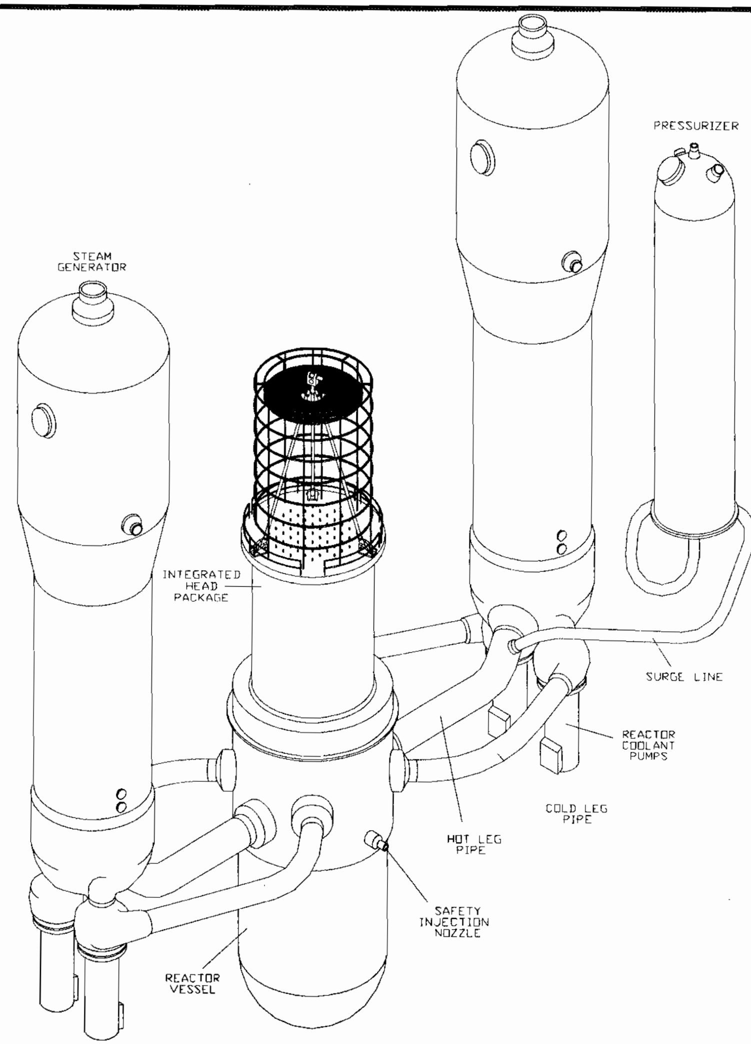


Comparison of Selected Parameters

PARAMETER	West 3XL	AP600	AP1000
Net Electric Output, MWe	985	610	1117
Reactor Power, MWt	2988	1933	3400
Hot Leg Temperature, °F	626	600	610
Number of Fuel Assemblies	157	145	157
Type of Fuel Assembly	17x17	17x17	17x17
Active Fuel Length, ft	14	12	14
Control Rods / Gray Rods	52 / 0	45 / 16	53 / 16
R/V I.D., inches	157	157	157
Vessel flow (Thermal Design)	295,500	194,200	300,000
Steam Generator Surface Area, ft^2	68,000	75,000	125,000
Pressurizer Volume, ft^3	1400	1600	2100

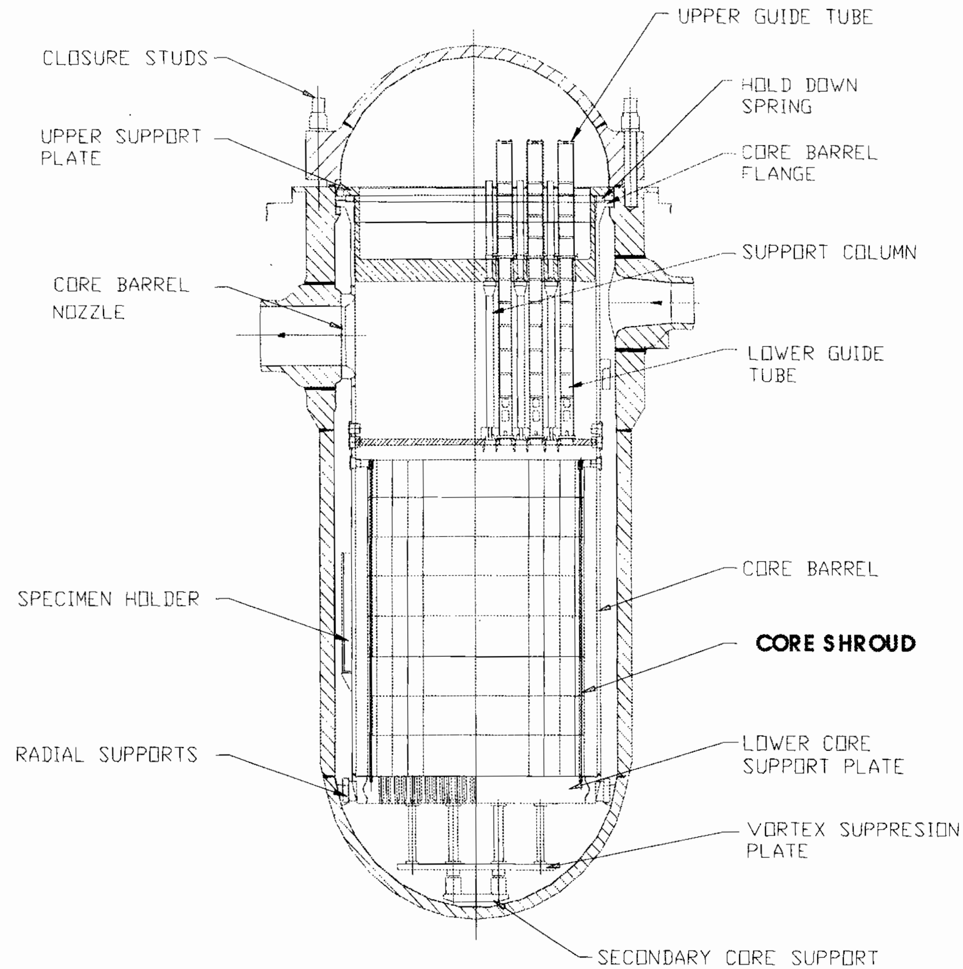
AP1000 Major Components

- **Reactor Vessel**
  - –W 3XL Vessel
  - –No bottom-mounted instrumentation
  - –Improved materials - 60 yr life
- **Δ125 Steam Generators**
  - –ANO RSG
- **Reactor Coolant Pump**
  - –Canned motor pumps
    - –Naval reactors; AP600; early commercial reactors
- **Simplified Main Loop**
  - –Same as AP600
  - –Reduced welds / supports

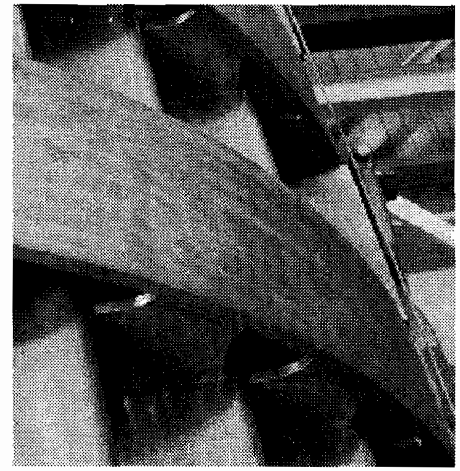


AP1000 Reactor Vessel

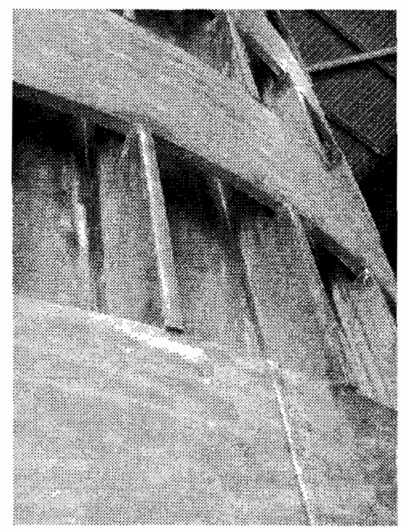
- **West. 3 Loop Reactor**
  - –157" ID, 157 fuel assemblies
  - –Ring forged construction
    - –No welds in core region
  - –Improved materials permit 60 yr design life
  - –W-CE type Core Shroud
    - –Replaces radial reflector
    - –All-welded design
  - –Top mounted incore I&C
    - –Fixed position, online readout
    - –No penetrations in bottom of vessel



YGN-5 Core Shroud



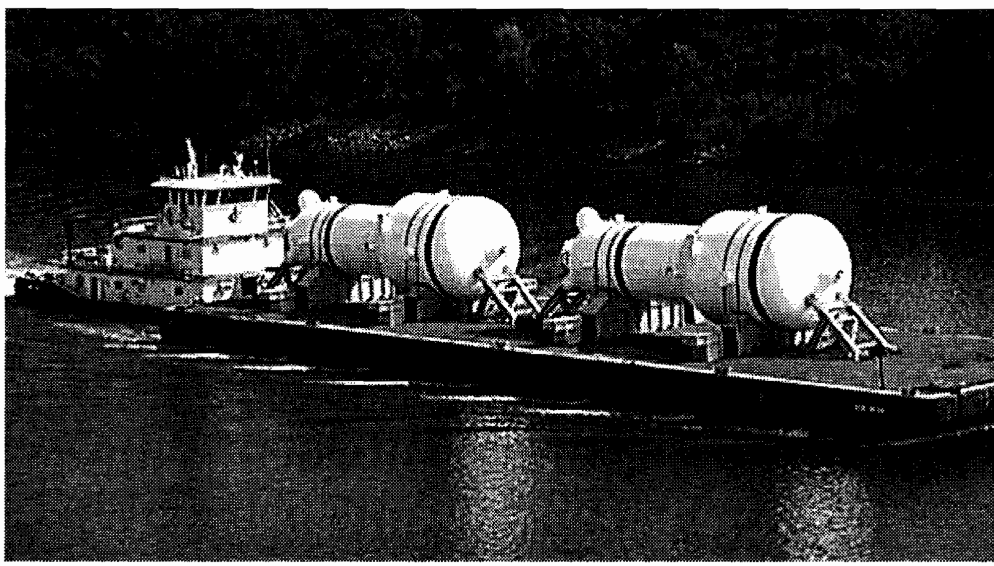
Y-Brace

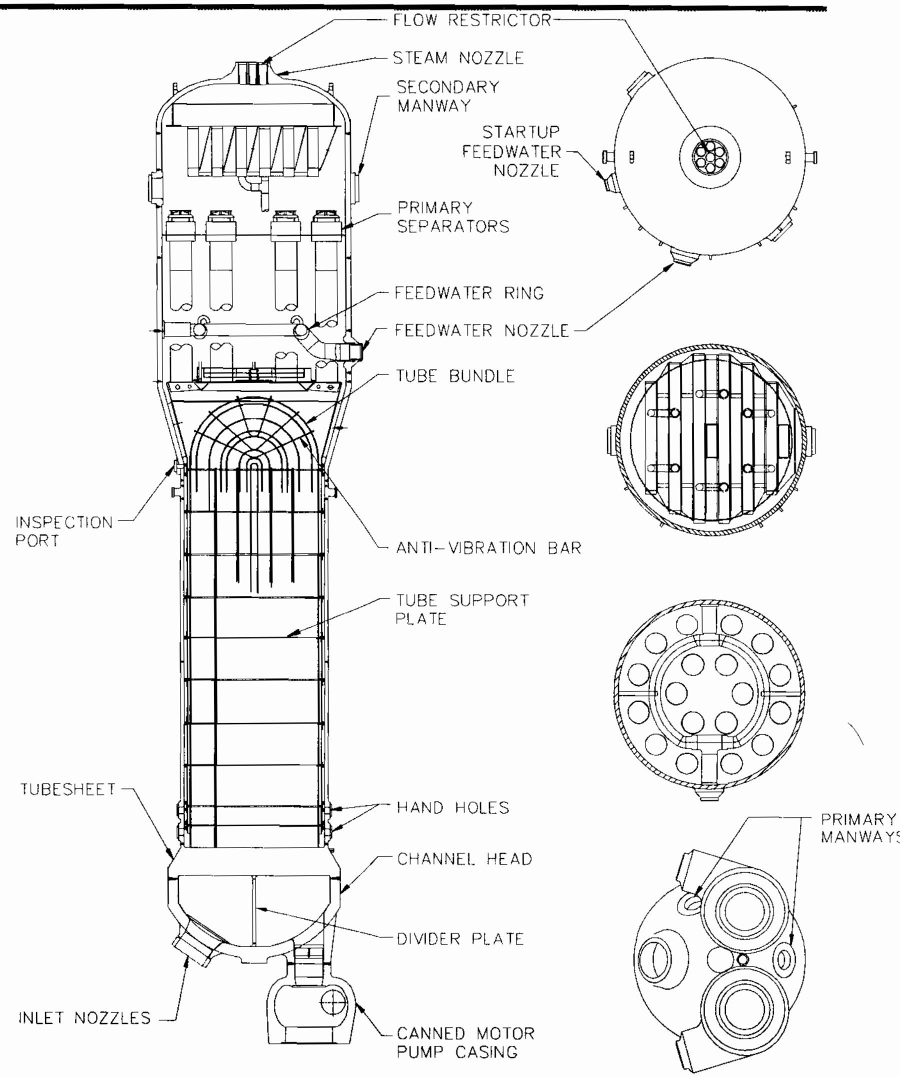


Rib

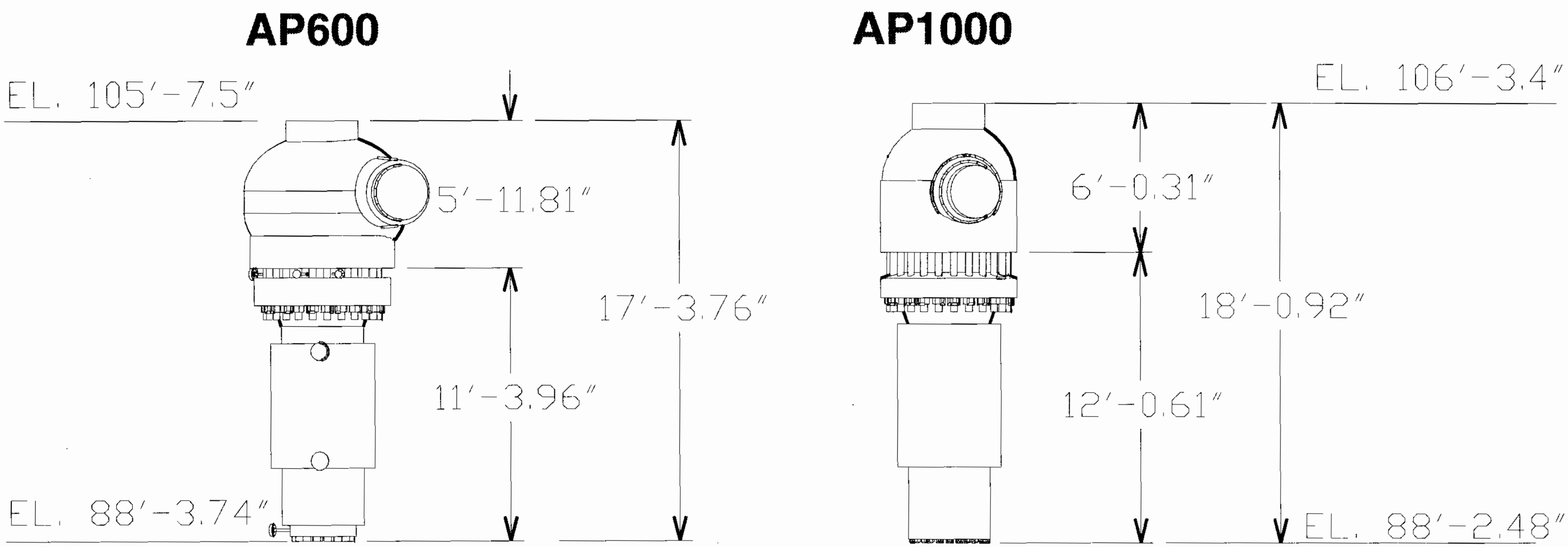
AP1000 Steam Generator

– **Based on Proven W Designs**
  – AP1000 design based on ANO RSG
  – Design Features
    – Inconel 690 TT tubes
    – Stainless steel support plates
    – Improved access
  – Excellent Operating Experience
  – Over 1200 SG years of operation
  – Less than 0.1% total tubes plugged





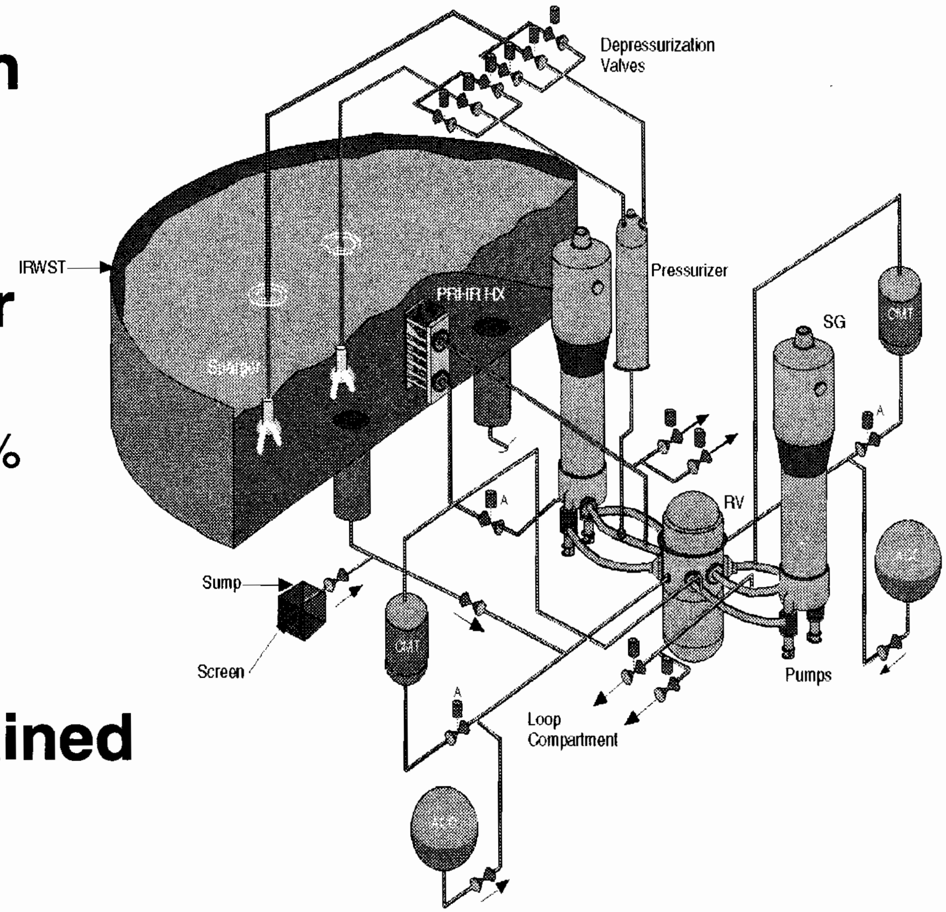
Reactor Coolant Pump



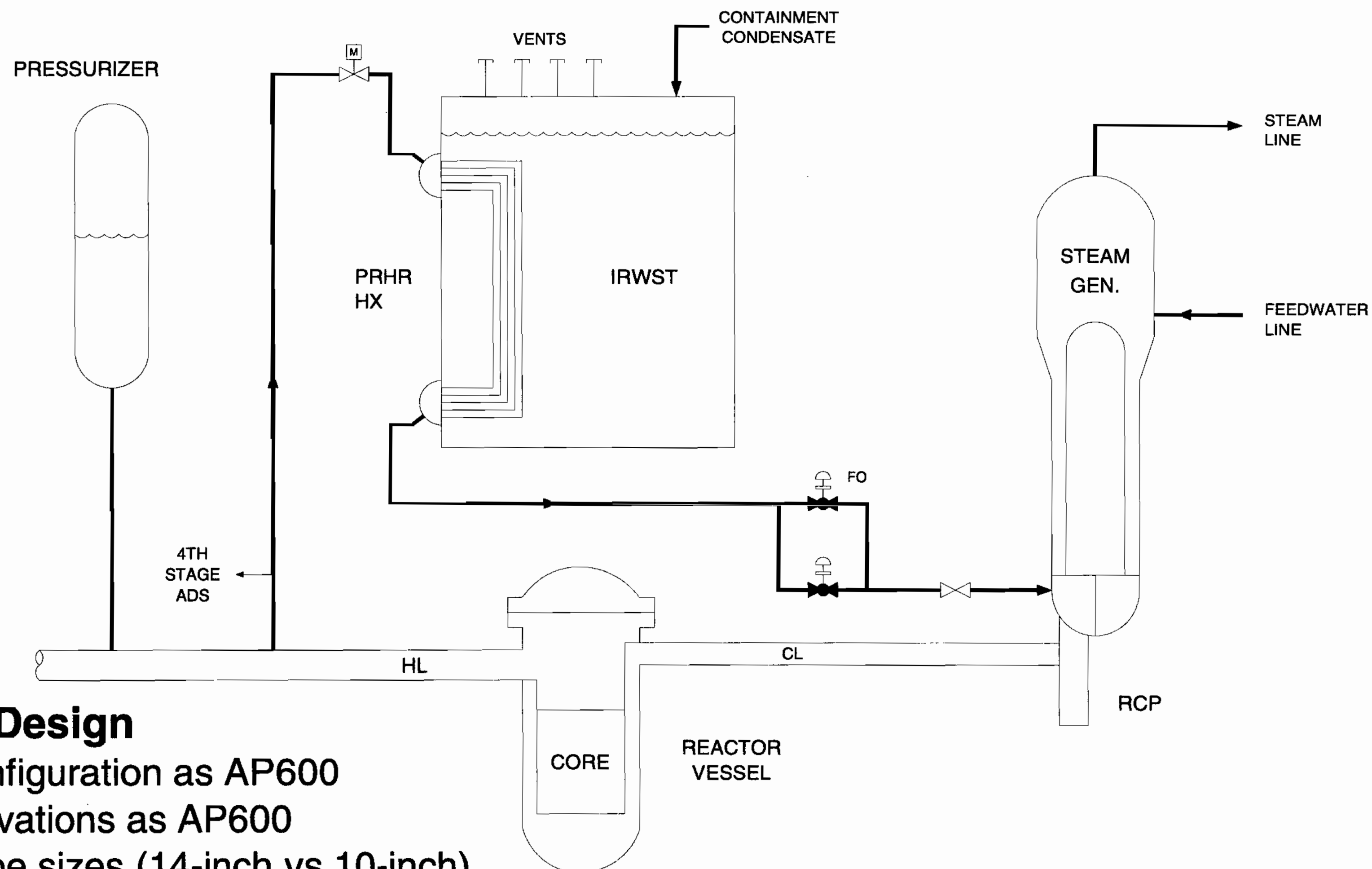
Parameter	AP600	AP1000
Design Flow, gpm	51,000	78,750
Design Head, ft	240	365
Rotating Inertia, lb-ft^2	5,000	16,500
Motor Rating, Hp	3200	7000

AP1000 Passive Core Cooling System

- **AP600 System Configuration Retained**
- **Capacities Increased to Accommodate Higher Power**
  - –PRHR HX Capacity Increased 72%
  - –CMT Volume & Flow Increased 25%
  - –ADS 4 Flow Increased 89%
  - –IRWST Injection Increased 84%
  - –Cont. Recirc. Increased 113%
- **System Performance Maintained**
  - –No core uncovery for SBLOCA
    - – DVI line break
  - –Large margin to PCT limit



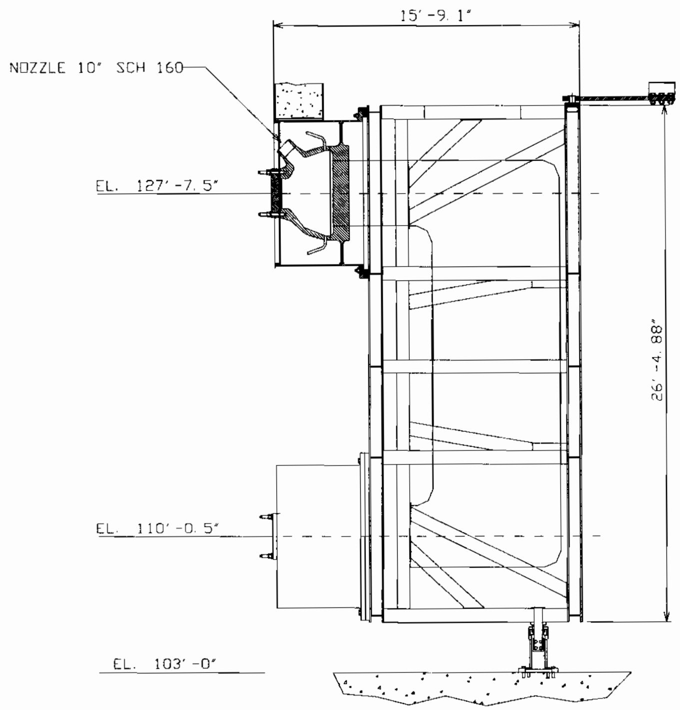
Passive Decay Heat Removal



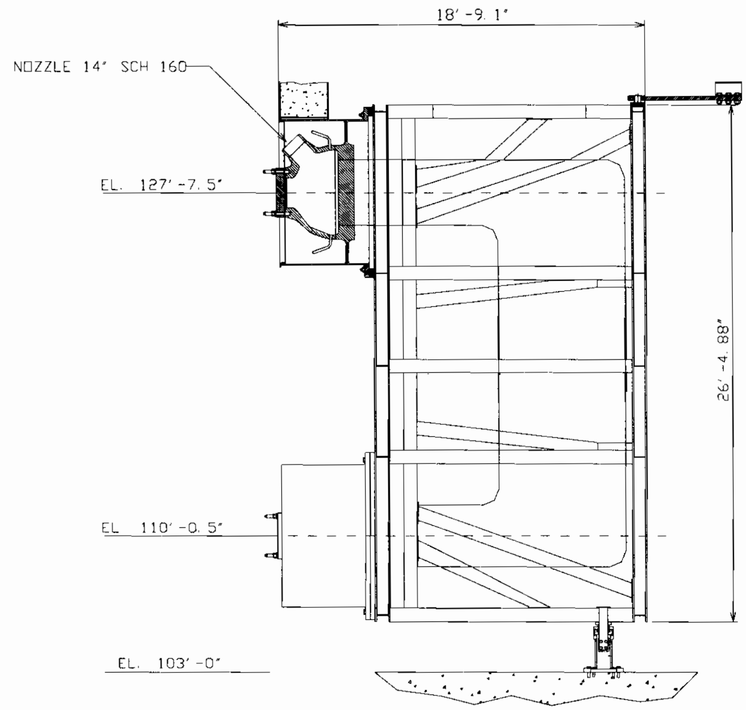
- **PRHR HX Design**
  - Same configuration as AP600
  - Same elevations as AP600
  - Larger pipe sizes (14-inch vs 10-inch)
  - Increased HX surface (more / longer horizontal tubes)

Passive RHR Heat Exchanger

AP600

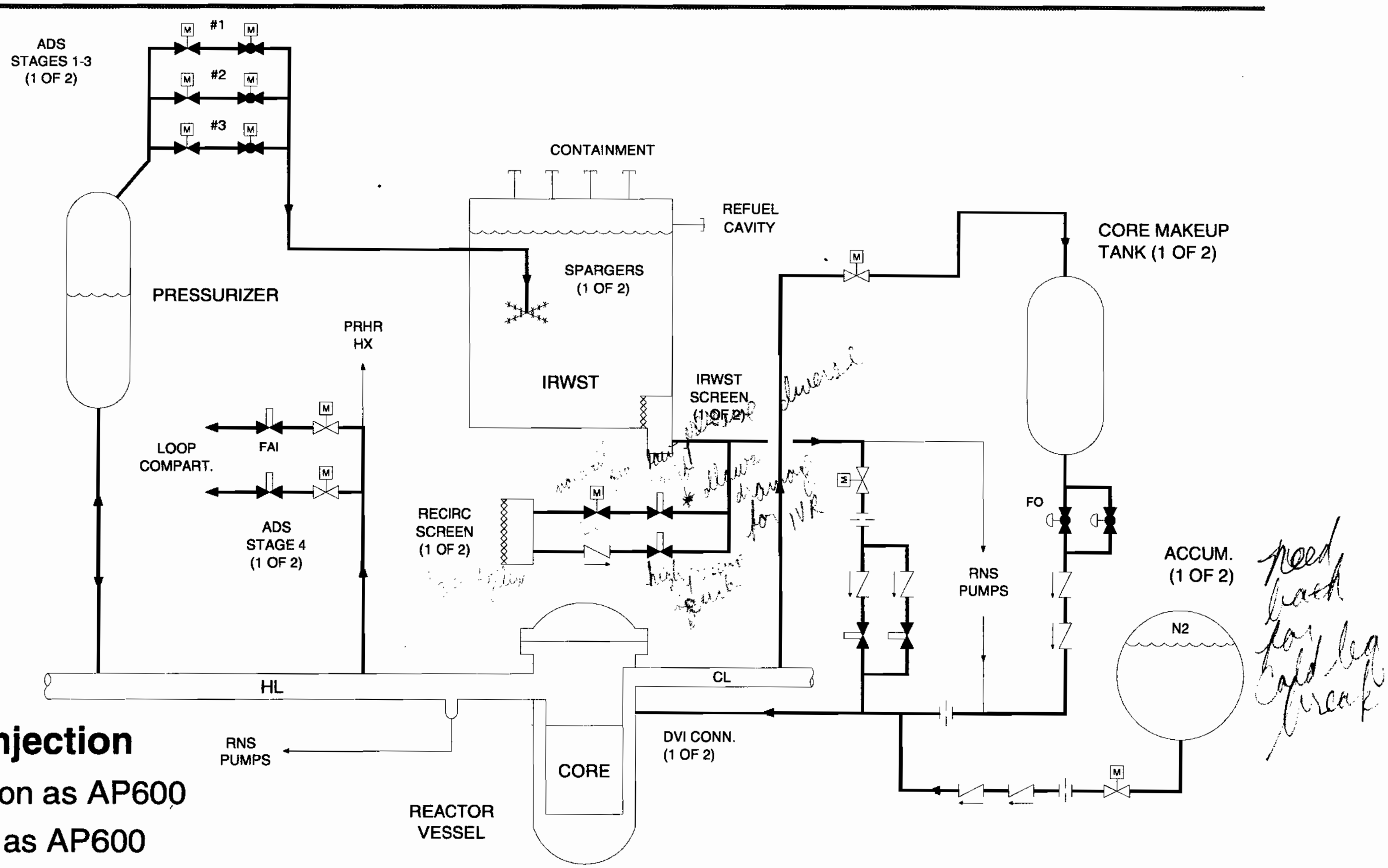


AP1000



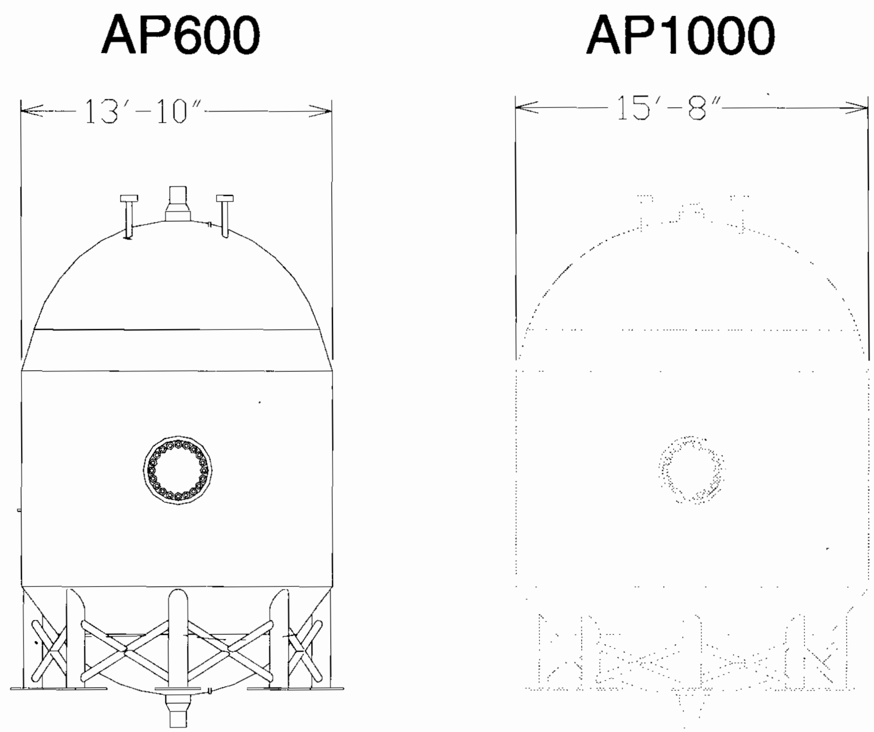
- PRHR Heat Transfer Capacity Increased 72%

AP1000 Passive Safety Injection



- **Passive Safety Injection**
  - Same configuration as AP600
  - Same elevations as AP600
  - Larger CMT and CMT flow tuning orifice
  - Larger IRWST, Recirc, ADS 4 pipe sizes

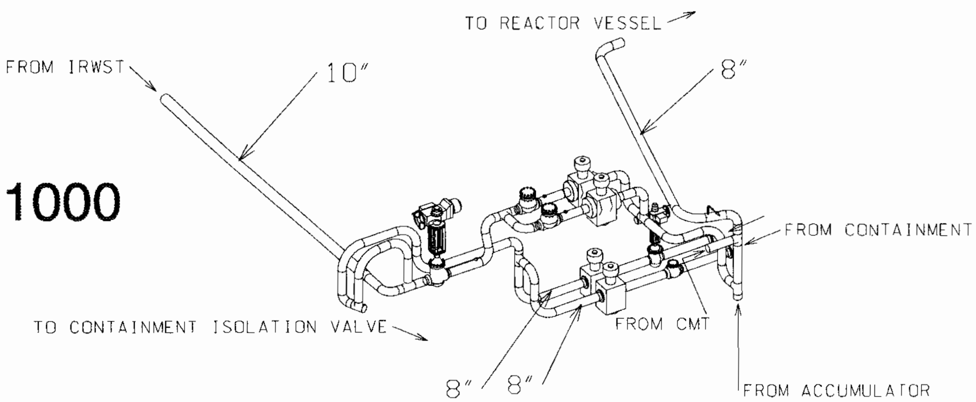
Core Makeup Tanks



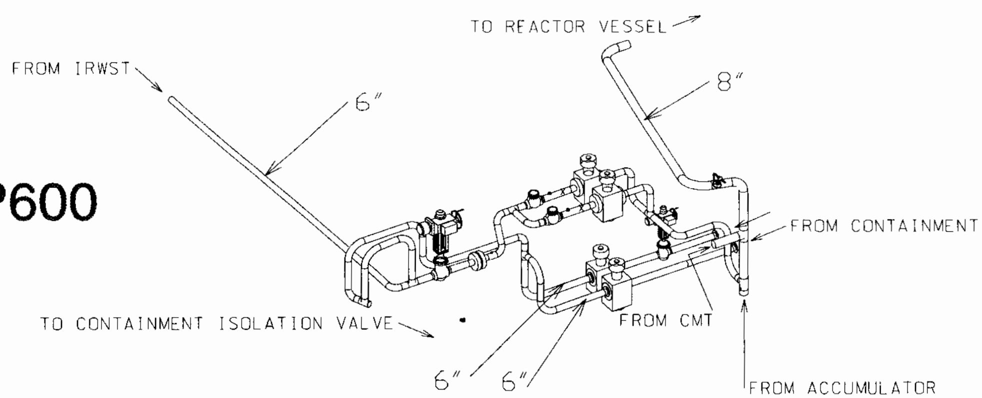
- Core Makeup Tank volume and flow rate increased 25%
  - Tank volume increased 2000 to 2500ft^3
  - Pipe / valves stay same, flow tuning orifice made less restrictive
  - Maintains same ADS timing

Comparison of IRWST Injection & Cont. Recirc.

AP1000



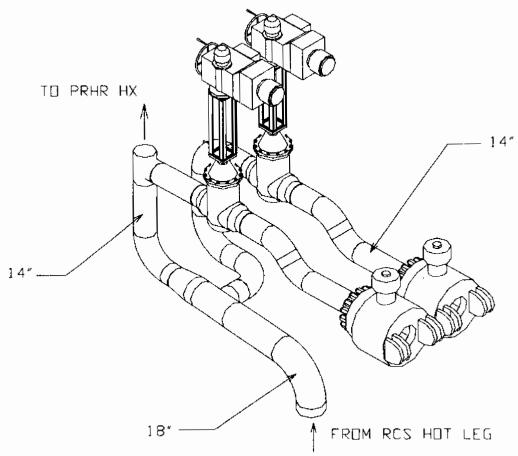
AP600



- **AP1000 IRWST Injection Capacity Increased**
  - Pipe / valves 6/8" > 8/10"
  - Initial water level increased
  - Flow capacity increased 84%
- **AP1000 Cont. Recirc. Capacity Increased**
  - Pipe / valves 6/8" > 8"
  - Initial IRWST level increased
  - Initial flooding of refueling cavity prevented > check valves
  - RNS suction from outside cont.
  - Flow capacity increased 113%

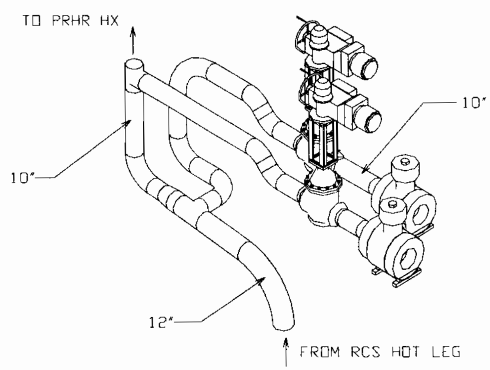
Comparison of 4th Stage ADS

AP1000



AP1000

AP600



AP600

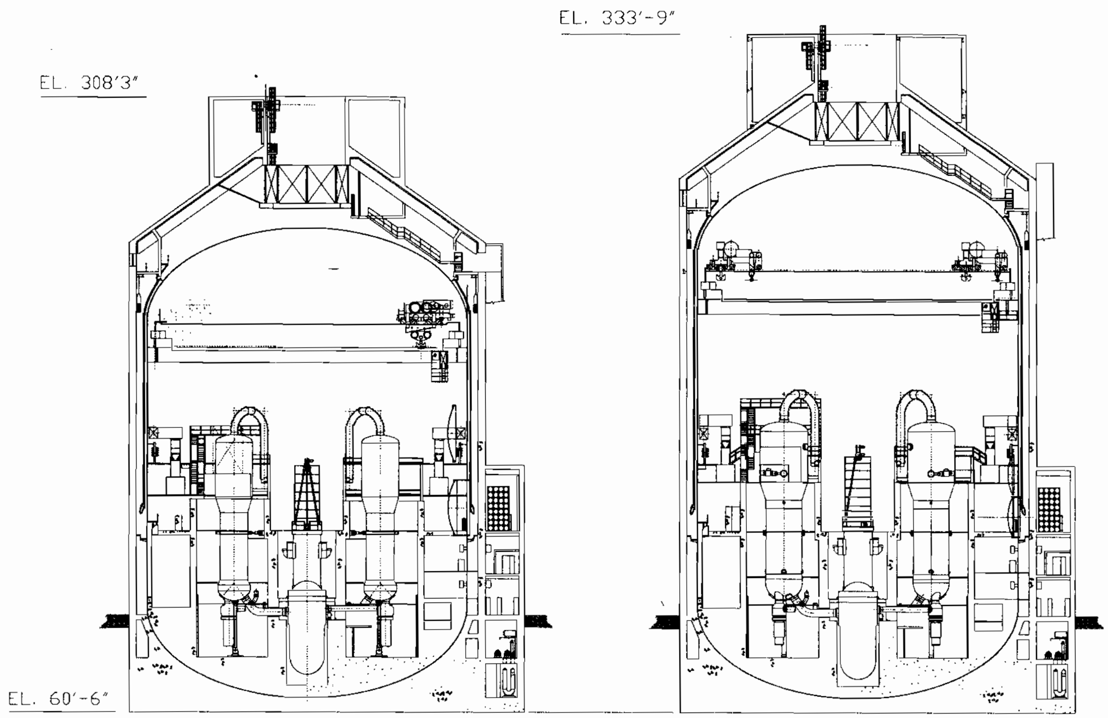
- **AP1000 ADS 4 Capacity Increased**
  - –Pipe / valves 10” > 14”
  - –Flow capacity increased 89%

AP1000 Safety Margins

	Typical Plant	AP600	AP1000
Loss Flow Margin to DNBR Limit	~ 1 – 5%	15.8%	~19%
Feedline Break Subcooling Margin	>0°F	~170°F	~140°F
SG Tube Rupture	Operator actions required in 10 min	Operator actions ***NOT*** required	Same as AP600
Small LOCA	3" LOCA core uncovers PCT ~1500 °F	≤ 8" LOCA ***NO*** core uncovery	Same as AP600
Large LOCA PCT (with uncertainty)	2000 – 2200°F	1676°F	~2120°F

AP1000 Containment Comparison

AP600



AP1000

	AP600	AP1000
Total Free Volume	100%	122%
Design Pressure, psig	45	59
Shell Thickness	1 5/8"	1 3/4"
Material	A537 Class 2	SA738 Grade B
PCS Water Drain Vol (72 hr)	100%	162%
Design – Peak Cont. Pres. (psi)	1.6 (LLOCA) 0.9 (MSLB)	3.6 (LLOCA) 1.7 (MSLB)

AP1000 Design Changes Considered in PRA

- **Core Power**
  - vs PXS / PCS increased capacities
  - Low boron core improves ATWT response
- **Larger SG**
  - More / longer SG tubes impacts SGTR IE Freq.
- **Variable Speed RCPs**
  - Only used during startup / shutdown conditions
- **More SG Safety Valves**
  - Impacts steam line break IE Freq.
- **Main Feedwater**
  - 3 constant speed pumps vs 2 variable speed pumps
- **Circulating Water Pumps**, 2 > 3 pumps

AP1000 Design Changes Considered in PRA

- **PXS Design Changes**
  - –PRHR, ADS 4, IRWST inject & Cont Recirc capacities increased by power ratio
    - –Verify success criteria with T&H analysis
    - –Larger PRHR HX could affect PRHR HX tube rupture IE Freq.
  - –CMT size & injection rate increased 25%
    - –Verify success criteria with T&H analysis
  - –Accum size not increased
    - * –Impacts success criteria for large CL LOCAs need both key change
    - –Verify success criteria with T&H analysis
  - –IRWST vents changed to prevent H2 release near containment wall

AP1000 Design Changes Considered in PRA

- **PCS Design Changes**
  - –PCS design pressure increased
    - –Thicker wall, higher strength steel, code changes
    - –Reduces overpressure margin for PRA
    - –Check performance without PCS water drain
      - –3rd / diverse PCS water drain valve reduces importance
- **Nonsafety Defense-In-Depth Systems**
  - –RNS injection water supply changed from IRWST to Cask Load Pit
    - –Outside containment water supply avoids possible adverse interaction
  - –Other DID system capacities increased to cover power increase
- **Electrical Bus / Component Train Assignments**

Summary

- **AP1000 Plant Is Very Similar To AP600**
  - –Same RCS configuration
  - –Same Safety system arrangement, capabilities
  - –Same Non-Safety DID system functions, arrangement, capabilities

- **AP1000 Uses AP600 PRA As Starting Point**
  - –Make few changes to account for differences

OVERVIEW OF AP1000 PRA

Selim Sancaktar
Fellow Engineer, Reliability and Risk Assessment
412-374-5983 - sancaks@westinghouse.com

OBJECTIVES

- The purpose of the AP1000 PRA is to provide inputs to the optimization of the AP1000 design and to verify that the US NRC PRA safety goals have been satisfied.
- As in the AP600, the PRA is being performed interactively with the design, analysis and operating procedures.

TECHNICAL SCOPE

- Since the configuration of the AP1000 reactor and safety systems is the same as the AP600, the AP600 PRA is used as the basis of the AP1000 PRA with relevant changes implemented in the model to reflect the AP1000 design changes.

TECHNICAL SCOPE

- AP1000 plant-specific T&H analyses are performed in order to determine the system success criteria.
- The CDF and LRF are calculated for internal events at-power. The off-site dose risk analysis is also performed. The external events and shutdown models are also assessed to derive plant insights and plant risk conclusions.

AP1000 PRA Report Outline

Level 1

CHAPTER 1	INTRODUCTION
CHAPTER 2	INTERNAL INITIATING EVENTS
CHAPTER 3	MODELING OF SPECIAL INITIATORS
CHAPTER 4	EVENT TREE MODELS
CHAPTER 6	SUCCESS CRITERIA ANALYSIS
CHAPTER 7	FAULT TREE GUIDELINES
CHAPTER 8	PASSIVE CORE COOLING SYSTEM – PASSIVE RESIDUAL HEAT REMOVAL
CHAPTER 9	PASSIVE CORE COOLING SYSTEM – CORE MAKEUP TANK
CHAPTER 10	PASSIVE CORE COOLING SYSTEM – ACCUMULATOR
CHAPTER 11	PASSIVE COOLING SYSTEM – AUTOMATIC DEPRESSURIZATION SYSTEM
CHAPTER 12	PASSIVE CORE COOLING SYSTEM – IN-CONTAINMENT REFUELING WATER STORAGE TANK
CHAPTER 13	PASSIVE CONTAINMENT COOLING SYSTEM
CHAPTER 14	MAIN AND STARTUP FEEDWATER SYSTEM
CHAPTER 15	CHEMICAL AND VOLUME CONTROL SYSTEM
CHAPTER 16	CONTAINMENT HYDROGEN CONTROL SYSTEM
CHAPTER 17	NORMAL RESIDUAL HEAT REMOVAL SYSTEM
CHAPTER 18	COMPONENT COOLING WATER SYSTEM
CHAPTER 19	SERVICE WATER SYSTEM
CHAPTER 20	CENTRAL CHILLED WATER SYSTEM
CHAPTER 21	AC POWER SYSTEM
CHAPTER 22	CLASS 1E DC AND UNINTERRUPTIBLE POWER SUPPLY SYSTEM
CHAPTER 23	NON-CLASS 1E DC AND UPS SYSTEM
CHAPTER 24	CONTAINMENT ISOLATION
CHAPTER 25	COMPRESSED AND INSTRUMENT AIR SYSTEM
CHAPTER 26	PROTECTION AND SAFETY MONITORING SYSTEM
CHAPTER 27	DIVERSE ACTUATION SYSTEM
CHAPTER 28	PLANT CONTROL SYSTEM
CHAPTER 29	COMMON-CAUSE ANALYSIS
CHAPTER 30	HUMAN RELIABILITY ANALYSIS
CHAPTER 31	OTHER EVENT TREE NODE PROBABILITIES
CHAPTER 32	DATA ANALYSIS AND MASTER DATA BANK
CHAPTER 33	FAULT TREE AND CORE DAMAGE QUANTIFICATION

AP1000 PRA Report Outline

Level 2+3

LEVEL I PRA

Selim Sancaktar

Fellow Engineer, Reliability and Risk Assessment

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CONTENTS

- Internal Events at Power
  - –Initiating Events
  - –Event trees and success criteria
  - –Systems Analysis
  - –CDF Quantification
  - –Sensitivity and Importance Analyses
  - –Uncertainty Analysis

CONTENTS

- Shutdown Events
- Internal Flooding and Fire
- Seismic Margins Evaluation
- Other External Events
- Summary of Results

large LOCA used NUREG-5750 use generic large break LOCA w. 10^{-6} initiating event frequency

Comparison of AP600 and AP1000 Initiating Events

was dominate large LOCA

new categories for AP1000

requires 2 out of 2 accumulators.

	Event Category	AP1000 Frequency	AP600 Frequency	Reason for Change
1	Large LOCA	5.04E-06	1.05E-04	use industry data
2	Large Spurious ADS Actuation	5.40E-05	N/A	new category - moved out of Large LOCA
3	Medium LOCA	4.36E-04	1.62E-04	use industry data
4	Core makeup tank line break	9.31E-05	8.94E-05	number of pipe segments
5	Safety injection line break	2.12E-04	1.04E-04	number of pipe segments
	Intermediate LOCA	N/A	0.00077	included in mloca
6	Small LOCA	5.00E-04	1.01E-04	use industry data
7	RCS leakage	6.20E-03	1.20E-02	use industry data
8	Passive residual heat removal tube rupture	1.34E-04	2.50E-04	number of pipe segments; updated database
9	Steam generator tube rupture	3.88E-03	5.20E-03	number of pipe segments; updated database
10	Reactor vessel rupture	1.00E-08	1.00E-08	
11	Interfacing system LOCA	5.00E-11	5.00E-11	
12	Transient with main FW	1.40E+00	1.44E+00	
13	Loss of RCS flow	1.80E-02	1.80E-02	
14	Loss of main FW to one steam generator	1.92E-01	1.92E-01	
15	Core power excursion	4.50E-03	4.50E-03	
16	Loss of component cooling water/service water	1.44E-01	1.44E-01	
17	Loss of main FW to both steam generators	3.35E-01	3.35E-01	
18	Loss of condenser	1.12E-01	1.12E-01	
19	Loss of compressed air	3.48E-02	3.48E-02	
20	Loss of offsite power	1.20E-01	1.20E-01	
21	Main steam line break downstream of MSIVs	5.96E-04	5.96E-04	
22	Main steam line break upstream of MSIVs	3.72E-04	3.72E-04	
23	Main steam line stuck-open valve	2.39E-03	1.21E-03	number of valves increased
24	ATWS precursor without main feedwater	4.81E-01	4.81E-01	
25	ATWS precursor With SI	1.48E-02	2.05E-02	contributing events changed
26	ATWS precursor with main feedwater	1.17E+00	1.17E+00	
	Total (Excluding ATWS precursors) =	2.38E+00	2.42E+00	

EVENT TREES AND SUCCESS CRITERIA

- For each initiating event, an event tree model is created.
- Differences between AP600 and AP1000 ET models are discussed in the next slide.
- Success criteria for each event tree node is reviewed and modified if needed.

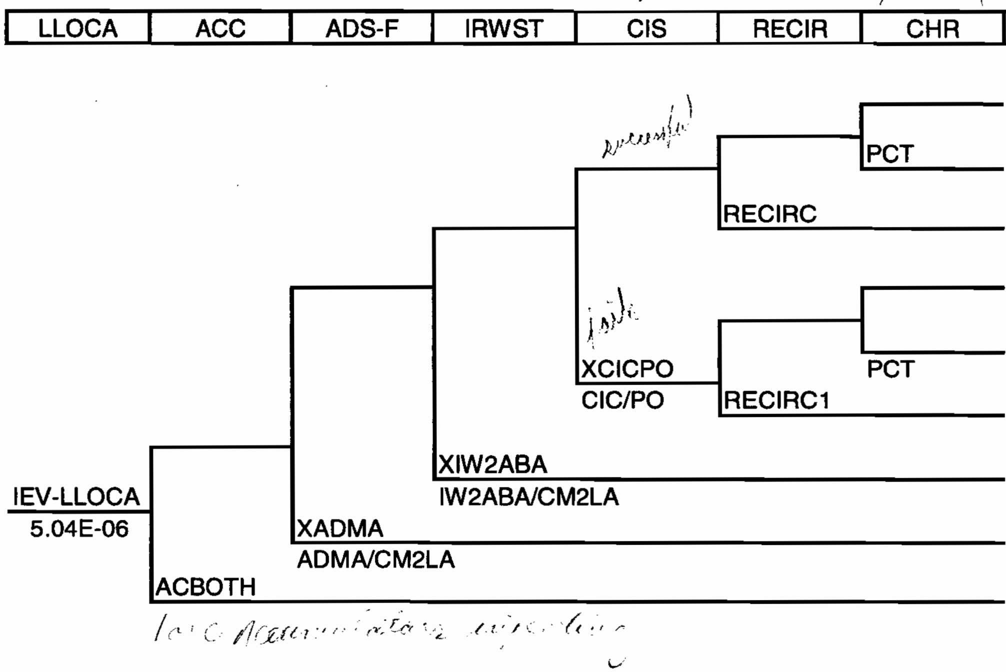
AP1000 Large LOCA Event Tree

New (?) larger ... need on stuck open HVAC penetration.

Point remains (?) decay heat with just air alone no PCS

In AP600 this was not modeled because air alone could remove all decay heat.

Late containment failure



	PDS	FREQ.		LCF
1	LLO-OK1			
2	LCF		lcllo-02	6.87E-12
3	3BL	8.43E-11	2lllo-03	
4	LLO-OK2			
5	LCF		lcllo-05	1.60E-12
6	3BL	1.77E-14	2lllo-06	
7	3BE	3.40E-10	2ello-07	
8	3D	1.97E-09	3dllo-08	
9	3BR	4.26E-08	2rllo-09	
Total CDF =		4.50E-08		8.47E-12

List of Top Events

Event	Description
LLOCA	Large LOCA Event Occurs
ACC	Accumulators Inject
ADS-F	Full RCS Depressurization by ADS occurs
IRWST	RCS Refill from IRWST by Gravity Injection Occurs
CIS	Containment Isolation Occurs
RECIR	Water Recirculation to RPV from the Sump Occurs
CHR	Containment Cooling is Established

Event Tree Node Success Criteria

Internal Events At-Power

Node	Node Description	Success Criteria
AC1A	FAILURE OF 1/1 ACCUMULATOR	Same as AP600
AC2AB	FAILURE OF 2/2 ACCUMULATORS	Same as AP600
ACBOTH	ANY ONE OF TWO ACCUMULATOR TRAINS FAIL	Same as AC2AB except failure of 1 of 2 accumulators - for Large LOCA
AD1	FAILURE OF PARTIAL ADS DEPRESSURIZATION	Same as AP600 except 2 stage 2/3 ADS lines need to open
AD1A	FAILURE OF PARTIAL ADS DEPRESSURIZATION	Same as AP600 except 2 stage 2/3 ADS lines need to open
ADA	FAILURE OF FULL ADS DEPRESSURIZATION	Same as AP600 except 3/4 stage 4 ADS lines need to open
ADAB	FAILURE OF FULL ADS DEPRESSURIZATION	Same as AP600 except 3/4 stage 4 ADS lines need to open
ADAL	FAILURE OF FULL ADS DEPRESSURIZATION	Same as AP600 except 3/4 stage 4 ADS lines need to open
ADB	FAILURE OF FULL ADS DEPRESSURIZATION	Same as AP600 except 3/4 stage 4 ADS lines need to open
ADF	FAILURE OF PARTIAL ADS DEPRESSURIZATION	Same as AP600
ADL	FAILURE OF FULL ADS DEPRESSURIZATION	Same as AP600 except 3/4 stage 4 ADS lines need to open
ADM	FAILURE OF FULL ADS DEPRESSURIZATION	Same as AP600 except 3/4 stage 4 ADS lines need to open
ADMA	FAILURE OF FULL ADS DEPRESSURIZATION	Same as AP600 except 3/4 stage 4 ADS lines need to open
ADQ	FAILURE OF FULL ADS DEPRESSURIZATION	Same as AP600 except 3/4 stage 4 ADS lines need to open
ADR	FAILURE OF PARTIAL ADS DEPRESSURIZATION	Same as AP600 except 2 stage 2/3 ADS lines need to open
ADRA	FAILURE OF PARTIAL ADS DEPRESSURIZATION	Same as AP600 except 2 stage 2/3 ADS lines need to open
ADS	FAILURE OF FULL ADS DEPRESSURIZATION	Same as AP600 except 3/4 stage 4 ADS lines need to open
ADT	FAILURE OF FULL ADS DEPRESSURIZATION	Same as AP600 except 3/4 stage 4 ADS lines need to open
ADU	FAILURE OF PARTIAL ADS DEPRESSURIZATION	Same as AP600 except 2 stage 2/3 ADS lines need to open
ADUM	FAILURE OF PARTIAL ADS DEPRESSURIZATION	Same as AP600 except 2 stage 2/3 ADS lines need to open
ADV	FAILURE OF PARTIAL ADS DEPRESSURIZATION	Same as AP600 except 2 stage 2/3 ADS lines need to open
ADW	FAILURE OF FULL ADS DEPRESSURIZATION	Same as AP600 except 3/4 stage 4 ADS lines need to open
ADZ	FAILURE OF PARTIAL ADS DEPRESSURIZATION	Same as AP600 except 2 stage 2/3 ADS lines need to open
BL	MSLB UPSTREAM OF MSIVs OUTSIDE CONTAINMENT	Same as AP600
CIA	FAILURE OF SG ISOLATION	Same as AP600
CIB	FAILURE OF SG ISOLATION	Same as AP600
CIC	CONTAINMENT ISOLATION FAILURE	Same as AP600

Event Tree Node Success Criteria

Internal Events At-Power

Node	Node Description	Success Criteria
CM1A	RCPS DO NOT TRIP OR 1 TRAIN OF C MT FAILS - 1/1 CMT TRAIN AVAILA.	Same as AP600
CM2AB	FAILURE OF TWO OF TWO CORE MAKEUP TANKS	Same as AP600
CM2LA	FAILURE TWO OF TWO CORE MAKEUP TANKS	Same as AP600
CM2NL	FAILURE OF 2/2 CMT	Same as AP600
CM2P	FAILURE OF TWO OF TWO CORE MAKEUP TANKS	Same as AP600
CM2SL	FAILURE OF TWO OF TWO CORE MAKEUP TANKS	Same as AP600
COND	TURBINE BYPASS TO MAIN CONDENSER (2/4 VALVES TO OPEN)	Same as AP600
COND1	FAILURE OF STEAM DUMP TO CONDENSER	Same as AP600
CSAX	FAILURE OF PRIMARY DEPRESSURIZATION BY CVS	Same as AP600
CSBOR1	FAILURE OF MANUAL BORATION BY CVS	Same as AP600
CSP	FAILURE OF CVS	Same as AP600
CVS1	FAILURE OF CVS	Same as AP600
DAS	FAILURE OF DIVERSE ACTUATION SYSTEM	Same as AP600
DAS1	FAILURE OF DIVERSE ACTUATION SYSTEM	Same as AP600
DGEN	FAILURE OF TWO OF TWO DIESEL GENERATORS	Same as AP600
FWF	MAIN FEEDWATER SYSTEM FAILS TO CONTINUE OPERATING	Same as AP600 - although 3 MFW pumps are available, only two are credited
FWT	FAILURE OF MFW SYSTEM	Same as AP600 - although 3 MFW pumps are available, only two are credited
IW1A	FAILURE OF ONE OF ONE IRWST INJECTION LINE	Same as AP600
IW1AM	FAILURE OF ONE OF ONE IRWST INJECTION LINE	Same as AP600
IW2AB	FAILURE OF TWO OF TWO IRWST INJECTION LINES	Same as AP600
IW2ABA	FAILURE OF TWO OF TWO IRWST INJECTION LINES	Same as AP600
IW2ABB	FAILURE OF TWO OF TWO IRWST INJECTION LINES	Same as AP600
IW2ABBM	FAILURE OF TWO OF TWO IRWST INJECTION LINES	Same as AP600
IW2ABM	FAILURE OF TWO OF TWO IRWST INJECTION LINES	Same as AP600
IW2ABP	FAILURE OF TWO OF TWO IRWST INJECTION LINES	Same as AP600
IW2ABPM	FAILURE OF TWO OF TWO IRWST INJECTION LINES	Same as AP600
MGSET	CONTROL ROD MG SETS FAIL TO TRIP	Same as AP600

Event Tree Node Success Criteria

Internal Events At-Power

Node	Node Description	Success Criteria
PCB	PASSIVE CONTAINMENT COOLING SYSTEM FAILS - AFTER SBO EVENT	same as PCT - with Station blackout power support system conditions
PCP	PASSIVE CONTAINMENT COOLING FAILS	Same as PCT - with loss of offsite power support system conditions
PCT	PASSIVE CONTAINMENT COOLING SYSTEM FAILS	1/3 PCS lines must inject; other wise the same as AP600
PO	PRE-EXISTING CONTAINMENT OPENING > 100 SQ CM	Same as AP600 (scalar quantity, no additional success criterion is needed.)
PRES	PRZ SV FAILURE FOR LOSS OF MFW ATWS, NO UET	Same as AP600
PRESU	INADEQUATE PRS RELIEF FOR LOSS OF MFW ATWS, WITH UET	Same as AP600 with UET = 0
PRI	PRHR ISOLATION FAILURE FOLLOWING PRHR TUBE RUPTURE	Same as AP600 and at least 1/2 gutter isolation valves must close
PRL	FAILURE OF PASSIVE RHR SYSTEM	Same as AP600 and at least 1/2 gutter isolation valves must close
PRP	FAILURE OF PASSIVE RHR SYSTEM	Same as AP600 and at least 1/2 gutter isolation valves must close
PRS	FAILURE OF PASSIVE RHR SYSTEM	Same as AP600 and at least 1/2 gutter isolation valves must close
PRSOV	EITHER PRZR SV FAILS TO RECLOSE	Same as AP600 and at least 1/2 gutter isolation valves must close
PRT	FAILURE OF PASSIVE RHR SYSTEM	Same as AP600 and at least 1/2 gutter isolation valves must close
PRTA	FAILURE OF PRHR	Same as AP600 and at least 1/2 gutter isolation valves must close
PRW	FAILURE OF PASSIVE RHR SYSTEM	Same as AP600 and at least 1/2 gutter isolation valves must close
R05	FAILURE TO RECOVER OFFSITE AC POWER IN 30 MINUTES	Same as AP600
RCL	FAILURE OF REACTOR COOLANT PUMPS TO TRIP	Same as AP600
RCN	FAILURE OF RCP TRIP	Same as AP600
RCT	FAILURE OF REACTOR COOLANT PUMPS TO TRIP	Same as AP600
RECIRC	FAILURE OF RECIRCULATION	Same as AP600
RECIRC1	CONTAINMENT SUMP RECIRCULATION FAILS	Same as RECIRC with 2/4 recirculation lines need to open (after failure of CIS)
RECIRC1B	RECIRC FAILS DUR BLACKOUT GIVEN CONTAIN ISOL. FAILS	Same as RECIRC with 2/4 recirculation lines need to open (after failure of CIS)
RECIRC1P	RECIRC FAILS DUR LOSP GIVEN CONT. ISOL. FAILS	Same as RECIRC with 2/4 recirculation lines need to open (after failure of CIS)
RECIRCB	CONTAINMENT SUMP RECIRCULATION FAILS	Same as AP600
RECIRCP	FAILURE OF RECIRCULATION	Same as AP600
RNH	RNS FAILS TO INJ ECT / RECIRCULATE / REMOVE HEAT	Same as RNR plus at least 1/2 CCW HX must remove decay heat

Event Tree Node Success Criteria

Internal Events At-Power

Node	Node Description	Success Criteria
RNHP	RNS FAILS TO INJECT /RECIRC/ REMOVE HEAT WITH LOSP	Same as RNH - with loss of offsite power support system conditions
RNN	RNS FAILS TO REMOVE DECAY HEAT FOR TRANS WITH PRHR SUCC	At least 1/2 RNS pump trains take suction from the IRWST and inject into the IRWST; the HX in the operating train(s) remove heat through CCW
RNNP	RNS FAILS TO REMOVE DECAY HEAT FOR TRANS WITH PRHR SUCC, FOLLOWING A LOSS OF OFFSITE POWER EVENT	Same as RNN - with loss of offsite power support system conditions
RNP	FAILURE OF NORMAL RHR IN INJECTION MODE	Same as RNR - with loss of offsite power support system conditions
RNR	FAILURE OF NORMAL RHR IN INJECTION / RECIRCULATION MODE	Same as AP600 with initial suction source coming from Cask loading pit.
RTPMS	FAILURE OF ROD INSERTION BY PMS	Same as AP600
RTPMS1	FAILURE OF ROD INSERTION BY PMS	Same as AP600
RTSTP	FAILURE OF MANUAL REACTOR TRIP	Same as AP600
SDMAN	OPERATOR FAILS TO PERFORM CNTRL REACTOR SHUTDOWN DURING ACCIDENT	Same as AP600
SFW	STARTUP FEEDWATER FAILS	Same as AP600
SFW1	FAILURE OF STARTUP FEEDWATER SYSTEM	Same as AP600
SFWA	FAILURE OF STARTUP FEEDWATER SYSTEM	Same as AP600
SFWM	FAILURE OF SFW SYSTEM WITH LOSS OF COMPRESSED AIR	Same as AP600
SFWP	FAILURE OF STARTUP FEEDWATER SYSTEM	Same as AP600
SFWT	FAILURE OF STARTUP FEEDWATER SYSTEM	Same as AP600
SGHL	FAILURE OF CVS AND STARTUP FEEDWATER ISOLATION	Same as AP600
SGTR	CONSEQUENTIAL SGTR OCCURS	Same as AP600
SGTR1	SINGLE CONSEQUENTIAL SGTR	Same as AP600
SLSOV	ANY SECOND. SIDE RELIEF VALVE FAILS TO RECLOSE (1 SV + PORV)	Same as AP600
SLSOV1	ANY SECOND. SIDE RELIEF VALVE FAILS TO CLOSE (2 SV + PORV)	Same as AP600
SLSOV2	ANY SECOND. SIDE RELIEF VALVE FAILS TO CLOSE (1 SV)	Same as AP600
SLSOV3	FAILURE TO RECLOSE OF SG PORV & 1 SG SV ON RUPTURED SG	Same as AP600

SYSTEMS ANALYSIS

- Fault tree models are used for systems analysis
- Same FT guidelines as AP600 PRA are used.
- Fault trees are modified as needed to reflect the current AP1000 design.

4 recirc paths
2 [illegible] CMC with high pressure squib valves see page 23

Typical AP1000 PRA System Failure Probabilities

Failure System/Function	Probability	Fault Tree Name
CMT Valve Signal	5.70E-07	CMT-IC11 (one train; auto and manual actuation)
PRHR Valve Signal	1.10E-06	RHR-IC01 (one train; auto and manual actuation)
Passive Cont. Cool.	1.80E-06	PCT
--		
Reactor Trip by PMS	1.20E-05	RTPMS (including operator actions)
Accumulators	6.90E-05	AC2AB
IRWST inj.	6.90E-05	IW2AB
ADS	9.30E-05	ADS (including operator actions)
--		
Passive PRHR	2.00E-04	PRT
Core Makeup Tanks	1.10E-04	CM2SL
125 vdc 1E Bus	3.10E-04	IDADS1 (one bus only)
DC Bus (Non-1E)	3.40E-04	ED1DS1 (one bus only)
RC Pump Trip	5.90E-04	RCT
--		
Chilled Water	1.40E-03	VWH
Containment Isol.	1.60E-03	CIC
Reactor Trip by DAS	1.70E-03	DAS (including operator action; excluding MGSET
failure))		
6900 vac Bus	3.20E-03	ECES1 (obe bus only)
CVS	3.40E-03	CVS1
480 vac Bus	5.90E-03	ECEK11 (one bus only)
Service Water	6.20E-03	SWT
Comp. Cooling Water	6.30E-03	CCT
--		
Diesel Generators	1.00E-02	DGEN
Startup Feedwater	1.70E-02	SFWT
Compressed Air	1.30E-02	CAIR
Condenser	2.40E-02	CDS
Main Feedwater	2.80E-02	FWT (including condenser)
RNS	9.10E-02	RNR
--		
Hydrogen Control	1.00E-01	VLH

driven by operator failure look into this

CDF QUANTIFICATION

- 26 event trees are quantified sequence by sequence
- 190 of the sequences had frequencies calculated for them
- The plant CDF is calculated to be 2.41e-07/year.

Contribution of Initiating Events to AP1000 CDF

gives you a feel for the amount of margin you have

	IEV CATEGORY	CONTRIBUTION TO CDF %	CONTRIBUTION TO CDF %	IEV FREQUENCY	NUMBER OF CUTSETS	CONDITIONAL CDF
1	IEV-SI-LB	9.50E-08	39.43	2.12E-04	1160	4.48E-04
2	IEV-LLOCA	4.50E-08	18.66	5.00E-06	286	8.99E-03
3	IEV-SPADS	2.96E-08	12.28	5.40E-05	1078	5.48E-04
4	IEV-SLOCA	1.81E-08	7.5	5.00E-04	1638	3.62E-05
5	IEV-MLOCA	1.61E-08	6.69	4.36E-04	1681	3.70E-05
6	IEV-RV-RP	1.00E-08	4.15	1.00E-08	1	1.00E+00
7	IEV-SGTR	6.79E-09	2.82	3.88E-03	3076	1.75E-06
8	IEV-CMTLB	3.68E-09	1.53	9.31E-05	987	3.95E-05
9	IEV-ATWS	3.61E-09	1.5	4.81E-01	136	7.49E-09
10	IEV-TRANS	3.08E-09	1.28	1.40E+00	1500	2.20E-09
11	IEV-RCSLK	1.71E-09	0.71	6.20E-03	1526	2.75E-07
12	IEV-POWEX	1.66E-09	0.69	4.50E-03	701	3.69E-07
13	IEV-LCOND	1.24E-09	0.52	1.12E-01	858	1.11E-08
14	IEV-LOSP	9.58E-10	0.4	1.20E-01	530	7.98E-09
15	IEV-LMFW	8.70E-10	0.36	3.35E-01	1334	2.60E-09
16	IEV-ATW-T	7.12E-10	0.3	1.17E+00	13	6.09E-10
17	IEV-LCAS	6.72E-10	0.28	3.48E-02	417	1.93E-08
18	IEV-SLB-V	6.06E-10	0.25	2.39E-03	305	2.54E-07
19	IEV-PRSTR	5.02E-10	0.21	1.34E-04	317	3.74E-06
20	IEV-LMFW1	4.53E-10	0.19	1.92E-01	763	2.36E-09
21	IEV-LCCW	3.23E-10	0.13	1.44E-01	690	2.24E-09
22	IEV-SLB-U	1.31E-10	0.05	3.72E-04	160	3.51E-07
23	IEV-ATW-S	1.11E-10	0.05	1.48E-02	55	7.48E-09
24	IEV-ISLOC	5.00E-11	0.02	5.00E-11	1	1.00E+00
25	IEV-LRCS	3.52E-11	0.01	1.80E-02	143	1.96E-09
26	IEV-SLB-D	9.15E-12	0	5.96E-04	18	1.54E-08
	Totals =	**2.41E-07**	**100**	**2.38***	**19374**	

Conditional CDF * IEV frequency = Contribution to CDF

* - The total initiating event frequency excludes the three ATWS precursor frequencies

Comparison of AP600 and AP1000 CDF

IEV CATEGORY	AP600 CONTRIBUTION TO CDF	AP600 CONTRIBUTION TO CDF (%)	AP600 IEV FREQUENCY	AP600 CONDITIONAL CDP (P1)	AP1000 CONTRIBUTION TO CDF	AP1000 CONTRIBUTION TO CDF (%)	AP1000 IEV FREQUENCY	AP1000 CONDITIONAL CDP (P2)	RATIO P2/P1
IEV-LLOCA	5.02E-08	29.67	1.05E-04	4.78E-04	4.50E-08	18.66	5.00E-06	8.99E-03	18.8
IEV-SI-LB	3.82E-08	22.58	1.04E-04	3.67E-04	9.50E-08	39.43	2.12E-04	4.48E-04	1.2
IEV-NLOCA	3.15E-08	18.63	7.70E-04	4.09E-05					
IEV-RV-RP	1.00E-08	5.91	1.00E-08	1.00E+00	1.00E-08	4.15	1.00E-08	1.00E+00	1
IEV-ATWS	8.98E-09	5.31	4.81E-01	1.87E-08	3.61E-09	1.5	4.81E-01	7.49E-09	0.4
IEV-MLOCA	6.23E-09	3.68	1.62E-04	3.85E-05	1.61E-08	6.69	4.36E-04	3.70E-05	1
IEV-SGTR	6.08E-09	3.6	5.20E-03	1.17E-06	6.79E-09	2.82	3.88E-03	1.75E-06	1.5
IEV-SLOCA	4.05E-09	2.4	1.01E-04	4.01E-05	1.81E-08	7.5	5.00E-04	3.62E-05	0.9
IEV-CMTLB	3.54E-09	2.09	8.94E-05	3.96E-05	3.68E-09	1.53	9.31E-05	3.95E-05	1
IEV-RCSLK	2.26E-09	1.34	1.20E-02	1.89E-07	1.71E-09	0.71	6.20E-03	2.75E-07	1.5
IEV-POWEX	1.83E-09	1.08	4.50E-03	4.07E-07	1.66E-09	0.69	4.50E-03	3.69E-07	0.9
IEV-TRANS	1.14E-09	0.67	1.40E+00	8.14E-10	3.08E-09	1.28	1.40E+00	2.20E-09	2.7
IEV-LCOND	1.03E-09	0.61	1.12E-01	9.23E-09	1.24E-09	0.52	1.12E-01	1.11E-08	1.2
IEV-LOSP	1.01E-09	0.6	1.20E-01	8.40E-09	9.58E-10	0.4	1.20E-01	7.98E-09	1
IEV-ATW-T	7.12E-10	0.42	1.17E+00	6.09E-10	7.12E-10	0.3	1.17E+00	6.09E-10	1
IEV-PRSTR	5.58E-10	0.33	2.50E-04	2.23E-06	5.02E-10	0.21	1.34E-04	3.74E-06	1.7
IEV-SLB-V	4.82E-10	0.28	1.21E-03	3.98E-07	6.06E-10	0.25	1.21E-03	5.01E-07	1.3
IEV-ATW-S	3.82E-10	0.23	2.05E-02	1.86E-08	1.11E-10	0.05	1.48E-02	7.48E-09	0.4
IEV-LMFW	3.03E-10	0.18	3.35E-01	9.04E-10	8.70E-10	0.36	3.35E-01	2.60E-09	2.9
IEV-LMFW1	1.76E-10	0.1	1.92E-01	9.16E-10	4.53E-10	0.19	1.92E-01	2.36E-09	2.6
IEV-LCAS	1.73E-10	0.1	3.48E-02	4.98E-09	6.72E-10	0.28	3.48E-02	1.93E-08	3.9
IEV-SLB-U	1.23E-10	0.07	3.72E-04	3.31E-07	1.31E-10	0.05	3.72E-04	3.51E-07	1.1
IEV-LCCW	1.23E-10	0.07	1.44E-01	8.52E-10	3.23E-10	0.13	1.44E-01	2.24E-09	2.6
IEV-ISLOC	5.00E-11	0.03	5.00E-11	1.00E+00	5.00E-11	0.02	5.00E-11	1.00E+00	1
IEV-LRCS	1.27E-11	0.01	1.80E-02	7.06E-10	3.52E-11	0.01	1.80E-02	1.96E-09	2.8
IEV-SLB-D	9.46E-12	0.01	5.96E-04	1.59E-08	9.15E-12	0	5.96E-04	1.54E-08	1
IEV-SPADS					2.96E-08	12.28	5.40E-05	5.48E-04	
Totals:	**1.69E-07**	**100**	**2.38E+00**		**2.41E-07**	**100**	**2.37E+00**		

need to add to LLOCA in AP600

AP1000 PRA Dominant CDF Sequences

	Sequence Frequency	to CDF % Contrib	Cum. % Contrib	Sequence ID	Sequence Description	Event Identifier
1	6.88E-08	28.52	28.52	2esil-07	SAFETY INJECTION LINE BREAK INITIATING EVENT OCCURS	IEV-SI-LB
					RCPS TRIP AND CMT INJECTION IS SUCCESSFUL - 1 OF 2 CMT TRAINS	DEL-XCM1A
					SUCCESS OF FULL ADS DEPRESSURIZATION	DEL-ADM
					FAILURE OF ONE OF ONE IRWST INJECTION LINE	SYS-IW1A
2	4.26E-08	17.66	46.18	2rllo-09	LARGE LOCA INITIATING EVENT OCCURS	IEV-LLOCA
					ANY ONE OF TWO ACCUMULATOR TRAINS FAIL	SYS-ACBOTH
3	2.13E-08	8.82	55	3dsad-08	SPURIOUS ADS INITIATING EVENT OCCURS	IEV-SPADS
					SUCCESS OF 1/2 OR 2/2 ACCUMULATORS	DEL-AC2AB
					FAILURE OF ADS OR CMT	SYS-XADMA
4	1.98E-08	8.23	63.23	3dsil-08	SAFETY INJECTION LINE BREAK INITIATING EVENT OCCURS	IEV-SI-LB
					RCPS TRIP AND CMT INJECTION IS SUCCESSFUL - 1 OF 2 CMT TRAINS	DEL-XCM1A
					FAILURE OF FULL ADS DEPRESSURIZATION	SYS-ADM
5	1.00E-08	4.15	67.38	3crvr-02	REACTOR VESSEL RUPTURE INITIATING EVENT OCCURS	IEV-RV-RP
6	8.44E-09	3.5	70.88	2lslo-05	SMALL LOCA INITIATING EVENT OCCURS	IEV-SLOCA
					SUCCESS OF CMT & RCP TRIP	DEL-XCM2SL
					SUCCESS OF PASSIVE RHR SYSTEM	DEL-PRL
					SUCCESS OF FULL ADS DEPRESSURIZATION	DEL-ADS
					FAILURE OF NORMAL RHR IN INJECTION MODE	SYS-RNR
					SUCCESS OF TWO OF TWO IRWST INJECTION LINES	DEL-IW2AB
					SUCCESS OF CIS & PRE-EXISTING CONTAINMENT OPENING	DEL-XCICPO
					FAILURE OF RECIRCULATION	SYS-RECIRC

AP1000 PRA Dominant CDF Sequences

	Sequence Frequency	% Contrib	Cum. % Contrib	Sequence ID	Sequence Description	Event Identifier
7	7.35E-09	3.05	73.93	2lmlo-05	MEDIUM LOCA INITIATING EVENT OCCURS	IEV-MLOCA
					SUCCESS OF CMT & RCP TRIP	DEL-XCM2NL
					SUCCESS OF FULL ADS DEPRESSURIZATION	DEL-ADM
					FAILURE OF NORMAL RHR IN INJECTION MODE	SYS-RNR
					SUCCESS OF TWO OF TWO IRWST INJECTION LINES	DEL-IW2AB
					SUCCESS OF CIS & PRE-EXISTING CONTAINMENT OPENING	DEL-XCICPO
					FAILURE OF RECIRCULATION	SYS-RECIRC
8	5.11E-09	2.12	76.05	3dmlo-12	SMALL LOCA INITIATING EVENT OCCURS	IEV-SLOCA
					SUCCESS OF CMT & RCP TRIP	DEL-XCM2SL
					SUCCESS OF PASSIVE RHR SYSTEM	DEL-PRL
					FAILURE OF FULL ADS DEPRESSURIZATION	SYS-ADS
					SUCCESS OF PARTIAL ADS DEPRESSURIZATION	DEL-ADV
					FAILURE OF NORMAL RHR IN INJECTION MODE	SYS-RNR
9	4.46E-09	1.85	77.9	3dmlo-12	MEDIUM LOCA INITIATING EVENT OCCURS	IEV-MLOCA
					SUCCESS OF CMT & RCP TRIP	DEL-XCM2NL
					FAILURE OF FULL ADS DEPRESSURIZATION	SYS-ADM
					SUCCESS OF PARTIAL ADS DEPRESSURIZATION	DEL-ADU
					FAILURE OF NORMAL RHR IN INJECTION MODE	SYS-RNR
10	3.72E-09	1.54	79.44	2rsad-09	SPURIOUS ADS INITIATING EVENT OCCURS	IEV-SPADS
					FAILURE OF 2/2 ACCUMULATORS	SYS-AC2AB
11	3.67E-09	1.52	80.96	2esad-07	SPURIOUS ADS INITIATING EVENT OCCURS	IEV-SPADS
					SUCCESS OF 1/2 OR 2/2 ACCUMULATORS	DEL-AC2AB
					SUCCESS OF ADS & CMT	DEL-XADMA
					FAILURE OF IRW OR CMT	SYS-XIW2ABA

AP1000 PRA Dominant CDF Sequences

	Sequence Frequency	% Contrib	Cum. % Contrib	Sequence ID	Sequence Description	Event Identifier
12	3.57E-09	1.48	82.44	2lsil-03	SAFETY INJECTION LINE BREAK INITIATING EVENT OCCURS	IEV-SI-LB
					RCPS TRIP AND CMT INJECTION IS SUCCESSFUL - 1 OF 2 CMT TRAINS	DEL-XCM1A
					SUCCESS OF FULL ADS DEPRESSURIZATION	DEL-ADM
					IRWST INJECTION IS SUCCESSFUL - 1 OF 1 TRAINS	DEL-IW1A
					SUCCESS OF CIS & PRE-EXISTING CONTAINMENT OPENING	DEL-XCICPO
					FAILURE OF RECIRCULATION	SYS-RECIRC
13	3.55E-09	1.47	83.91	6esgt-41	SGTR EVENT SEQUENCE CONTINUES	SYS-SGTRC
					FAILURE OF CMT OR RCP TRIP	SYS-XCM2SL
					SUCCESS OF PASSIVE RHR SYSTEM	DEL-PRL
					FAILURE OF FULL ADS DEPRESSURIZATION	SYS-ADT
					FAILURE OF PARTIAL ADS DEPRESSURIZATION	SYS-ADZ
14	3.31E-09	1.37	85.28	3aatw-23	ATWS PRECURSOR WITH NO MFW EVENT SEQUENCE CONTINUES	SYS-ATWSC
					SUCCESS OF SFW OR PRHR SYSTEM	DEL-XSRT
					SUCCESS OF MANUAL REACTOR TRIP	DEL-RTSTP
					FAILURE OF MANUAL BORATION BY CVS	SYS-CSBOR1
					FAILURE OF CMT OR RCP TRIP	SYS-XCM2AB
15	3.30E-09	1.37	86.65	2eslo-09	SMALL LOCA INITIATING EVENT OCCURS	IEV-SLOCA
					SUCCESS OF CMT & RCP TRIP	DEL-XCM2SL
					SUCCESS OF PASSIVE RHR SYSTEM	DEL-PRL
					SUCCESS OF FULL ADS DEPRESSURIZATION	DEL-ADS
					FAILURE OF NORMAL RHR IN INJECTION MODE	SYS-RNR
					FAILURE OF TWO OF TWO IRWST INJECTION LINES	SYS-IW2AB

AP1000 PRA Dominant CDF Sequences

	Sequence Frequency	% Contrib	Cum. % Contrib	Sequence ID	Sequence Description	Event Identifier
16	2.88E-09	1.19	87.84	2emlo-09	MEDIUM LOCA INITIATING EVENT OCCURS	IEV-MLOCA
					SUCCESS OF CMT & RCP TRIP	DEL-XCM2NL
					SUCCESS OF FULL ADS DEPRESSURIZATION	DEL-ADM
					FAILURE OF NORMAL RHR IN INJECTION MODE	SYS-RNR
					FAILURE OF TWO OF TWO IRWST INJECTION LINES	SYS-IW2AB
17	2.19E-09	0.91	88.75	6esgt-13	SGTR EVENT SEQUENCE CONTINUES	SYS-SGTRC
					SUCCESS OF CMT & RCP TRIP	DEL-XCM2SL
					SUCCESS OF PASSIVE RHR SYSTEM	DEL-PRL
					FAILURE OF FULL ADS DEPRESSURIZATION	SYS-ADS
					FAILURE OF PARTIAL ADS DEPRESSURIZATION	SYS-ADV
18	1.97E-09	0.82	89.57	3dllo-08	LARGE LOCA INITIATING EVENT OCCURS	IEV-LLOCA
					ACCUMULATOR INJECTION IS SUCCESS FUL - 2 OF 2 TRAINS	DEL-ACBOTH
					FAILURE OF ADS OR CMT	SYS-XADMA
19	1.57E-09	0.65	90.22	2lcmt-05	CMT LINE BREAK INITIATING EVENT OCCURS	IEV-CMTLB
					RCPS TRIP AND CMT INJECTION IS SUCCESSFUL - 1 OF 2 CMT TRAINS	DEL-XCM1A
					SUCCESS OF FULL ADS DEPRESSURIZATION	DEL-ADM
					FAILURE OF NORMAL RHR IN INJECTION MODE	SYS-RNR
					SUCCESS OF TWO OF TWO IRWST INJECTION LINES	DEL-IW2AB
					SUCCESS OF CIS & PRE-EXISTING CONTAINMENT OPENING	DEL-XCICPO
					FAILURE OF RECIRCULATION	SYS-RECIRC

AP1000 PRA Dominant CDF Sequences

	Sequence Frequency	% Contrib	Cum. % Contrib	Sequence ID	Sequence Description	Event Identifier
20	1.41E-09	0.59	90.81	1atra-17	TRANSIENT WITH MFW INITIATING EVENT OCCURS	IEV-TRANS
					FAILURE OF MFW & SFW & PRHR SYSTEMS	SYS-XSTW
					SUCCESS OF CMT & RCP TRIP	DEL-XCM2AB
					FAILURE OF FULL ADS DEPRESSURIZATION	SYS-ADA
					FAILURE OF PARTIAL ADS DEPRESSURIZATION	SYS-AD1A
21	1.29E-09	0.54	91.35	3dsil-16	SAFETY INJECTION LINE BREAK INITIATING EVENT OCCURS	IEV-SI-LB
					CMT INJECTION (1 OF 1 TRAINS) OR RCP TRIP FAILS	SYS-XCM1A
					SUCCESS OF FULL ADS DEPRESSURIZATION	DEL-ADQ
					FAILURE OF 1/1 ACCUMULATOR	SYS-AC1A
22	1.13E-09	0.47	91.82	6lsgtc05	CONSEQUENTIAL SGTR EVENT OCCURS	SYS-IECSGTR
					SUCCESS OF CMT & RCP TRIP	DEL-XCM2SL
					SUCCESS OF PASSIVE RHR SYSTEM	DEL-PRL
					SUCCESS OF FULL ADS DEPRESSURIZATION	DEL-ADS
					FAILURE OF NORMAL RHR IN INJECTION MODE	SYS-RNR
					SUCCESS OF TWO OF TWO IRWST INJECTION LINES	DEL-IW2AB
					SUCCESS OF CIS & PRE-EXISTING CONTAINMENT OPENING	DEL-XCICPO
					FAILURE OF RECIRCULATION	SYS-RECIRC
23	9.98E-10	0.41	92.23	3dsil-17	SAFETY INJECTION LINE BREAK INITIATING EVENT OCCURS	IEV-SI-LB
					CMT INJECTION (1 OF 1 TRAINS) OR RCP TRIP FAILS	SYS-XCM1A
					FAILURE OF FULL ADS DEPRESSURIZATION	SYS-ADQ

SENSITIVITY AND IMPORTANCE ANALYSES

- A set of sensitivity and importance analyses are made.

- The results of some of these analyses are provided in the attached tables.

AP1000 PRA System Importances

System Name	Case Description	Core Damage Frequency With System Failed (QNEW)	Core Damage Frequency Increasing Factor (R = QNEW / 2.41E-7)
PMS	No credit is taken for PMS in core damage sequences	1.59E-02	65878
DC-1E	No credit is taken for 1E DC Power in core damage sequences	5.65E-03	23454
IRWST-REC	No credit is taken for IRWST Recirculation in core damage sequences	1.47E-03	6119
ADS	No credit is taken for ADS in core damage sequences	1.46E-03	6040
IRWST-INJ	No credit is taken for IRWST Injection in core damage sequences	3.93E-04	1631
CMT	No credit is taken for CMT in core damage sequences	7.08E-05	294
ACC	No credit is taken for Accumulators in core damage sequences	6.01E-05	249
PRHR	No credit is taken for Passive RHR in core damage sequences	1.84E-05	76
PLS	No credit is taken for PLS in core damage sequences	9.00E-06	7
DC-Non 1E	No credit is taken for Non-1E DC Power in core damage sequences	6.56E-06	27
DAS	No credit is taken for DAS in core damage sequences	2.63E-05	16
AC	No credit is taken for AC Power in core damage sequences	2.36E-06	10
CAS	No credit is taken for CAS in core damage sequences	4.14E-07	1.7
N-RHR	No credit is taken for Normal RHR in core damage sequences	4.11E-07	1.7
SWS	No credit is taken for SWS in core damage sequences	4.00E-07	1.7
CCS	No credit is taken for CCS in core damage sequences	3.78E-07	1.6
SFW	No credit is taken for Startup Feedwater in core damage sequences	2.78E-07	1.2
DG	No credit is taken for Diesel Generators in core damage sequences	2.56E-07	1.1
MFW	No credit is taken for Main Feedwater in core damage sequences	2.54E-07	1.1
SG Overfill Protection	No credit is taken for SG Overfill Protection in core damage sequences	2.41E-07	1

Summary of Sensitivity Analysis Results

LCF?

Case Name	Case Description	Results
CASE 1	Initiating Event Importances Assuming AP600 Initiating Event Frequencies	LLOCA initiating event is the major contributor (88.23%) to CDF. CDF increases by a factor of 4.4.
CASE 2	Set LCF sequences to core damage	SI-LB initiating event is the major contributor (30.64%) to CDF. CDF increases by a factor of 1.3.
CASE 3	Initiating Event Importances	SI-LB (39.43) and LLOCA (18.66) initiating events are the major contributors to CDF
CASE 4	Accident Sequence Importances	IEV-SI-LB, DEL-XCM1A, DEL-ADM, SYS-IW1A (28.52%) is the major sequence contributor to CDF.
CASE 5	End State Importances	3BE (33.4%) and 3D+1D (23.9%) are the major contributors to CDF.
CASE 6	Common Cause Failure Importances	Software CCF of all cards and IRWST sump strainers plugging CCF are the major contributors to CDF.
CASE 7	Human Error Importances	Operator failure to diagnose SG tube rupture event is the major contributor to CDF.
CASE 8	Component Importances	IRWST strainer plugged, PRHR H/X plug/leak and IRWST tank failure are major contributors to CDF.
CASE 9	No credit is taken for ADS in core damage sequences	CDF increases by a factor of 6040.
CASE 10	No credit is taken for CMT in core damage sequences	CDF increases by a factor of 294.
CASE 11	No credit is taken for Accumulators in core damage sequences	CDF increases by a factor of 249.
CASE 12	No credit is taken for IRWST Injection in core damage sequences	CDF increases by a factor of 1631.

was not carried to Level 2

Summary of Sensitivity Analysis Results

Case Name	Case Description	Results
CASE 13	No credit is taken for IRWST Recirculation in core damage sequences	CDF increases by a factor of 6119.
CASE 14	No credit is taken for Passive RHR in core damage sequences	CDF increases by a factor of 76.
CASE 15	No credit is taken for PMS in core damage sequences	CDF increase by a factor of 65878.
CASE 16	No credit is taken for PLS in core damage sequences	CDF increases by a factor of 7.
CASE 17	No credit is taken for DAS in core damage sequences	CDF increases by a factor of 16.
CASE 18	No credit is taken for Normal RHR in core damage sequences	CDF increases by a factor of 1.7.
CASE 19	No credit is taken for SG Overfill Protection in core damage sequences	CDF increases by a factor of 1.0.
CASE 20	No credit is taken for Main Feedwater in core damage sequences	CDF increases by a factor of 1.1.
CASE 21	No credit is taken for Startup Feedwater in core damage sequences	CDF increases by a factor of 1.2.
CASE 22	No credit is taken for AC Power in core damage sequences	CDF increases by a factor of 10.
CASE 23	No credit is taken for Diesel Generators in core damage sequences	CDF increases by a factor of 1.1.
CASE 24	No credit is taken for 1E DC Power in core damage sequences	CDF increases by a factor of 23454.

Summary of Sensitivity Analysis Results

Case Name	Case Description	Results
CASE 25	No credit is taken for Non-1E DC Power in core damage sequences	CDF increases by a factor of 27.
CASE 26	No credit is taken for SWS in core damage sequences	CDF increases by a factor of 1.7.
CASE 27	No credit is taken for CCS in core damage sequences	CDF increases by a factor of 1.6.
CASE 28	No credit is taken for CAS in core damage sequences	CDF increases by a factor of 1.7.
CASE 29	Set HEPs to 1.0 in core damage output file (no credit for HEPs)	CDF increases by a factor of 57.
CASE 30	Set HEPs to 0.0 in core damage output file (perfect operator)	CDF decreases 8%.
CASE 31	Set HEPs to 0.1 in core damage output file	CDF increases by a factor of 6.5.
CASE 32	Impact of passive system check valve failure probabilities	CDF increases by a factor of 3.7.
CASE 33	Impact of explosive valve failure probabilities	CDF increases by a factor of 2.7.
CASE 34	Impact of reactor trip breaker failure probabilities	CDF has negligible increase.
CASE 35	Impact of RCP breaker failure probabilities	CDF increases by a factor of 1.2.
CASE 36	Sensitivity to standby non-safety systems (CVS,SFW,RNS,DAS,DG)	CDF increases by a factor of 31.

AP1000 Importance of Non-Safety Systems

- For AP600 - W determined safety importance of Non-Safety Systems
  - Part of the resolution of Regulatory Treatment of Non-Safety Systems (RTNSS) Policy Issue
  - Included PRA sensitivity studies / evaluations
    - IE Frequency Evaluations
    - Mitigation importance - Focused PRA, NSS importance sensitivity studies
  - DCD contains availability controls for selected NSS
- For AP1000 - W determined safety importance of Non-Safety systems
  - Non-safety systems have same functions, configurations, capabilities
  - PRA risk importance similar for both plants
    - PRA risk profile of AP1000 is similar to AP600
    - PRA system importance of NSS is similar to AP600
- AP1000 DCD contains same availability controls as AP600

List of Non-Safety Systems Covered by Availability Controls

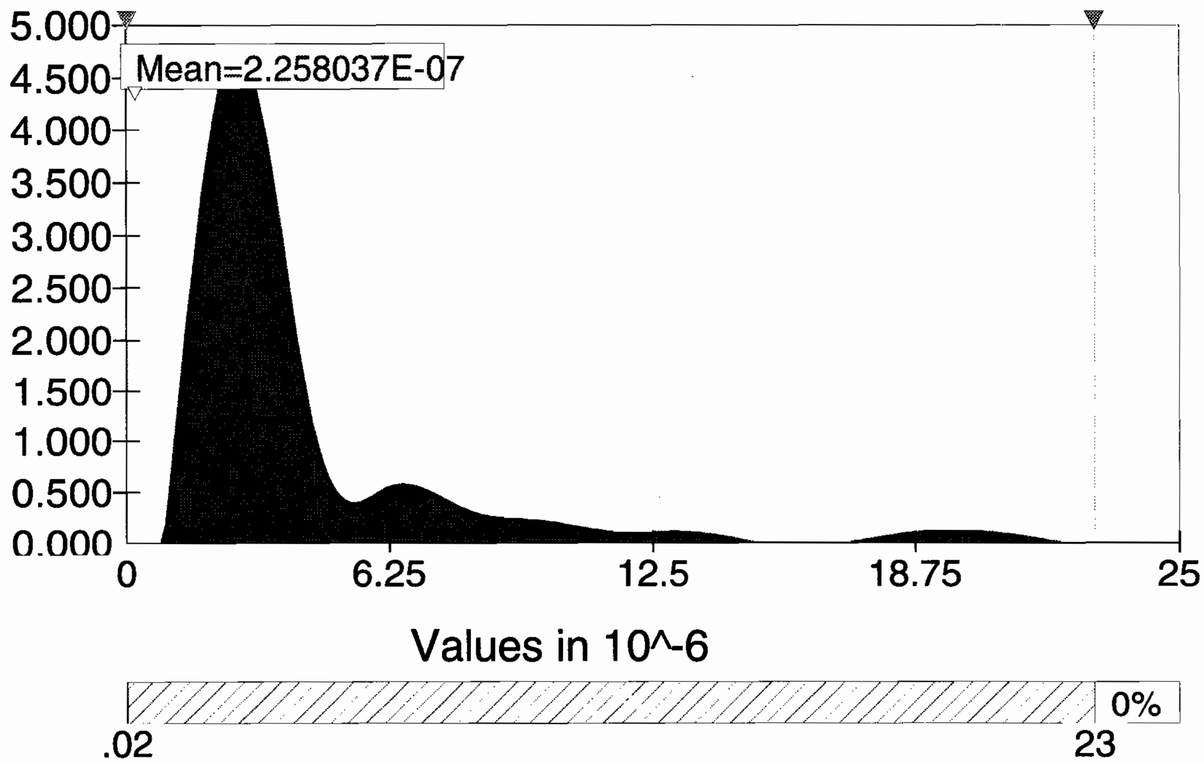
- Diverse Actuation System
- Normal Residual Heat Removal System
- Component Cooling Water System
- Service Water System
- PCS Ancillary Water Makeup
- MCR and I&C Room Ancillary Fans
- Hydrogen Igniters
- AC Power Supplies (Offsite and / or Standby Diesel Generators)
- Non-Class 1E DC and UPS

UNCERTAINTY ANALYSIS

- The plant CDF uncertainty range is found to be 7.3 E-07 – 2.1 E-08 for the 95% to 05 % interval.

- For a lognormal distribution, this would correspond to an error factor of 6, which can be considered as low for rare events.

Distribution for CDF-1000/B2233



Name	CDF-1000
Cell	B2233
Minimum	4.33E-09
Mean	2.26E-07
Maximum	2.11E-05
Std Dev	6.78E-07
Variance	4.60E-13
Skewness	16.32415
Kurtosis	386.4707
Mode	2.91E-08
Left X	1.78E-08
Left P	3%
Right X	0.000023
Right P	100%
Diff. X	2.30E-05
Diff. P	97%
5th Perc.	2.11E-08
95th Perc	7.29E-07
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Filter Min	
Filter Max	
#Filtered	0

uses @risk and excel spreadsheet

UNCERTAINTY ANALYSIS

- The mean values of the dominant accident sequence frequencies are close to the upper bound (95%) estimates;
- Among the initiating event categories, SI-LB has the highest 95-percentile CDF of 3.2E-07 /year.
- Among the dominant sequences, sequence # 07 of SI-LB event has the highest 95-percentile CDF of 2.1E-07/yr.

SHUTDOWN EVENTS

- A quantitative shutdown risk evaluation is performed for AP1000 for internal events.
- The risk profiles of AP1000 and AP600 for events during shutdown conditions are almost identical.
- 18 month refueling cycle
  The AP1000 Shutdown PRA has a CDF of 1.23E-07 events per year. This CDF is an 18% increase
  24 month refueling cycle
  of the AP600 Level 1 Shutdown CDF of 1.04E-07 events per year.

SHUTDOWN EVENTS

- The three events dominating the CDF for each plant are loss of component cooling / service water during drained condition, loss of offsite power during drained condition, and loss of RNS during drained condition.
- The initiating event CDF contributions show that the initiating event importance to be similar for the two plants.

SHUTDOWN EVENTS

- The twelve dominant accident sequences comprise 77 percent of the level 1 shutdown CDF. They consist of:

  – Loss of component cooling or service water system initiating event during drained condition with a contribution of 64 percent of the CDF

SHUTDOWN EVENTS

–Loss of RNS initiating event during drained condition with a contribution of 6 percent of the CDF

–Loss of offsite power initiating event during drained condition with a contribution of 5 percent of the CDF

–RCS overdraining event during drainage to mid-loop with a contribution of a 2 percent of the CDF.

INTERNAL FLOODING AND FIRE

- The internal flooding-induced CDF is estimated to be 8.8E-10 events per year for power operations.

- The CDF from flooding events at power is not an appreciable contributor to the overall AP1000 plant CDF.

INTERNAL FLOODING AND FIRE

- The top five at-power flooding scenarios comprise 91 percent of the at-power flooding-induced core damage frequency.
- These scenarios are for large pipe breaks in the turbine building with an initiating event frequency in the range of 1.4 – 2.0 E-03 / year, leading to a loss of CCW/SW event. Each scenario has a CDF of 1.2 – 1.8E-10/year.

INTERNAL FLOODING AND FIRE

- AP600 Fire PRA quantified with bounding focused PRA model (no credit for non safety related [illegible] or DC power [illegible] non safety related systems)
  - –CDF is 6.5E-07 /yr
  - –No credit for non-safety systems
- Extensive fire hazards analysis review completed for AP600 subsequent to fire AP600 PRA
  - –Fire separation improved
  - –Fire suppression features incorporated
  - –Design features incorporated to address hot-shorts

INTERNAL FLOODING AND FIRE

- Qualitative evaluation of risk from fire performed
  - –AP600 design features important for fire protection are included in the AP1000
    - –Fire separation / fire zones
    - –Systems used to achieve safe shutdown
    - –Fire suppression features
  - –AP1000 design is sufficiently robust that internal fires during power operation or shutdown do not represent a significant contribution to plant CDF

SEISMIC MARGINS EVALUATION

- The seismic margin analysis shows the systems, structures, and components required for safe shutdown. HCLPF values are greater than or equal to 0.50g
- This HCLPF is determined by the seismically induced failure of the fuel in the reactor vessel, core assembly failures, IRWST failure, or containment interior failures

SEISMIC MARGINS EVALUATION

- The SMA result assumes no credit for operator actions at the 0.50g review level earthquake, and assumes a loss of offsite power for all sequences

- The SMA shows the plant to be robust against seismic event sequences that contain station blackout coupled with other seismic or random failures

Comparison of Low HCLPF SSCs in AP1000 and AP600 Designs

Basic Event ID	Description	AP600 HCLPF	AP1000 HCLPF
EQ-CER-INSULATOR	Failure of Ceramic Insulators	0.09g	.09g
EQ-CORE-ASSEMBLY	Core Assembly Failure (not fuel)	0.50g	.50g
EQ-CV-INTER	Interior Containment	0.60g	.50g
EQ-IRWST-TANK	IRWST Failure	0.60g	.50g
EQ-RV-FUEL	Fuel Failure	0.50g	.50g
EQ-AB-EXTWALL	Aux. Building Exterior wall	0.58g	.51g
EQ-AB-FLOOR	Aux. Building Floor	0.58g	.51g
EQ-AB-INTWALL	Aux. Building Interior wall	0.58g	.51g
EQ-PCC-TANK	PCC Tank Failure	0.58g	.51g
EQ-SHDBLD-ROOF	Shield Building Roof	0.58g	.51g
EQ-SHDBLD-WALL	Shield Building Wall	0.58g	.51g
EQ-CABLETRAY	Cable trays - support controlled	0.54g	.54g
EQ-CMT-TANKS	Tank PXS 2A/B (Core Makeup Tank)	0.63g	.54g
EQ-SG-FAILS	Steam Generator Fails	0.65g	.54g
EQ-SGTR	Steam Generator Piping (one or a few)	0.65g	.54g
EQ-ACDISPANEL	120 vac distribution panel	0.51g	.55g
EQ-DC-SWBRD	125 vdc switchboard	0.51g	.55g
EQ-DCDISPANEL	125 vdc distribution panel	0.51g	.55g
EQ-PRZR-FAILS	Pressurizer Fails	0.67g	.55g
EQ-TRSFSWITCH	Transfer switch	0.51g	.55g

always in LOSP situation

OTHER EXTERNAL EVENTS

- High winds and tornadoes
- External floods
- Transportation and nearby facility accidents

  - As per the site selection criteria defined in Chapter 2 of the DCD, a frequency of occurrence of 10^{-6} per year, for an accident external to AP1000 that has a potential consequence serious enough to affect the safety of the plant according to 10 CFR 100 guidelines, is not exceeded.

Comparison of AP600 and AP1000 PRA Results

Scope	AP600	AP1000
Level 1 At-Power Internal Initiating Events	Quantification Performed CDF = 1.7E-07 Several additional cases quantified in response to NRC RAIs	Quantification Performed CDF = 2.4E-07 AP600 additional cases incorporated into the model
Level 2 At-Power Internal Initiating Events	Quantification Performed LRF = 1.8E-08 Containment Effectiveness = 89.5%	Quantification Performed LRF = 2.0E-08 Containment Effectiveness = 91.8%
Level 3 At-Power Internal Initiating Events	Quantification Performed	Quantification Performed
Internal Fire Events	Conservative (via focused PRA) Quantification Performed CDF = 6.5E-07 (internal) CDF = 3.5E-07 (shutdown)	Assessment Performed AP600 fire PRA quantification bounds AP1000
Internal Flooding Events	Quantification Performed CDF = 2.2E-10	Quantification Performed CDF = 8.8E-10
Shutdown Events	Quantification Performed for Level 1 and 2 CDF = 1.0E-07 LRF = 1.5 E-08 Several additional cases quantified in response to NRC RAIs	Quantitative Evaluation Performed CDF = 1.2E-07 AP600 additional cases incorporated into the estimation model
Focused PRA Internal Events At-Power	Quantification Performed CDF = 9.1E-06 LRF = 8.1E-07 Availability controls of NSS adopted	Sensitivity studies performed demonstrate that NSS are not important for AP1000 risk. Same availability controls on NSS adopted for AP1000

SUMMARY OF RESULTS

- The AP1000 PRA results show that
  - The very low risk of the AP600 has been maintained in the AP1000
  - The AP1000 PRA meets the US NRC safety goals with significant margin

AP1000 Level 2 PRA

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Purpose of Level 2 PRA

- Status of containment integrity
  - –system failures (unisolated SGTR, isolation failure)
  - –failure due to high energy phenomena
    - –induced tube rupture
    - –Steam Explosion (in-vessel and ex-vessel)
    - –Hydrogen Combustion
    - –High Pressure Melt Ejection / Direct Containment Heating
    - –Debris Impingement
    - –Core-Concrete Interaction
    - –Long Term Containment Pressurization from Decay Heat Steaming

Purpose of Level 2 PRA

- Quantify Magnitude and Timing of Offsite Release
  - –Accident Classes (same as AP600) PDS
  - –Release Categories (same as AP600, plus CFV)
  - –Source Terms (assumed same fractions as AP600)

AP1000 Containment Event Tree

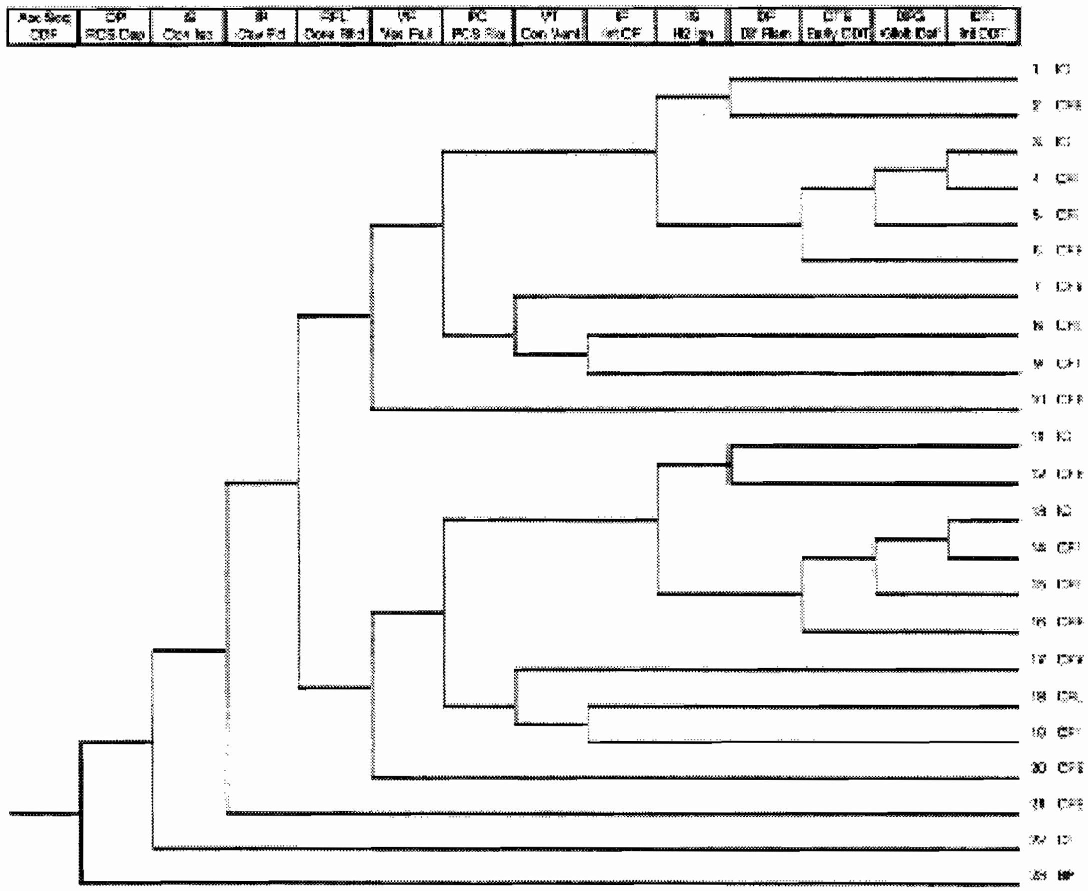
- Pretty much the same as AP600
- Added possibility of containment venting

  containment failure venting not the same as the vent in AP1000

  - added CFV release category
  - assumed failure probability of unity

Chapter 40 - describes venting.
If you get PCS back you will fail containment due to implosion

AP1000 Containment Event Tree



Containment Event Tree Quantification

- System Nodes
  - –Quantification
    - –Linked Fault Trees
    - –Scalars defined by accident class definition
  - –containment isolation
  - –cavity flooding
  - –PCS water cooling
  - –hydrogen control

Containment Event Tree Quantification

- Phenomenological Nodes
  - –Quantification
    - –Scalars defined by analysis of phenomena
    - –boundary conditions defined by accident class
  - –induced SGTR tube rupture
  - –core reflooding
  - –in-vessel retention of molten core debris
  - –hydrogen combustion
  - –containment integrity

AP600 AP1000
amount of H_2 630kg 800

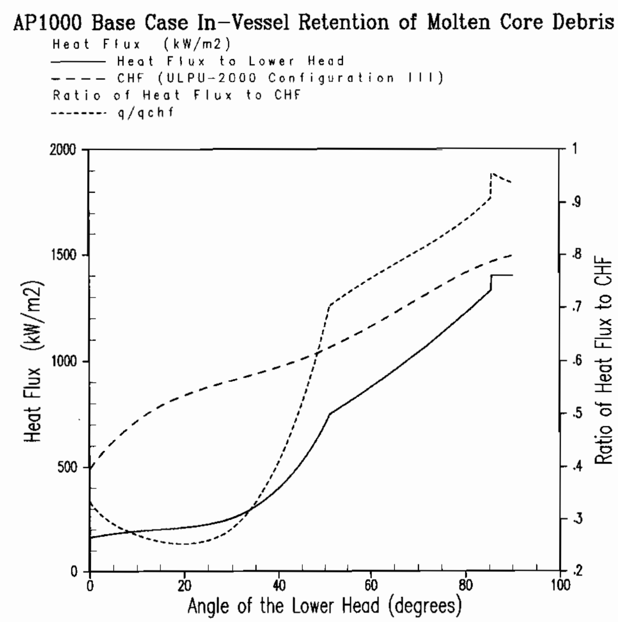
Retaining the 2 large PARs for Div-D
Removed the smaller PAR from the IRWST vents

In-Vessel Retention of Molten Core Debris

- Changes to the AP1000 that potentially impact IVR
  - –power is increased to 3400 MWt.
  - –157 14-ft fuel assemblies.
  - –core shroud instead of reflector
  - –lower core support plate is 1” thicker

In-Vessel Retention of Molten Core Debris

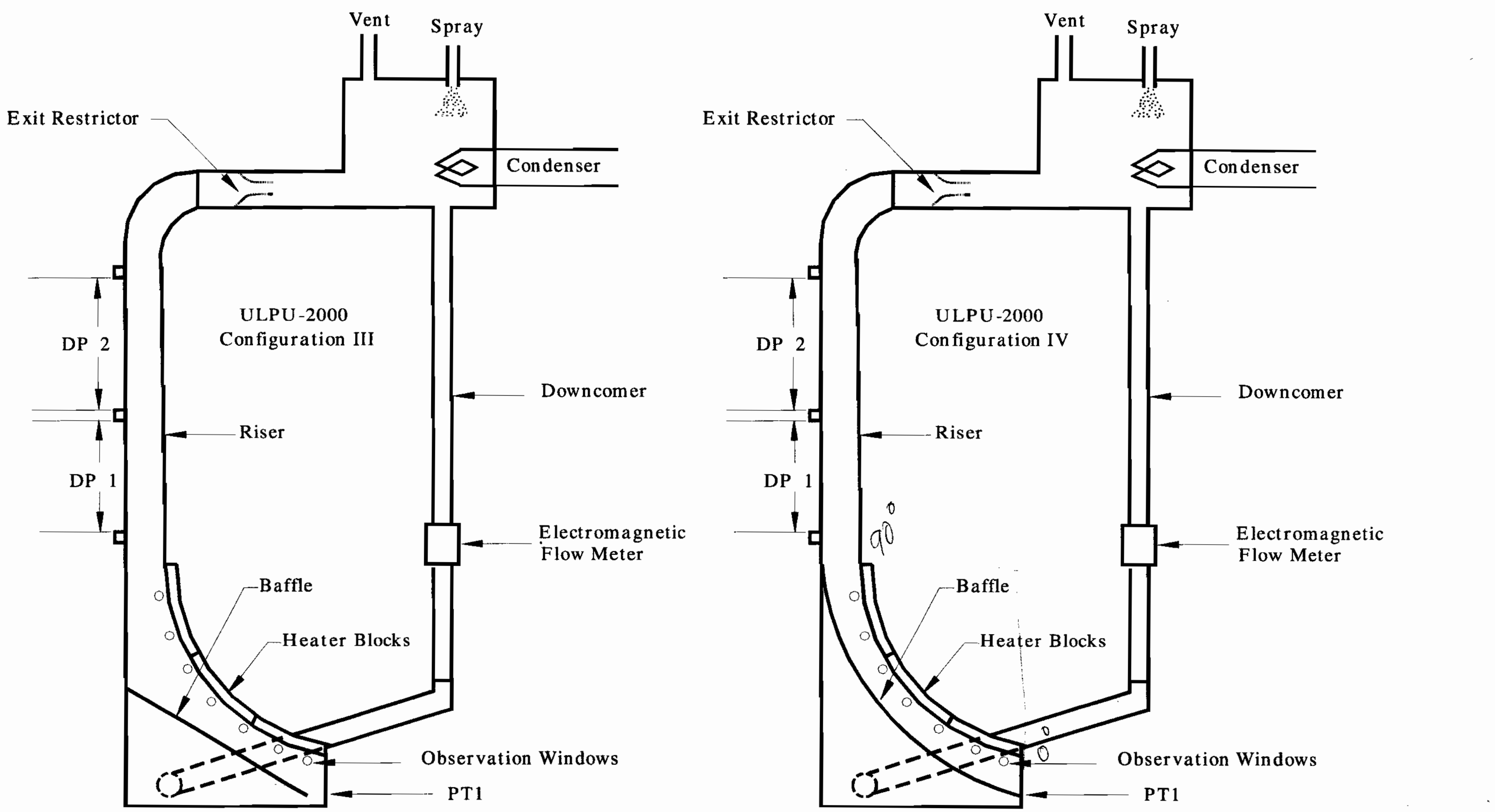
- DOE/ID-10460 methodology to quantify heat flux
  - AP600 CHF success criterion not sufficient for AP1000



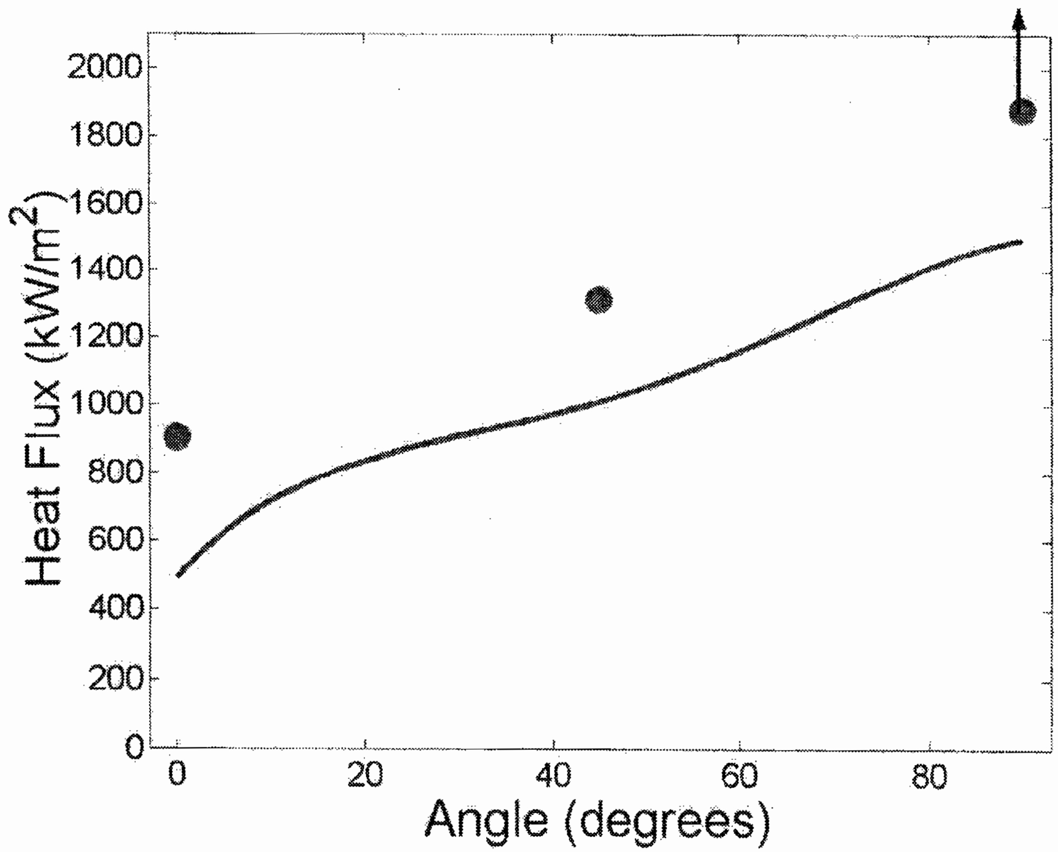
In-Vessel Retention of Molten Core Debris

- Effort to achieve margin similar to AP600
  - –testing
  - –design changes
- Performed UPLU Configuration IV Testing
  - –examined changes to increase CHF on vessel surface
    - –CHF increased more than 30%
  - –two-phase natural circulation is required
  - –insulation geometry and structure is important

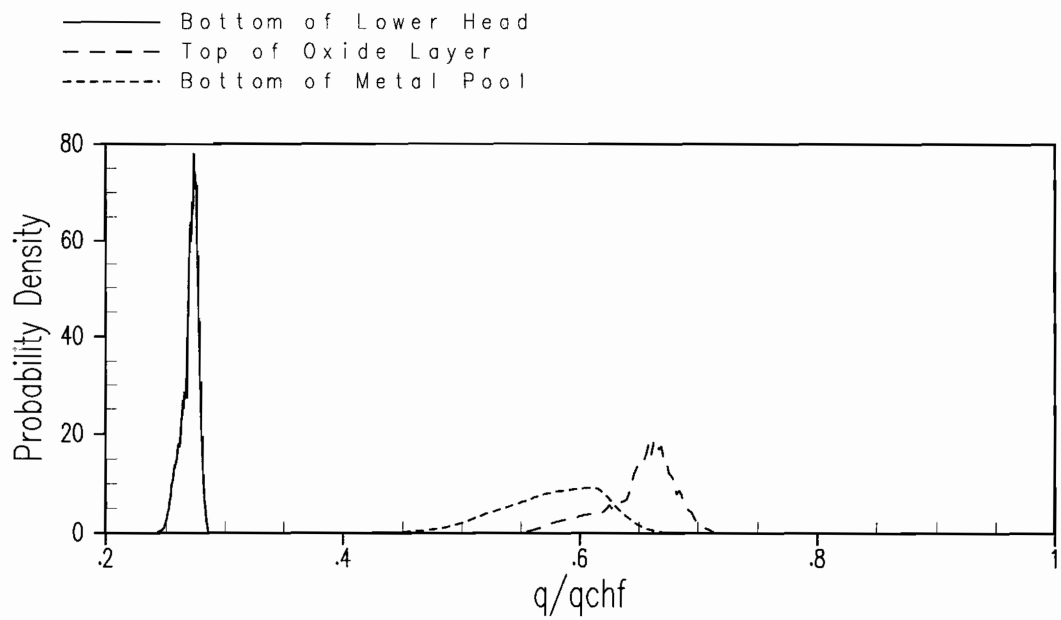
ULPU Configuration IV



ULPU Configuration IV Results



IVR Results



- Reclaim most of the margin of the AP600
- Continuing ULPU program to define design

In-Vessel Steam Explosion

- In-Vessel Steam Explosion Analysis for AP600
  - DOE/ID 10541 In-Vessel Steam Explosion ROAAM
  - Large margin to failure
- Debris relocation mechanism for AP1000 same as AP600
  - sideward failure
  - similar mass and superheat as AP600
- Geometry the same for AP1000 as AP600
- Extrapolate AP600 results to AP1000

Ex-Vessel Steam Explosion

- Prevented by IVR
- Vessel failure assumed to produce early containment failure on CET
- Assumed same vessel failure modes as AP600
  - similar mass and superheat in the debris
- AP600 ex-vessel steam explosion results extrapolated to AP1000

Hydrogen Combustion

- Hydrogen control system
  - –PARs (not credited in PRA)
  - –Igniters (same number and layout as AP600)
  - –If igniters working diffusion flame is only failure mode
- PCS water assumed to be working
  - –steam inert if PCS water not working
  - –no consideration of sprays
- Detonation assumed to fail containment

Early Hydrogen Burn

- During H_2 release
  - –before containment is well-mixed
- Diffusion Flames
  - –ADS stage 4 releases hydrogen in loop compts
  - –IRWST release hydrogen away from shell
    - –stand pipe vents near SG doghouse open preferentially
    - –CET failure probability defined by stuck open wall vents in accident classes with no ADS-4
  - –PXS compt vents hydrogen away from shell

Early Hydrogen Burn (continued)

- Deflagration to Detonation Transition
  - –igniter failure
  - –use AP1000 specific sequence conditional probabilities
  - –use AP600 DDT probabilities (Sherman-Berman Method)
  - –CET assumes containment failure if DDT occurs

Intermediate Hydrogen Burn

- Occurs before 24 hours
- Igniter failure
- Containment well-mixed
- Global deflagration
  - used AP600 probability distributions
    - hydrogen mass scaled up by mass ratio of active cladding
    - pre-burn pressure same as AP600
  - adiabatic peak pressure calculation
  - containment failure defined by containment fragility curve

Intermediate Hydrogen Combustion

- DDT
  - –acceleration of global burn in well-mixed compartments
  - –CMT compartment considered to be dry air
  - –DDT assumed to fail containment

High Pressure Melt Ejection

- SECY 93-087
  - –provide a reliable depressurization system
  - –provide debris retentive cavity design
- AP1000 has reliable 4 stage ADS
- Cavity layout is water-filled torturous pathway
  - –no direct pathway for debris impingement
- High pressure core melt assumed to fail SG tubes on containment event tree

Core Concrete Interaction

- Mitigated by IVR
- Vessel failure assumed to fail containment on CET
- AP1000 specific analysis
  - –basemat failure not expected within 24 hours
  - –containment overpressurization does not occur before basemat failure

Long Term Containment Pressurization

- Mitigated by PCS water cooling
  - * –added third diverse pathway CCFP=8.2%
  - –failure of water produces small probability of cnmt failure (2%)
- Containment pressurization with no PCS water
  - –nominal case
    - –Ambient Temp =80 F and best estimate ANS 79 decay heat
  - –bounding case
    - –Ambient Temp = 115 F and ANS 79 + 2 sigma decay heat
- Containment Fragility Curve defines failure

Long Term Containment Pressurization

[illegible] 20% [illegible] [illegible] of fragility curve [illegible] corresponds to 2% failure

- Assigned 0.02 failure probability for < 24 hours
  - –assumed failure probability = 1.0 after 24 hours
- Containment Venting
  - –investigated performance with various line sizes
  - –concluded that operator could vent through any line > 4”
  - –containment underpressure
  - –assigned venting failure probability of 1

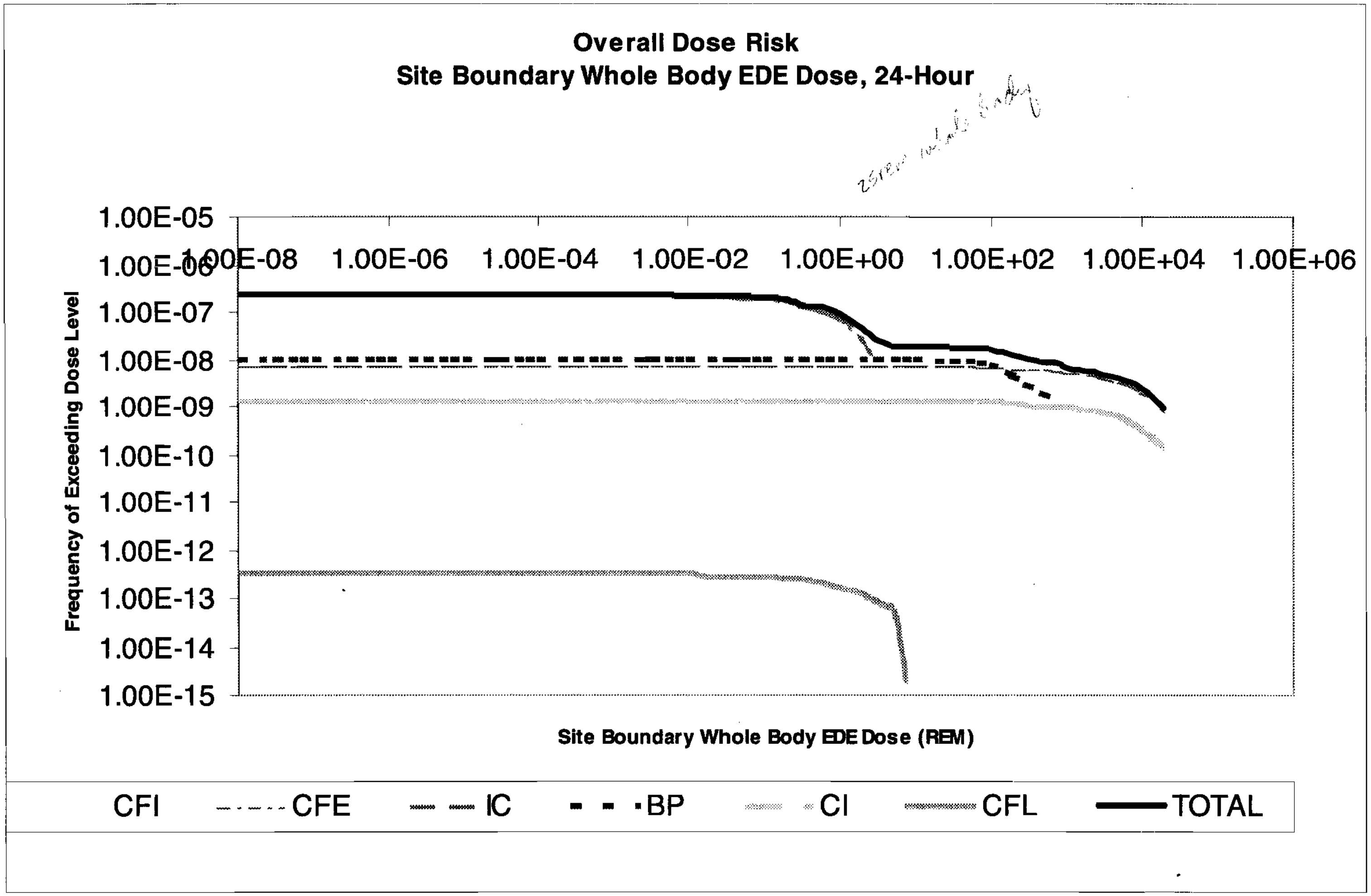
Level 2 At-Power Results

- Core Damage Frequency = 2.41×10^{-7} per year
- Large Release Frequency = 1.95×10^{-8} per year
- Frequency by Release Categories
  - Containment Bypass = 1.05×10^{-8}
  - Early Containment Failure = 7.47×10^{-9}

Level 3 PRA

- Used AP600 Release Fractions
  - AP1000 fission product inventories
- Calculated off site doses with MACCS2 1.12

Level 3 PRA Results



Success Criteria / Thermal-Hydraulic Analysis

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Overview

- **Success Criteria (Chapter 6 of PRA)**
  - –Changes in success criteria vs AP600
- **Thermal Hydraulic Analysis (Appendix A of PRA)**
  - –Analysis used, DCD, specific PRA, or other analysis / calculations
  - –Summary of results
  - –ADS analysis
  - –T&H Uncertainty Analysis

AP1000 Success Criteria

- **Based on AP600**
  - –Similar system design, arrangement, capabilities
- **Several Changes Made to the AP1000 Success Criteria**
  - –Due to increase in power vs capacity of mitigating features
  - –Due to design changes to accommodate the power increase
  - –Due to other factors
- **AP1000 Success Criteria More Conservative / Robust**
  - –Requires same or more equipment for success
    - –For example, requires 3/4 ADS 4 instead of 2/4 ADS 4
      - –Even though AP1000 ADS 4 is larger / MW
  - –Reduces T&H issues / uncertainty

AP1000 Success Criteria

Event	Same as AP600	Comments
Transients	yes	
ATWS	yes	UET = 0 because of low boron core
SLB (down MSIV)	yes	
SLB (up MSIV)	yes	
SGTR	yes	
RCS Leak	yes	
PRHR Tube Rupture	yes	
Small LOCA	no	See next slide
Medium LOCA	no	See next slide
CMT BL LOCA	no	See next slide
DVI LOCA	no	See next slide
Spurious ADS (Lg)	no	Requires ADS 4, cont. recirc (was part of AP600 Large LOCA)
Large LOCA	no	Requires 2/2 accum, ADS 4, cont. recirc.

Post ADS Success Criteria

- **Changes Made to Post ADS Success Criteria**
  - –Full ADS (IRWST) >> requires 3/4 ADS stage 4
    - –AP600 PRA used 2/4 ADS stage 4
      - –Later PRA T&H analysis showed that AP600 needed 3/4
    - –AP1000 ADS 4 capacity has been increased by more than power
  - –Partial ADS (RNS) >> requires 2 of 4 ADS stage 2 or 3
    - –AP600 PRA used 1/4 stage 2 or 3
    - –ADS stages 1, 2, 3 capacities not increased for AP1000
  - –Requires PRHR HX for MLOCAs with only Accum
    - –Provides operators more time (> 20 min) to take action
  - –Requires 2/4 Cont Recirc if Cont Isol fails
    - –1/4 Cont Recirc if Cont Isol works

Thermal Hydraulic Analysis

- **Thermal Hydraulic Analysis Has Been Performed to Determine the AP1000 Success Criteria**
  - –Use DCD analysis when applicable
  - –Otherwise
    - –Use special PRA analysis
    - –Use other calculations
    - –Use evaluations
- **Thermal Hydraulic Uncertainty Has Been Performed**
  - –Uses DCD analysis methods to bound T&H uncertainty for low margin / risk important accident scenarios
    - –Same approach as AP600

PRA T&H Analysis

- **Events That Utilize PRA Specific Analysis**
  - –ATWS > LOFTRAN analysis
  - –LTC > DCD analysis, AP600 lessons learned, AP1000 hand calculation (margin)
  - –Spurious ADS (large LOCA) > insights from operating plants (HL vs CL LOCAs), hand calculation (margin)
  - –LOCAs (other than LLOCAs) and Feed-Bleed Cooling > MAAP analysis

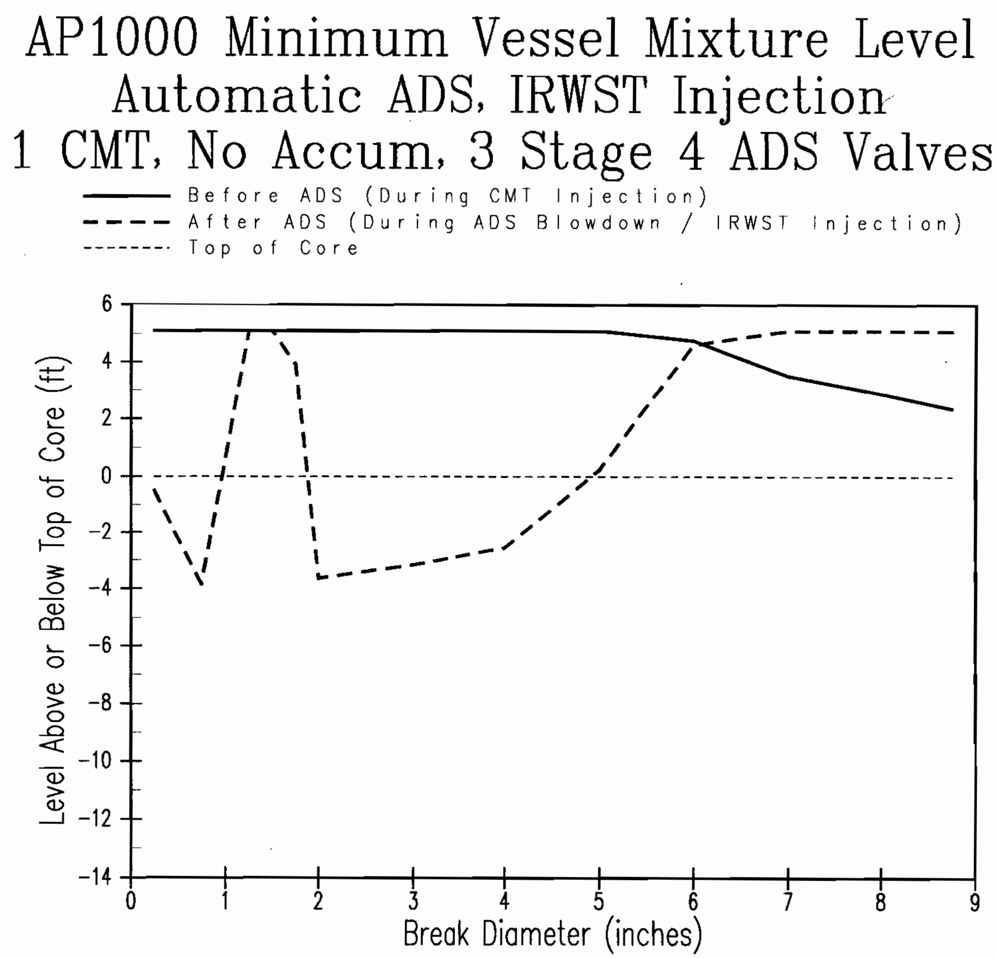
PRA T&H Analysis

- **LOCA and Feed-Bleed Cooling Analysis**
  - –Considers many different factors
    - –Initiating event, LOCA or Feed-Bleed Cooling after non-LOCA
    - –LOCA size and location
    - –Available mitigating equipment including CMT, Accum, RNS, PRHR HX, ADS, IRWST, Cont Recirc
  - –Made use of lessons learned from AP600
    - –Test results, DCD analysis, PRA analysis (both success criteria and T&H uncertainty
    - –Divided into four groups of analysis
      - –Automatic ADS with IRWST gravity injection or RNS injection
      - –Manual ADS with IRWST gravity injection or RNS injection

Auto ADS with IRWST Gravity Injection

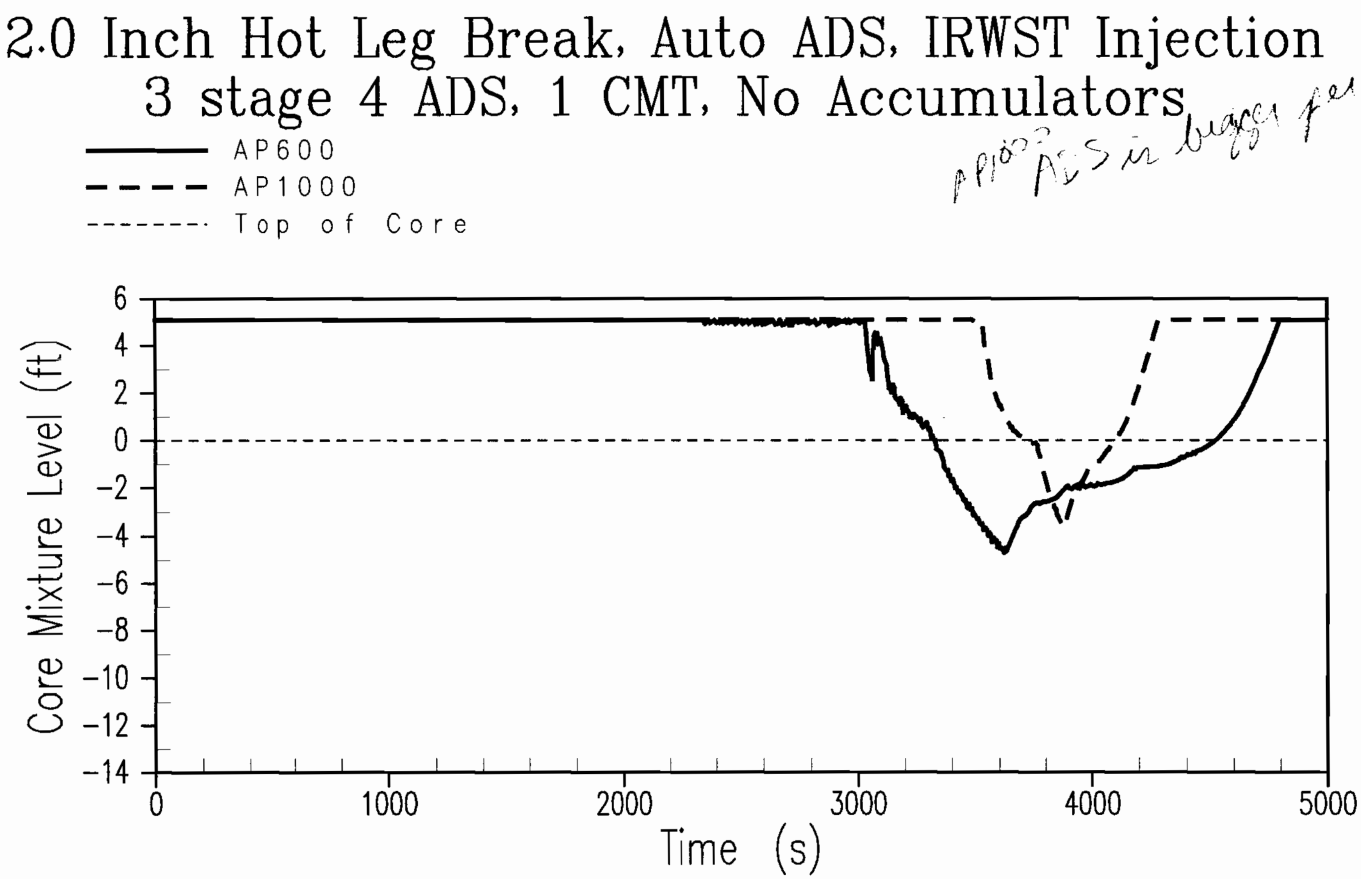
- **Limiting Equipment Assumed**
  - –Same as AP600
  - –One CMT, no Accum, 1 valve path in one IRWST injection line
  - –3/4 ADS stage 4 , no ADS stage 1/2/3, no PRHR HX
    - –For LOCAs < 2" some ADS 1/2/3 or PRHR HX required to reduce RCS pressure to below ADS 4 pressure interlock
  - –Containment isolation fails No credit for contmt backpressure
- **MAAP Analysis Was Performed**
  - –Break sizes 0.5" up to 8.75"
  - –Core uncovery depth and duration is less than AP600
    - –Increased capacity PXS, especially ADS 4, IRWST injection
  - –AP600 success criteria remains valid for AP1000

Auto ADS with IRWST Gravity Injection



Bigger break sizes get to ADS quicker that is the reason for the dip below of top of core.

Auto ADS with IRWST Gravity Injection

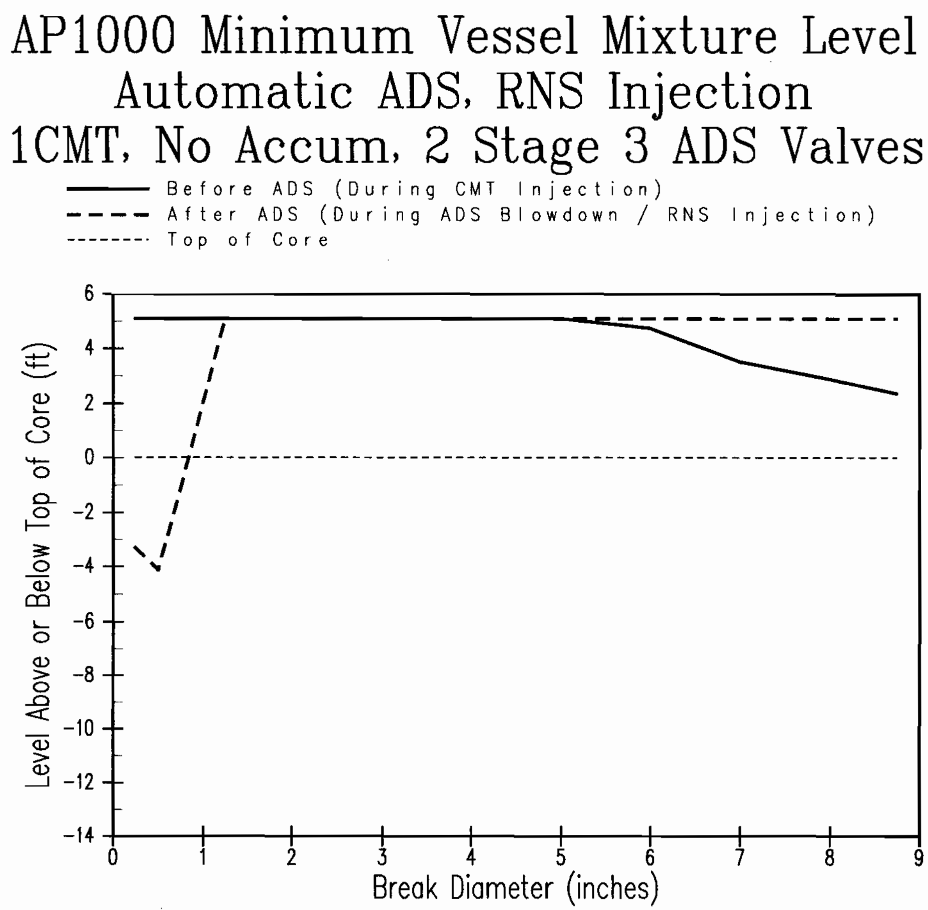


AP1000 ADS is bigger per MW

Auto ADS with Injection By RNS

- **Limiting Equipment Assumed**
  - –Same as AP600, except a second ADS stage 2/3 required
  - –One CMT, no Accum, no IRWST gravity inject line, one RNS pump
  - –2/4 ADS stage 2/3, no ADS stage 1/4, no PRHR HX
  - –Containment isolation fails
- **MAAP Analysis Was Performed**
  - –Break sizes 0.5" up to 8.75"
    - –DVI not analyzed, RNS not credited as success for this LOCA
  - –Core uncovery depth and duration less than with "full" ADS and IRWST gravity injection
  - –AP600 success criteria remains valid for AP1000

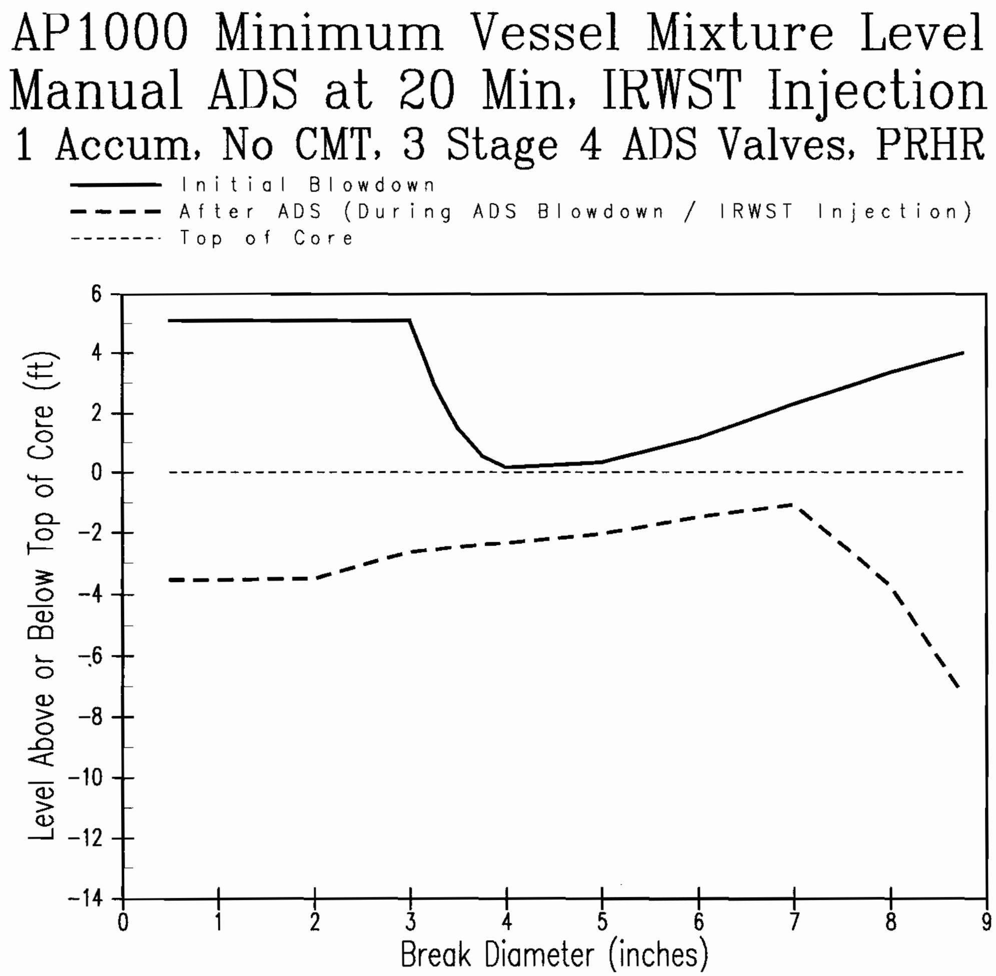
Auto ADS with Injection By RNS



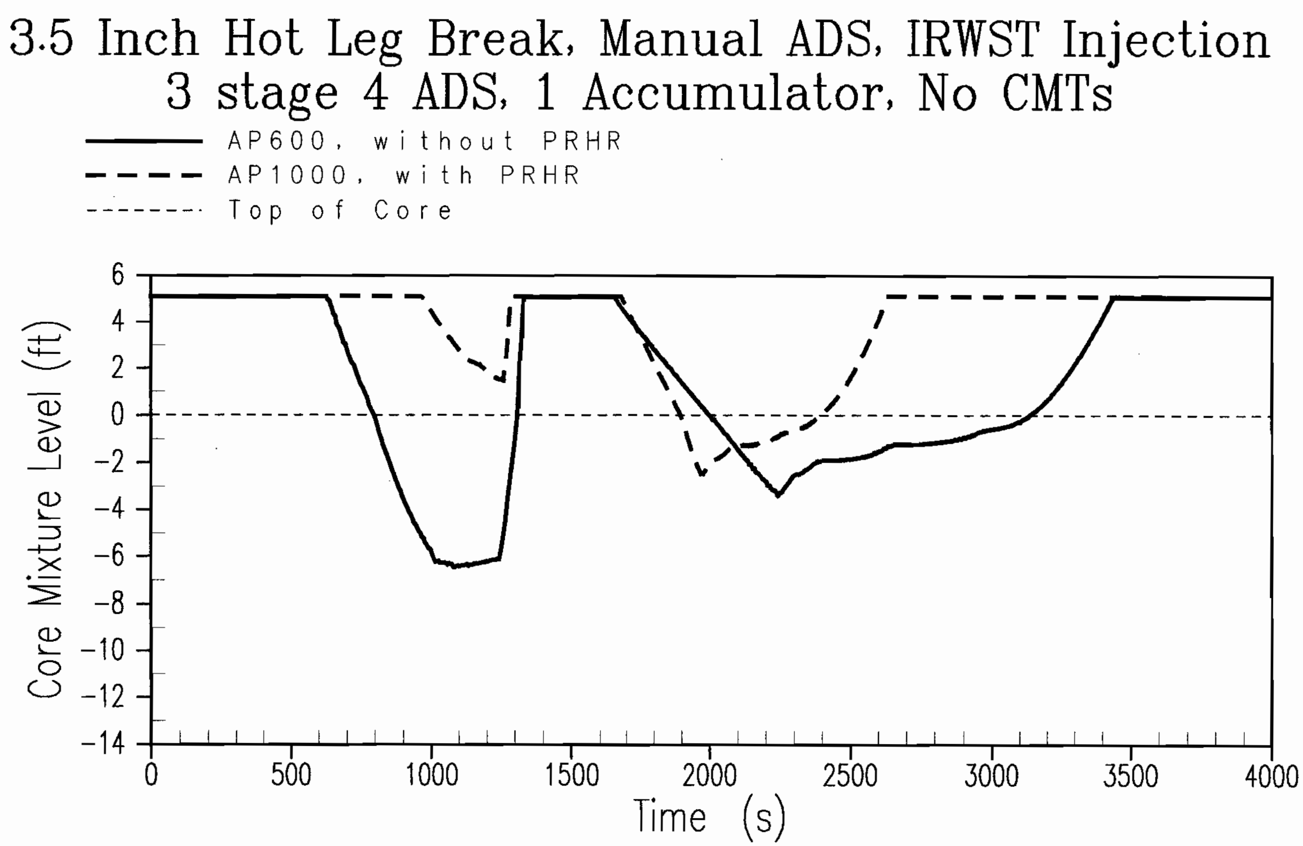
Manual ADS with IRWST Gravity Injection

- **Limiting Equipment Assumed**
  - –Same as AP600, except for use of PRHR HX for MLOCAs
  - –No CMT, 1 Accum, 1 valve path in one IRWST injection line
  - –3/4 ADS stage 4 , no ADS stage 1/2/3, no PRHR HX
    - –For MLOCAs 2" to 9" PRHR HX is required to give operators at least 20 min to open ADS 4
  - –Containment isolation fails
- **MAAP Analysis Was Performed**
  - –Break sizes 0.5" up to 8.75"
  - –Core uncovery depth and duration is similar to AP600
    - –Increased capacity PXS, especially ADS 4, IRWST injection, use of PRHR
  - –AP600 success criteria remains valid for AP1000

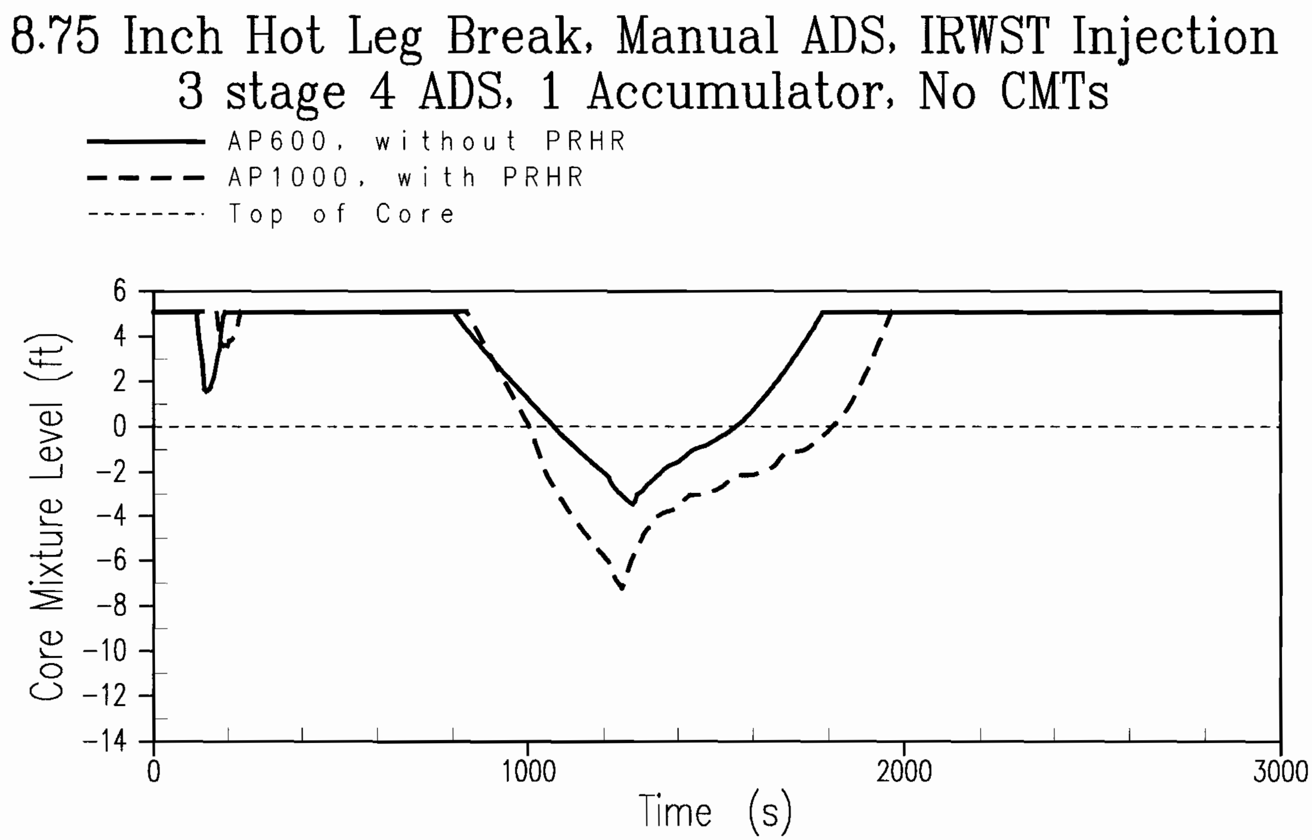
Manual ADS with IRWST Gravity Injection



Manual ADS with IRWST Gravity Injection



Manual ADS with IRWST Gravity Injection

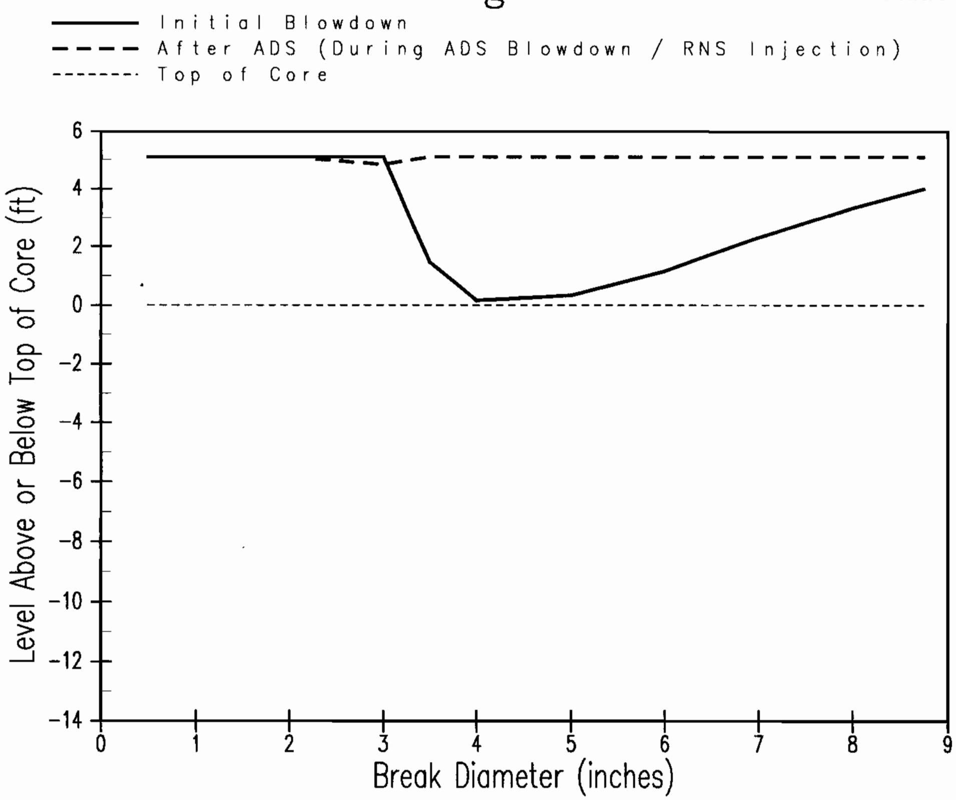


Manual ADS with Injection By RNS

- **Limiting Equipment Assumed**
  - –Same as AP600, except a second ADS stage 2/3 opens
  - –No CMT, 1 Accum, no IRWST gravity inject line, one RNS pump
  - –2/4 ADS stage 2/3, no ADS stage 1/4, no PRHR HX
    - –For MLOCAs 2" to 9" PRHR HX is required to give operators at least 20 min to open ADS 4
  - –Containment isolation fails
- **MAAP Analysis Was Performed**
  - –Break sizes 0.5" up to 8.75"
    - –DVI not analyzed, RNS not credited as success for this LOCA
  - –No core uncovery is calculated for these events
  - –AP600 success criteria remains valid for AP1000

Manual ADS with Injection By RNS

AP1000 Minimum Vessel Mixture Level
Manual ADS at 20 Min, RNS Injection
1 Accum, No CMT, 2 Stage 3 ADS Valves, PRHR



Large LOCA Success Criteria

- **Large LOCA Pipe Breaks**
  - –Use DCD LLOCA analysis since use 2/2 Accum
- **Spurious ADS (Large HL LOCA)**
  - –Uses 1/2 Accum > need specific PRA justification
  - –Less severe than CL large LOCAs, much lower PCT end of blowdown

	AP600	AP1000
Break Location	CL	HL
Break size	DECL	4 x ADS-4 valves
PCT at end of blowdown (°F)	1000	500
Number accumulator injecting	1	1
Core heatup time (sec)	106	120
Core linear power (kw/ft)	4.100	5.707
PCT increase (°F)	786	1239
PCT without uncertainty (°F)	1786	1739
PCT uncertainty (°F)	244	251
PCT with uncertainty (°F)	2030	1990

WCOBT / TRAC
[illegible]
2 acc

Post ADS Long Term Cooling

- **For AP600**
  - –T&H uncertainty LTC analysis showed acceptable results
- **For AP1000**
  - –LTC features all improved by more than power increase
    - –Power increase ~ 72%
    - –ADS 4 capacity ~ 89%
    - –IRWST injection ~ 84%
    - –Containment recirc ~113%
  - –AP600 success criteria remains valid for AP1000

T&H Uncertainty

- **AP600 Approach**
  - –Bounds T&H uncertainty
    - –Selected high risk / low margin cases
      - –6 LOCAs, 3 LLOCAs, 3 LTC
    - –Analyzed with DCD codes / assumptions
      - –Conservative decay heat, line resistances, ...
- **AP1000 Approach**
  - –Same as AP600
    - –Because of similarity of designs and PRA results
      - –Considered same T&H uncertainty cases
      - –Some no longer apply because of changes in success criteria

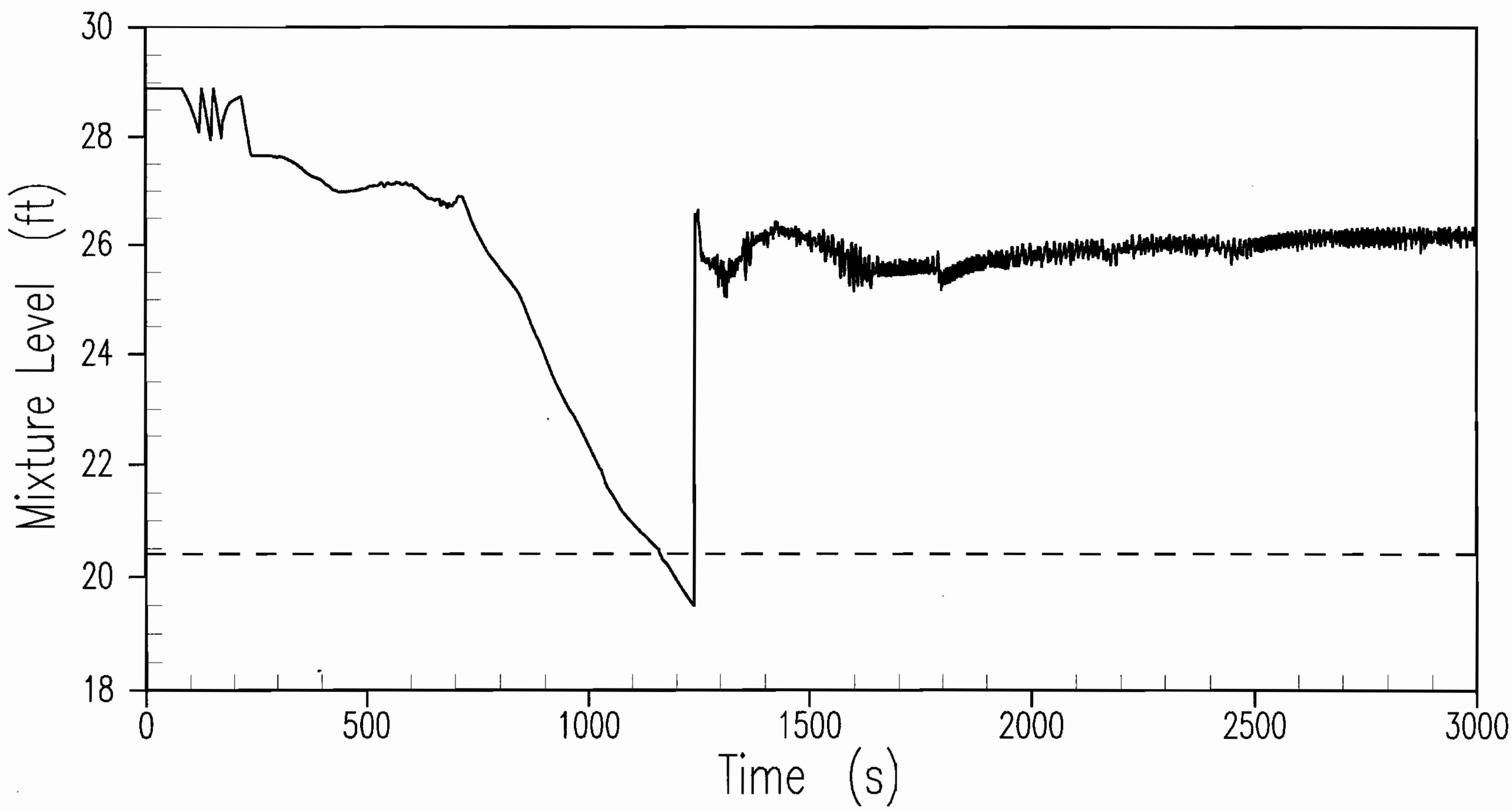
Small LOCA T&H Uncertainty

- **AP600 Small LOCA T&H Uncertainty Cases**
  - –First 3 cases also apply to AP1000
    - –Add PRHR to cases 1 & 2 since required by AP1000 success criteria
  - –Last 3 cases not required for AP1000
    - –Success criteria has changed > these cases would not be success

AP600 T&H Uncertainty Cases / Equipment Availability										
	Cont Isol	ADS 1/2/3	ADS 4	CMT	Acc	IRWST Val/Path	Recirc Val/Path	Applicable to AP1000	AP600 PCT (F)	AP1000 PCT (F)
Small LOCAs										
1. 3.25" HL	yes	0	4	0	1	1/1	na	yes	1157	719
2. DE CMT inlet	yes	0	4	0	2	1/1	na	yes	none	none
3. DE DVI	no	0	3	1	0	1/1	na	yes	1435	1570
4. DE DVI	no	0	2	1	1	1/1	na	no	1235	--
5. 2" HL	yes	0	2	1	1	1/1	na	no	none	--
6. 9" HL	yes	0	0	2	2	1/1	na	no	none	--

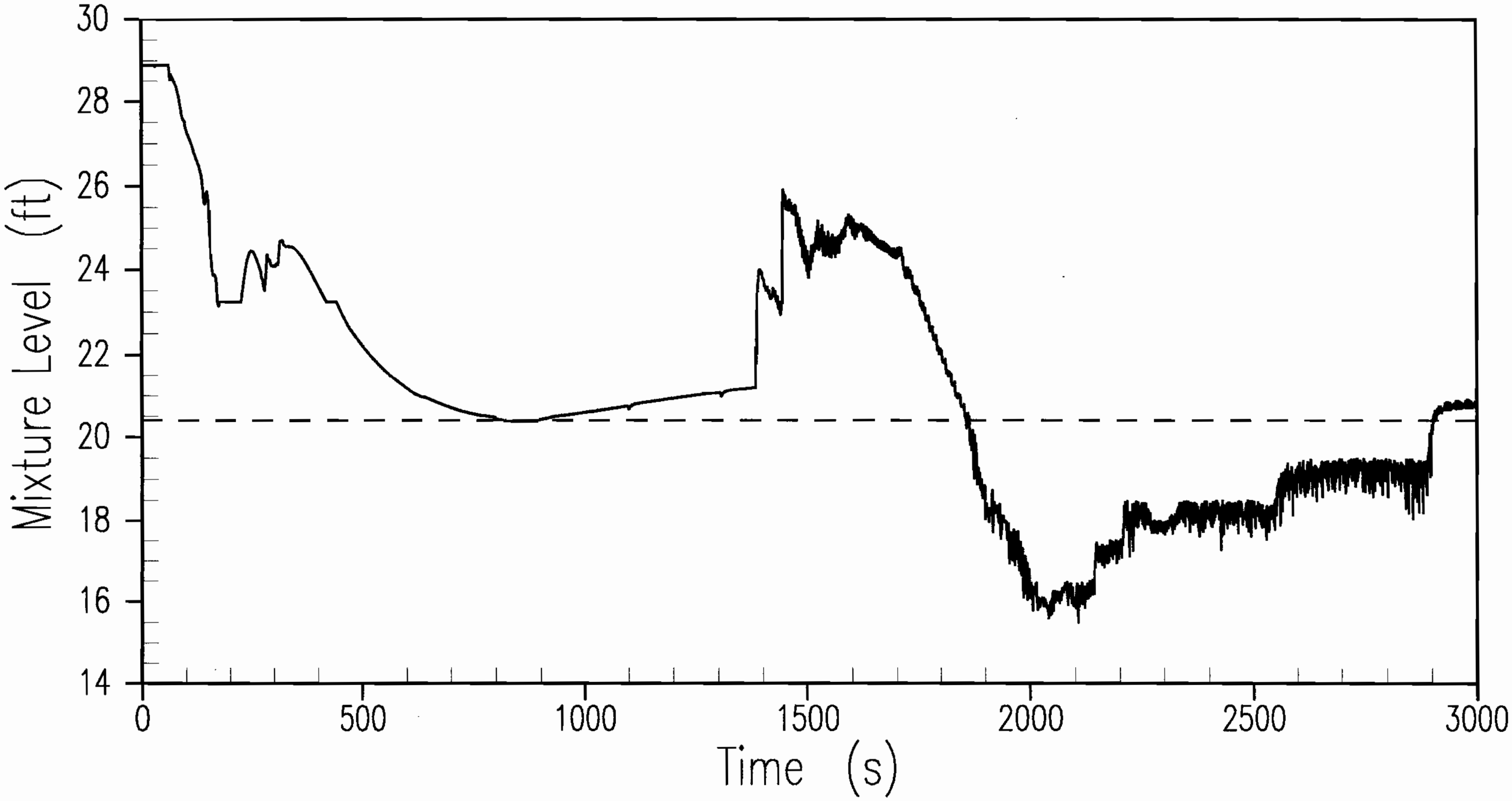
T&H Uncertainty

3.25 Inch Hot Leg Break/Manual ADS4/No Stage 1-2-3 ADS/No CMTs



T&H Uncertainty

DE DVI Break/Auto ADSS4, 1/2 CMTs, 0/2 ACCs, No Stage 1-3 ADS



Large LOCA T&H Uncertainty

- **AP600 Large LOCA T&H Uncertainty Cases**
  - –Performed to verify 1 / 2 Accum was success
  - –AP1000 requires 2 / 2 Accum and 3 / 4 ADS 4
    - –Verified by DCD analysis
    - –T&H uncertainty analysis not required

AP600 T&H Uncertainty Cases / Equipment Availability								Applicable to AP1000	AP600 PCT (F)	AP1000 PCT (F)
	Cont Isol	ADS 1/2/3	ADS 4	CMT	Acc	IRWST Val/Path	Recirc Val/Path			
Large LOCAs										
1. DE CL	yes	na	na	1	1	na	na	no	2017	--
2. Split CL	yes	na	na	1	1	na	na	no	2030	--
3. DE CL	no	na	na	1	2	na	na	no	1925	--

Long Term Cooling T&H Uncertainty

- **AP600 LTC T&H Uncertainty Cases**
  - –3 cases with 2 windows per case
  - –2" LOCAs not limiting > not needed for AP1000
  - –Other cases covered by AP1000 DCD analysis and extra margin provided in ADS 4 / IRWST inject / Cont Rcirc

AP600 T&H Uncertainty Cases / Equipment Availability										
	Cont Isol	ADS 1/2/3	ADS 4	CMT	Acc	IRWST Val/Path	Recirc Val/Path	Applicable to AP1000	AP600 PCT (F)	AP1000 PCT (F)
Long Term Cool										
1. 2" CL, IRWST	yes	0	3	0	1	1/1	na	no	na	--
2. 2" CL, recirc	yes	0	3	0	1	1/1	1/1	no	na	--
3. DVI, IRWST	yes	0	3	0	1	2/1	na	no	na	--
4. DVI, recirc	yes	0	3	0	1	2/1	1/1	no	na	--
5. DVI, IRWST	no	0	4	1	1	2/1	na	no	na	--
6. DVI, recirc	no	0	4	1	1	2/1	2/1	no	na	--

AP1000 Success Criteria / T&H Analysis

- **Success Criteria Made More Robust**
  - –Minimizes T&H issues / uncertainty
- **Success Criteria Verified**
  - –DCD analysis, specific PRA analysis or other calculations
  - –Specific PRA Analysis Performed
    - –Used insights / lessons learned from AP600
    - –Analysis shows similar / less severe results than AP600
- **T&H Uncertainty Bounded**
  - –DCD analysis methods used to bound T&H uncertainty for low margin / risk important accident scenarios
  - –Fewer cases analyzed because of more conservative success criteria

/

July 23, 1998

The Honorable Shirley Ann Jackson
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Dear Chairman Jackson:

SUBJECT: REPORT ON THE SAFETY ASPECTS OF THE WESTINGHOUSE ELECTRIC COMPANY APPLICATION FOR CERTIFICATION OF THE AP600 PASSIVE PLANT DESIGN

During the 454th meeting of the Advisory Committee on Reactor Safeguards, July 8-10, 1998, we completed our safety review of the Westinghouse Electric Company application for certification of its AP600 passive plant design. This report is intended to fulfill the requirement of 10 CFR 52.53 that "the ACRS shall report on those portions of the application which concern safety." During our review, we had the benefit of discussions with representatives of Westinghouse and its consultants, and the NRC staff. We also had the benefit of the documents referenced.

AP600 Application

On June 26, 1992, Westinghouse tendered its application to the NRC for certification of the AP600 design. This application was submitted in accordance with Subpart B, "Standard Design Certifications," of 10 CFR Part 52, "Early Site Permits; Standard Design Certifications; and Combined Licenses for Nuclear Power Plants," and Appendix O, "Standardization of Design: Staff Review of Standard Designs." The application was docketed on December 31, 1992, and assigned Docket Number 52-003.

The application consists of the AP600 Standard Safety Analysis Report (SSAR), the Tier 1 Material, and the probabilistic risk assessment (PRA). On June 26, 1992, Westinghouse submitted the SSAR and the PRA. In December 1992, Westinghouse submitted the Tier 1 Material, which contains inspections, tests, analyses, and acceptance criteria (ITAAC) and Tier 1 design descriptions. Design certification is sought for the power generation complex, excluding those elements and features considered site-specific. All safety-related structures, systems, and components (SSCs) are located on the nuclear island and are to be included in the design certification.

Three aspects of the plant design (i.e., instrumentation and control (I&C) systems, human factors engineering, and some piping) will be completed by the combined license (COL) applicant using the design processes described in the SSAR and ITAAC.

The staff issued a Draft Safety Evaluation Report (DSER) on November 30, 1994, a supplement to the DSER in April 1996, and an Advance Final Safety Evaluation Report on May 2, 1998. Our activities related to the review of the AP600 design are listed in the Attachment. As a result of our review, we issued three interim letters identifying several issues. The resolution proposed by Westinghouse to these issues is acceptable, pending staff review and approval.

AP600 Design Description

The AP600 plant is designed for use at either single-unit or multiple-unit sites. The scope of t design is complete except for site-specific elements. The AP600 design has a nuclear steam supply system rating of 1933 MWt, with an electrical output of at least 600 MWe. The plant has a design objective of 60 years without a planned replacement of the reactor vessel. The design does provide, however, for the replacement of other major components, including the steam generators.

The primary objective of the AP600 design is to meet safety requirements and goals defined for advanced light-water reactors with passive safety features as specified in the Electric Power Research Institute Utility Requirements Document. An additional objective is to provide a greatly simplified plant with respect to design, licensing, construction, operation, inspection, maintenance.

The plant arrangement consists of five principal structures; the nuclear island, the turbine building, the annex building, the diesel generator building, and the radwaste building.

The nuclear island, which includes all safety-related or seismic Category I structures, is designed to withstand the effects of natural phenomena and postulated events. It consists of a containment building, a concrete shield building, and an auxiliary building, which are described below.

- The containment building consists of a free-standing steel containment vessel which has a design pressure of 45 psig and associated internal structures. The vessel performs the function of containing the release of radioactivity to the atmosphere following postulated design-basis accidents. The vessel is also part of the passive containment cooling system.

- The shield building comprises the structure and annulus area that surrounds the containment building. In the event of an accident, the passive containment cooling system releases water that runs down the outside of the containment vessel to enhance heat removal.

- The auxiliary building is designed to provide protection and separation for the seismic Category 1 mechanical and electrical equipment located outside the containment building. The building also provides protection for safety-related equipment against the consequences of internal or external events. The main control room, Class 1E I&C systems, Class 1E electrical systems, and reactor fuel handling area are contained in the auxiliary building.

The turbine building houses the main turbine generator and associated fluid and electrical systems. The annex building includes the health physics area, the technical support center, access control, and personnel facilities. The diesel generator building houses two diesel generators and their associated support systems. The radwaste building contains facilities for the handling, processing, and storing of radioactive wastes.

The overall plant arrangement utilizes building configurations and structural designs to minimize the building volumes and quantities of bulk materials (concrete, structural steel, rebar) consistent with safety, operational, maintenance, and structural needs. The plant arrangement provides separation between safety and nonsafety equipment and systems to preclude adverse interactions among them. Separation between redundant safety equipment and systems provides confidence that the safety functions can be performed. In general, this separation is provided by concrete walls.

The ITAAC program is intended to ensure that the plant, when built, conforms to the design parameters and assumptions that existed at the time of design certification. For example, the efficacy of the passive emergency core cooling system depends on the flow resistances of piping segments, relief valves, and other components. The flow resistances will be measured in the as-built plant to ensure that they conform with the values derived and validated by the test and analysis program.

Safety Enhancement Features

The AP600 design contains many features that are not found in current operating plants. For example, a variety of engineering and operational improvements provide additional safety margins and comply with the Commission's Severe Accident, Safety Goal, and Standardization Policy Statements. Unique features of the AP600 design include an improved reactor core design, a large reactor vessel, a large pressurizer, an in-containment refueling water storage tank (IRWST), an automatic depressurization system, a digital microprocessor-based I&C system, hermetically sealed canned rotor coolant pumps mounted to the steam generator, and increased battery capacity.

The AP600 design represents a significant departure from previous commercial nuclear reactor technology in that it places more dependence on passive systems for accident response. Passive systems depend on gravity, condensation, and small pressure differences to prevent or mitigate damage to the core and to ensure containment of radioactive fission products in the event of accidents. Active systems, on the other hand, employ flow loops and pumps that require electrical or other sources of motive power. The performance of active systems is, in general, better known because of existing test data and extensive operating experience. Passive systems, although not tested under full-scale conditions, are more likely to ensure safety functions, especially under conditions where external or emergency motive power could be compromised.

The AP600 I&C systems are significantly different from those in current operating plants. The primary differences result from using software-based digital systems with multiplexed and fiber optics data links in place of the analog systems. The use of digital systems with multiplex and fiber optics data links reduces the amount of cabling in the plant, thereby reducing configuratic complexity and fire hazards.

The AP600 design does not require Class 1E electrical power except that provided by the Class 1E dc batteries and their inverters. This feature significantly reduces the complexity of the pl electrical systems and the reliance on safety-grade diesel generators.

The AP600 plant includes an innovative security plan which features the use of defensive capabilities at various vital area access points. This feature results in elimination of the protective area boundary and associated security attributes used at current operating nuclear power plants.

AP600 Test and Analysis Program

Westinghouse conducted an extensive test and analysis program, utilizing separate-effects and integral-system facilities both to investigate the behavior of the AP600 passive safety systems and to develop a database for validation of the computer codes used to perform accident and transient analyses. Key aspects of the test and analysis program include:

- Core Makeup Tank (CMT) Test Program to characterize the CMT over an extended range of thermal-hydraulic conditions.
- Automatic Depressurization System (ADS) Test Program, both to characterize the steam flow through the IRWST sparger and to test the thermal-hydraulic behavior of the ADS piping network.
- Passive Residual Heat Removal (PRHR) System Test Program to generate data for design and characterization of the AP600 PRHR heat exchanger.
- Oregon State University Advanced Plant Experiment (APEX) Test Program to obtain integral-systems data for code validation; emphasis was placed on low-pressure and long-term core cooling behavior for design-basis, small-break loss-of-coolant accidents (LOCAs).
- SPES-2 High-Pressure, Full-Height Integral-Systems Test Program to obtain integral-systems data for code validation; the particular focus was on accident progression from initiation to establishment of stable IRWST injection.
- Passive Containment Cooling System Test Program to obtain integral-systems test data on the thermal-hydraulic performance of this system to support code validation.

This extensive test and analysis program was necessary to validate the accident analysis codes applied to new, passive emergency core cooling systems for which there is not a significant experience base. The accident analysis codes used by Westinghouse included:

- LOFTRAN/LOFTTR2 for analyses of non-LOCA transients
- NOTRUMP for evaluation-model analyses of small-break LOCAs
- WCOBRA/TRAC for best-estimate analyses of large-break LOCAs
- WCOBRA/TRAC for analyses of long-term core cooling
- WGOTHIC for design-basis accident analyses of the containment

To ensure that the test and analysis program adequately addressed important phenomena with respect to the passive systems and that the results would scale to the prototype size, Westinghouse developed a phenomena identification and ranking table and performed a scaling analysis for both the primary coolant system and the containment.

In addition, the NRC staff performed confirmatory experimental and analytical programs in support of the AP600 design certification review. These programs included the integral-systems testing performed at the Japan Atomic Energy Research Institute ROSA-AP600 facility, and follow-on testing performed at the Oregon State University APEX facility. The NRC staff also performed confirmatory analyses utilizing the NRC codes RELAP-5 and CONTAIN. The results of the staff's programs significantly aided our review of the Westinghouse test and analysis program.

During the extensive reviews of the Westinghouse test and analysis program, we raised numerous issues. These issues have been documented in our interim letters and meeting minutes. Based on discussions with representatives of Westinghouse and the NRC staff, all of our issues pertaining to the Westinghouse test and analysis program have been adequately resolved.

There are, however, a number of issues that arose during our review that, while not directly affecting the acceptability of the AP600 test and analysis program, should be considered in the context of future design certification reviews. We plan to address these issues in a future lett pertaining to lessons learned from the AP600 design certification review.

Probabilistic Risk Assessment

The AP600 design certification application included a PRA, in accordance with regulatory requirements. This PRA was done well and rigorous methods were used to quantify risk metrics, including core damage frequency (CDF) and large, early release frequency (LERF). Point estimates of the risk metrics are:

CDF = 2 X 10-7 per reactor year
LERF = 2 X 10-8 per reactor year

These risk metrics are low compared to those estimated for existing nuclear power plants. The PRA was an integral part of the design process. This contributed significantly to design modifications, which resulted in the low CDF and LERF.

The PRA addressed passive safety systems and software-based digital I&C systems. Qualitative analyses and extensive sensitivity studies were used to compensate for incomplete modeling of these important features of the plant. In addition, the concept of the "focused" PRA was introduced to reduce uncertainties in the estimated performance of passive systems. The objective of the "focused" PRA was to determine whether the goals for CDF and LERF could be met without the support of the nonsafety-related systems. The regulatory treatment of nonsafety systems (RTNSS) process was used to impose special requirements on some nonsafety systems to ensure, with high confidence, that they would be available when needed. For example, Westinghouse used the RTNSS process to impose administrative controls on the availability of the engineered safety feature actuation function of the diverse actuation system order to reduce uncertainties associated with the digital system software. The RTNSS process is an excellent example of a good risk-informed and performance-based approach.

We applaud the use of the "focused" PRA and the RTNSS process in developing defense-in-depth measures. But, we caution against establishing the practice of comparing the results of "focused" PRAs with Safety Goals. These Goals apply to a plant as it is designed and operated. Comparison of these Goals with results of analyses, restricted to include only safety systems, would amount to the imposition of a new goal that does not appear in the Commission's Safety Goal Policy Statement.

Additional Observations

Westinghouse's approach for quantifying digital systems software in the PRA is consistent with the guidance in Regulatory Guide 1.152, "Criteria for Digital Computers in Safety Systems in Nuclear Power Plants." This approach provides a method for identifying and assessing design strengths and weaknesses.

The AP600 plant will use passive autocatalytic recombiners to maintain hydrogen concentrations below the flammability limit within the containment following design-basis accidents. We agree, in principle, that these devices are improvements over hydrogen recombiners used in existing plants. The COL applicant is responsible for qualifying passive autocatalytic recombiners. The present regulatory requirements for qualifying mechanical equipment are insufficient to ensure continued passive autocatalytic recombiner operation for the expected duty cycle.

The AP600 reactor containment is a steel shell. It has been designed to meet Service Level C of the ASME Boiler and Pressure Vessel Code. The containment meets all regulatory requirements. Testing has shown that steel shell containments are susceptible to catastrophic failure when overpressurized. For the AP600 design, however, under the peak pressure calculated in the Level 2 PRA for severe accident conditions, the probability of failure of the containment is estimated to be approximately 0.01. Deformation of the pressurized containment vessel and its interaction with the shield building could also induce leakage and further reduce the likelihood of failure. In any event, we have not been able to identify significant risks associated with possible catastrophic failure modes of the AP600 containment.

Westinghouse has concluded that external reactor vessel cooling will prevent core debris from penetrating the reactor vessel. This conclusion is based on a scenario for degradation of the core that avoids consideration of direct contact by metallic core debris with the reactor vessel. The NRC staff has concluded that reactor vessel failure is not precluded and has required that Westinghouse consider ex-vessel core debris interactions. Westinghouse performed these evaluations and found that the AP600 containment performs satisfactorily under these severe conditions.

ACRS Conclusion Concerning AP600 Design

Based on our review of those portions of the AP600 application which concern safety, we believe that acceptable bases and requirements have been established to ensure that the AP600 design can be used to engineer and construct plants that with reasonable assurance can be operated without undue risk to the health and safety of the public.

Dr. Thomas S. Kress did not participate in the Committee's deliberation regarding external reactor vessel cooling.

Dr. Dana A. Powers did not participate in the Committee's deliberation regarding the AP600 source term or the results of Sandia National Laboratories tests on containment structural integrity and on environmental qualification of passive autocatalytic recombiners.

Dr. George Apostolakis did not participate in the Committee's deliberation regarding the AP600 passive system reliability assessment or the analyses performed by the Idaho Engineering and Environmental Laboratory concerning the use of the WCOBRA/TRAC code and external reactor vessel cooling.

Sincerely,

/s/

R. L. Seale
Chairman

References:

1. U.S. Nuclear Regulatory Commission, "Advance Final Safety Evaluation Report Related to the Certification of the AP600 Design," May 1998 (Predecisional Information).
2. U.S. Department of Energy Report DE-AC03-90SF18495, dated June 26, 1992, prepared by Westinghouse Electric Corporation, "AP600 Standard Safety Analysis Report," updated through Revision 23 (issued May 18, 1998).
3. U.S. Department of Energy Report DE-AC03-90SF18495, dated June 26, 1992, prepared by Westinghouse Electric Company, "AP600 Probabilistic Risk Assessment," updated through Revision 11 (issued March 1998).
4. U.S. Department of Energy Report DE-AC03-90SF18495, December 1992, "AP600 Tier 1 Material," updated through Revision 5 (issued May 7, 1998).
5. Letter dated February 19, 1998, from R. L. Seale, Chairman, ACRS, to L. Joseph Callan, Executive Director for Operations, NRC, Subject: Interim Letter on the Safety Aspects of the Westinghouse Electric Company Application for Certification of the AP600 Plant Design.
6. Letter dated April 9, 1998, from R. L. Seale, Chairman, ACRS, to L. Joseph Callan, Executive Director for Operations, NRC, Subject: The Safety Aspects of the Westinghouse Electric Company Application for Certification of the AP600 Plant Design-Interim Letter 2.
7. Letter dated June 15, 1998, from R. L. Seale, Chairman, ACRS, to L. Joseph Callan, Executive Director for Operations, NRC, Subject: The Safety Aspects of the Westinghouse Electric Company Application for Certification of the AP600 Plant Design-Interim Letter 3.
8. U.S. Nuclear Regulatory Commission, Draft NUREG-1512, "Draft Safety Evaluation Report Related to the Certification of the AP600 Design," November 1994.
9. U.S. Nuclear Regulatory Commission, Supplement to the "Draft Safety Evaluation Report Related to the Certification of the AP600 Design," April 1996.
10. Westinghouse Electric Corporation, WCAP-14812, Revision 2, "Accident Specification and Phenomena Evaluation for AP600 Passive Containment Cooling System," April 1998 (Proprietary).
11. Westinghouse Electric Corporation, WCAP-14845, Revision 3, "Scaling Analysis for AP600 Containment Pressure During Design Basis Accidents," March 1998 (Proprietary).
12. Westinghouse Electric Corporation, WCAP-14326, Revision 3, "Experimental Basis for the AP600 Containment Vessel Heat and Mass Transfer Correlations," May 1998 (Proprietary).
13. Westinghouse Electric Corporation, WCAP-14135, Revision 1, "Final Data Report for PCS Large-Scale Tests, Phase 2 and Phase 3," April 1997 (Proprietary).
14. Westinghouse Electric Corporation, WCAP-14382, Revision 0, "WGOTHIC Code Description and Validation," May 1995 (Proprietary).
15. Westinghouse Electric Corporation, WCAP-14407, Revision 3, "WGOTHIC Application to AP600," April 1998 (Proprietary).
16. Office of Nuclear Regulatory Research (RES) Report, RPSB-98-04, "Phenomenology Observed in the AP600 Integral Systems Test Programs Conducted in the ROSA-AP600, APEX, and the SPES-2 Facilities," D. Bessette, RES, M. DiMarzo, University of Maryland, P. Griffith, Massachusetts Institute of Technology, April 1998.
17. Letter dated July 1, 1998, from B. McIntyre, Westinghouse, to NRC: Attention: J. Larkins, ACRS, transmitting Response to ACRS Request for NOTRUMP Break Area Sensitivity Study.

18. Letter dated July 2, 1998, from B. McIntyre, Westinghouse, to NRC: Attention: J. Larkins, ACRS, Subject: Responses to ACRS Reactor Coolant System Issues.
19. Letter dated June 25, 1998, from B. McIntyre, Westinghouse, to NRC: Attention: J. Larkins, ACRS, Subject: Closure of ACRS Thermal Hydraulic Subcommittee Items for June 11-12, 1998 Meeting.

Attachment: Chronology of the ACRS Review of the Westinghouse Application for the AP600 Passive Plant Design Certification

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ATTACHMENT

CHRONOLOGY OF THE ACRS REVIEW OF THE WESTINGHOUSE APPLICATION FOR THE AP600 PASSIVE PLANT DESIGN CERTIFICATION

The extensive ACRS review of the AP600 design and its interactions with representatives of the NRC staff and Westinghouse are discussed in the minutes of the following ACRS meetings. The questions raised by ACRS members during meetings which were not formally documented in ACRS reports and letters were answered during subsequent discussions.

ACRS MEETING/DATES	SUBJECT
Thermal Hydraulic Phenomena 12/17/91	Proposed Commission Paper on Need for Full-Height, Full-Pressure Integral System Testing of AP600 Design
Thermal Hydraulic Phenomena 3/3/92	Integral System Testing Requirements for AP600 Design
Thermal Hydraulic Phenomena 6/23-24/92	Integral System Testing Requirements for AP600 Design
Thermal Hydraulic Phenomena 3/4-5/93	Office of Nuclear Regulatory Research (RES) RELAP5/MOD3 Code
Thermal Hydraulic Phenomena 7/22-23/93	Westinghouse Test and Analysis Program (TAP)
Thermal Hydraulic Phenomena 9/21/93	TAP - Oregon State University APEX Test Facility
Thermal Hydraulic Phenomena 10/28/93	RES - ROSA-V (ROSA-AP600) Confirmatory Test Program
Thermal Hydraulic Phenomena 1/4-5/94	RES - RELAP5/MOD3 Code
Thermal Hydraulic Phenomena 3/15-16/94	TAP - Core Makeup Tank Test Facility, Passive Containment Cooling System
Thermal Hydraulic Phenomena 5/18-19/94	TAP - WCOBRA/TRAC Code
Thermal Hydraulic Phenomena 8/25-26/94	RES - Confirmatory Test Programs
W Standard Plants Designs 1/11/95	Overview and General Description of the AP600 Plant Design
Thermal Hydraulic Phenomena 2/15-16/95	TAP - WCOBRA/TRAC Code
Thermal Hydraulic Phenomena 3/27-28/95	RES - Phenomena Identification and Ranking Table (PIRT) for RELAP5 Code
Thermal Hydraulic Phenomena 3/29-30/95	TAP - Passive Containment Cooling System

W Standard Plant Designs 5/31/95	Commission Paper on Status of Ten Key Technical and Policy Issues
Thermal Hydraulic Phenomena 7/26-27/95	Qualification Document for the WCOBRA/TRAC Code
Thermal Hydraulic Phenomena 1/18-19/96	Qualification Document for the WCOBRA/TRAC Code
Thermal Hydraulic Phenomena 2/22-23/96	RES Program for Demonstrating Adequacy of the RELAP5/MOD3 Code to Assess Behavior of AP600 Design
Thermal Hydraulic Phenomena 5/9-10/96	TAP - Overview
Severe Accidents 6/5/96	Probabilistic Risk Assessment of Severe Accidents
W Standard Plant Designs 7/19/96	SECY-96-128, "Policy and Key Technical Issues Pertaining to the AP600 Design"
433rd ACRS Meeting 8/8/96	SECY-96-128, "Policy and Key Technical Issues Pertaining to the AP600 Design" ACRS Report Issued 8/15/96
W Standard Plant Designs 12/4/96	Chap. 4: Reactor Chap. 5: Reactor Coolant System and Connected Systems Chap. 9: Auxiliary Systems Chap. 11: Radioactive Waste Management
Thermal Hydraulic Phenomena 12/18-19/96	TAP - Scaling and PIRT Closure Report
Thermal Hydraulic Phenomena 2/12-14/97	RES Program for Demonstrating Adequacy of the RELAP5/MOD3 Code to Assess Behavior of AP600 Design
Thermal Hydraulic Phenomena 2/19/97	RES - ROSA-AP600 Confirmatory Test Program
Thermal Hydraulic Phenomena 3/28/97	TAP - Long-Term Cooling with WCOBRA/TRAC Code
442nd ACRS Meeting 6/13/97	AP600 Containment Spray System ACRS Report issued 6/17/97
Thermal Hydraulic Phenomena 7/29-30/97	TAP - NOTRUMP Small-Break LOCA Code
Thermal Hydraulic Phenomena 9/29-30/97	TAP - Passive Containment Cooling System
Thermal Hydraulic Phenomena 12/9-10/97	TAP - PIRT; Scaling of Reactor Coolant System; NOTRUMP Code
Thermal Hydraulic Phenomena 12/11-12/97	TAP - WGOTHIC Containment System Code
Advanced Reactor Designs 2/3-4/98	Chap. 7: Instrumentation and Controls Chap. 8: Electrical Power Chap. 13: Conduct of Operations Chap. 18: Human Factors Engineering
448th ACRS Meeting 2/5/98	TAP Chap. 1: Introduction and General Discussion Chap. 4: Reactor Chap. 5: Reactor Coolant System and Connected Systems Chap. 7: Instrumentation and Controls Chap. 8: Electrical Power

	Chap. 11: Radioactive Waste Management Chap. 13: Conduct of Operations Chap. 18: Human Factors Engineering Interim ACRS letter issued 2/19/98
Advanced Reactor Designs 3/30 - 4/1/98	Chap. 2: Site Characteristics Chap. 9: Auxiliary Systems Chap. 10: Steam and Power Conversion Chap. 12: Radiation Protection Chap. 13: Conduct of Operations (Security) Chap. 15: Accident Analyses
451st ACRS Meeting 4/2/98	TAP Chap. 2: Site Characteristics Chap. 9: Auxiliary Systems Chap. 10: Steam and Power Conversion Chap. 12: Radiation Protection Chap. 13: Conduct of Operations (Security) Chap. 15: Accident Analyses Interim ACRS Letter 2 Issued April 9, 1998
Thermal Hydraulic Phenomena 5/11-12/98	TAP - Primary Coolant System
Advanced Reactor Designs 5/13-15/98	Chap. 1: Introduction and General Discussion Chap. 6: Engineered Safety Features Chap. 14: Initial Test Program Chap. 16: Technical Specifications Chap. 17: Quality Assurance Levels 2 and 3 PRA Regulatory Treatment of Nonsafety Systems
453rd ACRS Meeting 6/3/98	TAP Chap. 3: Design of Structures, Components, Equipment, and Systems Chap. 6: Engineered Safety Features Chap. 9: Appendix A - Fire Protection Analysis Chap. 14: Initial Test Program Chap. 16: Technical Specifications Chap. 17: Quality Assurance PRA Interim ACRS Letter 3 Issued June 15, 1998
Thermal Hydraulic Phenomena 6/11-12/98	TAP - Passive Containment Cooling System
Advanced Reactor Designs 6/17-18/98	ITAAC; Level 1 PRA; Adverse Interaction Evaluation Report; and Containment Spray System
Advanced Reactor Designs 7/7/98	TAP and Responses to ACRS Questions

The Nuclear Regulatory Commission (NRC) staff identified the following issues as needing further review during the design certification phase in the March 25, 2002, AP1000 pre-application review assessment (these issues are documented on page 2 of this assessment). Some of these issues were also raised in the NRC staff RAIs as discussed below. Original wording from pre-application review assessment is in bold.

(1) **Westinghouse has not demonstrated that the existing AP600 integral tests provide data over the range of conditions necessary to validate entrainment models in the NOTRUMP and WCOBRA/TRAC codes that they intend to use. In particular, the NOTRUMP code lacks acceptable models for liquid entrainment in the upper plenum or from a horizontal stratified water level in the hot legs during the ADS-4 actuation.**

NRC Staff position: Westinghouse should provide appropriate test data over the range of conditions necessary to validate entrainment models in NOTRUMP and WCOBRA/TRAC that they intend to use for the AP1000.

The NRC staff documented its questions associated with this issue in requests for additional information (RAIs) 440.149 through 440.173. Westinghouse provided its responses to these RAIs via letters dated October 2, 2002 (RAIs 440.149, 150, and 167), October 18, 2002 (RAIs 440.152, 153, 156, and 169), November 1, 2002 (RAIs 440.151, 158, 159, 160, 164, 170, 171, and 172), November 15, 2002 (RAIs 440.154, 161, 165, and 166), November 26, 2002 (RAIs 440.155, 157, 163, 168, and 173), and December 2, 2002 (RAI 440.162). In response to identification of this issue in the pre-application review, Westinghouse submitted WCAP-15833, "WCOBRA/TRAC AP1000 ADS-4/IRWST [automatic depressurization syestem phase 4/in-containment refueling water storage tank] Phase Modeling." In its RAI responses, Westinghouse proposed changes to WCAP-15833 to address issues raised by the NRC staff. (The NRC staff has not yet received the next revision of WCAP-15833.) The NRC staff has reviewed the design certification application, the RAI responses, and WCAP-15833, and has determined that the information submitted to date does not address the issues raised in the RAIs with respect to upper plenum liquid entrainment. The NRC staff believes that Westinghouse should provide appropriate test data over the range of conditions necessary to validate entrainment models in NOTRUMP and WCOBRA/TRAC that they intend to use to support the AP1000 design certification application. (Submission of new test data is necessary to resolve the issues of modeling entrainment phenomena, including upper plenum entrainment, that occur during a small-break loss-of-coolant accident (SBLOCA).) The NRC staff is in the process of documenting this position in a letter to Westinghouse (the NRC staff will deliver a copy of this letter to the ACRS members).

As reflected in the staff's letter dated February 28, 2003, to Westinghouse, the staff considers the following RAIs to be unresolved: 440.151, 152, 154, 157, 160, 161, 162, 164, 169, 171, and 173 (these represent potential open items in the draft safety evaluation report [DSER]).

(2) **The review of the ability of the LOFTRAN code to evaluate potential steam voids within the reactor system following a main steamline break (MSLB) will be deferred to Phase 3, since Westinghouse did not provide an MSLB analysis for the AP1000 plant design.**

NRC staff position: Westinghouse should demonstrate that voids are not produced in the reactor coolant loops following a MSLB beyond the capability of the LOFTRAN code.

Westinghouse submitted its MSLB analysis in its design certification application dated March 28, 2002 (Section 15.1.5). The NRC staff documented its questions associated with this issue in RAI 440.054. Westinghouse provided its response to RAI 440.054 on November 1, 2002.

Main steam line break analyses were performed with and without off site power present. Voiding was not calculated to occur in the reactor coolant loops.

The NRC staff has reviewed these documents and considers this issue (and RAI 440.054) to be resolved by the docketed information (see letter to Westinghouse dated February 28, 2003).

(3) **Westinghouse needs to qualify the penalty factor used with the NOTRUMP passive residual heat removal (PRHR) heat exchanger (HX) model. Existing PRHR HX test data show the boiling heat transfer correlation used in NOTRUMP to be non-conservative at high heat fluxes. The difference between the correlation predictions and test data becomes significant for the PRHRHX heat fluxes predicted for the AP1000, which are larger than those predicted for the AP600 standard plant design.**

NRC staff position: Westinghouse needs to justify PRHR HX for high heat flows.

The staff's RAI assoicated with this issue is RAI 440.054. Westinghouse responded to this RAI on November 1, 2002. Westinghouse reduces the PRHR HX heat transfer area by 50% in all NOTRUMP calculations for AP1000. A calculation was performed showing that the 50% reduction produced conservative results in comparison to test data.

The NRC staff has reviewed these documents and considers this issue (and RAI 440.054) to be resolved by the docketed information (see letter to Westinghouse dated February 28, 2003).

(4) **Westinghouse did not justify that the increased flow area of the ADS-4 would support the liquid expulsion to avoid boron precipitation in the vessel during long-term cooling.**

NRC staff position: Westinghouse needs to justify ADS-4 liquid entrainment model under long-term cooling conditions and to quantify the expulsion of liquid from the vessel.

This issue was addressed in RAI 440.091. Westinghouse provided its response on December 2, 2002. Qualification of the WCOBRA/TRAC model is needed to calculate liquid entrainment for the AP1000 ADS4 configuration. Also needed is the quantification of liquid flow from the vessel.

The staff considers RAI 440.091 to be unresolved (additional information is necessary to resolve the issue raised in the RAI). This represents a potential open item in the DSER

(5) **Westinghouse did not justify the methodology used to calculate peak clad temperature (PCT) in the event that the core becomes uncovered during a small break LOCA.**

NRC staff position: Additional SB LOCA break sizes need to be analyzed. If core uncovery is predicted, the staff must review core heatup codes for AP1000.

This issue was addressed in RAI 440.098. Westinghouse provided its response on November 1, 2002. Additional break sizes were analyzed. No core uncovery was calculated.

The staff considers RAI 440.098 to be resolved.

(6) **Westinghouse did not properly scale the containment large scale test (LST) for transients, and the test is only valid for steady-state conditions. This limitation was identified during the AP600 review and also applies to the AP1000 design. However, the LST does support the mass and heat transfer correlations used in the WGOTHIC code for the AP600 and the AP1000. Westinghouse needs to perform the WGOTHIC containment analyses with an evaluation model and appropriate boundary conditions to ensure that the mass and heat transfer correlations remain valid for the AP1000 design.**

This issue was addressed in Westinghouse's design certification application dated March 28, 2002. The WGOTHIC licensing evaluation model was developed to account for scaling related issues identified with the Large Scale Test (LST) facility. Conservative modeling was developed to address mixing and stratification, as well as the mass and heat transfer correlations. The AP1000 WGOTHIC model is now consistent with the staff approved model.

No RAIs were necessary and this issue is considered resolved.

WESTINGHOUSE AP1000

DESIGN CERTIFICATION REVIEW

SPECIFIC UNRESOLVED ISSUES FROM

NRC STAFF'S REQUEST FOR ADDITIONAL INFORMATION

CONCERNING

PROBABILISTIC RISK ASSESSMENT

TO BE DISCUSSED AT

FEBRUARY 26, 2003

PUBLIC MEETING

RAI 720.038

As discussed in RAI number 720.038, an important objective in the AP1000 design certification is to identify important PRA insights and assumptions to ensure that they have been addressed in ITAAC and D-RAP and COL action items. The following questions concern the lack of documentation regarding shutdown risk significant assumptions and features of the AP1000 design. Given the updated common cause analyses for the HP and LP squib valves and the shorter operator response times, the staff needs importance and sensitivity analyses for the AP1000 shutdown PRA documented in the AP1000 shutdown PRA rather than referring the reader to the AP600 Shutdown PRA. The sensitivity analyses should include:

a. The AP1000 Shutdown CDF based on a licensee following the minimum compliance with Technical Specifications which includes the licensee having only one IRWST injection and recirculation path operable during modes 5 and 6.
b. The results of the focused AP1000 shutdown PRA.
c. The AP1000 Shutdown CDF assuming all human error probabilities (HEPs) are set to .5 which includes HEPs RCS-MANODS1 and RCSMANODS2.

2 Please justify in the AP1000 DCD section 19.E.4.8.1 why the NOTRUMP analyses performed for the AP600 plant to show the plant response to a loss of RNS cooling in Mode 4 with the RCS intact and in Mode 5 with the RCS is acceptable for the AP1000 design.
covered by 440.119

3. Vacuum refill of the RCS from drained conditions was mentioned; however, no risk assessment was done for this plant configuration. Passive RHR should be operable according to the AP1000 Technical Specifications during this plant configuration since the RCS would be closed which should reduce risk. However, Westinghouse should document in the AP1000 Shutdown PRA the additional plant risk occurring from vacuum refill of the RCS during drained conditions and how this risk affects the AP1000 shutdown PRA results.

4. Based on the RAI response to 720.065, the staff does not believe that the ability to close the containment in the AP1000 is the same for AP600 because the time to boiling is reduced from 17 to 10 minutes. The staff also noted that shutdown LRF frequencies were reported in AP1000 Implementation of the Regulatory Treatment of Nonsafety-Related Systems Process (on Table 2-2). However, there is no discussion of shutdown LRF in the AP1000 shutdown PRA, nor is there a discussion regarding the failure likelihood of closing containment given a severe accident at shutdown. The staff is requesting Westinghouse to document in the AP1000 shutdown PRA : (1) the assessment used to estimate that likelihood that the operators could fail to close containment during shutdown, and (2) a corresponding discussion of the shutdown LRF frequencies in the AP1000 shutdown PRA.

5. Westinghouse's response to RAI 720.070 is not adequate for the staff to derive AP1000 risk insights regarding shutdown fires, shutdown floods and seismic events at shutdown.

a. As reported in RAI 720.070, the AP600 shutdown fire frequency is comparable to the AP600 at power fire frequency. Thus, Westinghouse is requested to document changes in the AP1000 shutdown fire assessment from the AP600 shutdown fire assessment. Specifically, Westinghouse needs to identify and document in the AP1000 Shutdown PRA: (1) any differences in equipment locations in the various fire areas and zones with respect to the AP600 design and (2) the qualitative or quantitative impacts (if necessary) of such differences on shutdown risk results and shutdown risk insights.

b. The staff did not find any AP1000 PRA based insights regarding how transient combustibles will be controlled at shutdown to maintain the assumed shutdown ignition frequencies. Westinghouse needs to document in the AP1000 shutdown PRA how transient combustibles at shutdown will be controlled.

c. Considering the updated common cause analysis for the HP and LP squib valves and the revised shutdown initiating event frequencies based on an 18 month refueling cycle, Westinghouse is requested to provide the dominant AP1000 shutdown fire scenarios in the AP1000 shutdown PRA.

d. Westinghouse is requested to document in the AP1000 Shutdown PRA any changes from the AP600 shutdown internal floods assessment that could impact AP1000 shutdown risk insights.

e. Westinghouse is also requested to document in the AP1000 Shutdown PRA the dominant AP1000 shutdown flooding scenarios.

DISCUSSION TOPICS RELATED TO RESPONSES TO RAI's TO BE DISCUSSED AT THE 2/26/2003 MEETING BETWEEN NRC AND WESTINGHOUSE

RAI 720.027

The staff had requested additional information about several differences in initiating event category frequencies used in the AP600 and the AP1000 probabilistic risk assessments (PRAs). These differences are related to (a) various loss of coolant accident (LOCA) categories, (b) steam generator tube rupture (SGTR) accidents, and (c) passive residual heat removal (PRHR) tube rupture accidents. Westinghouse's response need further clarification in the following areas:

1. For LOCA categories, Westinghouse states that in the AP1000 PRA "operating experience" data reported in NUREG/CR-5750 for pipe breaks as opposed to data from a pipe break analysis used in the AP600 PRA were used. However, the NUREG/CR-5750 data rely on expert opinion and include significant uncertainty. In addition, since NUREG/CR-5750 was published additional information (e.g., Davis Besse finding) is available. The impact of this uncertainty on results and conclusions, especially in combination with other outstanding issues (e.g., late containment failure modeling and common cause failure probabilities of squib valves), needs to be investigated. In particular, the combined effect of such uncertainties on the process used to identify "low margin risk significant" sequences for bounding thermal-hydraulic (T-H) uncertainty and on the regulatory treatment of non-safety systems (RTNSS) process should be investigated. Westinghouse's response to this RAI refers to other RAI responses which do not really include a response to the staff's question (e.g., about the combined impact of this uncertainties on the RTNSS process).

2. It is stated that the frequency of SGTR events assumed in the AP1000 PRA is based on a more recent calculation that was performed in conjunction with a replacement steam generator project which is proprietary to Westinghouse. The staff would be interested in reviewing this information.

RAI 720.028

The staff requested additional information about the impact of two potentially significant differences between the AP600 and the AP1000 PRAs in the categorization of LOCA initiating events on the approach used to identify "low margin risk significant sequences" and address T-H uncertainty. One difference involves combining two AP600 PRA LOCA categories (i.e, the medium LOCA and the intermedium LOCA) into one AP1000 PRA category (labeled medium LOCA). The other difference involves the splitting of the AP600 PRA large LOCA category into two categories, the large LOCA category (pipe breaks) and the spurious opening of the automatic depressurization system (ADS) valves category (SPADS).

Westinghouse's response covers the major part of the question. However, the staff would like to get further clarification on the following points:

1. The SPADS category includes a sequence with one accumulator available and only three ADS valves opening (sequence 8 in expanded event tree) which is assumed to be a success (i.e., no core damage). This frequency of this sequence was estimated to be 1.07E-9 (could be significantly higher if uncertainty associated with the CCF probability of squib valves is considered) which does not appear to be bounded. Please clarify.

2. In the RAI response it is stated: "NLOCA required one ADS stage 2/3 valve to open to allow RNS injection and MLOCA did not require any of these valves. In the AP1000, the more restrictive NLOCA success criteria was applied to all breaks in the MLOCA range..." However, in the expanded MLOCA event tree for AP1000 sequences with no ADS 2/3 success are shown as Ok (e.g., sequences 2 and 4).

RAI 720.029

The staff requested clarification about the basis for time windows available for several operator actions associated with specific LOCA sequences. Westinghouse's response needs further clarification regarding the following items:

1. Table provided in the response does not appear to provide adequate information for the reader to understand what human actions are involved and what assumptions are made in calculating the human error probabilities involved. First column, labeled "Event" reports the initiating event category, the second column, labeled "Time Window" does not discuss time windows.

2. Discuss whether the T-H analyses used as the basis for calculating the time windows used in the HRA include T-H uncertainties.

3. The time window for event RHN-MAN01 was revised in Table 6-3 to 10 minutes. However, it appears that the HEP calculated in Chapter 30 of the PRA is based on a time window of 20 minutes.

4. It is stated in Table 6-3 that the time window for human action CMN-MAN01 is consistent with associated recognition action but it is not stated what the time window is. Also, in Chapter 30 where the HEP is calculated, event CIT-MAN0S is listed instead of event CMN-MAN01.

5. In the revised Section 6.3.2.5 under "Medium LOCA," successful PRHR is required for successful operator action. However, PRHR does not appear at all in the MLOCA and CMT line break event trees or in the tables where the steps for calculating HEP are reported.

RAI 720.030

The AP1000 PRA event trees include a top event for containment cooling (event CHR). It is stated that *"For success paths that result in steam release to the containment, the success of containment cooling (PCS or RNS) is modeled. If containment cooling is successful, then the* *path ends in an OK state. If PCS water cooling is not successful, then the path goes to a special OK end state to allow containment integrity sensitivity studies to be made."* This "special OK" end state is labeled "late containment failure (LCF)" end state on page 4-141 and defined as an end state *"...where the containment heat removal by either passive containment cooling system (PCS) or component cooling water (CCS) heat exchangers via normal residual heat removal (RHR) fails."*

The staff requested clarification about the meaning of the "special Ok" status. Westinghouse responded that a sensitivity study shows that even if the LCF state is considered to be a core damage, the plant CDF would increase by only 29%. The staff needs further clarification about the following:

1. The major contributors to the 29% CDF increase and how this impacts the LRF.

2. The impact of this assumption on the focused PRA where no credit is taken for the non-safety related systems and the RTNSS process.

RAI 720.033

The staff requested clarification on several statements and common cause failure (CCF) probabilities related to explosive (squib) and check valves, included in Chapter 12 on Passive Core Cooling/In-Containment Refueling Water Storage Tank and in Chapter 29 on Common Cause Failure Analysis. Westinghouse's response states that the same group of valves are available following a safety injection (SI) line break as when there is no SI line break. The staff does not agree with this assumption because when an injection line fails the valves in that line are obsolete. In addition, the calculation of the common cause failure probabilities for two and three-out-of-six valves needs clarification (e.g., number of valves in injection vs. recirculation lines required to mitigate an SI break accident).

RAI 720.035

The staff requested Westinghouse to explain the process that will be used to verify that a PMS designed with the "Common Q" option will have equivalent or better reliability than the system modeled in the PRA and how the introduction of the "Common Q" option will affect important PRA-based insights about the PMS. Westinghouse responded that "the PRA results are not sensitive to small changes in PMS failure probabilities" and "The general architecture of the Common Q PMS is similar to that modeled in the AP1000 PRA and includes the features listed above." The staff needs further clarification, including a direct comparison of the design features found to be important in the PRA between the "Common Q" option and the PMS modeled in the PRA. In addition, a direct comparison of the "design certification requirements" for the two cases can help clarify the issue. Based on the results of these comparisons, the identification of new "design certification requirements" to ensure PMS reliability may be required. The same comments apply also for DAS and PLS designed with the "commercial off-the-shelf hardware and software current at the time of construction" option.

RAI 720.037

The staff requested the use a systematic approach to identify "risk significant low margin" sequences for detailed T-H uncertainty assessment. Westinghouse provided such an approach. However, there are some points that need further clarification. Examples are:

1. Impact of open issues on the frequency of analyzed sequences.

2. Scope of expanded event trees. It appears that there are some gaps in the rational used to limit the event trees that were expanded. The staff needs more details about the reasons for limiting the number of the event trees that were expanded and analyzed.

3. Discussion of investigation performed to ensure that there are no adverse system interactions between passive and active systems.

4. Explanation of LLOCA sequences 7 to 10. On what basis are these sequences found to be successful even when T-H uncertainties are included? Why sequence 6 (2 CMTs and one Acc available) leads to core damage but sequences 7 to 10 (1 CMT and two accumulators available) are ok?

5. Sequence 21 of LLOCA classified as UC8 but was not analyzed because the calculated frequency is less than 1E-9/year. However, if a more conservative frequency for large breaks and a more conservative probability for the failure to isolate the containment are considered, the frequency of this sequence will be higher than the cutoff frequency.

6. Discussion of impact of timing of operator actions on T-H analysis (e.g., for sequences requiring manual ADS actuation).

7. Discuss the basis for assuming that the SI-LB sequences 18 to 21, involving failure to isolate the containment and availability of only one CMT and one Acc, are Ok. Similarly for sequence 23 (one CMT and no Acc. available) and sequence 28 (no CMT and only one Acc. available).

8. If credit is taken in the T-H analysis for the PRHR, this system needs to be included in the appropriate event trees.

9. Discussion of the impact of T-H uncertainty on passive containment cooling success criteria assumed in the PRA.

RAI 720.038

Westinghouse identified important PRA insights and assumptions and provided a list of design certification requirements, such as requirements for inspection, tests, analyses and acceptance criteria (ITAAC), the requirement for a design reliability assurance program (D-RAP) and combined operating license (COL) action items. However, the staff cannot close this issue until all other outstanding issues are closed and significant progress in preparing the final safety evaluation report (FSER) and the design control document (DCD) is being made.

RAI 720.039

The staff requested Westinghouse to provide all important steps in the process of using PRA results to identify systems, structures and components (SSCs) for regulatory oversight as well as the type and level of such oversight for non-safety-related systems. This information should account for uncertainties in the AP1000 PRA so that it can be used by the staff to make similar conclusions, about the need for non-safety-system oversight, to those made for the AP600 design (e.g., as documented in the AP600 FSER Chapter 19.1.7 "PRA input to the RTNSS Process.)" Westinghouse did not provide this information with its response.

Review of W Responses to RAIs Related to Level 2 and 3 PRA and Severe Accidents (RAIs 720.041 - 720.063)

RAIs 720.041 through 720.063 address concerns regarding the Level 2 and 3 PRA, portions of the deterministic analyses of severe accidents, and the evaluation of severe accident mitigation design alternatives (SAMDAs) for AP1000. Additional aspects of the deterministic analyses of severe accidents (e.g., fuel coolant interactions) are addressed in other 720-series RAIs and are not addressed below.

The additional information provided by W is generally responsive to the concerns raised in the RAIs. However, for many of the RAIs, portions of the requested information was not provided. Those RAIs which have not been fully addressed are:

RAIs 720.042, 043, 046, 048, 050, 053, 055, 056, 058, and 060. Those aspects of the RAI requiring additional information from W are summarized below.

720.042 The RAI requested AP1000-specific analyses and stated that the information provided "should include a comparison of event timing, fraction of core melted, hydrogen generation rates and quantities, mass and superheat characteristics of debris relocating into the lower plenum, and fission product release histories for representative sequences in each accident class." This has not been provided.

720.043 The RAI noted that time windows available for operator actions in AP1000 are shorter than for AP600 and requested that W provide an assessment of the shorter times on human error probabilities and containment performance. The response addressed these impacts for 1 operator action, but 3 additional actions have shorter times in AP1000 and were not addressed.

720.046 The RAI questioned the completeness of the containment isolation fault tree success criteria tables in Chapter 24. The response explained why some of the valves are not included, but certain disparities still exist. Specifically, the following valves appear to be modelled for "CI" but are not listed in Table 24-8: V058B, V074A, V075A, AOV250A. Startup feedwater penetration check valves 256A and B are mentioned in the response, but are not shown in any of the containment isolation valve tables.

720.048 The RAI requested that W provide AP1000-specific assessments for each of the alternate debris configurations identified in INEEL's review of external reactor vessel cooling for AP600. W did not provide these assessments in their response, but appears to have performed such analyses based on information they presented during a 1/24/2003 meeting with ACRS.

720.050 The RAI requested that W either: (1) establish the applicability of the ULPU Configuration III test results to AP1000, or (2) develop AP1000-specific test data based on the prototypical insulation and flow conditions for AP1000. This was not addressed in the response.

720.053 The RAI noted that events with core damage could result in higher containment pressures than the sequence on which the probability value for node "IF" is based. The response provided qualitative arguments regarding sequence selection. Request that W provide pressure histories for core melt sequences representative of those evaluated at node "IF" to confirm.

720.055 The RAI requested a deterministic assessment of DCH pressure loads based on the methodology developed as part of DCH issue resolution. The response repeated the same qualitative arguments contained in PRA, Appendix B.

720.056 The RAI noted a number of inconsistencies between the offsite consequence estimates for AP1000 and AP600. AP1000-specific results have now been provided and the noted inconsistencies have been eliminated. However, two new inconsistencies are noted -- the dose for release category "CI" is identical to the AP600 value, and the dose for release category "CFE" is less than the AP600 value. The reasons need to be explained.

720.058 The RAI requested that W provide an assessment of the impact on basemat melt-through times and containment pressure (for both limestone and basaltic concretes) assuming that oxide/metallic separation does not occur, in order to confirm their conclusion regarding basemat failure and the adequacy of the sump curb design. In their response, W indicated that the sump curb height will be increased, but they did not provide the requested assessment.

720.060 The evaluation of SAMDAs was omitted from the PRA/DCD and submitted in response to this RAI. The evaluation does not address a number of items called out in the RAI and has several additional deficiencies, as summarized below:

- the cost benefit methodology appears to be based on an outdated guidance document (NUREG/CR-3568, 1983). The current guidance for regulatory analysis contained in NUREG/BR-0184 (1997) and NUREG/BR-0058 (2000) should be applied.
- replacement power costs were omitted. These averted onsite costs need to be included consistent with SECY-99-169.
- the CDF and population dose values used in the evaluation only reflect internal events. The contribution to CDF and population dose from shutdown and fire events should also be included.
- the RAI requested an explanation of how insights from the AP1000-specific PRA and supporting risk analyses for external and shutdown events, including importance analyses and cutset screening, were used to identify potential plant improvements. This was not addressed in the response.
- the RAI requested justification that the potential improvements identified through a systematic process (as suggested above) are included within the set of 15 SAMDAs identified in Appendix 1B of the AP1000 DCD. This was not addressed in the response.

RAIs 720.82, 85, 88/89, 92, 95, 96 - need clarification (see below follow-up questions)

RAI 720.082:

1. Please clarify the following aspects of sequence #20 (1ATRA-17) of the AP1000 PRA. If the cutsets comprising this sequence include substantial relative contributions with different characteristics regarding these points, also specify approximate frequency contributions.

a. Although IVR by means of lower head cooling cannot be credited for this high-pressure sequence, what is the status of cavity flooding from the IRWST?

b. Are the gutter drain valves assumed to close successfully? (i.e., is condensate from the containment directed to the containment sump or to the IRWST in this scenario?)

RAI 720.085:

The AP600 in-vessel steam explosion analysis that was cited in support of the AP1000 neglects the possibility of initially small FCIs (with little energetic potential) being a driver for larger melt crucible failures that would increase the melt pour rate. Please elaborate on how these events were considered or bounded for RPV survival in-vessel?

RAI 720.088-089:

1. A detailed description of the finite-difference model that has been used to perform calculations to support the side failure and melt relocation arguments. This should include the description of the model assumptions, pedigree, and their experimental basis.

2. A demonstration of its technical position by considering a wider range of phenomenological uncertainties including the effects of other debris relocation alternatives, the metallic layer depths, and the melt pool stratification and layering, on the AP1000 lower head integrity. In any reanalysis, please consider the uncertainties associated with the measured critical heat flux on the outside surface of the AP1000 lower head.

RAI 720.092:

The Westinghouse design criteria do not address how many igniters should be placed within each AP1000 compartment. The issue of igniter spacing has not been addressed for the AP600 or AP1000 plants. Please provide the technical basis for the numbers and the placement of igniters in AP1000 containment.

RAI 720.095:

1.What are the mixture compositions within the AP1000 containment for a representative accident with 100% active cladding reaction throughout the entire sequence, including times beyond the intermediate time frame?

2.What is the probability of DDT for mixture compositions beyond the intermediate time frame and when the entire containment is treated as an individual room for the purposes of the global burn?

RAI 720.096

Please provide a detailed sample calculation for the problem of solving the AICC pressure (equations shown in Section 41.9.2 (Revision 0)). It is presumed that this is the procedure used to produce the values in Table 41-4 and the basis for the values reported in Section 41.11. For example, using Equation 41-2 and the values given below the equation, it is not possible to obtain the same values for gas masses shown in Table 41-4. Furthermore, Equation 41-6 lists four gas constituents yet Table 41-4 lists five. If one uses the values provided in Section 41.9.2, one would get estimates of the AICC pressure that exceed the ASME service level C stress intensity limit of 91 psig.