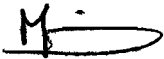




UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C. 20555-0001

October 10, 2003

MEMORANDUM TO: ACRS Members

FROM: Med El-Zeftawy, Senior Staff Engineer 
ACRS

SUBJECT: CERTIFIED MINUTES OF THE ACRS SUBCOMMITTEE MEETING ON
FUTURE PLANT DESIGNS - JULY 17-18, 2003, MONROEVILLE, PA

The minutes of the subject meeting, issued on August 22, 2003, have been certified as the official record of the proceedings of that meeting. A copy of the certified minutes is attached.

Attachment: As stated

cc: J. Larkins
S. Bahadur
R. Savio
H. Larson
M. Kelton

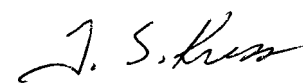
September 11, 2003

MEMORANDUM TO: M. El-Zeftawy, Senior Staff Engineer
ACRS

FROM: Thomas S. Kress, Chairman
Future Plant Designs Subcommittee

SUBJECT: CERTIFICATION OF THE MINUTES OF THE MEETING OF THE
ACRS SUBCOMMITTEE ON FUTURE PLANT DESIGNS , JULY
17-18, 2003. MONROEVILLE, PA.

I do hereby certify that, to the best of my knowledge and belief, the minutes of the subject meeting held on July 17-18, 2003, are an accurate record of the proceedings for that meeting.

 9/11/03

Thomas S. Kress Date
Subcommittee Chairman



UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C. 20555-0001

*Advanced Pressurized
Water Reactor*

August 22, 2003

MEMORANDUM TO: Dr. Thomas S. Kress, Chairman
Future Plant Designs Subcommittee

FROM: Med El-Zeftawy, Senior Staff Engineer *M.E.*
ACRS

SUBJECT: WORKING COPY OF THE MINUTES OF THE ACRS
SUBCOMMITTEE ON FUTURE PLANT DESIGNS, JULY 17-18,
2003-- MONROEVILLE, PA

A working copy of the minutes for the subject meeting is attached for your review. Please review and comment on them at your earliest convenience. Also attached is a certification sheet.

Attachment: Minutes.

Cc: Future Plant Designs Subcommittee Members
J. Larkins
S. Bahadur
H. Larson

*Med:
But minutes I have seen
can*

CERTIFIED

9/11/03

By Thomas S. Kress

Issued 8/22/03

**ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
MEETING OF THE ACRS SUBCOMMITTEE ON
FUTURE PLANT DESIGNS
JULY 17-18, 2003
MONROEVILLE, PA**

INTRODUCTION

The ACRS Subcommittee on Future Plant Designs held a meeting on July 17-18, 2003, in Monroeville, PA at the Westinghouse facilities. The purpose of the meeting was to discuss with representatives of the Westinghouse and the NRC staff issues related to the design certification of the AP1000 design. The meeting was open to public attendance. Med El-Zeftawy was the Designated Federal Official for this meeting. Request for time to make oral statements was received from Dr. Susan G. Sterrett, Assistant Professor of Philosophy- Duke University/NC. The meeting was convened by the Subcommittee Chairman at 1:00 p.m on July 17, 2003, recessed at 6:22 p.m, and reconvened at 8:30 a.m and adjourned at 12:50 p.m. on July 18, 2003.

ATTENDEES

ACRS

T. Kress, Subcommittee Chairman
F. P. Ford, Member
G. Leitch, Member
V. Ransom, Member

J. Sieber, Member
G. Wallis, Member
M. El-Zeftawy, Senior Staff (DFO)
R. Caruso, Senior Staff

Principal NRC Speakers

J. Segala, NRR
J. Starefos, NRR
S. Basu, RES

Principal Industry Speakers

E. Cummins, Westinghouse
M. Corletti, Westinghouse
T. Schulz, Westinghouse
S. Sancaktar, Westinghouse
T. Hayes, Westinghouse

D. Frederick, CONAX FL CO.
J. Li, Polestar App. Tech, Inc.
J. Scobel, Westinghouse
H. Esmaili, ERI
M. Khatib-Rahbar, ERI

M. Zavisca, ERI
S. Sterrett, Duke University
R. Fuld, Westinghouse

R. Orr, Westinghouse
W. Bamford, Westinghouse

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

Dr. Thomas S. Kress, Chairman of the ACRS Subcommittee on Future Plant Designs convened the meeting at 1:00 p.m on July 17, 2003. Dr. Kress stated that the purpose of this meeting was to review and discuss issues associated with the AP1000 design certification. Such issues include instrumentation and control, man-machine interface, design acceptance criteria, human factors, squib valve reliability, and the NRC draft safety evaluation report open items regarding the design reviews. The Subcommittee will gather information, analyze relevant issues and facts, and formulate proposed positions and actions as appropriate, for deliberation for the full Committee regarding the AP1000 design.

DISCUSSION OF AGENDA ITEMS

Mr. Terry Schulz, Westinghouse, briefed the Subcommittee regarding the AP1000 passive core cooling system (PXS) and the automatic depressurization system (ADS) design features. The ADS valves are part of the reactor coolant system (RCS) and interface with the PXS. The ADS is divided into two groups and four depressurization stages, with a total of 20 valves. These stages connect to the RCS at different locations. The first, second, and third stage valves are included as part of the pressurizer safety and relief valve (PSARV) module and are connected to nozzles on top of the pressurizer. The fourth stage valves connect to the hot leg of each reactor coolant loop.

Opening of the ADS valves is required for the PXS to function as required to provide emergency core cooling following postulated accidents. The first stage ADS valves are two motor operated (4 in.) valves in series. The second and third stage ADS valves each have two motor operated (8 in.) valves in series. The fourth stage ADS valves are (14 in.) Squib valves arranged in series, with normally open dc-powered motor operated valves. The fourth stage ADS valves are interlocked so that they cannot be opened until the RCS pressure has been substantially reduced.

Squib valves were selected for ADS stage 4. Mr. Schulz stated that these valves are very reliable to open on demand (better than air-operated valves AOVs, and motor-operated valves MOVs). The squib valves are diverse from ADS stage 1/2/3 MOVs; have a very low chance of inadvertent opening; zero leakage during normal operation; simplified in-service testing and inspection; and reduced capital cost. The squib valves are supported by U.S. utilities.

Mr. Schulz indicated that the ADS 4 in-service tests are in accordance with ASME. This include removing 20% of charges every 2 years; fire charges in test fixture to demonstrate that valve would have operated; and the use of staggered testing. In addition, verifying the continuity of

ADS 4 circuit after disconnecting and re-connecting wires; and verifying the valve position sensor every 2 years. The ADS 4 in-service inspections are in accordance with ASME. This include every ten years measuring shear cap dimensions to ensure no thinning; performing dye penetrant test to ensure no cracking; and the use of staggered testing.

Mr. Tom Hayes, Westinghouse, briefed the Subcommittee regarding the AP1000 squib valve actuation circuits. Each ADS 4 squib valve can be actuated by either of two protection and monitoring system (PMS) , auto/manual; and diverse actuation system (DAS), manual. The squib valve actuation circuits are energize-to-actuate. Each squib valve controller has two inputs—"Arm" and "Fire". To actuate the squib valve, the "Arm" circuit must be energized while the "Fire" circuit remains un-energized. This permits a capacitor to charge. Full charge is indicated and alarmed to the operator. The "Arm" circuit must then be de-energized followed by energizing the "Fire" circuit. Actuation of either "Arm" or "Fire" alone will not result in valve actuation. Simultaneous actuation of both "Arm" and "Fire" circuits will not result in valve actuation.

Failures 'upstream' or 'downstream' of the squib valve controller have a low probability of causing a spurious actuation. Failures of the squib valve controller have a low probability of causing a spurious actuation. Under normal conditions the squib valve controller has no power and no stored energy. No credible failure of the squib valve controller will result in valve actuation. For fire induced spurious actuation, there are two coordinated 'Arm' and 'Fire' circuits implemented in different cabinets. A fire is assumed to start in one cabinet, and before it can spread to a second cabinet the operators are required to remove power. Downstream of the valve controller there are no active components, and no adjacent cables with sufficient energy for actuation.

Mr. Dan Frederick, CONAX Florida Corporation, briefed the Subcommittee regarding the development of the squib valve. Based upon Pyronetics 2 inch ID flow passage valve, General Electric concluded that Pyronetics was capable of designing, manufacturing and testing squib valve (7 inch ID). Westinghouse AP600 ADS stage 4 is the same as GE valve. Westinghouse AP1000 ADS stage 4 is simply scaled-up (14 inch ID) AP600/GE SBWR valve. The GE SBWR ADS valve testing included seismic and other dynamic loads that evaluated vibration testing, and actuation. For the AP1000 ADS 4 valve, the planned tests include charge sizing, inspection, hydrostatic and leak testing, vibration, and actuation. The design shear section is standard pyrovalve design feature that meets ASME code Section III, Class I. 316 L material was selected for the shear section. This material is not subject to stress corrosion cracking like numerous other materials under high pressure and temperature, and is designed for ease of removal for inspection. Mr. Frederick stated that squib valves have high inherent reliability (0.9998169 at 90% confidence). The reliability for smaller valve is applicable for larger valves.

Mr. Selim Sancaktar, Westinghouse, briefed the Subcommittee regarding the reliability of large squib valves in the AP1000 design. There are two types of failures of the ADS 4 squib valves that were modeled. These are failure to open after receiving a signal to open ($5.8E-04$ /demand used), and opens spuriously ($5.4E-05$ /year is used leading to large LOCA). Three different sources of failure probability were used to establish the AP1000 squib valve reliability in the fail to open mode. These are the ALWR Utility Requirements Document (URD) indicating a failure to operate probability of $3E-03$ per demand; Sandia Laboratories failure probabilities of $2.0E-04$

and $3.2\text{E-}04$ per demand; and a geometric mean of the URD and Sandia for a failure probability of $5.8\text{E-}04$ per demand. This is the value used in the AP1000. Mr. Sancaktar stated that the valve reliability information provided by the squib valve vendor CONAX indicated that the squib valves have high inherent reliability and the reliability for smaller valves is applicable to larger valves. In addition, no failures associated with shear section cracking under constant high pressure and temperature are expected.

For spurious opening failure, the dominate cause on ADS 4 valve is considered a spurious signal. Structural failure of the valve is deemed to be much less likely and is not estimated in the AP1000 PRA. The failure frequency of one or more ADS stage 4 squib valves opening due to spurious signal generation is estimated to be $5.4\text{E-}05/\text{year}$. The fact that hardware failure of the valve leading to gross leakage is considered small compared to this value implies that the contribution of failures from this failure mode is deemed to be less than $5\text{E-}06/\text{year}$, or $5.7\text{E-}10/\text{hour}$ for four valves. This translates to $1.4\text{E-}10/\text{hour}$ failure rate for a single valve.

Westinghouse used a similar estimate of structural valve failure as a pipe segment. In AP1000 significant leakage from a pipe segment is assigned a failure rate of $8.5\text{E-}09/\text{hour}$. In general, Mr. Sancaktar indicated that if the squib valve failure probability to open is doubled, then the plant CDF for internal events at power goes from $2.41\text{E-}07/\text{year}$ to $2.77\text{E-}07/\text{year}$ -- a 15% increase; and if the spurious opening of the ADS failure probability is doubled, the plant CDF will increase by 12.3%.

Dr. Jun Li, Polestar Applied Technology, Inc., described the post-LOCA design basis aerosol deposition in AP1000 containment. As part of the AP600 design certification, Polestar performed a QA calculation of radiological design basis fission product aerosol removal rates (λ) in containment by natural processes. Similar to the AP600, the AP1000 containment has a large steel shell cooled on the outside, leading to higher heat transfer rate and higher natural aerosol removal rate for fission product aerosols than would exist from sedimentation alone. Since AP1000 and AP600 have a similar design, the calculation is a repetition of the AP600 calculation with AP1000 parameters and thermal hydraulics. The AP600 sensitivity study was also referenced to assess possible variation of AP1000 λ s. The AP1000 has, compared to AP600, 75% higher thermal power, 20% larger containment by volume, and 75% more aerosol mass.

Dr. Li indicated that the calculation procedure included selection of the AP1000 sequence that has relatively high probability and has timing that is similar to the NRC specified Regulatory Guide 1.183 timing for PWR fission product release; calculate containment thermal hydraulic conditions for selected sequence using MAAP4 code; and calculate containment aerosol removal rates for MAAP4 thermal hydraulic conditions and aerosol assumptions using the Polestar QA code STARNAUA. The STARNAUA code has been documented and benchmarked against experiment. The expected results, compared to AP600, higher diffusion and thermophoresis due to higher heat transfer to containment shell, similar sedimentation due to similar concentration and well-mixed assumption (conservative), and higher containment λ . Average λ is 1.1 per hour.

Mr. James Scobel, Westinghouse, described the thermal hydraulic conditions for the AP1000 λ calculation. The severe accident environment was generated with MAAP4 with

conservative conditions for lambda calculation. The dominant core damage sequence from PRA included break in a direct vessel injection line (fails one train of passive injection), full reactor coolant system depressurization, failure of gravity injection, successful cavity flooding and in-vessel retention of core debris, vessel reflooding through break, and hydrogen igniters.

Mr. H. Esmaili and Mr. M. Khatib-Rahbar, Energy Research, Inc.(ERI), described the in-vessel retention (IVR) and ex-vessel fuel coolant interaction (FCI) for the AP1000 design. The objectives are the assessment of IVR to determine the likelihood and location of vessel breach, and the formulation of FCI scenarios and quantification of FCI impulse loads using an approach similar to that used by ERI/NRC for the AP600 design. Mr. Khatib stated that ERI has considered two bounding configurations for the IVR. Melt configuration I is molten ceramic pool with an overlaying molten light metallic layer. Melt configuration II is molten ceramic pool between a bottom heavy metallic layer and a top light metallic layer. The ERI model only accounts for thermal interactions. Chemical reactions with vessel wall was not considered (i.e., in the absence of a crust, the potential for chemical reactions and dissolution could be important). The solution method is based on non-linear Newton-Raphson method that allows for temperature dependence of viscosity. The material properties are based on the INEEL report used for the AP600 design. Results indicated that decay heat partitioning is based on the amount of uranium in the bottom layer. The ERI results indicated that there is a potential for the side failure of the lower head as a result of the focusing effect for melt configuration I (estimated failure likelihood ranges from 0.04 to 0.30). Failure of the bottom head at the bottom location is not likely based on the results of ERI melt configuration II parametric calculations.

For the AP1000 ex-vessel FCI, ERI applied initial and boundary conditions based on plant-specific MELCOR calculations and IVR analysis. ERI assumed melt initial conditions in the lower plenum at vessel breach, cavity condition at vessel breach (deep water, fully submerged lower head), side failure, and containment pressure and temperature at time of vessel breach. ERI claims that the ex-vessel FCI loads for side failure in the AP1000 remain lower than those for the AP600. A sensitivity calculation assuming the bottom failure of the lower head results in a lower impulse loads as compared to cases involving side failure.

Mr. M. Zavisca and Mr. Khatib-Rahbar outlined ERI's MELCOR analysis of selected severe accident sequences. The objectives are to derive initial and boundary conditions for analysis of IVR and ex-vessel FCI issues; calculate extent and consequences of molten core-concrete interactions (MCCI); provide information regarding potential containment challenges from hydrogen combustion; and verify expected changes in overall accident progression relative to AP600. The MELCOR model consisted of 44 control volumes and 75 flow paths. The core has 10 axial and 5 radial nodes. All safety systems relevant to severe accidents are modeled, including core make-up tanks (CMTs), passive core cooling system (PCCS), ADS, passive residual heat removal heat exchanger, and cavity flooding. Melting of the core shroud was not included. Mixing of the molten debris in lower plenum was modeled with enhanced conductivity.

For the MCCI, scenarios included limestone, basalt, and limestone/sand aggregate concrete types, flooded and dry cavity. ERI concluded that global deflagrations do not challenge the AP1000 containment integrity. PCCS is successful in preventing over pressure with or without MCCI. Concrete basement penetration is not expected within 3 days. The accident timing and containment response is generally similar to AP600, scaled to changes in power, core mass, and containment volume.

Mr. Richard Orr, Westinghouse, outlined the AP1000 seismic and structural design. Some of the AP1000 structural changes from the AP600 design included increased capacity of the PCS water storage tank volume from 540,000 gallons to 800,000 gallons, PCS air inlets reconfigured to 12 feet by 6.5 feet, containment vessel is raised by 25 feet and 6 inches, polar crane is raised with increased capacity, RCS equipment increased in size, steam generator and pressurizer compartment walls are raised, and fuel pit floor elevations lowered by 18.5 feet. The containment vessel general outline is 130 feet in diameter, and 215 feet 4 inches in height, ASME III design code, design pressure of 59 psig, design temperature of 300 degrees F, and design external pressure of 2.9 psid.

For the seismic design basis, Westinghouse selected 0.30 g for safe shutdown earthquake (SSE) at foundation level and hard rock foundation sites. The SSE design response spectra was developed from Regulatory Guide (RG) 1.60 with enrichment in the frequency range from 15 to 33 Hz. Specific percentage of critical damping values in the seismic analysis of AP600 that conform to the guidelines of RG 1.61 and industry practice was used.

For the nuclear island seismic models, finite element shell models of buildings for static analyses and for generation of simplified stick models were developed. Simplified finite stick models of buildings for time history dynamic analyses were developed for the auxiliary and shield building, containment internal structures, containment vessel and polar crane, and reactor coolant loop. Concrete properties accounted for limited amount of cracking. Static analyses of shell models were developed to obtain member forces in walls and floors.

Currently, Westinghouse agreed on level of detail comparable to AP600 for seismic analyses and structural design. There are 5 open items on COL information for Geotech, 7 open items on seismic analysis, and 14 items on structural design. Design calculations for critical sections are being reconciled to the results of the updated seismic analyses. High confidence low probability of failure (HCLPF) values are calculated for structures, systems, and components (e.g., containment vessel, primary components, mechanical equipment, valves, and electrical equipment).

Mr. Orr indicated that seismic margins (HCLPF) are generally slightly lower than the AP600 plant since the design is similar and the AP1000 seismic response is higher. All HCLPF values are above the review level earthquake of 0.5 g. Some of the COL actions regarding seismic margins include confirming the use of seismically robust electro-mechanical relays, and conforming that the as-built plant conforms to the design. Westinghouse is in the process of responding to the NRC current open items.

Ms. Joelle Starefos, Project Manager/NRR, briefed the Subcommittee regarding the AP1000 draft safety evaluation report (DSER) open items. The NRC staff issued the DSER on June 16, 2003, with 174 (as compared to 1300 for AP600) open items. Currently, the staff is engaged with Westinghouse representatives on 82 open items, resolved 5 open items, and is satisfied with Westinghouse's response to 31 open items (confirmatory). The DSER open items include testing and computer code evaluation, initial test program, security, leak-before-break, and wind and tornado loading. The current estimate for the NRC staff to issue the final safety evaluation report (FSER) is September 2004.

Under the broad-scope revision to 10 CFR Part 50, Appendix A, GDC 4, the NRC allows the use

of analyses to exclude from the design basis consideration of the dynamic effects of pipe ruptures in nuclear power plants, provided it is demonstrated that the probability of pipe rupture is extremely low under conditions consistent with the design basis for the piping. The demonstration of low probability of pipe rupture utilizes deterministic fracture mechanics and leakage analyses that evaluate the stability of a postulated flaw and the ability to detect leakage before the flaw could grow and break the pipe (leak-before-break, LBB). Westinghouse has identified 26 AP1000 piping systems or subsystems for LBB application.

The NRC staff currently has an open item regarding LBB that deals with the Alloy 690/52/152 susceptibility to primary water stress corrosion cracking (PWSCC). The staff is discussing with Westinghouse representatives the need for inspections and sensitivity study margins to provide sufficient defense-in-depth. In addition, the staff is working to determine if the appropriate bounding limits are established using preliminary analysis results during the design certification phase.

Other open items include sump performance, structural/seismic design, liquid entrainment, and PRA related items. For the sump performance, the staff is concerned regarding debris loading of IRWST screens, recirculation screens, and debris through reactor coolant system break. The staff audited the associated Westinghouse calculation and identified assumptions of debris size, density, and porosity that are not consistent with industry practices. For the structural/seismic

open items, the staff identified 38 items, many of which would require audit of specific Westinghouse calculations. Numerous discussions were held regarding the liquid entrainment, long term cooling, and boron precipitation issues and the staff 's review will continue with independent analyses and review of Westinghouse submittals.

Mr. Mike Corletti, Westinghouse, discussed some of the DSER open items and the planned resolution paths. The DSER open items already discussed with ACRS are thermal-hydraulic issues, PRA, and seismic and structural design. Other open items that need to be discussed include LBB, sump performance issues, security, control room X/Q, 10 CFR 50.44, and ITAAC. The security is largely the responsibility of the combined license (COL) applicant and will be handled separately through compliance to design related requirements contained in the revised design basis threat and interim compensatory measures.

For the Ap1000 control room X/Q was performed using the same methodology as applied to AP600 design. The DSER requests a comparison of the AP1000 calculation to the recently (June 2003) RG 1.193. Westinghouse is preparing a compliance summary to RG 1.193. For 10 CFR 50.44, standards for combustible gas control system in LWR, the NRC is considering a revision in its regulation (e.g., relaxation of the requirements regarding Hydrogen recombiners). For the AP1000 design, the hydrogen recombiners are not required based on new regulation, however, they are still provided for defense-in-depth.

Dr. Susan G. Sterrett, Assistant Professor of Philosophy- Duke University/Durham, North Carolina, expressed concerns regarding the NRC's review of the AP1000 fluid systems design and the QA procedures [see the attached two letters: 1) AP1000 Fluid Systems Design & QA Procedures, dated July 30, 2003; and 2) Heat of Solar Radiation and AP1000 Ultimate Heat Sink, dated July 31, 2003].

Mr. Warren Bamford, Westinghouse, briefed the Subcommittee regarding the LBB issue. He stated that the AP1000 will use piping design acceptance criteria (DAC) in lieu of detailed piping design and analysis. This is the same approach used for the ABWR and System 80+ designs. In addition, the AP1000 piping configuration is based largely on the AP600 design. The final piping design and analysis will be completed during the COL stage and will be verified by ITAAC during construction.

The DESR contains two open items related to LBB. The first open item requests Westinghouse to include COL applicant commitment to implement inspection plans, evaluation criteria, and other types of measures imposed on or adopted by operating PWRs with currently approved LBB applications as part of the resolution of concerns regarding the potential for PWSCC in those units. The second open item requests sensitivity studies to be performed to address uncertainties related to PWSCC and the possible impact on LBB piping.

Mr. Bamford indicated that in view of the continuing occurrence of PWSCC of Alloy 600, and its associated welds Alloys 82 and 182, the decision was taken to preclude use of these materials in the AP1000 design. The materials selected for these applications are Alloys 690, 52 and 152. Alloys 52 and 152 welds have been shown to exhibit excellent resistance to PWSCC, both in the laboratory and field experience. In addition, the completed stress analyses for the AP600 demonstrate the feasibility that the AP1000 piping systems can be designed to meet bounding analysis curves.

Mr. Terry Schulz, Westinghouse, discussed the AP1000 containment recirculation screen performance. He stated that the AP1000 has robust containment recirc. Screen design. There are long times (2 to 4 hours) before recirculation starts allowing time for debris settling. The AP1000 design has a deep floodup levels, tall screens (13 feet vs. 6 for typical PWRs), bottom of screen is well above floor, and lower recirculation flow rates (85% less than current plants).

Protective plates are provided over screens to allow debris to settle out before reaching screens. No fibrous debris is generated by LOCA conditions due to the use of metal reflective insulation used in LOCA blowdown damage zone. Coating debris may be generated, but will settle out due to high density nonsafety coatings specified inside containment. Westinghouse assumes a conservative total amount of resident debris of 500 lb (half fiber and half particle), with all resident debris to be transported to screens.

Mr. Hayes and Mr. Fuld, Westinghouse, in preparation for tour of plant automation headquarters, described the process for the instrumentation and control (I&C) system and the human factors engineering for the AP1000 design. Mr. Hayes stated that for the I&C and human factors, Westinghouse plans to use the design acceptance criteria (DAC) process. Westinghouse requests only the certification of the functional requirements on the design when it is completed. The COL applicant will be the one to fulfill the obligation to prove that the final design meets those requirements. The automation products are not just used for the AP600/AP1000 designs. They are also used in other plants.

SUBCOMMITTEE COMMENTS, CONCERNS, AND FOLLOW-UP

The Subcommittee Members raised the following significant points during their discussion with Westinghouse representatives and the NRC staff:

- In the thermal-hydraulic area, more discussion is in progress between the NRC staff and Westinghouse regarding entrainment, level swell, and boron precipitation.
- In the severe accident environment generated with the MAAP 4 code, more clarification is needed for the methodology used for the AP1000 environment for Lambda calculation.
- Clarification of the dominant core damage sequence from PRA.
- The vessel retention issue-- where and how it breaks through the vessel and how is that relates to the FCI?
- The squib valve reliability and the lack of actual valve testing.
- The performance of the sump screens and the acceptance of the AP1000 design to tolerate resident debris on screens.

SUBCOMMITTEE's ACTION

The Subcommittee members determined that a presentation by Westinghouse representatives and the NRC staff during the ACRS full Committee meeting in September 2003 would be scheduled.

BACKGROUND MATERIALS PROVIDED TO THE SUBCOMMITTEE PRIOR TO THIS MEETING

1. Subcommittee status report, including agenda.
2. Report dated March 14, 2002, from George A. Apostolakis, Chairman, ACRS, to Richard A. Meserve, Chairman, NRC, Subject: Phase 2 Pre-Application Review for AP1000 Passive Plant Design.
3. Letter dated March 25, 2002, from James E. Lyons, NRR, to W.E. Cummins, Westinghouse, Subject: Applicability of AP600 Standard Plant Design Analysis Codes, Test Program and Exemptions to the AP1000 Standard Plant Design.
4. Draft Safety Evaluation Report (DSER), dated June 16, 2003 (CD Format).
5. DSER Open Items Sections.

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-1323 Rhode Island Avenue, NW, Washington, DC 20005 (202) 234-4433.

NATIONAL FOUNDATION ON THE ARTS AND HUMANITIES**National Council on the Humanities; Meeting**

Pursuant to the provisions of the Federal Advisory Committee Act (Public L. 92-463, as amended) notice is hereby given the National Council on the Humanities will meet in Washington, DC on July 24-25, 2003.

The purpose of the meeting is to advise the Chairman of the National Endowment for the Humanities with respect to policies, programs, and procedures for carrying out his functions, and to review applications for financial support from and gifts offered to the Endowment and to make recommendations thereon to the Chairman.

The meeting will be held in the Old Post Office Building, 1100 Pennsylvania Avenue, NW., Washington, DC. A portion of the morning and afternoon sessions on July 24-25, 2003, will not be open to the public pursuant to subsections (c)(4), (c)(6) and (c)(9)(B) of section 552b of Title 5, United States Code because the Council will consider information that may disclose: trade secrets and commercial or financial information obtained from a person and privileged or confidential; information of a personal nature the disclosure of which would constitute a clearly unwarranted invasion of personal privacy; and information the premature disclosure of which would be likely to significantly frustrate implementation of proposed agency action. I have made this determination under the authority granted me by the Chairman's Delegation of Authority dated July 19, 1993.

The agenda for the session on July 24, 2003 will be as follows:

Committee Meetings

(Open to the Public)

Policy Discussion

9-10:30 a.m.

Education Programs—Room M-07
Public Programs—Room 420
Federal/State Partnership and
Challenge Grants—Room 507

(Closed to the Public)

Discussion of Specific Grant
Applications and Programs Before the
Council

10:30 a.m. until adjourned

Education Programs—Room M-07
Public Programs—Room 420
Federal/State Partnership and
Challenge Grants—Room 507

2-3:30 p.m.

National Humanities Medals—Room
527

The morning session on July 25, 2003 will convene at 9 a.m., in the 1st Floor Council Room M-09, and will be open to the public, as set out below. The agenda for the morning session will be as follows:

A. Minutes of the Previous Meeting**B. Reports**

1. Introductory Remarks
2. Staff Report
3. Congressional Report
4. Reports on Policy and General Matters
 - a. Overview
 - b. Education Programs
 - c. Public Programs
 - d. Challenge Grants
 - e. Federal/State Partnership
 - f. National Humanities Medals

The remainder of the proposed meeting will be given to the consideration of specific applications and closed to the public for the reasons stated above.

Further information about this meeting can be obtained from Mr. Daniel C. Schneider, Advisory Committee Management Officer, National Endowment for the Humanities, 1100 Pennsylvania Avenue, NW., Washington, DC 20506, or by calling (202) 606-8322, TDD (202) 606-8282. Advance notice of any special needs or accommodations is appreciated.

Daniel C. Schneider,
Advisory Committee Management Officer.

[FR Doc. 03-17165 Filed 7-7-03; 8:45 am]

BILLING CODE 7536-01-P

NATIONAL TRANSPORTATION SAFETY BOARD**Sunshine Act Meetings; Agenda**

TIME AND DATE: 1 p.m., Tuesday, July 15, 2003.

PLACE: NTSB Conference Center, 429 L'Enfant Plaza SW., Washington, DC 20594.

STATUS: The one item is Open to the Public.

MATTERS TO BE CONSIDERED:

7567 Highway Accident Report—15 Passenger Van Single-Vehicle Rollover Accidents, Henrietta, Texas, May 8, 2001, and Randleman, North Carolina, July 1, 2001.

News Media Contact: Telephone: (202) 314-6100. Individuals requesting specific accommodations should contact Ms. Carolyn Dargan at (202) 314-6305 by Friday, July 11, 2003.

FOR FURTHER INFORMATION CONTACT:
Vicky D'Onofrio, (202) 314-6410.

Dated: July 3, 2003.

Vicky D'Onofrio,

Federal Register Liaison Officer.

[FR Doc. 03-17352 Filed 7-3-03; 2:11 pm]

BILLING CODE 7533-01-M

NUCLEAR REGULATORY COMMISSION**Advisory Committee on Reactor Safeguards; Subcommittee Meeting on Thermal-Hydraulic Phenomena; Notice of Meeting**

The ACRS Subcommittee on Thermal-Hydraulic Phenomena will hold a closed meeting on July 16-17, 2003, at Westinghouse Electric Company, 4350 Northern Pike, Monroeville, Pennsylvania.

The entire meeting will be closed to public attendance to discuss Westinghouse Electric Company LLC proprietary information per 5 U.S.C. 552b(c)(4).

The agenda for the subject meeting shall be as follows:

Wednesday, July 16, 2003—8:30 a.m.

until the conclusion of business.

Thursday, July 17, 2003—8:30 a.m. until 12 noon.

The purpose of this meeting is to review the thermal-hydraulic aspects of the AP1000 design. The Subcommittee will hear presentations by and hold discussions with representatives of the NRC staff, Westinghouse Electric Company LLC, and other interested persons regarding this matter. The Subcommittee will gather information, analyze relevant issues and facts, and formulate proposed positions and actions, as appropriate, for deliberation by the full Committee.

For Further Information Contact: Mr. Ralph Caruso, Designated Federal Official (Telephone: 301-415-8065) between 7:30 a.m. and 4:15 p.m. (e.t.).

Dated: July 1, 2003.

Sher Bahadur,

Associate Director for Technical Support, ACRS/ACNW.

[FR Doc. 03-17185 Filed 7-7-03; 8:45 am]

BILLING CODE 7590-01-P

NUCLEAR REGULATORY COMMISSION*** Advisory Committee on Reactor Safeguards Subcommittee Meeting on Future Plant Designs; Notice of Meeting**

The ACRS Subcommittee on Future Plant Designs will hold a meeting on July 17-18, 2003, at Westinghouse

Draft of Remarks by Dr. S. G. Sterrett

501st ACRS meeting, April 11th, 2003

Rockville, MD

I'm Susan G. Sterrett. I am currently a professor at Duke University in Durham, North Carolina. I should perhaps mention that, prior to my academic career, I worked as a design engineer in the commercial nuclear power plant industry, ^{including as a fluid system designer on the AP60 in the mid-nineties.} I am making these remarks as a member of the public, unaffiliated with any organization.

I'm here today because I have some questions about the NRC's review of the AP1000. Put briefly, my question is whether the NRC verifies or asks for proof that the system parameters reported in the AP1000 design certification application (and used in the analyses) are actually justified by a detailed design, as opposed to the AP1000 system designs being at the stage of conceptual system design or justified only by preliminary equipment sizing calculations. I'd like a few minutes to explain the relevance and the significance of the question.

According to the rules under which the AP1000 is being licensed by the NRC, the level of design information required in a design certification application is, with a few explicit exceptions, the level of information that was required at the operating license stage under the previous two-step licensing process. I think this requirement makes sense, too, inasmuch as what the NRC is licensing in approving the AP1000 is an actual plant design that is certified to be constructed and operated.

In following some of the AP1000 licensing activities via the NRC's website, I have noticed that much is often made of the similarities between the AP1000 systems and the AP600 systems. This can be misleading: the performance of the various fluid

systems in the plant -- that is, the flows, temperatures, and pressures that obtain at various points within a system are affected by many kinds of differences in a plant design. As I am sure everyone here realizes:

-- Anytime a system flowrate changes, pressure drops in the system will change.

-- Likewise, anytime the pressure at some point in a system changes, flowrates in it or some other system can be affected.

--- Thus, even for those systems that are exactly the same physically speaking (i.e., same pipe size and layout) for the AP1000 as for the AP600, there is still the question of whether there are differences in the inlet or outlet pressures in a system or piece of equipment to which it connects. Different inlet or outlet pressures will result in differences in fluid system performance.

For example, suppose the main steam system pressure is different on the AP1000; then, on the AP1000, there would be a different driving head for lines connected to it than there was on the AP600. So, even if the system hardware and layout of a system connected to the main steam system, say, is exactly the same on the AP1000 as it was for the AP600, the resulting values of major fluid system parameters -- e.g., the mass and volume flowrates and the pressures that result -- could be quite different.

Obviously the effects on things like the flow capability of relief valve piping and valve arrangements would need to be looked at. Accommodating these changes could require resizing piping or control valves in order to achieve the flowrate claimed for the system.

I've given the main steam system as an example, but the general point holds for every system in the plant. To infer from the fact that the hardware and layout on an AP1000 system is exactly the same as on the AP600, to the conclusion that the performance is the same, is incorrect. The various AP1000 analyses now under review are only as valid as the assumptions made in them about the performance of the plant systems.

What does this point mean for the review of the AP1000 design, which makes frequent appeal to the certified AP600 design? In many aspects of the safety analyses, the NRC has been very alert to the differences between the AP1000 and the AP600. The point of my examples is that this awareness ought to be extended to plant fluid system performance, specifically, that some reassurances should be sought that the fluid system design details for all the plant systems have been properly attended to, and that, given that the level of detail required at this stage is supposed to be the same as that at the operating license stage, these should not be just preliminary sizing calculations. I worry about the complacency with which the AP600 design is referenced in justifying the AP1000 system designs.

The AP1000 is sometimes referred to as an uprating of the AP600 design. Of course this would be significantly larger than any uprating that the NRC has licensed so far, and of course it differs from most upratings in that there is no AP600 operating experience to draw upon. To the extent that thinking of the AP1000 as an uprating of the AP600 is appropriate, however, it would make sense to require that all the plant system reviews that would be required for an extended power uprating be performed for the AP1000. As there is now a draft review standard for extended power uprates that could be used to guide such a review of the AP1000 (RS-001, dated December 2002), this seems a natural thing to do. I wonder whether there has in fact been a review of this sort for the AP1000. So let me ask: has there?

that meet their stated functional requirements in terms of flowrates, pressures, and temperatures, even if the piping layout for the certified design may not be final in every detail.

In conclusion, I am asking whether the review of the AP1000 design has included ensuring that the design details upon which the analyses that the ACRS has been reviewing depend, have in fact been attended to. In particular, I think it is clear that L/D criteria should be provided at this stage for systems whose layout is to be finalized at a later date, and "proof-of-design" calculations be provided for those whose layout is determined at this stage. Otherwise, there is no assurance that the analyses you are reviewing so carefully and thoughtfully apply to the plant design you are certifying.

Thank you for listening.

Respectfully submitted,



Dr. Susan G. Sterrett

Duke University

Durham, NC 27708

sterrett@duke.edu

919-660-3054 (office & voicemail)

919-660-3050 (receptionist)

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AP1000 DSER Open Items



July 18, 2003

ACRS Future Plant Subcommittee Briefing

Joelle Starefos, Project Manager
Office of Nuclear Reactor Regulation

NRC Current Review Status

- Issued DSER on June 16, 2003, with 174 Open Items
- Working to Resolve Open Items
 - Engaged Westinghouse (W) on 82 Open Items
 - Resolved 5 Open Items
 - Satisfied with W response to 31 Open Items (Confirmatory)
- Schedule for Final Safety Analysis Report (FSER)
 - NRC Issue FSER: September 2004
 - Reassessing Schedule

DSER Open Items

174 Open
Items

■ (as compared
to over 1300
for AP600
DSER)

■ Chapter 1 (Introduction) - - - - -	3
■ Chapter 2 (Site Envelope Char) - - - - -	6
■ Chapter 3 (Structures, Comp., Equip.) - - -	30
■ Chapter 4 (Reactor) - - - - -	3
■ Chapter 5 (Reactor Coolant System) - - - -	3
■ Chapter 6 (Engineered Safety Features) - -	9
■ Chapter 7 (I & C) - - - - -	0
■ Chapter 8 (Electric Power Systems) - - - -	1
■ Chapter 9 (Auxiliary Systems) - - - - -	7
■ Chapter 10 (Steam and Power Conv.) - - - -	3
■ Chapter 11 (Radioactive Waste Man.) - - - -	0
■ Chapter 12 (Radiation Protection) - - - - -	0
■ Chapter 13 (Conduct of Ops) - - - - -	3
■ Chapter 14 (Verification Progs) - - - - -	43
■ Chapter 15 (Transient & Acc. Anal.) - - - - -	6
■ Chapter 16 (Technical Specs) - - - - -	3
■ Chapter 17 (Quality Assurance) - - - - -	5
■ Chapter 18 (Human Factors) - - - - -	7
■ Chapter 19 (Severe Accidents) - - - - -	36
■ Chapter 20 (Generic Issues) - - - - -	2
■ Chapter 21 (Testing & Comp Code Eval.) - -	4
■ Chapter 22 (RTNSS) - - - - -	0
■ Chapter 23 (Review by the ACRS) - - - - -	0
■ Chapter 24 (Conclusions) - - - - -	0

3

DSER Open Issues

- Supplemental DSER Sections
 - Chapter 21 – Testing and Computer Code Evaluation
 - Section 14.2 – Initial Test Program
 - Section 13.6 – Security
 - Section 3.6.3.4 – Leak-Before-Break
 - Section 3.3 – Wind and Tornado Loadings

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DSEER Open Issues (cont'd)

- *General* Open Items from DSEER Chapter 1
 - DSEER based upon Design Control Document (DCD) Revision 3...all revisions will be reviewed prior to FSER
 - Identification of Tier 2* information needs to be completed prior to FSER issuance
 - Identification and incorporation of combined license (COL) action items in FSER/DCD

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DSEER Open Issues (cont'd)

- Combined License (COL) Action Items
 - Many open items proposed new COL action items or change to existing COL action items
 - Additional items may be identified as reviews are completed
- Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC)
 - Approximately 15 of the 35 items are open; remaining are resolved or confirmatory

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DSER Open Issues (cont'd)

- Quality Assurance (QA)
 - QA Test Control Implementation Inspection at Oregon State University
 - Inspection of the Implementation of the Project-Specific Quality Plan at Westinghouse

7

DSER Open Issues (cont'd)

- Security
 - April & May 2003 – Interim Compensatory Measures (ICMs) and revised Design Basis Threat (DBT) provided to Westinghouse (W)
 - May 2003 – W revised Design Control Document (DCD) to defer Security Plan to COL Applicant
 - June 2003 – W letter provided assessment of the impact of ICMs and revised DBT to AP1000
 - Staff is reviewing DCD against 10 CFR Part 73, ICMs, and revised DBT for design implications that are not site specific

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DSER Open Issues (cont'd)

- Leak-Before-Break (LBB)
 - Alloy 690/52/152 susceptibility to PWSCC
 - Limited Test Data and Operating Experience
 - Staff discussed with W the need for inspections and an understanding of sensitivity study margins to provide sufficient defense-in-depth to address uncertainty of PWSCC
 - Piping stress analysis for most limiting LBB systems
 - Staff is working to determine if the appropriate bounding limits are established using preliminary analysis results during the design certification phase
 - The information that the staff needs to make a safety conclusion regarding the preliminary analysis results has not yet been provided
 - Planning future meetings with W to address this issue

9

DSER Open Issues (cont'd)

- Sump Performance
 - Staff concerns regarding debris loading of IRWST screens, recirculation screens, and debris through reactor coolant system break
 - The staff audited the associated W calculation and identified assumptions of debris size, density, and porosity that are not consistent with industry practices

10

DSER Open Issues (cont'd)

- Structural / Seismic Design
 - The staff identified 38 structural/seismic items, many of which require audit of specific W calculations
 - Containment design was not completed at April 2003 audit

11

DSER Open Issues (cont'd)

- Liquid entrainment, long term cooling, core swell, and boron precipitation
 - Numerous discussion at ACRS Thermal Hydraulic Subcommittee meeting on July 16 & 17, 2003, provided a good technical exchange for the staff regarding these issues
 - Staff review will continue with independent analyses and review of W submittals

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DSER Open Issues (cont'd)

- Probabilistic Risk Assessment (PRA)
 - January 23-24, 2003 - ACRS PRA Subcommittee Meeting
 - Raised ADS-4 Squib Valve reliability issue discussed yesterday
 - February 26, 2003 - PRA Meeting to discuss requests for additional information (RAIs) with W
 - PRA related open items: 24
 - Staff still reviewing W responses to the OIs

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AP1000 Summary

- Resolving DSER Open Items
- Questions?

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AP1000 Instrumentation and Control Design Overview

July 18, 2003

Tom Hayes
Passive Plant Projects
412-374-4420 - hayestp@westinghouse.com

ACRS VISIT TO WESTINGHOUSE 286

- I&C product presentation and demonstration
- Advanced Control Room Development Facility
- Not AP1000-specific

AP1000 I&C Design Overview

- I&C scope of Design Certification.
 - Design process is certified.
 - Functional requirements are certified.
 - Functions (Automatic, Manual -- safety, non-safety)
 - Arrangement
 - Qualification (Seismic, EMI/RFI, Environmental)
 - Power supplies
 - Independence (Separation, Isolation)
 - Design details are not certified -- COL requirement.
- Information presented at 286 site will generally be beyond Design Certification (i.e., COL items).

AP1000 I&C Design Overview

- The I&C scope for AP1000 was selected due to the rapid changes in I&C products.
 - The intent is to allow use of current technology.
 - Same as AP600 and other certified designs.
 - 'Current' I&C product for AP600 (Class 1E) was Eagle.
 - Eagle is used in other plants (Sizewell, Temelin, upgrades).
 - 'Current' I&C product for AP1000 (Class 1E) is Common Q.
 - Common Q is used in other plants (future Korean plants, upgrades).
 - 'Current' I&C product for both AP600 & AP1000 (non-Class 1E) is Ovation.
 - AC450 is also a 'current' non-Class 1E product.

Human Factors Engineering Design Acceptance Criteria for AP1000

July 18, 2003

Robert B. Fuld, CHFP
(860) 731-6168

Design Acceptance Criteria

- ITAAC includes
 - Design Commitment
 - Inspections/Tests/Analyses
 - Design Acceptance Criteria
- Firm commitment to auditable results & acceptable conclusions (DCD Tier 1)
- Brings closure to One-step licensing process for ALWR
- COL applicant is responsible, though vendor may satisfy

Human Factors Engineering

- Heterogeneous discipline addressing usability of system design
- In 1-step NRC licensing, emphasizes review of design process
- HFE process review guided by NUREG-0711 => classic systems approach
- HFE product review guided by NUREG-0700 & validation test results

HFE DAC

- Confirms closure of open elements of HFE plan
- Confirms specific aspects of design description / requirements for MCR, RSR, & local control stations

HFE DAC for AP1000

- Table 3.2-1 has 13 design commitments from Tier 1 HFE Design Description (see DCD Sec.3.2 handout)
- Typically state that, "A report exists and concludes that..." or "[required inspection confirms that a specified item] includes..."
- COL applicant is responsible

Conclusion

- HFE DAC states COL commitment in DCD Tier 1 to have auditable results & acceptable conclusions for design
- Commitments include HFE V&V of design
- Brings closure to HFE design process prior to operation

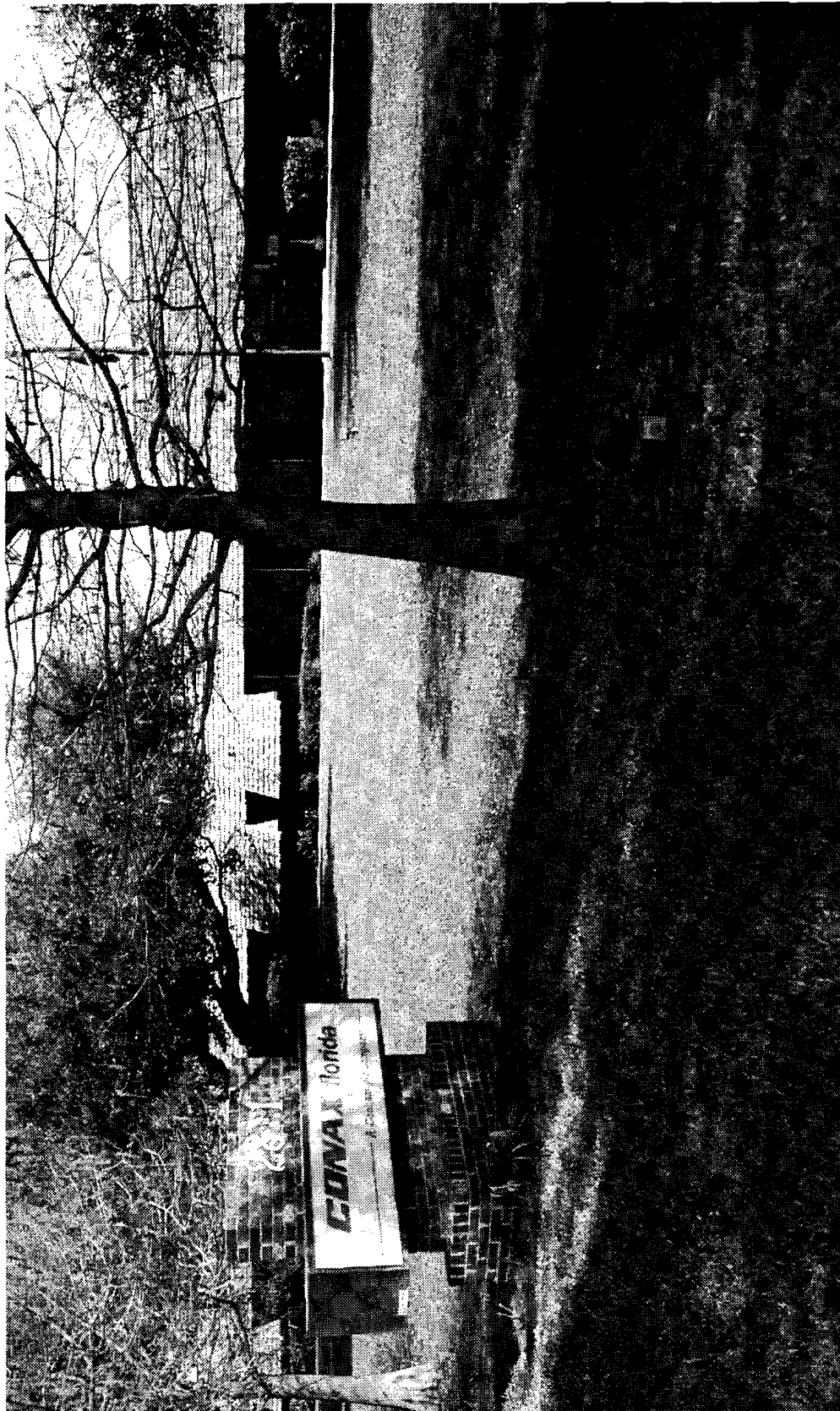
SQUIB VALVE



AGENDA

- Overview of Conax Florida Corporation
- GE SBWR ADS (AP600 ADS 4) Valve Development
- AP1000 ADS 4 Valve
- Squib Valve Reliability

OVERVIEW OF CONAX FLORIDA CORP.



CONAX florida
A Cobham plc company



CORPORATE OVERVIEW

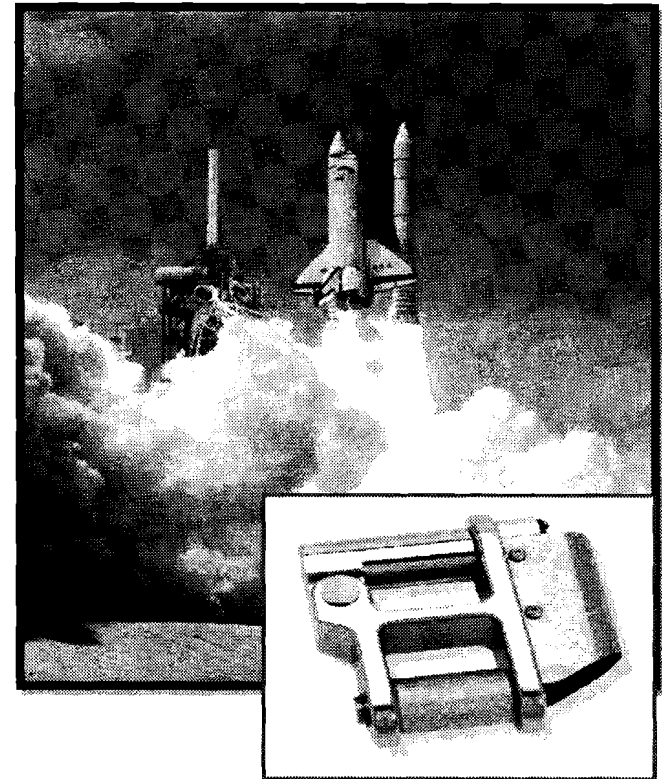
- Conax was founded in 1948.
- First developed electro-explosive devices in early 1950's.
- Conax Florida subsidiary was formed and moved to Florida in 1982.
- Facility was ISO 9001 certified in September 1997.
- Conax was purchased by Cobham plc of Dorset, England in 1998.
- Employ approximately 150 people.
- \$30 million in annual sales.
- Located in St. Petersburg, Florida

Aerospace Systems

- Utilize Conax's proprietary electro-explosive technology to actuate and control critical systems.

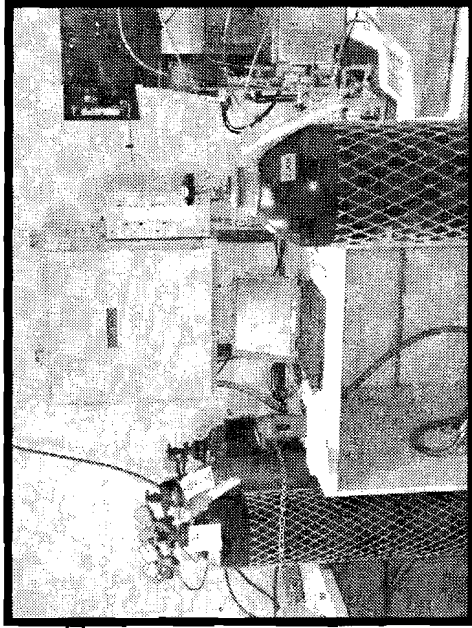
- Conax Products Include:
 - Pyrovalves
 - Stored Gas Systems
 - Water Activated Systems Systems
 - Pin Pullers & Cutters
 - Actuation Systems
 - Complex "Build to Print"

- Advantages:
 - Fast Acting
 - Solid Metal Seal
 - Reliable & Environmentally Durable
 - NASA Sponsored & Qualified

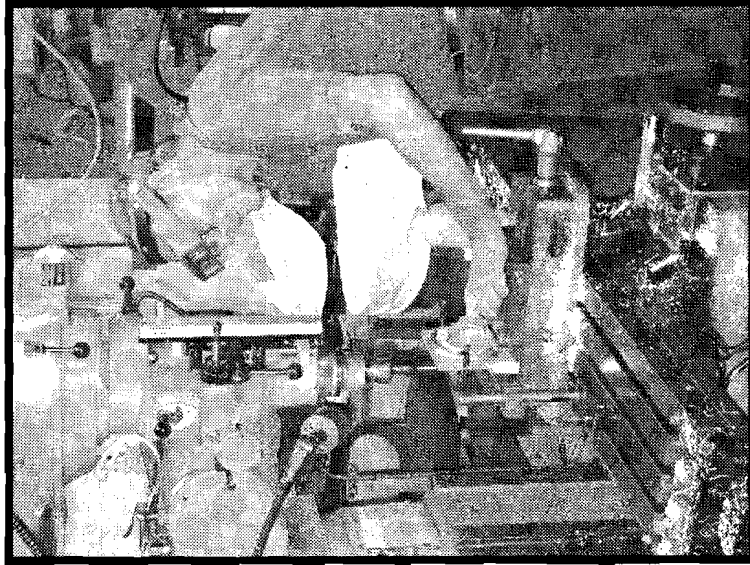




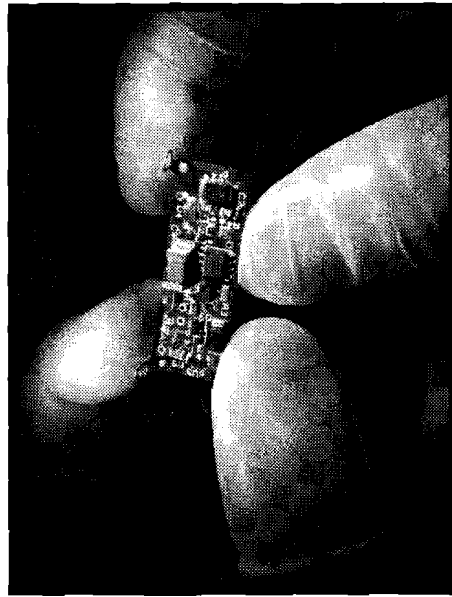
Class 10,000 Clean Room



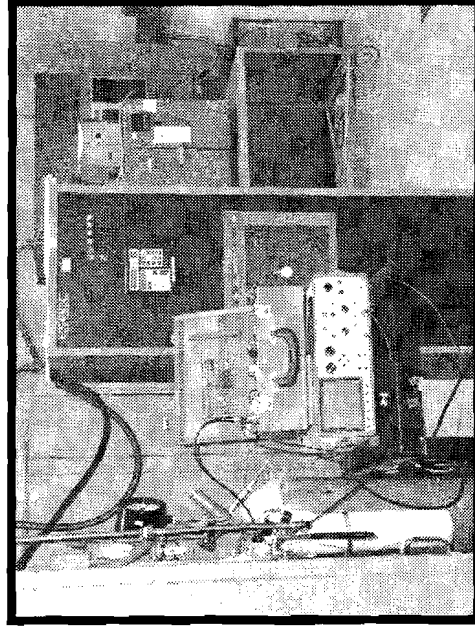
Gas Purity Testing



Prototype Shop



Electronic Assembly



Environmental Testing

CONAX florida
A Cobham plc company

CONAX MAJOR PROGRAMS

Space Systems

Atlas
Titan
Space Shuttle
Centaur
EELV
Delta
Taurus
Pegasus
AXAF
A2100
HS601
HS702
Mars Surveyor
PanAm Sat
NSTARC

Under Sea

MK-46 Torpedo
MK-48 Torpedo
MK-50 Torpedo
MK-67 (SLMM)
ADCAP Mine Torpedo
Arctic Buoy
CCAPS

Build To Print

Acoustic Counter-Measure
Submarine Separable Cover
Javelin Flex Assy.
Torpedo Nose Assy.
Smoke Generator

Missile Systems

Javelin
EKV
BAT
THAAD
Tomahawk
Tactical Tomahawk
Standard Missile
Minuteman
MX



CONAX florida
A Cobham plc company

CONAX QUALITY ASSURANCE

Certification

- Quality Assurance System ISO-9001 Certified
 - November 1997
 - January 2001
- Quality System in Compliance with AS-9100
 - International Aerospace Quality Group

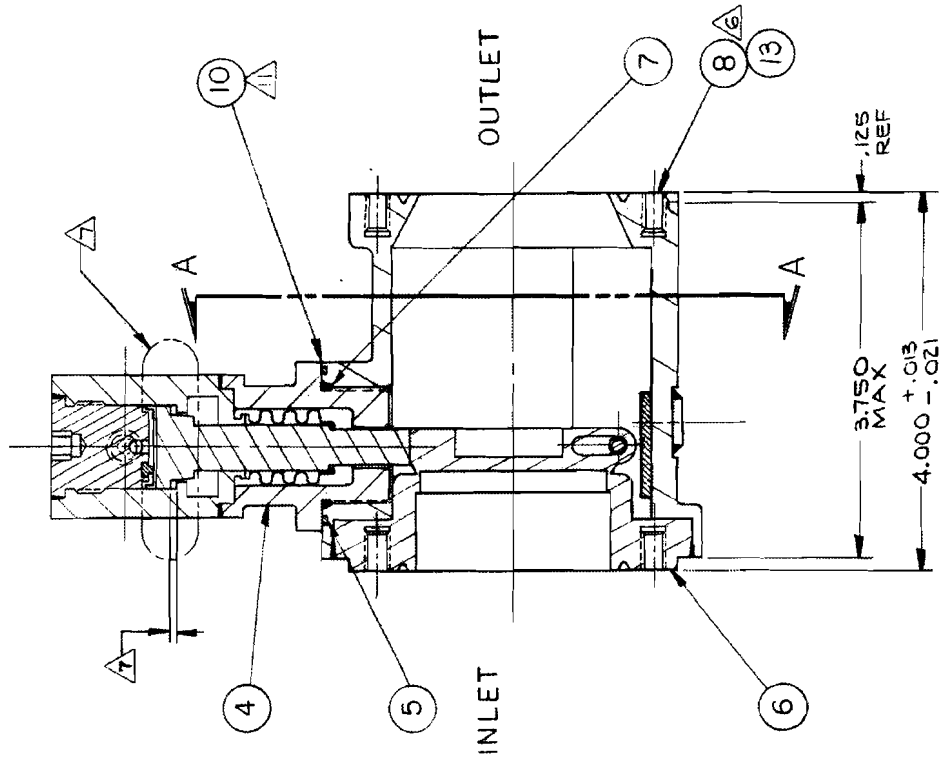
GE SBWR ADS VALVE DEVELOPMENT

- General Electric initiated contact with established valve suppliers throughout the world (7 bidders)
- Pyronetics proposed valve simplest design approach
 - Based upon Pyronetics 2" ID flow passage valve
- General Electric provided a list of Pyronetics customers
 - Some of Pyronetics customers contacted
- General Electric concluded OEA Pyronetics was capable of designing, manufacturing and testing squib valves (7" ID)
- Contract awarded to Pyronetics
- Westinghouse AP600 ADS 4 is same ID as GE valve
- Westinghouse AP1000 ADS 4 is simply scaled-up AP600 / GE SBWR valve



2" ID OEA DESIGN

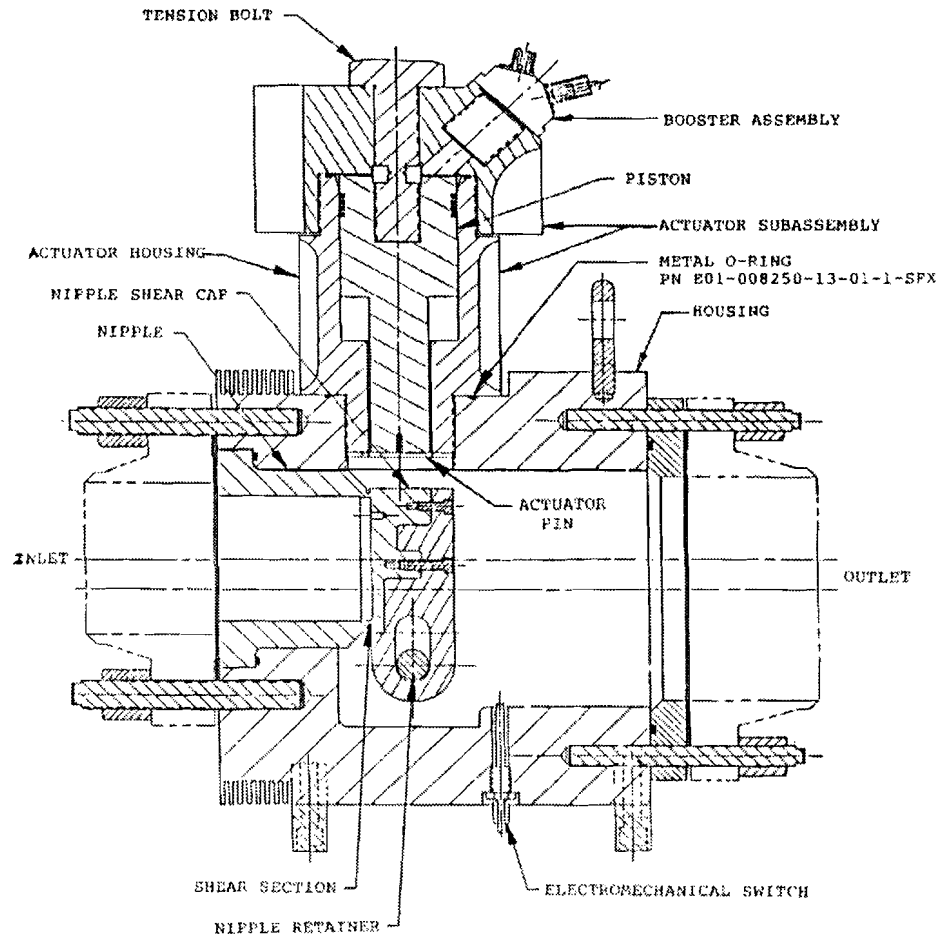
Design of 2" ID flow passage valve



CONMAX florida
A Cobham plc company

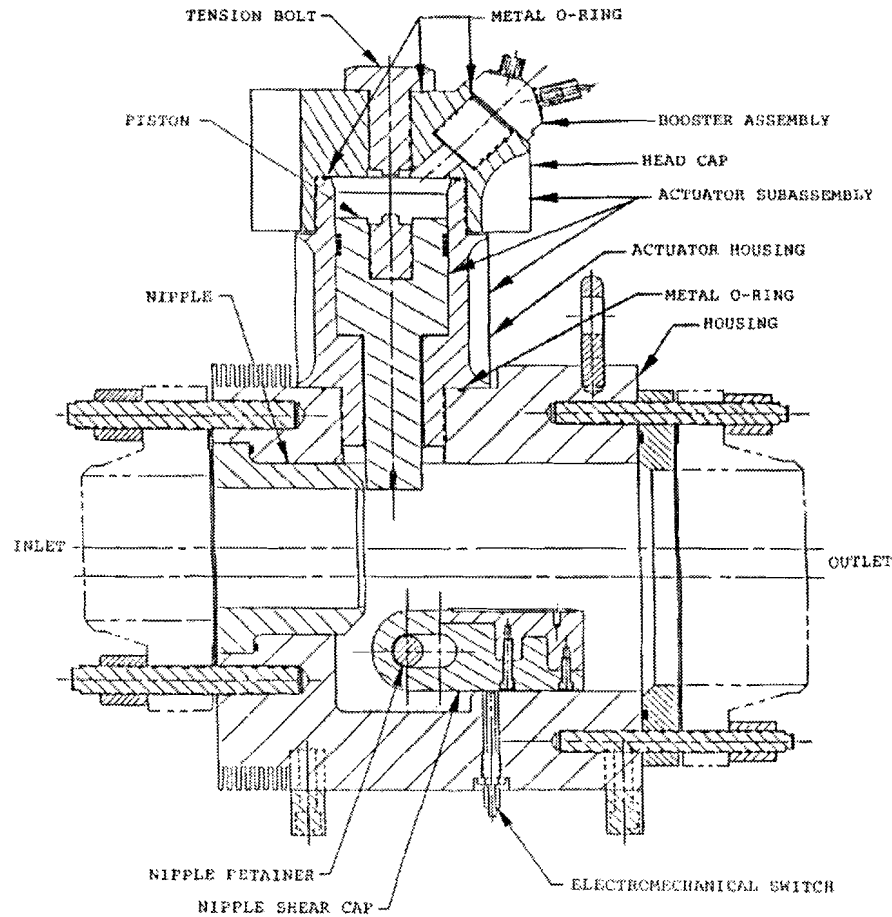
GE SBWR ADS DESIGN

Design of SBWR ADS (7" ID) - closed



GE SBWR ADS DESIGN

Design of SBWR ADS (7" ID) - open



TEST PROGRAM PERFORMED

Acceptance Tests

- Examination of Product (full size, pressure rating)
- Hydrostatic Testing
- Leakage (inlet pressurized)
- Thermal Exposure – Inlet conditioned to 550°F and surrounding air temperature 190°F (actuator met <280°F requirement)
- Cleanliness Verification

Valve Development Test Sequence

Development Tests ⁽¹⁾	SERNO 1	SERNO 2
First Actuation	1	1
Second Actuation	2	

⁽¹⁾ Both units delivered to GE for additional testing subsequent to Pyronetics testing

TEST PROGRAM PERFORMED ON INITIATOR BOOSTER

Closed Bomb Testing:

- At temperature
- 100% and 80% booster load

Lab Samples Testing:

- Thermo Gravimetric Analysis (TGA) scans to determine thermal degradation as a function of time and temperature,
- Differential Scanning Calorimetry (DSC) tests to determine change of state temperature and temperature regimes of interest,
- Isothermal tests to determine the amount of weight loss.

Radiation Testing:

- Boosters: 2.31×10^7 rads Total Integrated Dose (TID)
 - Position switch and cables 5.6×10^7 Rads TID
-

INITIATOR/BOOSTER TESTING (con't)

Accelerated Thermal Aging:

- Boosters: 25 days at 360°F (simulating four year normal life)
- Cable Assemblies: 69 days at 374°F (simulating ten year normal life)
 - Did not meet requirement. Design changed to use different epoxy.
- Position Switch: 54 days at 360°F (simulating ten year normal life)

Reliability Testing:

- All units met performance requirements





SBWR ADS VALVE TESTING

Seismic and other dynamic loads evaluated

- Vibration Testing
- Actuation / Flow

AP1000 ADS 4 VALVE

EDNAX florida
A Cobham plc company

PERFORMANCE REQUIREMENTS

	AP1000 ADS 4	SBWR ADS
Nominal Size	Inlet: 14 inch Min ID: 9.24	Inlet: 8 inch Min ID: 7.00
Valve Body Material	SS (316L)	SS (304L)
Safety/Seismic Class	1 / 1	1 / 1
Design Pressure	Inlet: 2485 psig Temp: 650°F	Inlet: 1500 psi Temp: 595°F
Seat DP, Forward	Inlet: 2500 psig	Inlet: 1500 psig
External Temperature Range, Normal	120°F to 50°F	190°F
Radiation Level	.5x10 ⁷ RAD (ten years)	2.31x10 ⁷ RADS (four years)
Open Pressure Range	Inlet: 1 to 2500 psig	Inlet: 1 to 1500 psig
Design Life of Booster	8 years target	4 years

AP1000 ADS 4 VALVE DEVELOPMENT

1. Design

- Scale up design from SBWR ADS valve
- Analysis Design Report (ASME Code)

2. Development/Prototype

- Testing similar to previous SBWR ADS program.
- Tests planned:
 - Charge sizing
 - Inspection
 - Hydrostatic and Leak Testing
 - Vibration
 - Actuation (over and under loaded boosters)

RELIABILITY

- **High Reliability Requirements**
- **FMEA (preliminary sample)**
- **Design Shear Section**
- **Reliability of Squib Valve**
- **IST Replacement of Charges**

HIGH RELIABILITY IS REQUIRED

- Our customers require high reliability
 - Life Support programs
 - Aerospace programs
 - Consequences of failures are high

- Conax procedures control high reliability

- Custom Valve Designs / Up-Scaling is Standard Process
 - Simple valve design reduces problems
 - Development process is able to deliver highly reliable valves



HIGH RELIABILITY IS BUILT IN

Design / Analysis

- Past experience with similar products (lessons learned)
- Design analysis of new design
- Examination and analysis of drawings
- FMEA
- Reliability analysis

Testing

- Development and prototypes units
- Margin testing (over and under loaded boosters)
- Acceptance and qualification
- On propellants (powder form and in initiators and booster)

Quality Assurance: ISO 9001 Certification

DESIGN SHEAR SECTION

Design:

- Shear section is standard pyrovalve design feature
- Concept proven on thousands of valves
 - No leakage ever reported on delivered product
 - Thickness as small as 0.003 inch and thicker
- Concept same as for AP600 / SBWR ADS
- Designs meets ASME code Section III, Class I
 - NB3200, Design by analysis

DESIGN SHEAR SECTION

Corrosion Affects

- 316 L Material selected for the con-o-cap (i.e. shear section):
 - Material compatible with planned system media
 - Not subject to stress corrosion cracking like numerous other materials

In Service Inspection

- Designed for ease of removal for inspection

RELIABILITY SUMMARY

- **UPCO reliability of:**
 - .999568 at 90% confidence
 - .999437 at 95% confidence
 - Valves manufactured: 64,690
 - Total quantity fired: 5,324

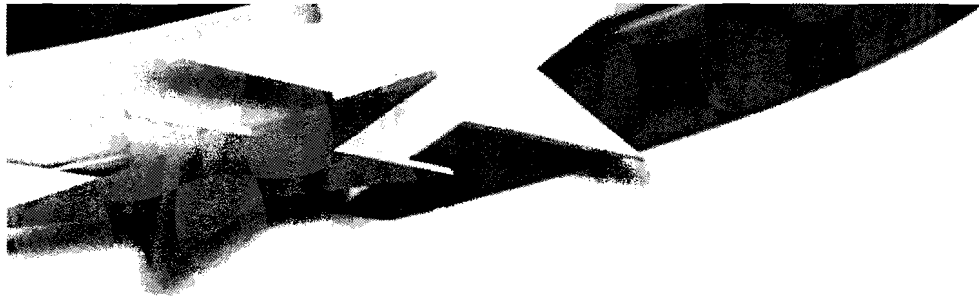
 - **Conax reliability of:**
 - .9998169 at 90% confidence
 - Based upon:
 - Total initiators >25,000
 - Valves manufactured: >25,000

 - **Sandia reliability of:**
 - .999839 at 90% confidence
 - Based upon:
 - Initiators manufactured: 25,000
-

RELIABILITY

- Squib valves have high inherent reliability
 - Reliability for smaller valves applicable for larger valves
 - Same design standards established
 - > Engineering Analysis
 - > Proof and Leak Testing
 - > Over and under loaded boosters used during testing
 - > Design concept similar (shearing material) in all cases.
 - No failures associated with shear section cracking under constant high pressure and temperature





Squib Valve Preliminary Reliability Evaluation - Extracts

**Center for System Reliability
Sandia National Laboratories
PO Box 5800, MS 1176
Albuquerque, NM 87185-1176**

July 14, 2003





Squib Valve Reliability Overview

- Squib valves are high precision devices with an operational reliability typically over 0.999, even after long periods of dormancy. Characteristics supporting reliability include:
 - High energy delivered per unit weight
 - Small volume, compact
 - Ability to valve very high pressure working media
 - Long-term storable energy
 - Rapid actuation time
 - Relative simplicity
 - Controllable initiation and output energies



Reliability Calculations

Valve Assembly Operational Reliability: $R_V = (R_C)(R_S)(R_M)$



Component
Reliability:

R_C

R_S

R_M

Operational Reliability:

$$R_V = (.999908)(.999990)(.999941) = .999839$$

Mission Reliability:

$$R_{MV} = (R_V)(R_{DV})$$

$$R_{MV} = (.999839)(.999842) = .999681$$



Case Study

Study by NASA Langley Research Center showed:

- “Blowby” (failure mode) is a major safety concern
- “Blowby” cannot be prevented by single or dual o-rings in pyrotechnically actuated piston/cylinder configurations.
- The metal-to-metal seal design employed by Conax (models 1832-191-01 and 1802011-01) completely prevented blowby under conditions that were more severe than the o-ring sealed valve designs.

Valve Test Data (1996)

		Product D-Tests		Stockpile Evaluation	
MC No.	Development Tests	Lot No.	Product No.	NMLT/SLT	NMFT/SFT
3006	143	107	547	161	100
3205	354	45	318	355	86
3206	194	33	210	223	86
3294	97	71	194	31	59
3784		37	78		
3295	41	44	116	31	72
3297	66	14	227	31	72
3785		35	415		
3298	65	36	174	31	36
All †	29		279	421	48
3570	59	34	76	78	48
4232			13		
4241	0	30	56	0	0
TOTALS:	1048		2703	1362	607
GRAND TOTAL WITHOUT DEVELOPMENT TESTS =					4672

No Observed Failures

Current Failure Rate Assessment = 0.00021

† Includes 4 valves: MC3425/A, MC3427/A, MC3428/A, MC3604/A



Valve Test Data (2002)

MC No.	System	Development Tests	Product D-Tests		Stockpile Evaluation	
			Lot No.	No.	NMLT/SLT	NMFT/SFT
3006	W76	143	110	554	192	126
3205	W79	354	45	318	311	83
3206	W79	194	33	210	164	88
3294	B83	97	72	162	47	85
3784			37	79	1	
3295	B83	41	45	113	47	101
3297	B83	70	14	227	47	101
3785			40	422		
3298	B83	65	38	176	48	60
All †	W84	60		604	311	80
3570	W87	59	40	87	119	54
4232			13	46		
4241	W87	0	33	59	37	14
	TOTALS:	1083		3057	1324	792
GRAND TOTAL WITHOUT DEVELOPMENT TESTS =						5173

No Observed Failures

Current Failure Rate Assessment : 0.00013

† Includes 4 valves: MC3425/A, MC3427/A, MC3428/A, MC3604/A





Conclusions

- The AP1000 valve design is a basic design that has been used extensively for many smaller squib valves.
- Existing fleets of smaller valves have been successfully utilized in similar and harsher environments than that proposed for use in the AP1000.
- Evaluations by subject matter experts indicate that some reliability information can be inferred from smaller valves to larger valves when scaling the design.
- Given that standard mechanical engineering design practices are followed, the reliability performance of the larger valve will be consistent with that of smaller valves.



AP1000



AP1000 Design Certification Review
Westinghouse Electric Company

Presentation to
Advisory Committee on Reactor Safeguards
Future Plant Sub-Committee
July 17 - 18, 2003



Agenda

Thursday July 17, 2003

-
- | | | |
|--|---------------------------------|----------------|
| ● Introduction | Tom Kress, ACRS | 1:00 PM |
| ● ADS-4 Squib Valve Reliability | | |
| - System Design | Terry Schulz, Westinghouse | 1:05 PM |
| - ADS 4 Controls | Tom Hayes, Westinghouse | |
| - Valve Design | Dan Frederick, Conax | |
| - Modeling in AP1000 PRA | Selim Sancaktar, Westinghouse | |
| BREAK | | 2:45 PM |
| ● Post-LOCA Aerosol Deposition on Containment | Jun Li, Polestar | 3:00 PM |
| | Jim Scobel, Westinghouse | |
| ● Summary of NRC Severe Accident Calculations | M. Khatib-Rahbar, ERI | 3:45 PM |
| ● AP1000 Seismic and Structural Design | R. Orr, W | 4:30 PM |
| ADJOURN | | 5:30 PM |

Agenda

Thursday July 17, 2003

-
- | | | |
|---|--------------------------------|-----------------|
| ● Introduction | Tom Kress, ACRS | 8:30 AM |
| ● DSER Open Items | | |
| - NRC Overview | Joelle Starefos, NRC | 8:35 AM |
| - Westinghouse Resolution Paths | Mike Corletti, Westinghouse | 9:30 AM |
| BREAK | | 10:00 AM |
| ● Leak-Before Break Issue | Warren Bamford, West. | 10:15 AM |
| ● Sump Performance Issues | Terry Schulz, West. | 10:45 AM |
| ● AP1000 I&C Design Overview | Tom Hayes, Westinghouse | 11:15 AM |
| - I&C Overview | | |
| - Human Factors Engineering DAC | | |
| ● General Discussion | Tom Kress | 11:45 PM |
| ADJOURN | | 12:30 PM |

Tour of Plant Automation Headquarters

AP1000

ADS-4 Squib Valve Reliability

July 17, 2003



Outline

- **PXS/ADS System Design Features** T. Schulz
- **ADS 4 Controls** T. Hayes
- **Valve Design Features, Reliability** Valve Vendor
- **Modeling of ADS 4 in AP1000 PRA** S. Sancaktar

AP1000

AP1000 PXS/ADS Design Features

July 17, 2003

Terry Schulz
Advisory Engineer

412-374-5120 - schulztl@westinghouse.com



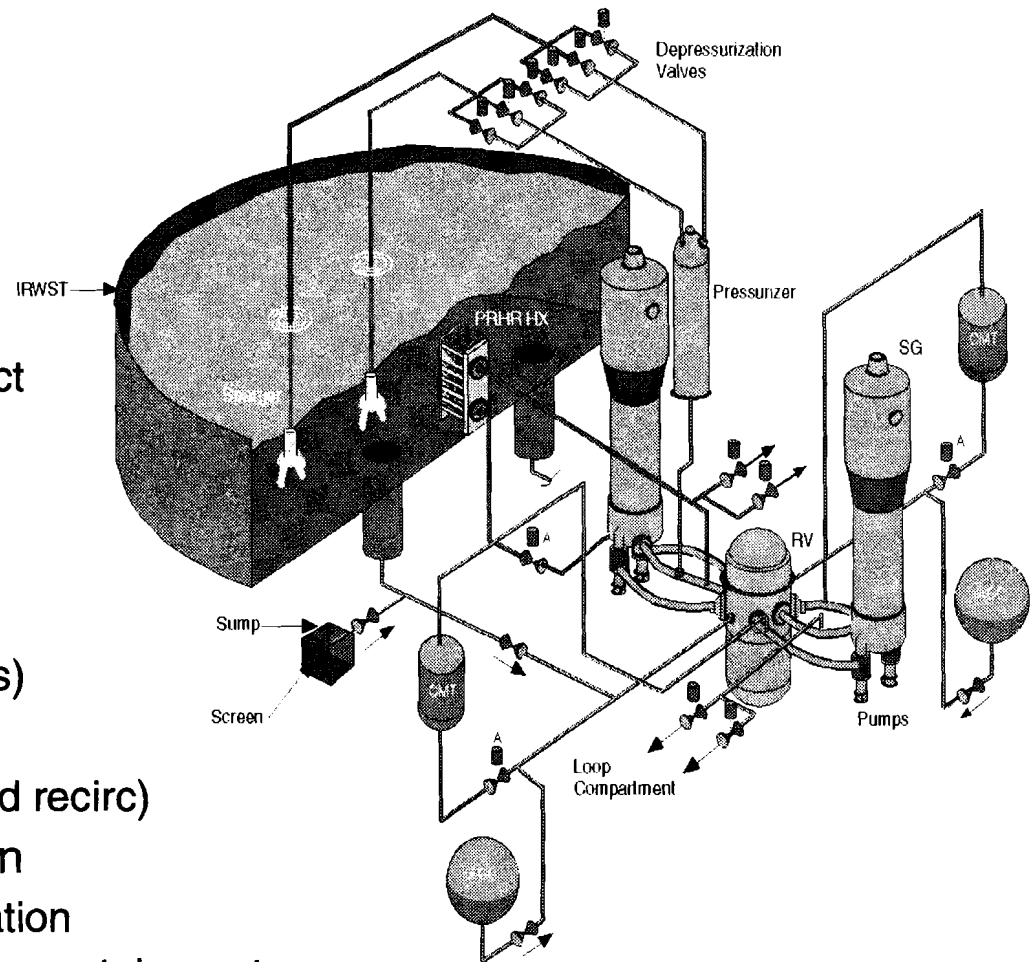
AP1000 Passive Core Cooling System

- **PRHR HX**

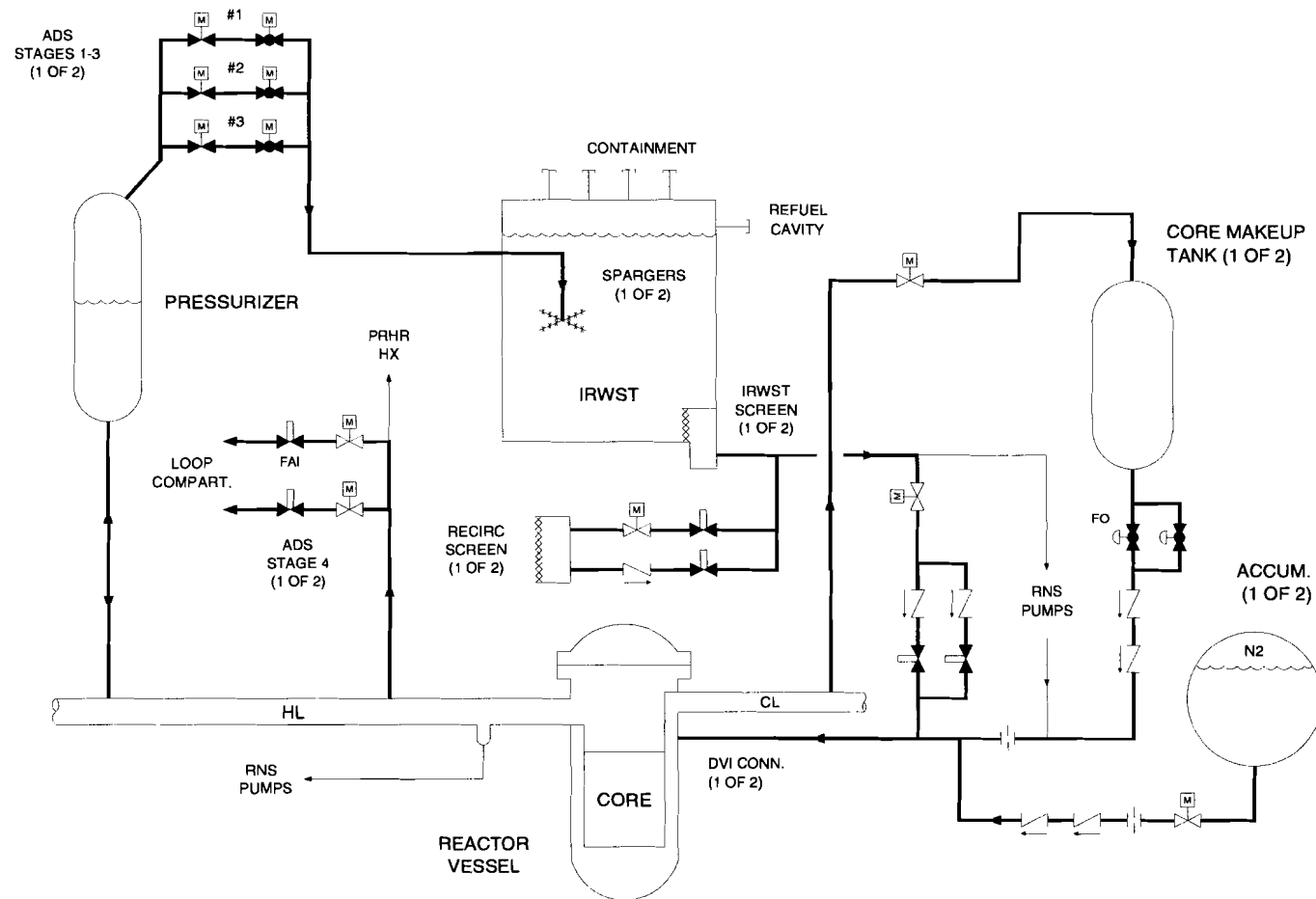
- Natural circ. heat removal
- Replaces AFW pumps

- **Passive Safety Injection**

- Core Makeup Tanks
- Full RCS pres, natural circ. inject
- Replaces HHSI pumps
- Accumulators
- Similar to current plants
- IRWST Injection
- Low pres (replaces LHSI pumps)
- Containment Recirculation
- Gravity recirc. (replaces pumped recirc)
- Automatic RCS Depressurization
- Staged, controlled depressurization
- Stages 1-3 to IRWST, stage 4 to containment



Passive Safety Injection



AP1000 ADS

- **Provides Controlled RCS Depressurization**

- For LOCA mitigation
 - Required to allow injection from Accum, IRWST, Contain Recirc
 - Multiple stages / smaller high pres sizes reduce impact on RCS
- Automatic actuation via PMS
 - ADS 1 (2x4") – low1 CMT level (2/4 sensors in either CMT)
 - ADS 2 (2x8") – ADS 1 plus timer
 - ADS 3 (2x8") – ADS 1 plus timer
 - ADS 4a (2x14") – low2 CMT level (2/4 sensors in either CMT)
 - ADS 4b (2x14") – ADS 4 plus timer
- Manual controls via PMS and DAS
 - Requires 2 switches in PMS or DAS

ADS 4 Valve Control

- **ADS 4 Valve Power / Actuation Trains**

- Each ADS 4 valve is actuated by two PMS divisions (A/C or B/D)
 - These PMS divisions are located by pairs inside containment
- Each ADS 4 valve is also actuated manually by DAS

- **PMS Automatic Actuation Requires**

- CMT actuation signal
- CMT low 1 level plus time delays
- CMT low 2 level
 - Low2 level not reached if RNS is started per emergency procedures
- RCS low pressure

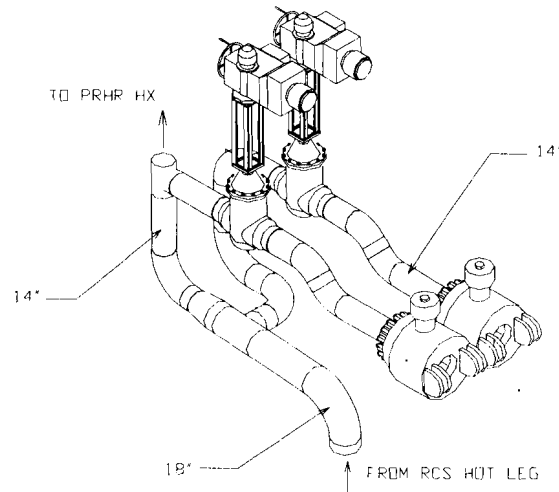
- **PMS and DAS Manual Controls**

- Requires actuation of 2 switches

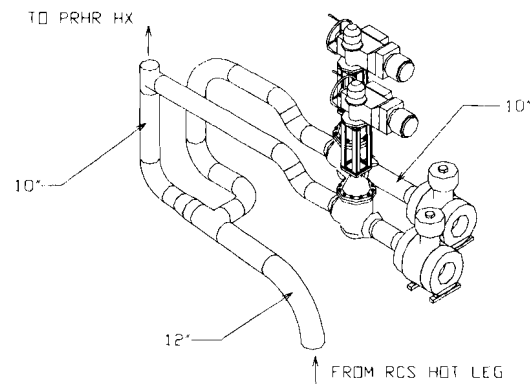
- **Chance of Inadvertent Actuation is Very Low**

Comparison of 4th Stage ADS

AP1000



AP600



ADS 4 Valve Requirements

	A P 1 0 0 0
Nominal Size	14 in
- Flow ID	9.24 in
Safety / Seismic Class	1 / 1
Design Pres. / Temp.	2485 / 650 psia / F
Seat DP , Opening Max.	2500 psi
, Opening Min.	1 psi
, Opening Expected	100 psi
Normal, fluid temp (nominal)	500 F
Normal, fluid temp (max)	615 F
Connection to Piping, Upstream	Flanged
, Downstream	na
Material of Construction	SS

ADS 4 Valve Type Selection

- **Squib Valves Were Selected for ADS Stage 4**
 - Very reliable to open on demand (better than AOVs, MOVs)
 - Ability to add independent actuation circuits (use 2 PMS, 1 DAS)
 - Diverse from ADS stage 1/2/3 MOVs
 - Very low chance of inadvertent opening
 - Zero leakage during normal operation
 - Simplified IST, ISI, maintenance
 - Reduced capital cost
 - Reduced development costs / uncertainties
 - Compact size, weight
 - Supported by U.S. utilities

ADS 4 Inservice Tests

- **In Accordance With ASME**

- Remove 20% of charges every 2 years
 - Fire charges in test fixture to demonstrate that valve would have operated
 - Use staggered testing
- Verify continuity of ADS 4 circuit after disconnecting and re-connecting wires
- Verify valve position sensor every 2 years

ADS 4 Inservice Inspections

- **In Accordance With ASME**

- Every ten years perform the following
 - Measure shear cap dimensions to ensure no thinning
 - Perform dye penetrant test to ensure no cracking
 - Use staggered testing

AP1000

AP1000 Squib Valve Actuation Circuits

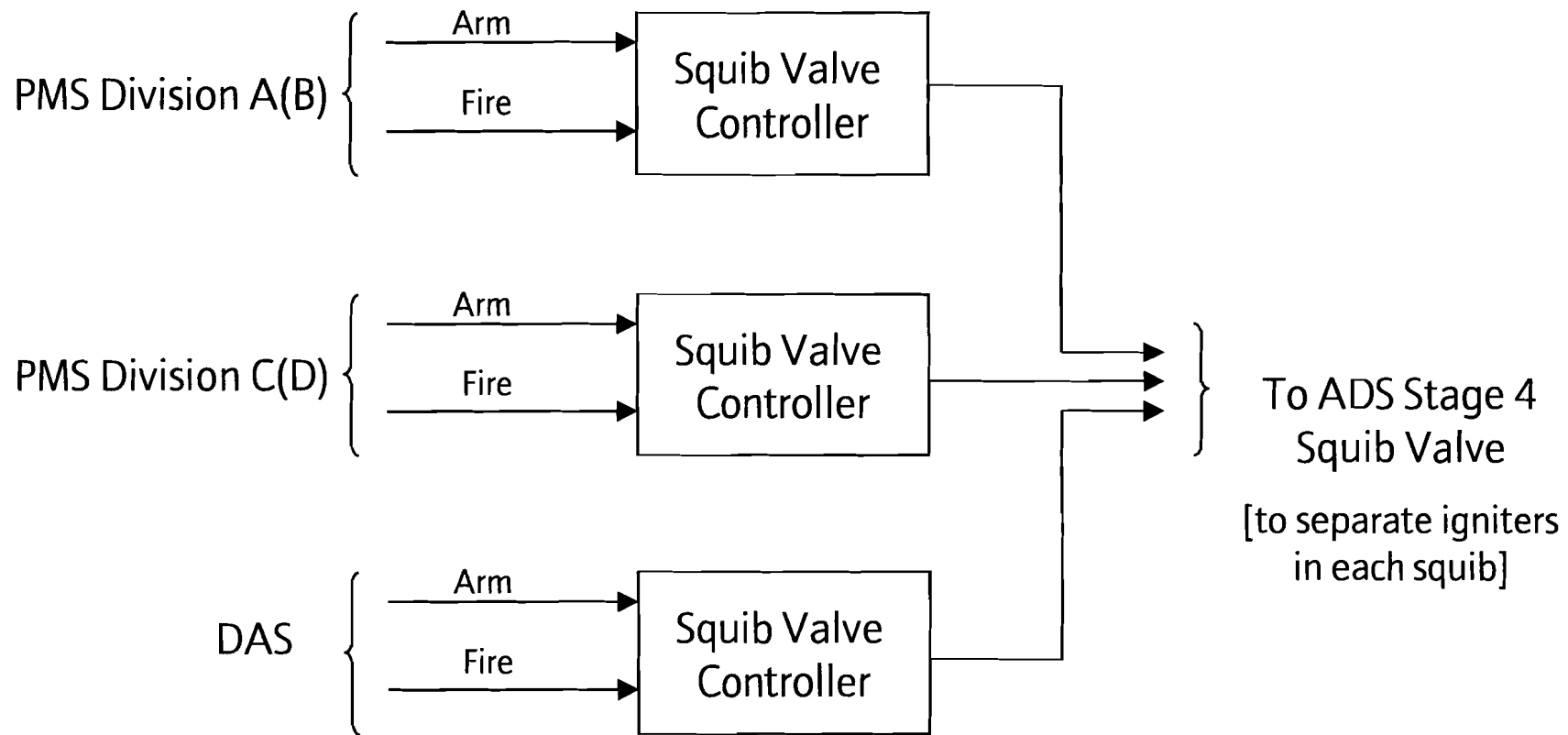
Tom Hayes
Passive Plant Projects
412-374-4420 - hayestp@westinghouse.com

AP1000 Squib Valve Actuation Circuits

- Each ADS 4 squib valve can be actuated by:
 - Either of two Protection System (PMS) divisions, auto / manual
 - Diverse Actuation System (DAS), manual.
- Squib valve actuation circuits are energize-to-actuate.

AP1000 Squib Valve Actuation Circuits

Each squib valve actuation circuit utilizes a squib valve controller



AP1000 Squib Valve Actuation Circuits

- Each squib valve controller has 2 inputs -- 'Arm' and 'Fire'
 - To actuate the squib valve, the 'Arm' circuit must be energized while the 'Fire' circuit remains un-energized. This permits a capacitor to charge.
 - Full charge is indicated and alarmed to the operator.
 - The 'Arm' circuit must then be de-energized followed by energizing the 'Fire' circuit.
 - If the 'Fire' circuit actuation is delayed, the capacitor will discharge and the 'Arm' cycle must be repeated for actuation.

AP1000 Squib Valve Actuation Circuits

- Each squib valve controller has 2 inputs -- 'Arm' and 'Fire' (continued)
 - Actuation of either 'Arm' or 'Fire' alone will not result in valve actuation.
 - Simultaneous actuation of both 'Arm' and 'Fire' circuits will not result in valve actuation.
 - Each 'Arm' and 'Fire' circuit is two-pole, energize-to-actuate.

AP1000 Squib Valve Actuation Circuits

- Failures 'upstream' of the squib valve controller have a low probability of causing a spurious actuation.
 - 'Arm' and 'Fire' signals come from different PMS output cabinets/DAS switches.
 - PMS ADS-4 automatic actuation utilizes 2-out-of-4 logic and several interlocks (CMT actuation, low RCS pressure).
 - PMS and DAS manual actuation from dedicated switches in the MCR requires action at two locations.
 - PMS uses high quality software (Class 1E).

AP1000 Squib Valve Actuation Circuits

- Failures of the squib valve controller have a low probability of causing a spurious actuation.
 - Under normal conditions the squib valve controller has no power and no stored energy.
 - No credible failure of the squib valve controller will result in valve actuation.

AP1000 Squib Valve Actuation Circuits

- Failures 'downstream' of the squib valve controller have a low probability of causing a spurious actuation.
 - Two-pole, energize-to-actuate circuits
 - Squib valve actuation cables use less than 50 volts and are routed in instrument trays.
 - Between the controller and the valve (primarily in containment), adjacent cables do not have sufficient energy to actuate valve.

AP1000 Squib Valve Actuation Circuits

- Fire Induced Spurious Actuation
 - Upstream of controller
 - Two *coordinated* 'Arm' and 'Fire' circuits are implemented in different cabinets
 - A fire is assumed to start in one cabinet
 - Before it can spread to a second cabinet the operators are required to remove power
 - Control room dedicated PMS/DAS switches are disabled following evacuation of control room.
 - Squib Valve Controller
 - Squib valve controller has no power available during normal operation.
 - Downstream of controller
 - No adjacent cables with sufficient energy for actuation

AP1000 Squib Valve Actuation Circuits

● SUMMARY

- ADS 4 controlled by 2 *coordinated* circuits ('Arm' and 'Fire')
 - 2-out-of-4 logic and high-quality software for automatic actuation
 - Squib valve controller has no power available during normal operation.
- Downstream of controller
 - No active components
 - No adjacent cables with sufficient energy for actuation



AP1000

Reliability of Large Squib Valves in AP1000 Design

Selim Sancaktar

Fellow Engineer

Reliability and Risk Assessment

Squib Valves In AP1000 PRA

- **Three types of Squib valves are modeled in the AP1000 PRA. The four Squib valves used on the fourth stage ADS are the subject of this presentation.**
- **Squib valves are used in the AP1000 design because they are more reliable than MOVs or AOVs.**
- **The Squib valve design is very simple and there are very few ways for the valve to fail. AOVs or MOVs valves have more moving parts that can fail and prevent the valve from actuating.**
- **Two types of failures of the ADS fourth stage Squib valves are modeled:**
 1. **The valve fails to open after receiving a signal to open (5.8E-04/demand used).**
 2. **The valve opens spuriously (5.4E-05/year is used leading to large LOCA)**



Fail to Open Mode

- Three different sources of failure probability were used to establish the AP1000 Squib valve reliability:
- The ALWR URD indicates a failure to operate probability for Squib valves of 3E-03 per demand. This may be because the basis for the URD value is a small population of valves and extrapolation from older, less relevant data.
- Sandia Laboratories have worked on designs of weapons systems and space systems where Squib valves are commonly used. They were consulted to verify the URD failure probability. Two sources at Sandia produced failure data based on a large population of Squib valves. The data produced failure probabilities of 2.0E-04 per demand and 3.2E-04 per demand.
- Each of these values is relevant to the AP1000 Squib valve failures. A geometric mean of the URD value and the Sandia values produces a failure probability of 5.8E-04 per demand [$\{(3.0E-03) * (2.0E-04) * (3.2E-04)\}^{1/3}$]. This is the value used in the AP1000 PRA.

Valve Reliability Data From Conax

- **The valve reliability information provided by the Squib valve vendor CONAX indicates that:**
 - Squib valves have high inherent reliability
 - Reliability for smaller valves is applicable to larger valves
 - No failures associated with shear section cracking under constant high pressure and temperature expected.

- **UPCO reliability of:**

- 0.999568 at 90% confidence Failure probability = 4.32E-04
- 0.999437 at 95% confidence
 - Based on Valves manufactured: 64,690;
 - Total quantity fired: 5,324

- **Conax reliability of:**

- 0.9998169 at 90% confidence Failure probability = 1.83E-04
 - Based on Total Initiators: > 25,000;
 - Valves Manufactured: >25,000

- **This data supports the high reliability of these valves**

- **Further justifies the failure to open probability used in
AP1000 PRA**

Spurious Opening Failure

- **The dominate cause of spurious opening on an ADS 4 valve is considered a spurious signal**
 - Structural failure of the valve is deemed to be much less likely and is not estimated in the AP1000 PRA
- **The failure frequency of one or more ADS stage 4 Squib valves opening due to spurious signal generation is estimated to be 5.4E-05/year in AP600/AP1000.**
- **The fact that hardware failure of the valve leading to gross leakage is considered small compared to this value implies that the contribution of failures from this failure mode is deemed to be less than 5E-06/year, or 5.7E-10/hour for four valves. This translates to 1.4E-10/hr failure rate for a single valve.**

Estimate of Structural Valve Failure

- **Squib valve is considered to similar to a pipe segment.**
- **In AP1000 significant leakage from a pipe segment is assigned a failure rate of 8.5E-09/hr.**
 - A piping section is defined as a segment of pipe between major discontinuities (i.e., pumps, valves, elbows, bends, etc.,) up to 10 ft in length.
 - The fraction of this rate corresponding to leakage severe enough to constitute a rupture should be taken as 5%. The 5% value is consistent with WASH-1400.
 - Of this 5%, 10% should be taken to be essentially a complete break; 30% should be taken to be a large rupture; and the remaining 60% a small rupture.
 - This approach has been used in operating plant PRA that have been approved by NRC (See Reference 1, Annex A, Note 10 to Table A2-1).
 - Squib valve contribution to Large LOCA yearly frequency would be:
 - $4 * 8.5E-09 * 8760 * 0.05 * 0.4 = 6E-06/\text{year}$

Reference 1: Advanced Light Water Reactor Requirements Document, Volume III, Appendix A to Chapter 1, "PRA Key Assumptions and Groundrules," EPRI, Rev. 5 & 6, December 1993.

PRA Sensitivity to Squib Valve Reliability

- **If the Squib valve failure probability (FTO) is doubled, then the plant CDF for internal events at power goes from 2.41E-07/yr to 2.77E-07/yr - a 15% increase**
- **The contribution of spurious ADS opening initiating event to plant CDF is 12.3%. If the spurious opening of ADS failure probability is doubled, the plant CDF will increase by 12.3%**
- **The importance of ADS Squib valves in AP1000 plant risk is recognized and design and operational precautions are already built in to assure their reliability**

Conclusions

- **Recent information obtained from different sources support the AP1000 PRA treatment of the ADS 4 squib valves**
 - Failure probabilities used for squib valves in AP1000 PRA are reasonable
 - Are consistent with operating experience
 - Are expected to be achievable for the AP1000 squib valves
 - Up-scaling of squib valves would not negatively affect their historical high reliability
 - Operational environment is not seen as an important contributor to valve failure probability
 - The conclusions of the AP1000 PRA with respect to spurious failure of ADS 4 squib valves are valid
 - As a side point, in case of a squib valve structural failure, the ADS line can be isolated, terminating the ensuing LOCA

Thermal Hydraulic Conditions for AP1000 Lambda Calculation

James H. Scobel

Containment and Radiological Analysis

412-374-5030 scobeljh@westinghouse.com



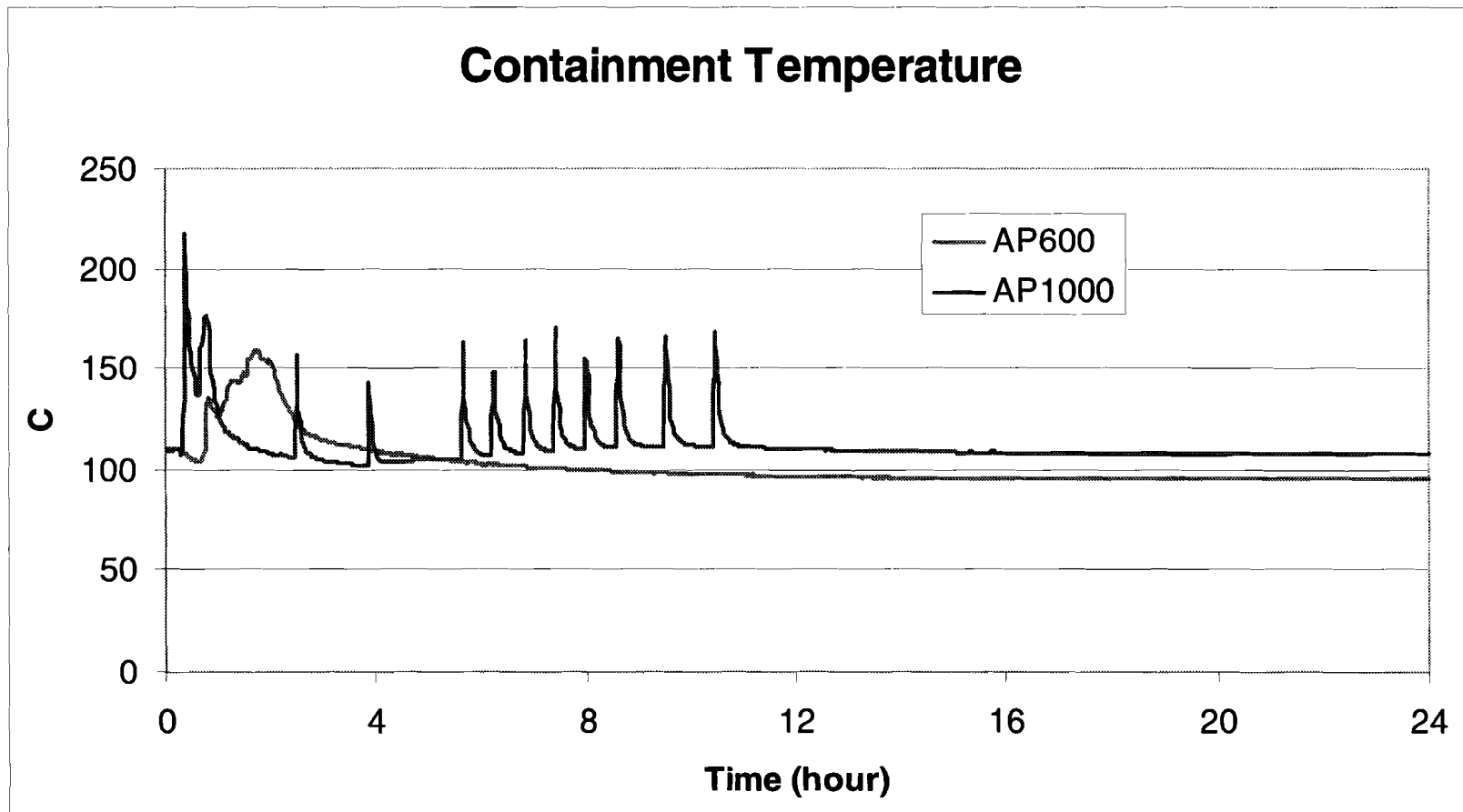
Accident Sequence Definition

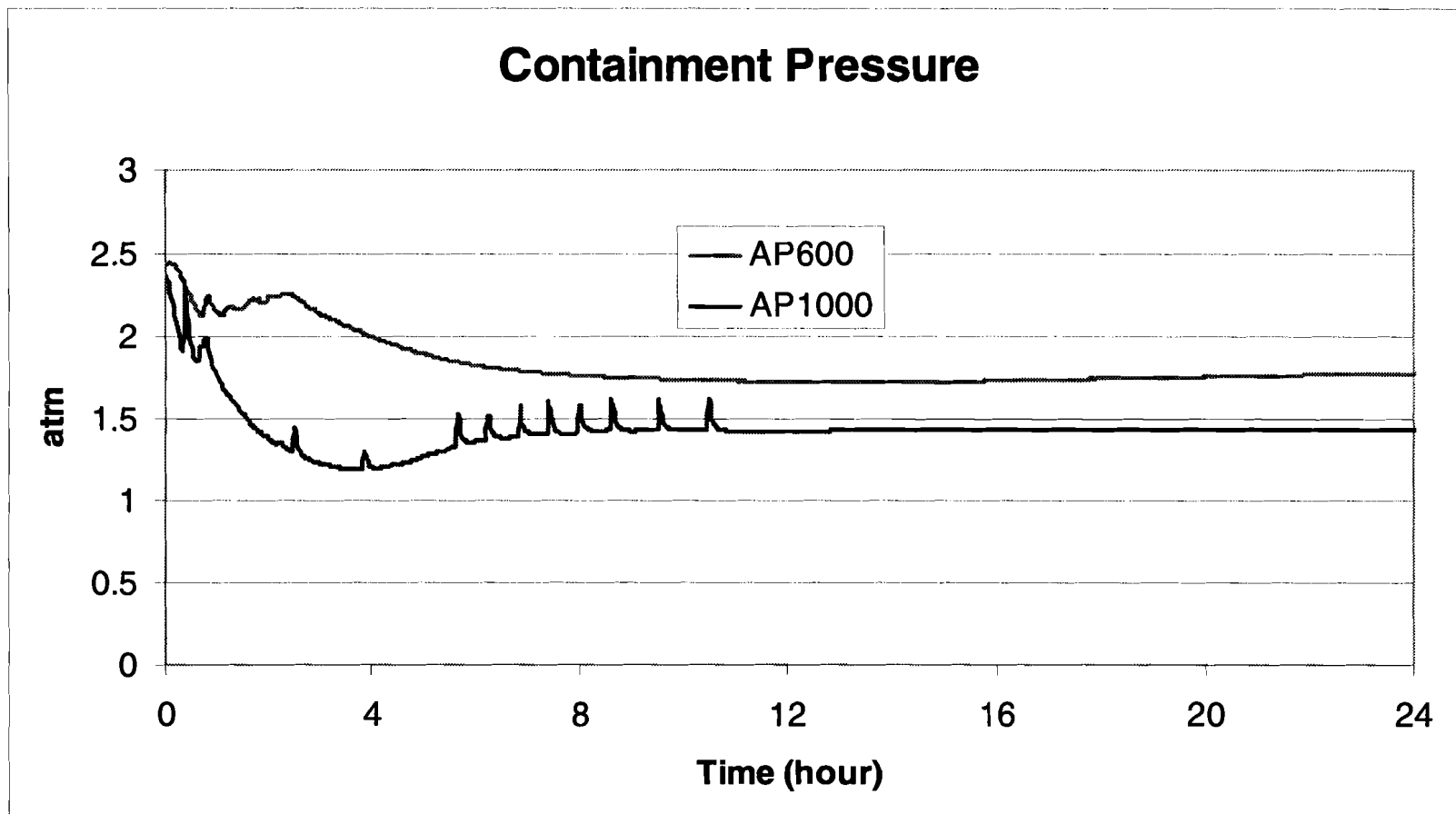
- **Severe accident environment generated with MAAP4**
 - conservative conditions for lambda calculation
 - methodology used for AP600 Environment for Lambda calculation
- **Dominant core damage sequence from PRA**
 - Break in a direct vessel injection line (fails 1 train of passive injection)
 - Full RCS depressurization
 - failure of gravity injection
 - successful cavity flooding and in-vessel retention of core debris
 - vessel reflooding through break
 - hydrogen igniters

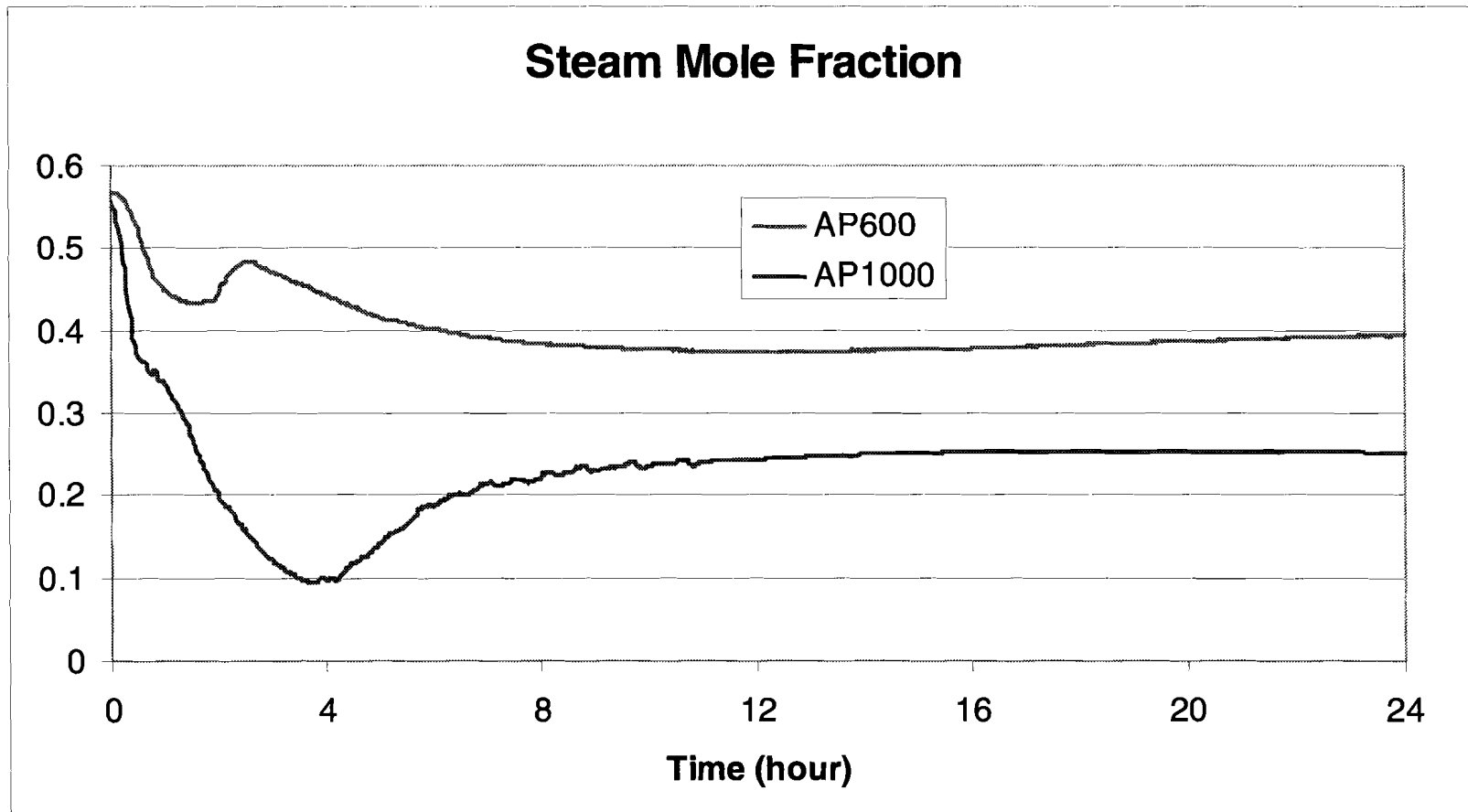
Parameters for Lambda Calculation

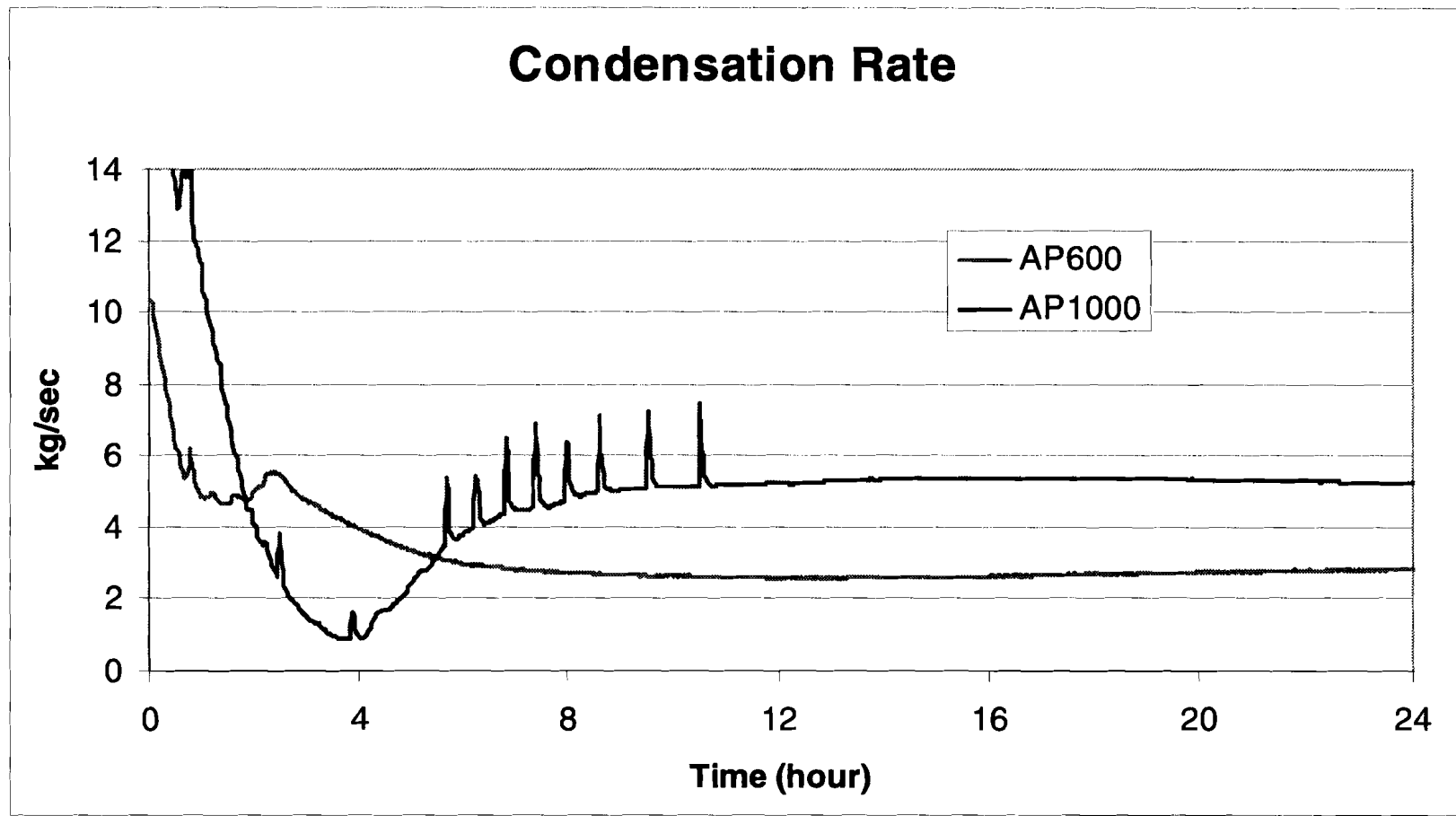
- **NAUA code T&H input parameters**

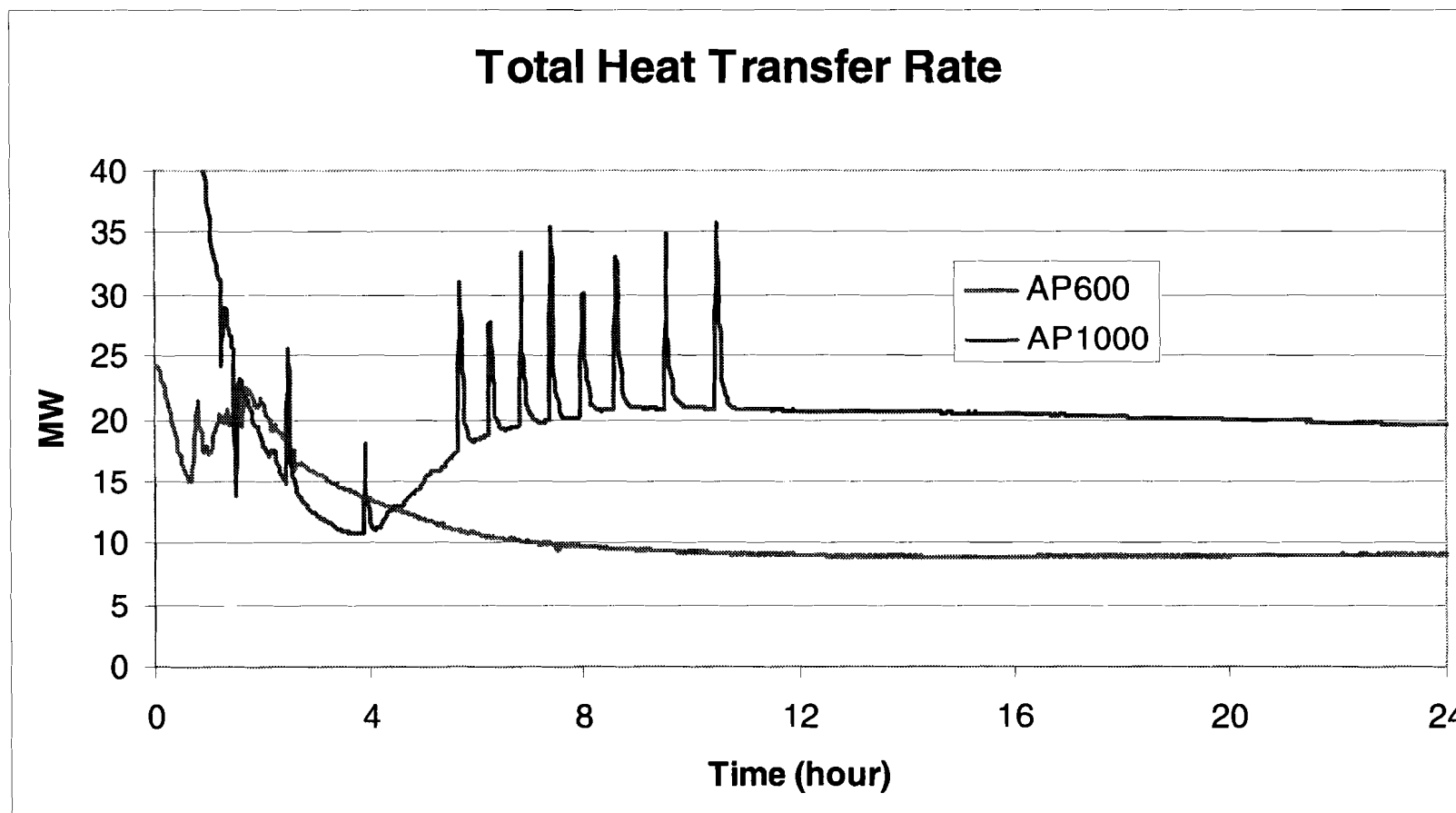
- containment pressure
- containment gas temperature
- containment steam mole fraction
- condensation rate on heat sinks
- total heat transfer to heat sinks

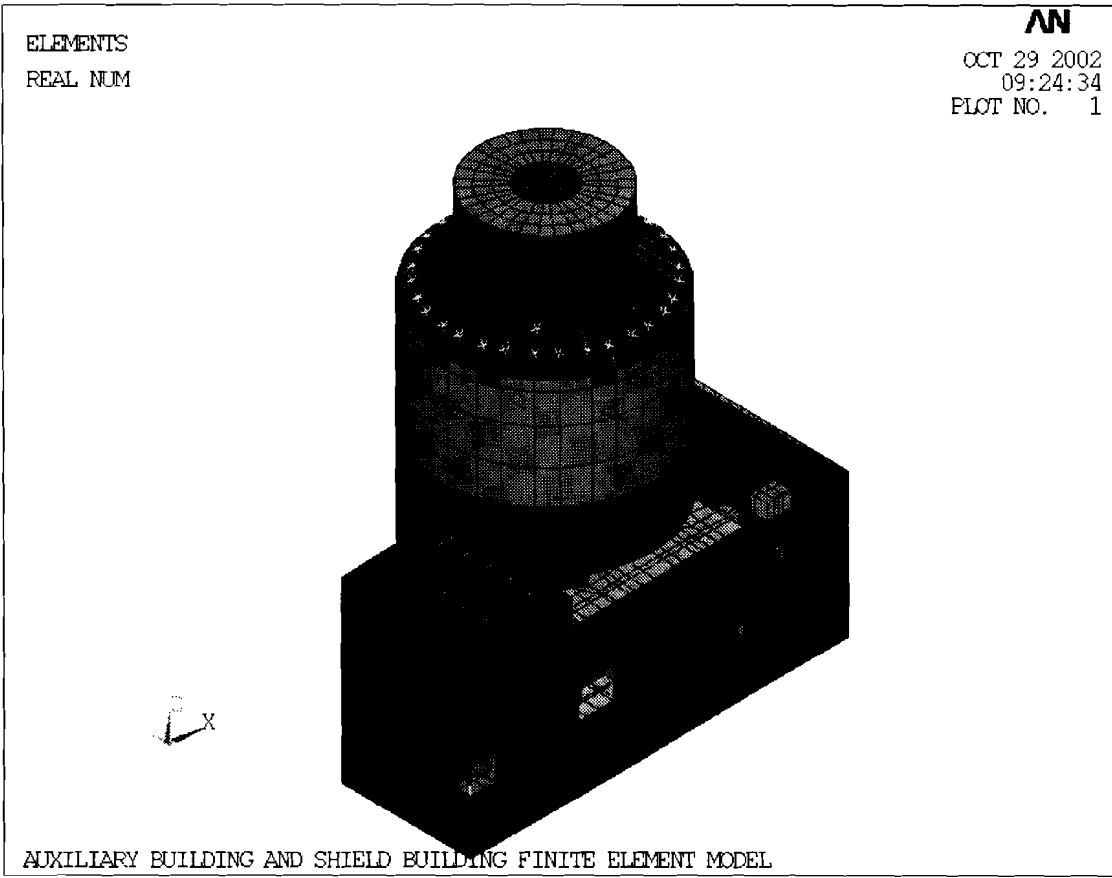












AP1000 Seismic and Structural Design

Richard Orr

Consulting Engineer, AP600 & AP1000 projects

412-374-5924 - orrs@westinghouse.com



Outline

- **Structural Configuration Changes from AP600**
- **Seismic Design Basis**
- **Seismic Analyses of Nuclear Island Structures**
- **Structural Design of Critical Sections**
- **NRC Staff Review and DSER Open Items**
- **Seismic Margins**

AP1000 vs AP600 Structures

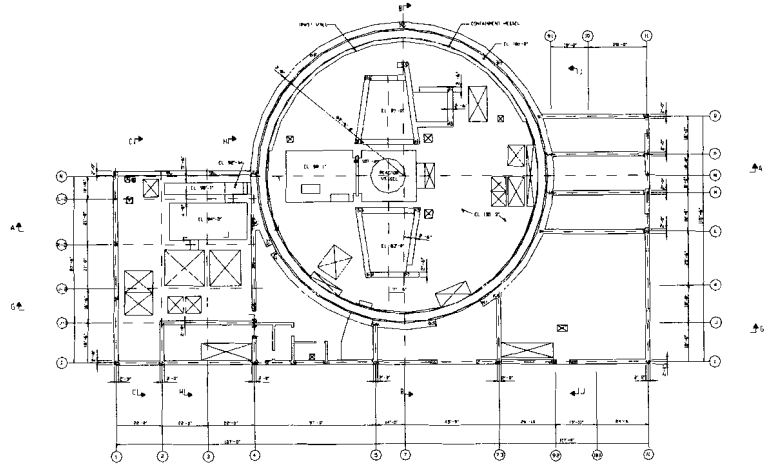
- **Shield building raised by 25'6"**
- **PCS tank capacity increased to 800,000 gal.**
- **PCS air inlets reconfigured to 12' x 6.5'**
- **Containment vessel raised by 25'6"**
- **Polar crane raised and capacity increased**
- **RCS equipment increased in size**
- **Steam generator and pressurizer compartment walls raised**
- **Fuel pit floor elevations lowered by 18.5"**

AP1000 Structural Arrangement

AP1000

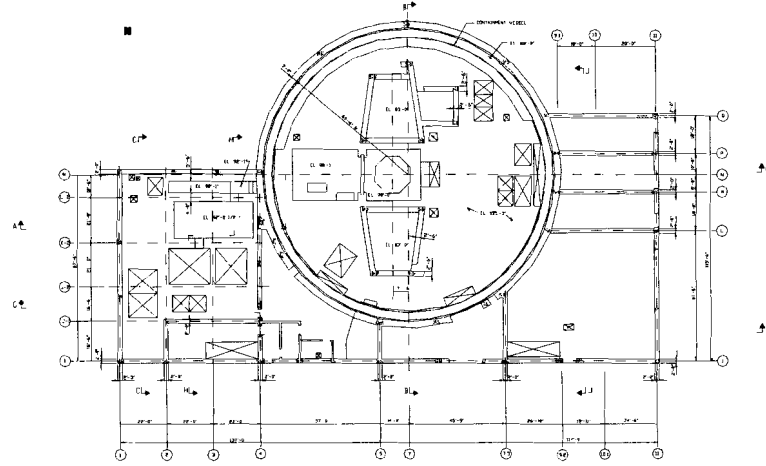
Plan at Elevation 135'

AP600



AP600

AP1000



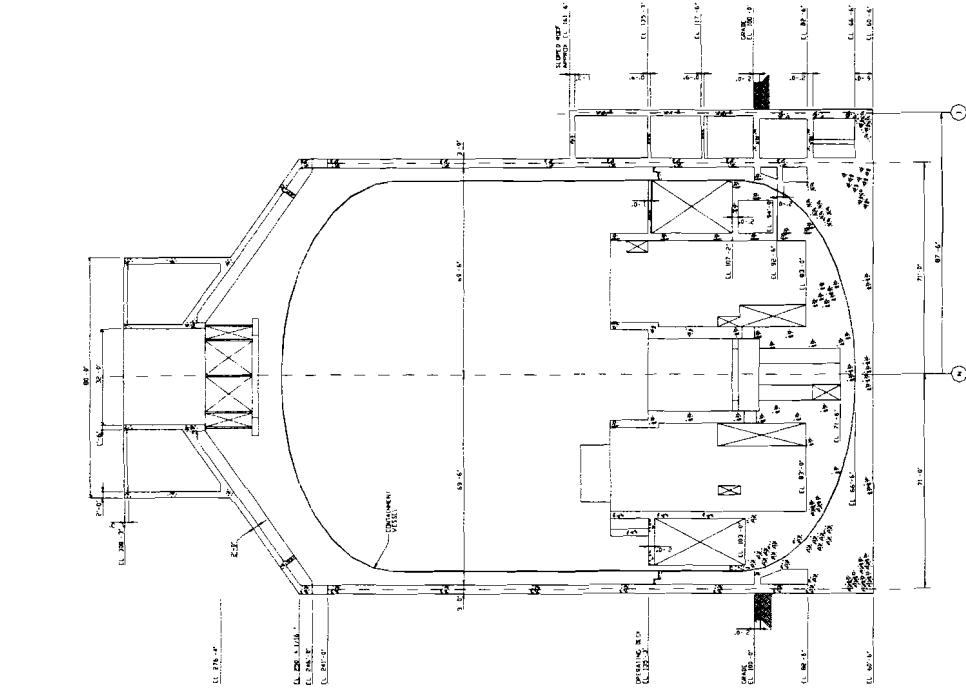
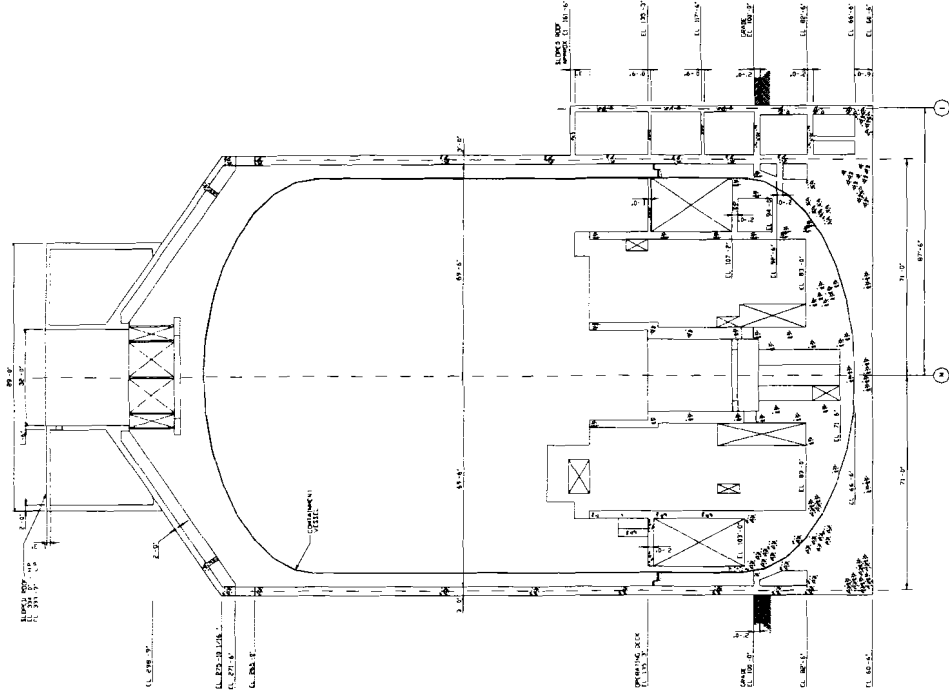
AP1000

EL. 135'-3"

AP1000 Structural Arrangement

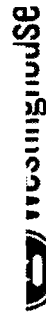
AP1000

Containment Section View



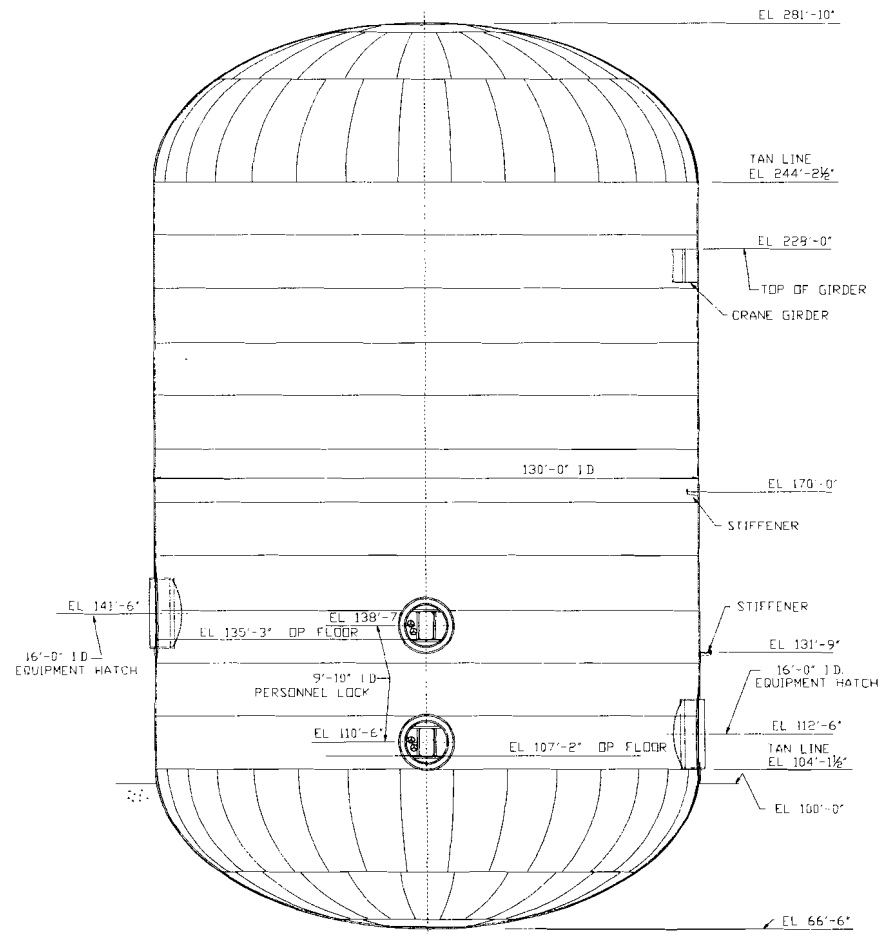
AP1000

AP600



Containment Vessel General Outline

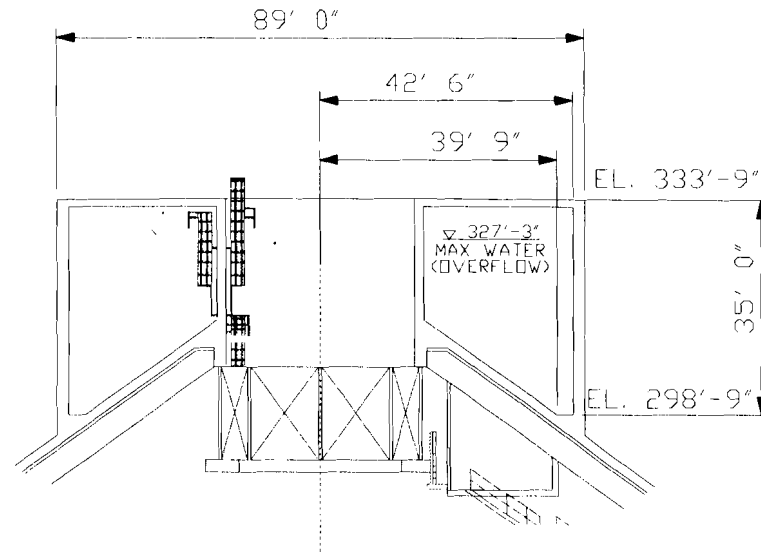
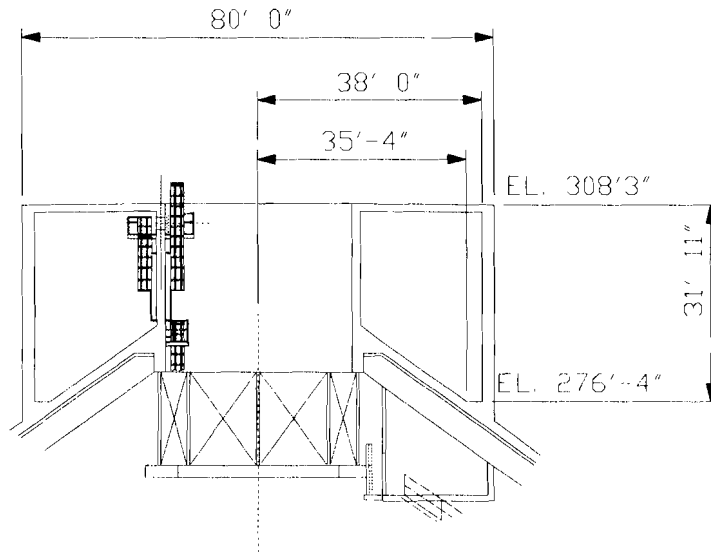
Diameter: 130 feet
 Height: 215 feet 4 inches
 Design Code: ASME III, Div. 1
 Material: SA738, Grade B
 Design Pressure: 59 psig
 Design Temperature: 300°F
 Design External Pressure: 2.9 psid



PCS Water Storage Tank

AP600

AP1000

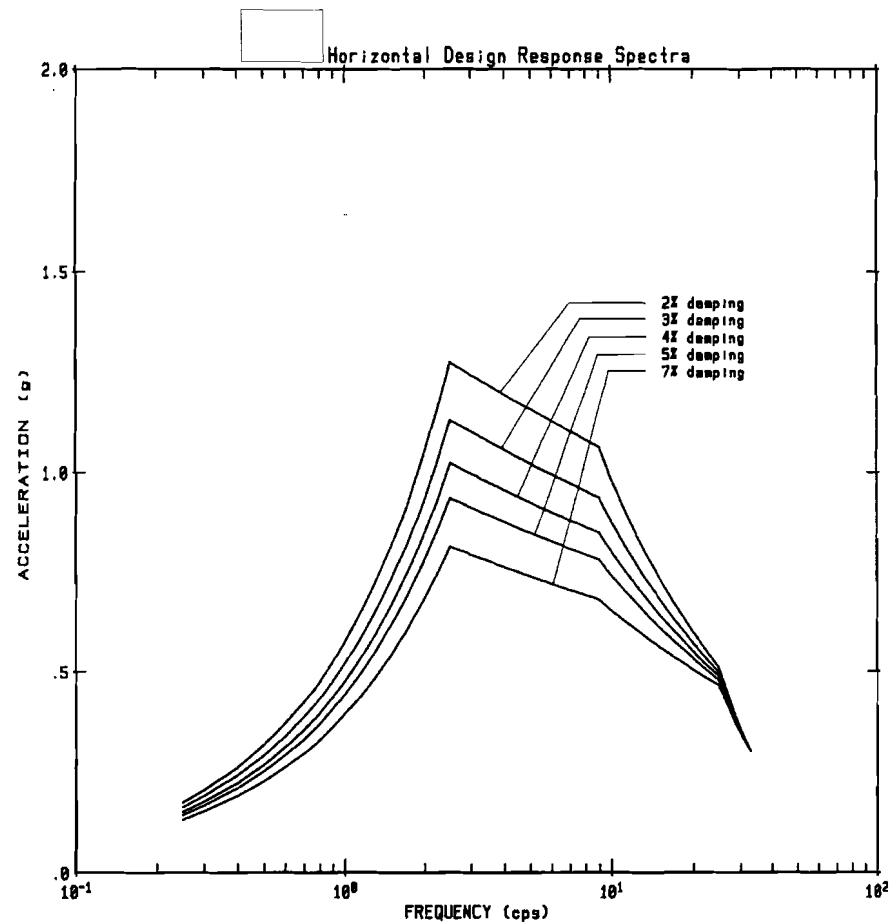


- **Passive Containment Cooling Water Storage Tank volume increased from 540,000 gallons to 800,000 gallons**



Seismic Design Basis

- 0.30 g SSE at foundation level
- Hard Rock foundation
- RG 1.60 response spectrum amplified at high frequencies

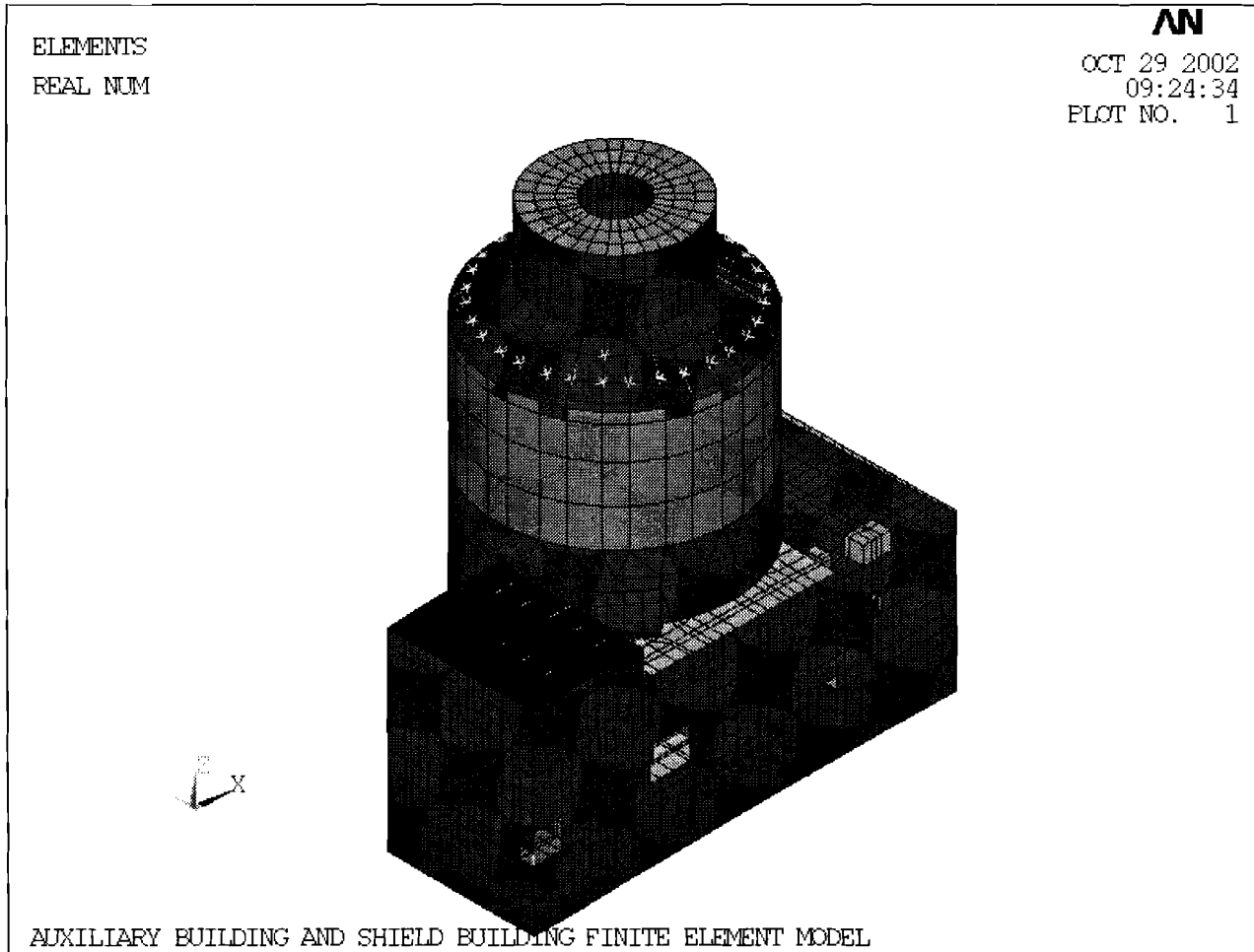


Nuclear Island Seismic Models

- **Finite element shell models of buildings for static analyses and for generation of simplified stick models**
- **Simplified finite element stick models of buildings for time history dynamic analyses**
 - Auxiliary and shield building
 - Containment Internal Structures
 - Containment vessel and polar crane
 - Reactor coolant loop

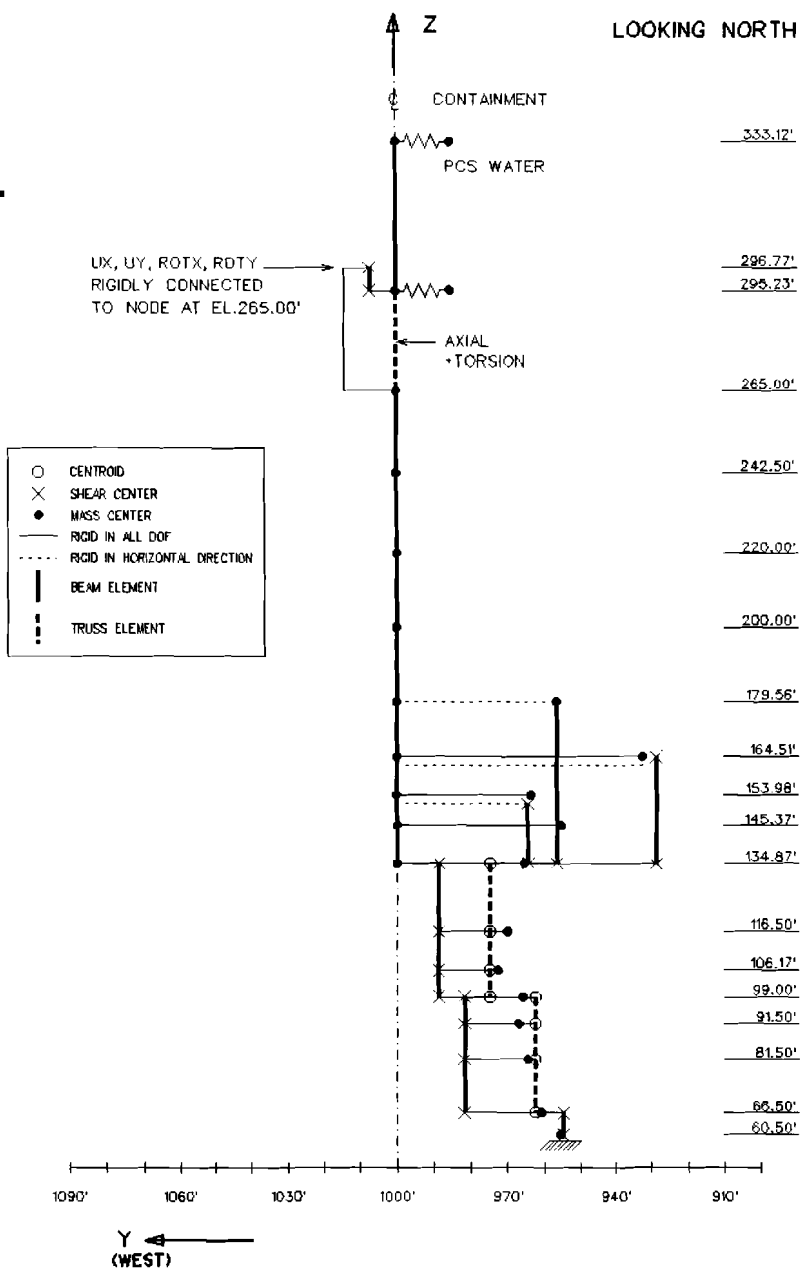


Finite Element Model Of Nuclear Island

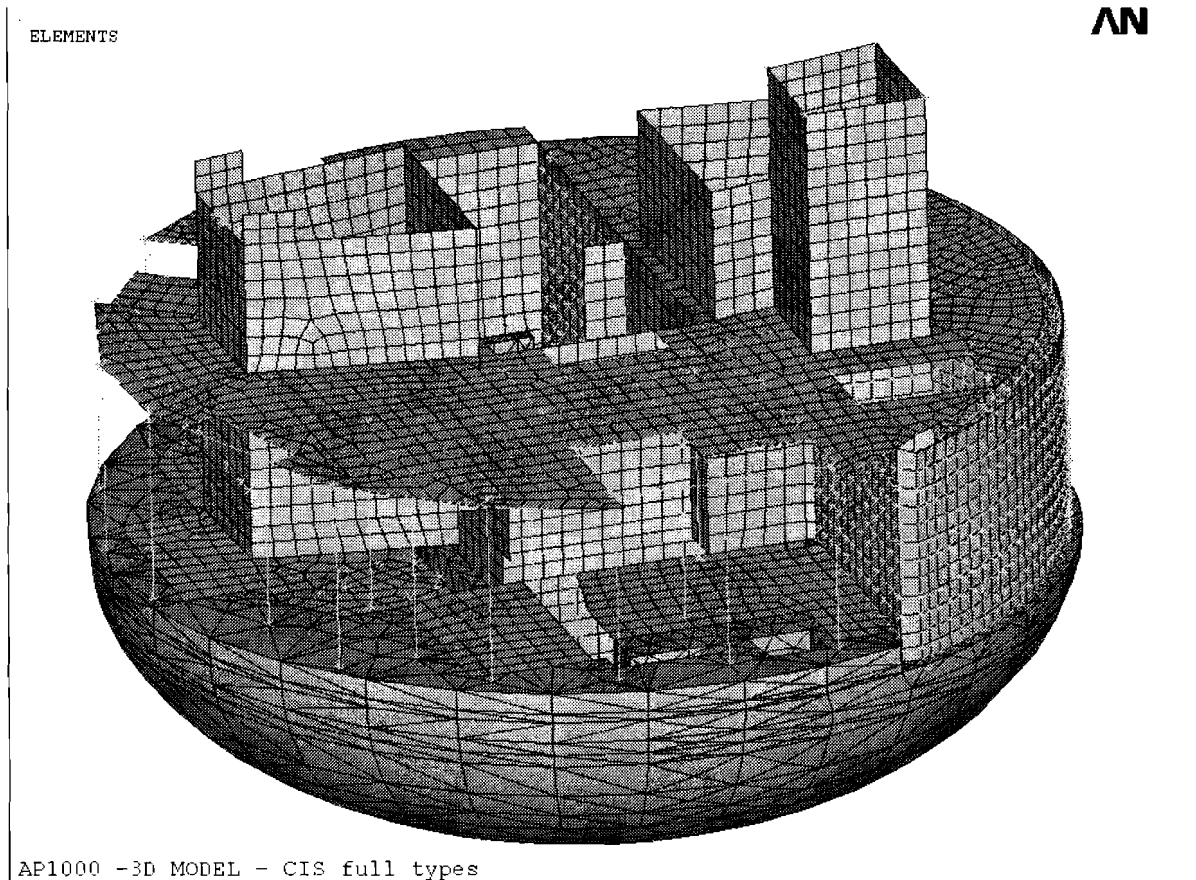


ASB Stick Model

AP1000



CIS Finite Element Model



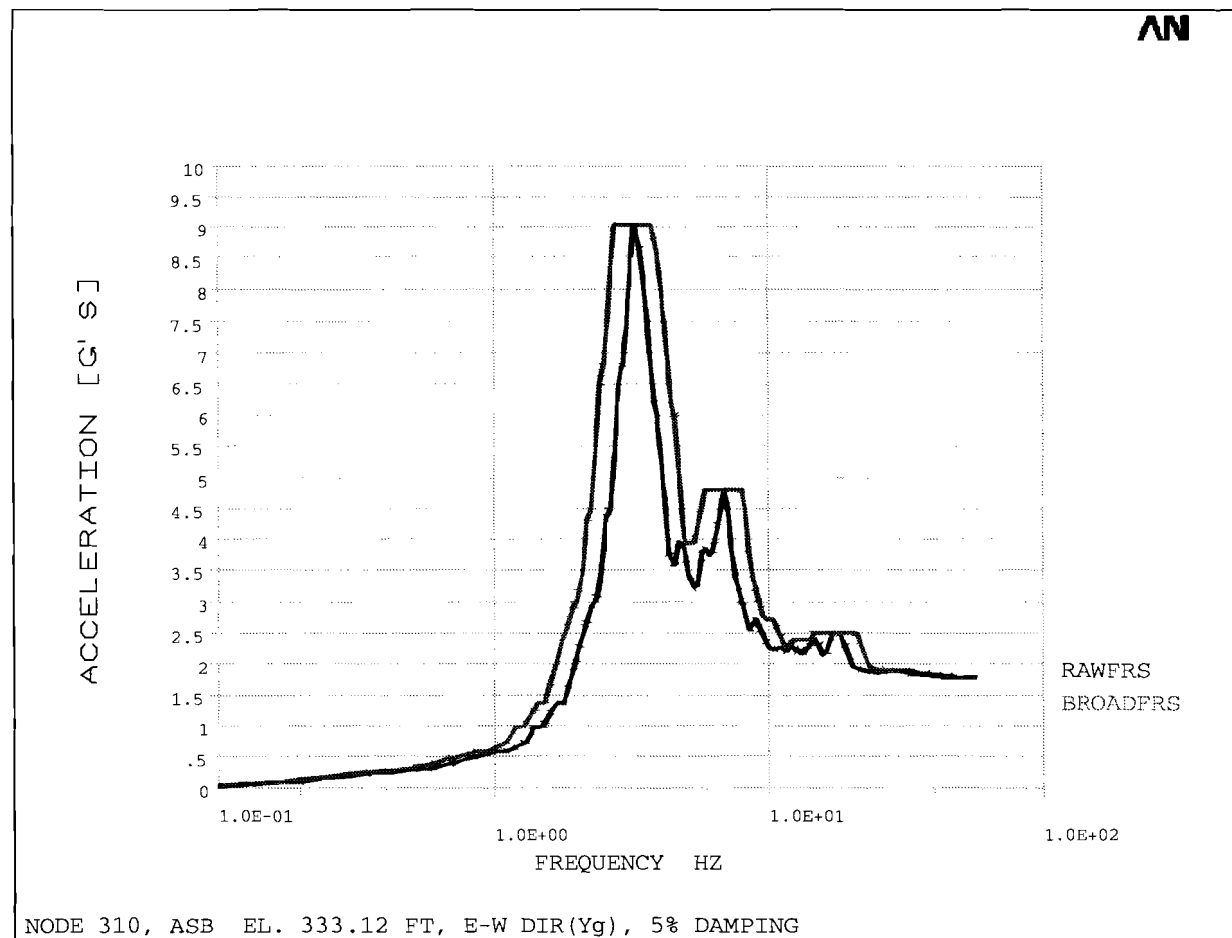
Model Without the NI Basemat Looking from North

Nuclear Island Seismic Analyses

- **Modal analyses of finite element shell models to obtain dynamic properties**
- **Time history dynamic analyses of stick models to obtain accelerations and floor response spectra. Concrete properties account for limited amount of cracking.**
- **Static analyses of shell models to obtain member forces in walls and floors.**

Typical Floor Response Spectra Top of Shield Building Roof

AP1000

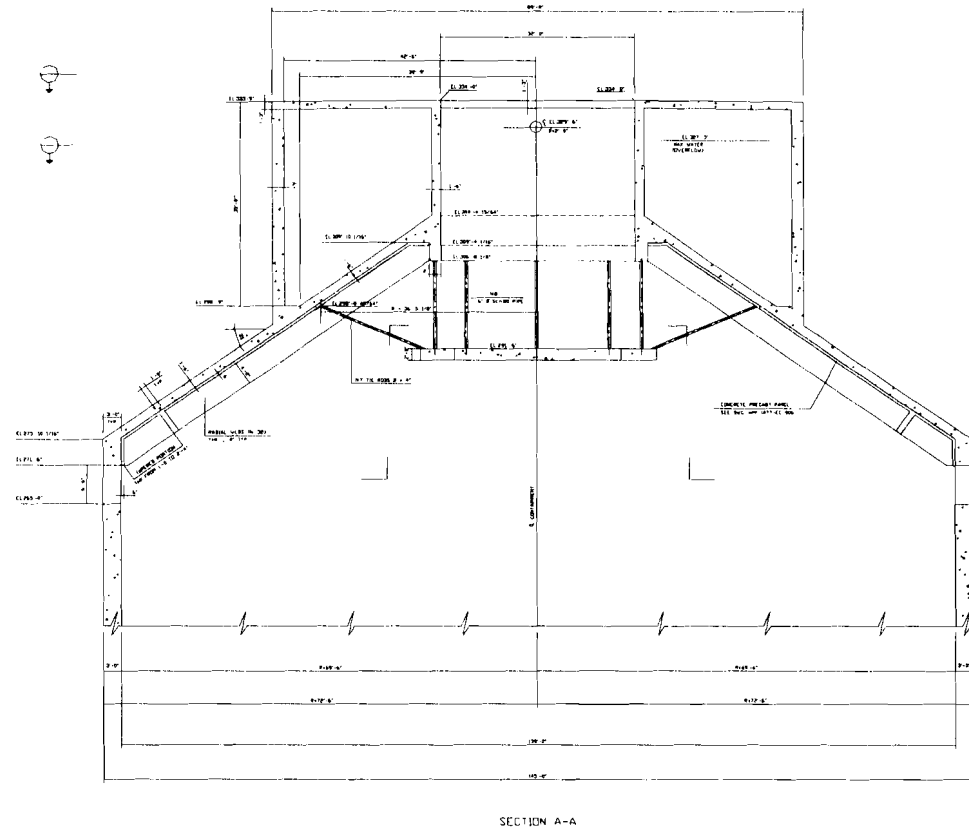


Structural design of critical sections

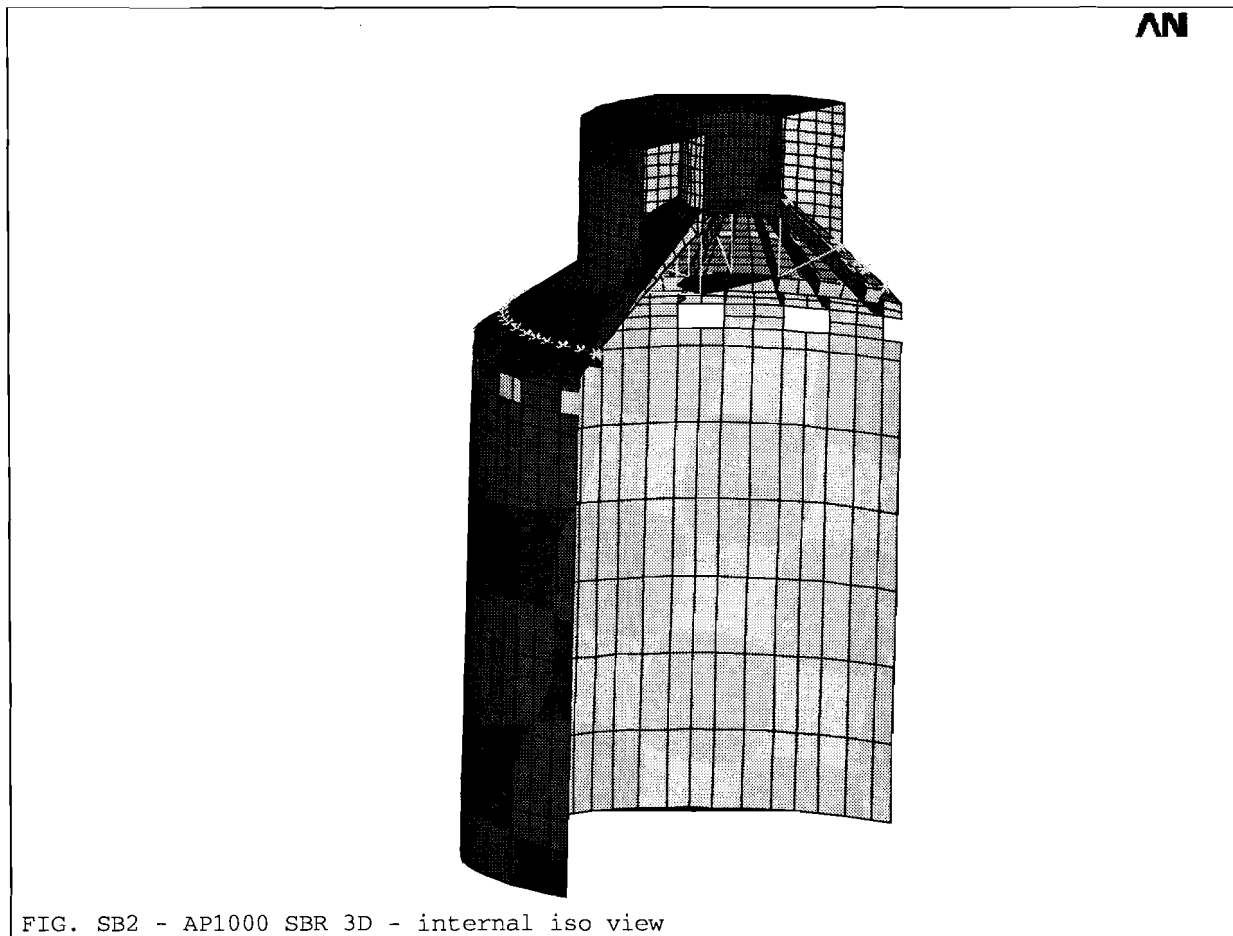
Detail design calculations prepared and reviewed by NRC staff for 22 critical sections

Equivalent static accelerations are applied to detailed finite element structural models

Element member forces are used to size reinforcement



Shield Building Roof 180 Degree Model



APP-1277-S2C-002

NRC Review

- **At conclusion of the pre-application review Westinghouse agreed on level of detail comparable to AP600 for seismic analyses and structural design.**
- **Meetings were held in November, 2002 and April, 2003 with NRC staff to review RAI responses and to audit the seismic analyses and structural design described in DCD section 3.7 and 3.8.**

DSER Open Items

- **DSER has:**
 - 5 Open Items on COL Information for Geotech. (DCD section 2.5),
 - 7 Open Items on seismic analysis (DCD section 3.7) and
 - 14 Open Items on structural design.
- **Westinghouse has transmitted additional information related to all the DSER Open Items.**
- **Seismic analyses were rerun to use a reduced concrete stiffness. Results were updated in DCD Revision 6.**



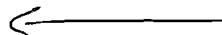
DSEER Open Items (continued)

- **Design calculations for critical sections are being reconciled to the results of the updated seismic analyses.**
- **Additional detail analyses have been completed for the containment vessel and will be reviewed by NRC.**

Seismic Margins

- **High Confidence Low Probability of Failure (HCLPF) values are calculated for structures, systems, and components**
 - **Buildings/structures**
 - **Shield Building**
 - **Containment Vessel**
 - **Interior Containment Structure & IRWST Tank**
 - **Primary Components and Support**
 - **Mechanical Equipment**
 - **Valves**
 - **Electrical Equipment**

Seismic Margins - HCLPF

- Seismic Margins (HCLPF) are generally slightly lower than the AP600 plant since the design is similar and the AP1000 seismic response is higher
- All HCLPF values are above the Review Level Earthquake of 0.5g  (Fuel)?

Seismic Margins - COL Actions

- **Demonstrate that the seismic response is equal to or less than that used to calculate HCLPF values**
- **Confirm the use of seismically robust electro-mechanical relays**
- **Confirm that the as-built plant conforms to the design**
- **Perform a verification walkdown identifying any differences in the as-built conditions from the design and ensuring that no additional vulnerabilities were created**

Seismic Margins – On Going Actions

- **Responding to DSER Open Items**
- **Reconciling Seismic Margin HCLPFs against seismic analysis results included in DCD Revision 6**
- **Reconciling Seismic Margin HCLPFs against structural design changes included in DCD Revision 6.**

AP1000 Containment Recirc Screen Performance

Terry Schulz

Advisory Engineer

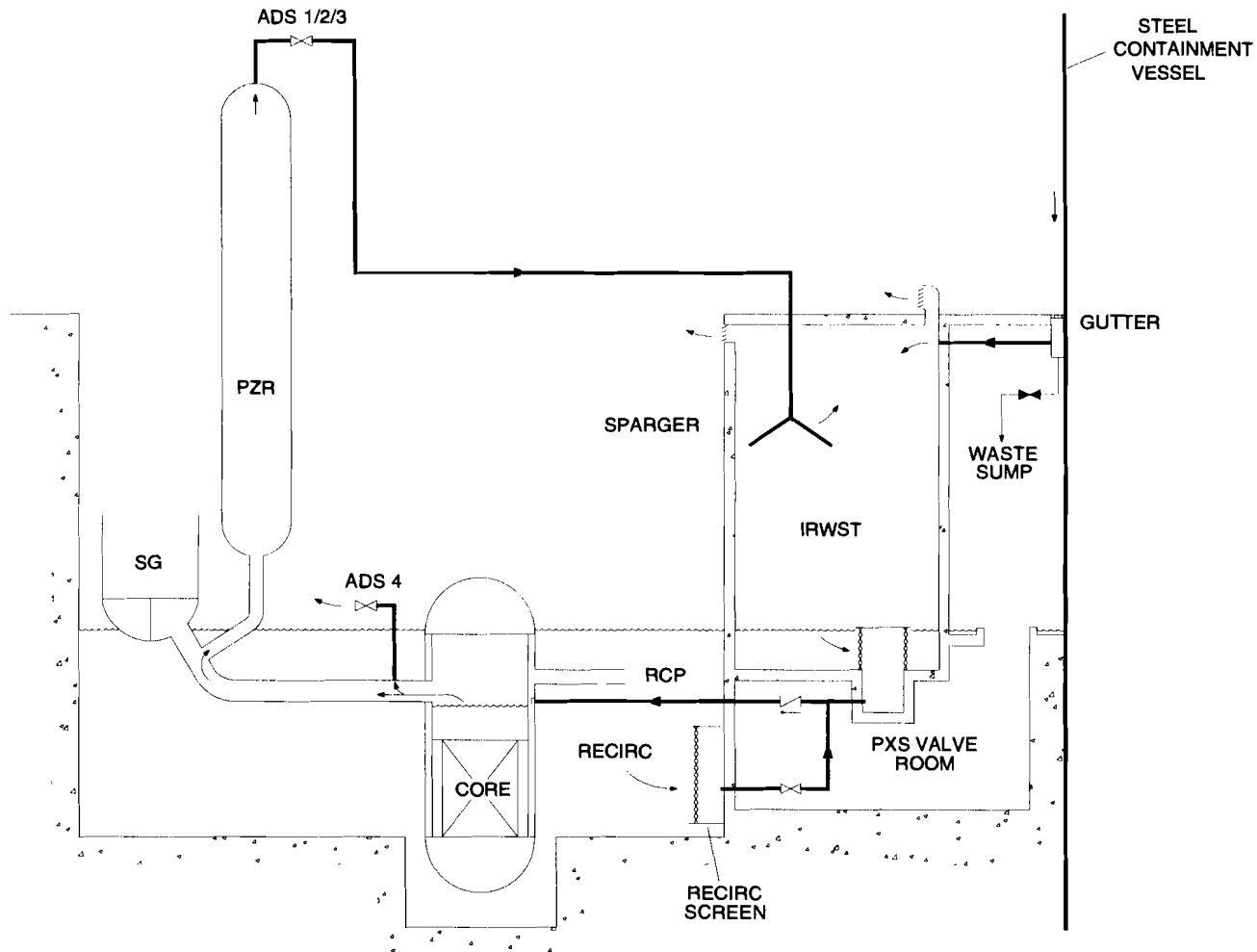
412-374-5120 - schulztl@westinghouse.com



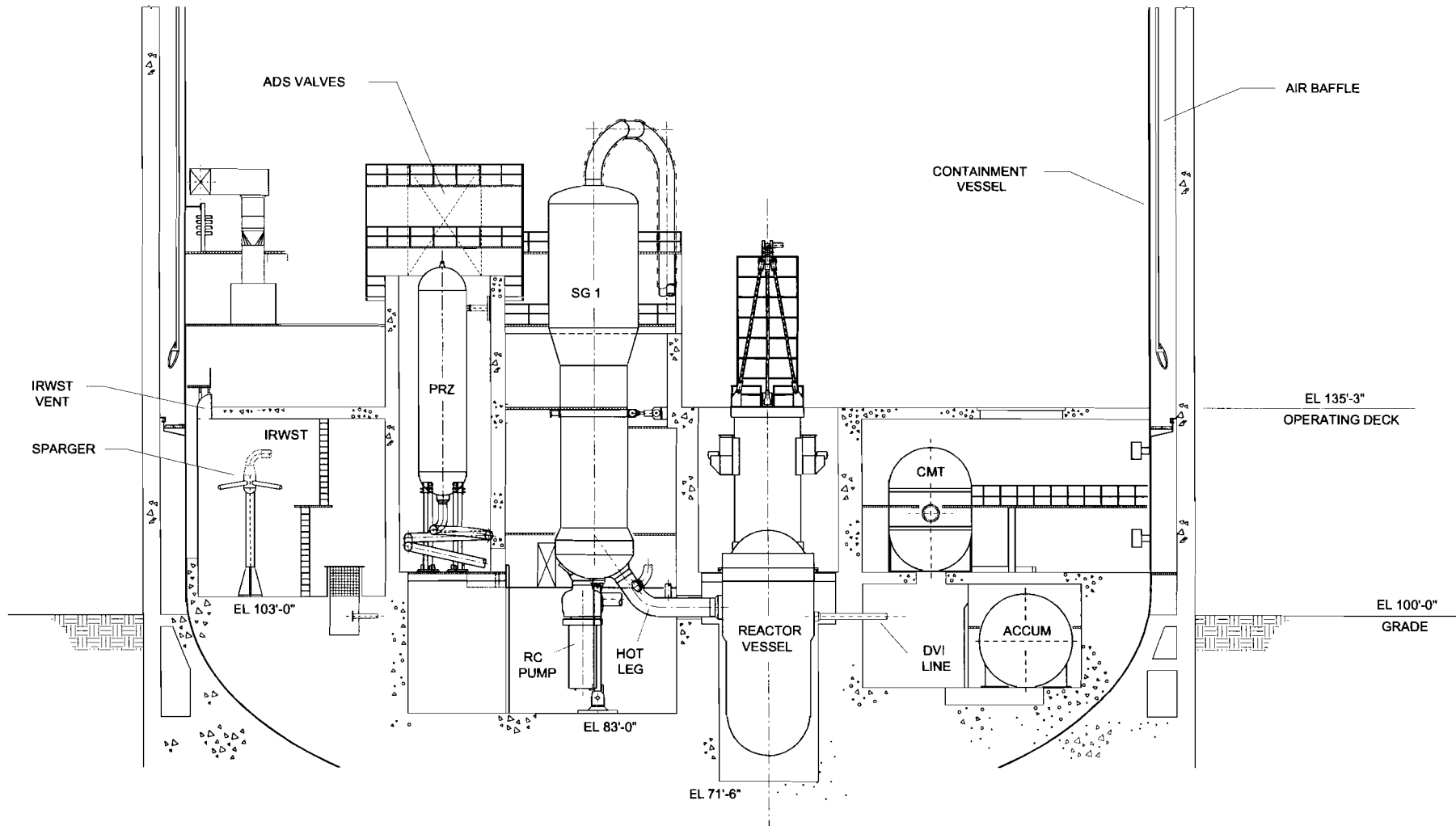
Containment Recirc. Screens

- **AP1000 Has Robust Containment Recirc. Screen Design**
 - Favorable post-LOCA conditions
 - Long times (2 - 4 hr) before recirculation starts, time for debris settling
 - Deep floodup levels (24')
 - Tall screens (13')
 - Bottom of screen well above floor (2')
 - Lower recirc flow rates (85% less than current plants)
 - Protective plates provided over screens
 - Debris settles out before reaching screens
 - No fibrous debris generated by LOCA conditions
 - Metal reflective insulation used in LOCA blowdown damage zone
 - Coating debris may be generated, but will settle out
 - High density nonsafety coatings specified inside containment

LOCA Long Term Cooling

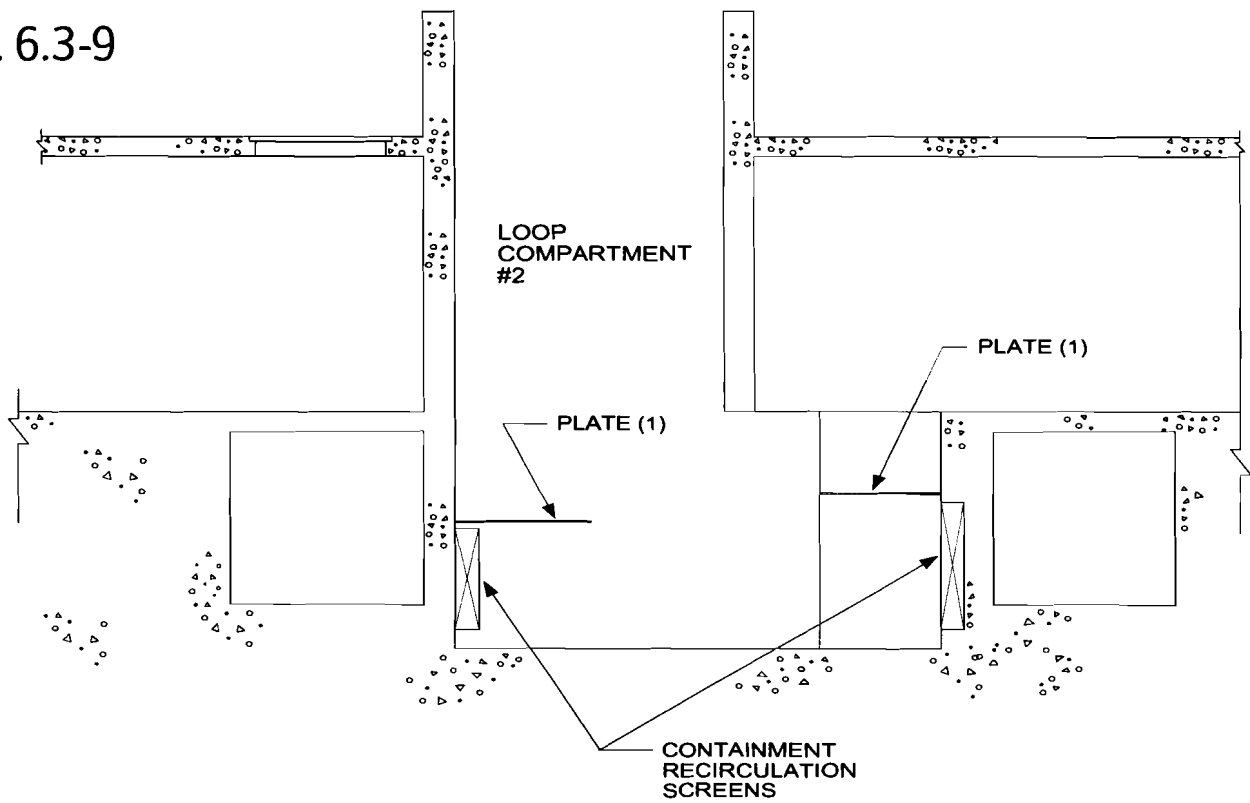


PXS Equipment Layout



Recirc Screen Section View

DCD Fig. 6.3-9



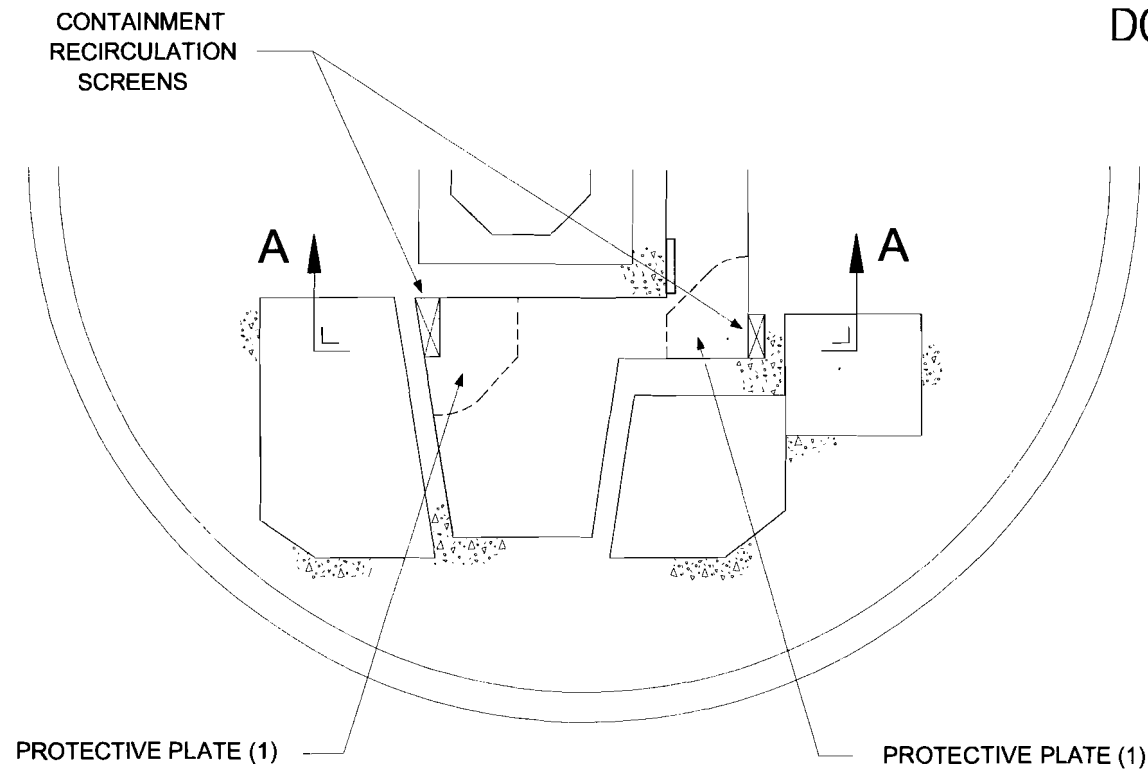
SECTION A-A

NOTE 1 - MINIMUM PLATE SIZE AND ELEVATION LIMITS
ARE DEFINED IN SUBSECTION 6.3.2.2.7.1.



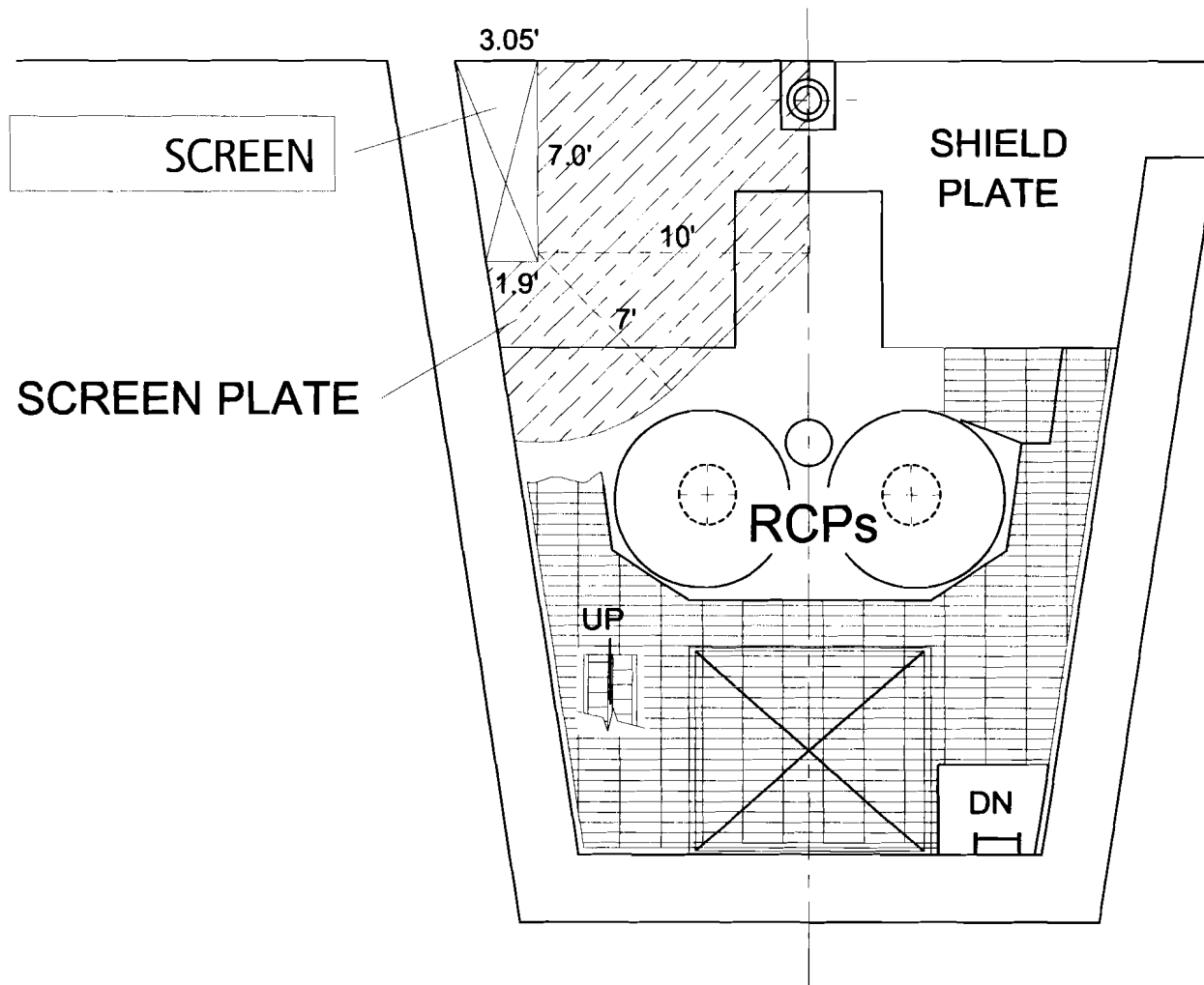
Recirc Screen Plan View

DCD Fig. 6.3-8

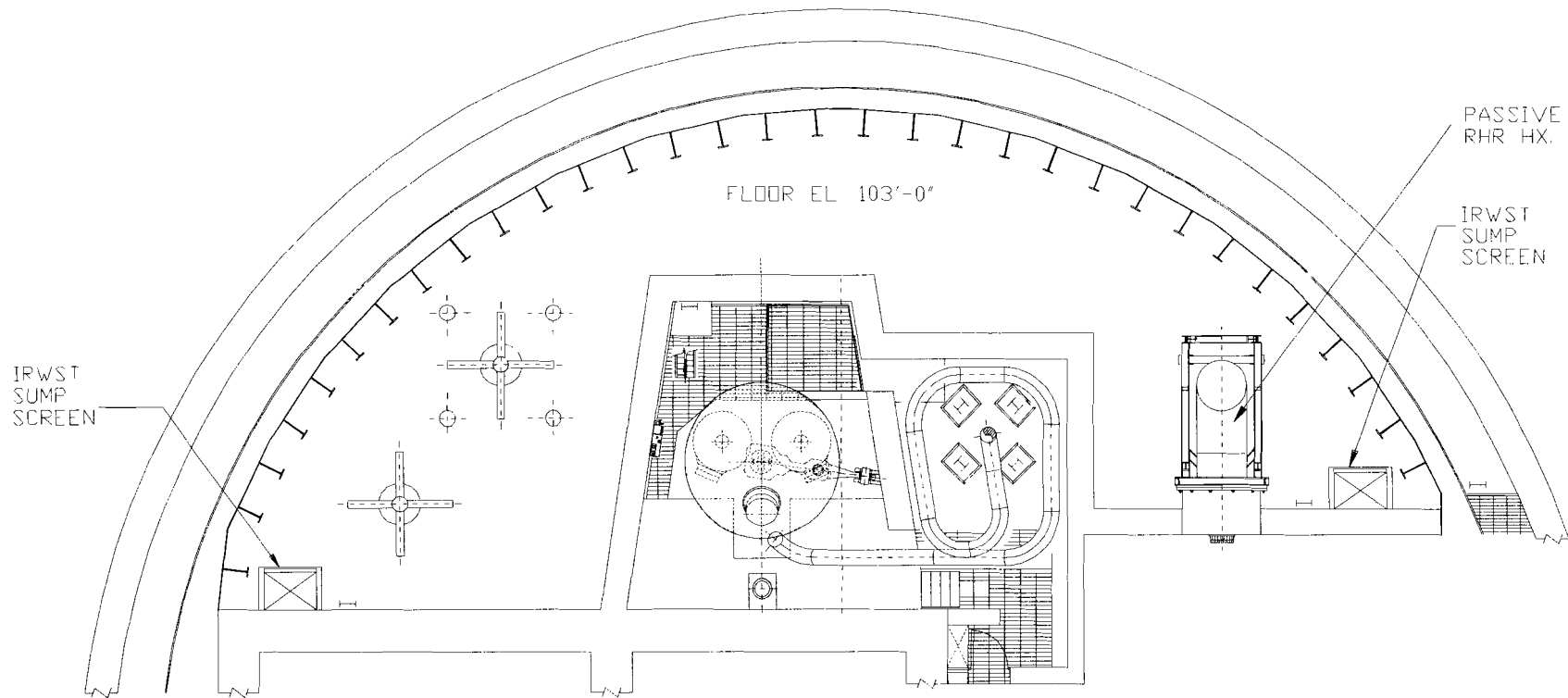


NOTE (1) MINIMUM PLATE SIZE AND ELEVATION LIMITS ARE DEFINED IN SUBSECTION 6.3.2.2.7.1

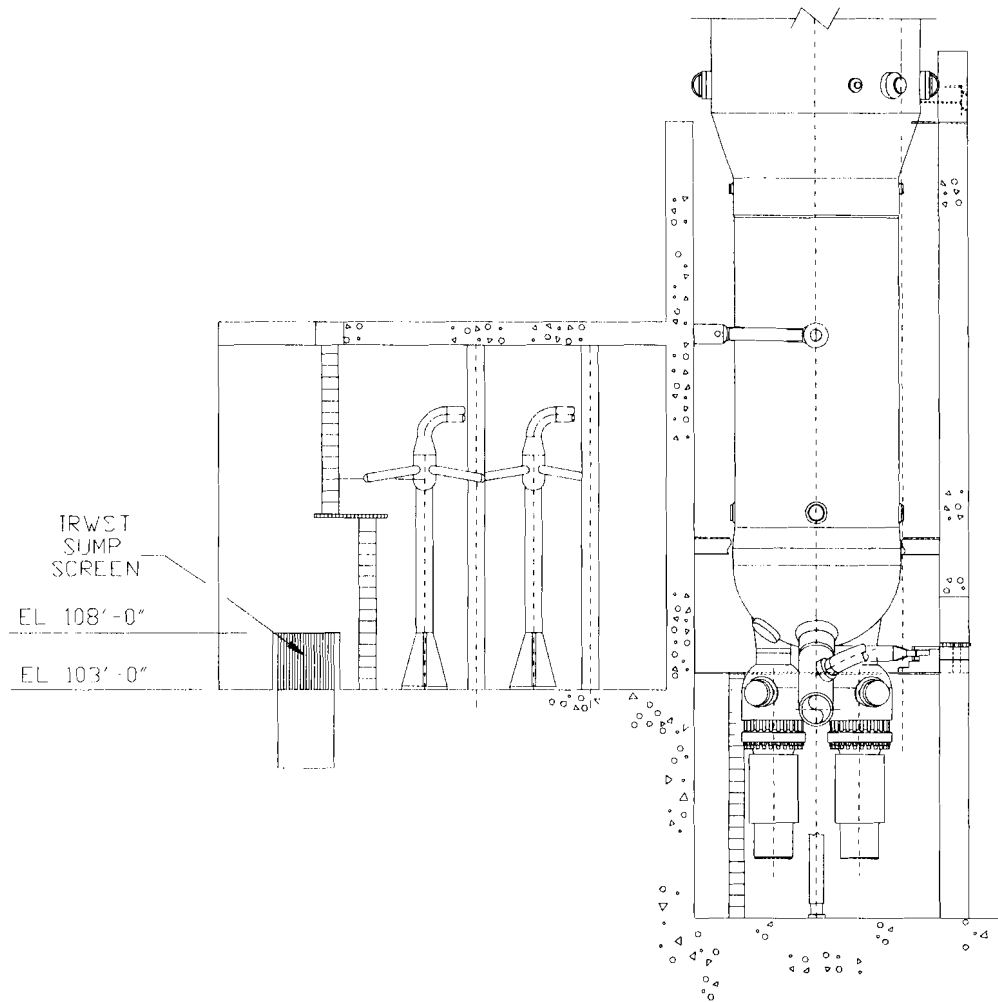
Recirc Screen Plate



IRWST Screen, Plan



IRWST Screen (Section)



Screen Comparison

	Typical PWR with HHSI, LHSI, spray pumps	AP1000 with RNS with PXS Pumps	
Screen area (ft ²)	200	70	
Screen height (ft)	6	13	
Bottom screen above floor (ft)	0.5	2	
Flood depth above floor (ft)	7	25	
Flood depth above top screen (ft)	1	10	
Time to start of recirc (hr)	< 1	4 (2 DVI)	
Max recirc flow (gpm)	10,000	1,600	1,330
Water velocity			
- at screen face (ft/sec)	0.11	0.052	0.023
- 10 ft from screen (ft/sec)	0.59	0.007	0.003

↑
0

Fibrous Debris

- **No Fibrous Insulation Debris Generation**
 - Use metal reflective insulation subject to LOCA blowdown damage
- **No Other Fibrous Debris Generation**
 - Fire barriers utilize steel/cement composite panels
 - Any fibers will be bound to high density steel or cement and settle out
- **Evaluated Impact of 'Resident' Debris**
 - Limited by COL cleanliness program
 - Performed bounding evaluation
 - Impact on PXS performance is acceptable

Resident Debris Evaluation

- **Conservative Assumptions**

- Total amount of resident debris is 500 lb
 - Half fiber and half particle
- All resident debris is transported to screens
- Conservative DP calculation
 - NUREG-6224, "Parametric Study of the Potential for BWR ECCS Strainer Blockage Due to LOCA Debris Generation" z

- **Evaluated Impact on**

- IRWST screens
- Containment recirc screens
- Core (assuming debris enters RCS via flooded break)

Resident Debris Evaluation

● IRWST Screens

- 500 lb debris split between both screens (DVI LOCA)
- Calc DP is 0.24 psi with 1170 gpm flow through screen connected to intact DVI line
- DP represents a 4% increase in the injection line DP, insignificant

*Not in
NOTRUMP
analysis*

● Containment Recirc Screens

- 500 lb debris applied to one screen (DVI LOCA)
- Calc DP is 0.58 psi with 1300 gpm flow through screen
- DP represents a 20% increase in the recirc line DP, recirc flow could decrease 10%
 - Use of best estimate line resistance would compensate for debris

● Core

- Some neutrally buoyant fibrous debris may enter RCS through break and collect on lower core support plate or in the fuel grids
- 500 lb debris split between recirc screens (40%) and core (60%)
- Calc DP is ~ 1 psi across core debris at 1300 gpm
- Will raise downcomer level ~ 29 in, still below DVI connection

→ GET-191

Coatings Used Inside Containment

- **Coatings Expected to Remain In Place**

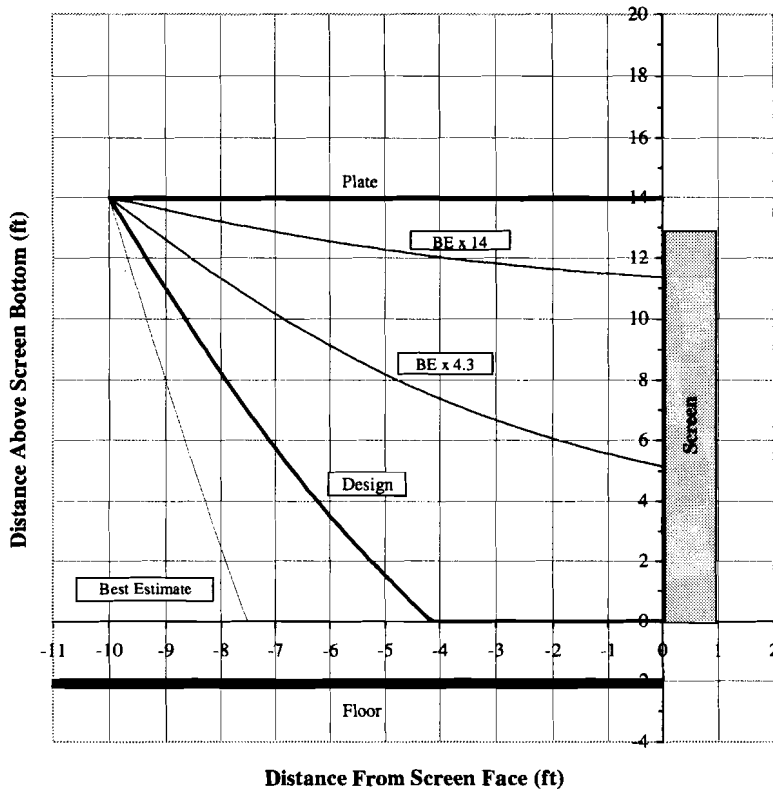
- Coatings will be qualified for post accident conditions
- Application and inspections will not be safety classified

- **Failure of Coatings Can Be Tolerated**

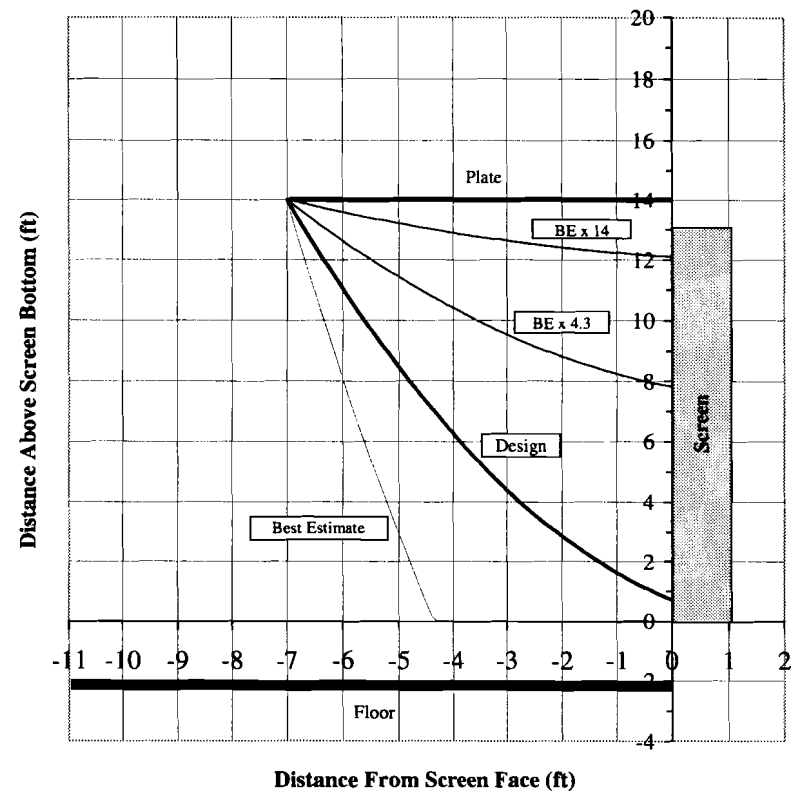
- Coatings are specified to be high density ($> 100 \text{ lb/ft}^3$ dry film)
- Will settle out in AP1000 post LOCA conditions
 - Long injection time (delayed recirc time)
 - Low velocities
 - Tall screens
 - Screen protective plates

Corridor Screen Performance

Case 1 - Higher Flow From Front



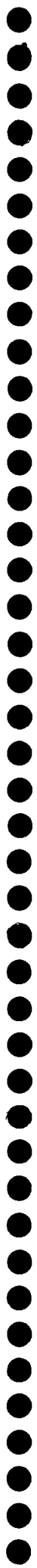
Case 2 - Uniform Flow (Side View)



Summary

- **AP1000 Has Robust Contain Recirc Screen Design**

- Delayed recirc, deep flood levels, low velocities
- Tall recirc screens located well above floors
- Protective plates provided above recirc screens
- Use of metal reflective insulation
 - In LOCA blowdown damage zone
- High density, nonsafety coatings
- Plant can tolerate 'resident' debris on screens
 - Assuming conservative, bounding evaluation



AP1000 Draft Safety Evaluation Report Open Items Related to LBB

**Presentation to ACRS
Pittsburgh, PA**

Warren Bamford

AP1000 Piping Design

- **AP1000 will use piping Design Acceptance Criteria in lieu of detailed piping design and analysis**
 - Same approach as ABWR and System 80+
 - AP1000 piping configuration based largely on AP600
 - Line routings are the same
 - Some sizes changed
- **Final piping design and analysis is completed during COL stage**
- **Final piping design and analysis verified by ITAAC during construction**

AP1000 Draft Safety Evaluation Report

- **Two DSER Open Items Related to LBB**

- 3.6.3.4-1
- 3.6.3.4-2

- **PWSCC Open Item 3.6.3.4-1**

- Requests W to include Combined Operating License applicant commitment to implement inspection plans, evaluation criteria, and other types of measures imposed on or adopted by operating PWRs with currently approved leak-before-break (LBB) applications as part of the resolution of concerns regarding the potential for PWSCC in those units
- Westinghouse has incorporated the COL item in the AP1000 DCD

AP1000 Draft Safety Evaluation Report

- **Open Item 3.6.3.4-2**

- Requests sensitivity studies be performed to address uncertainties related to PWSCC and the possible impact on LBB piping
 - Evaluate TGSCC crack morphology parameters as a surrogate for PWSCC and assess impacts on the LBB analyses

Westinghouse - NRC Recent Meeting

- Meeting held at NRC on 7-11-03
- Useful discussions held
- Each side presented ideas to resolve these issues
- Follow-up discussions planned
- Key issues will be discussed briefly here

Alloy 690, Alloy 52, Alloy 152 in AP1000

- In view of the continuing occurrence of primary water stress corrosion cracking [PWSCC] of Alloy 600, and its associated welds Alloys 82 and 182, the decision was taken to preclude use of these materials in the AP1000 design
- The materials selected for these applications are Alloys 690, 52 and 152, respectively
- The recent cracking experience in Alloy 600 and associated welds in operating PWRs therefore has no direct relevance to the AP1000

Alloy 690 - Historical Perspective

- Thermally treated Alloy 690 [A690 TT] was adopted as the preferred alloy for SG heat transfer tubing applications in 1986
- A690 TT also began service as mechanical SG tube plugs at approximately the same time
- Since the initial replacement SG startup at D.C. Cook Unit 2 in May 1989, A690 TT is now in service at more than fifty PWRs worldwide
- Applications of A690 TT have since been extended to include SG divider plates, pressurizer heater sleeve penetrations, RV head penetrations (including CRDM pipes), and other small-bore instrument penetrations

Alloy 690 - Experience (Cont'd.)

- Several of the CE-repaired components, with A690 TT as the replacement material, have been in service since approximately 1989
- With over fourteen years of SG operating experience, at temperatures exceeding 328°C [622.4°F], and nearly sixteen years in pressurizer penetration applications at 343°C [650°F] there has not been a single incidence of environmental degradation of A690 TT

Alloys 52 and 152

- With the extension of A690 TT applications to SG divider plates, RV and pressurizer penetrations, and other applications requiring welding, the A690 weld metal analogs Alloys 52 and 152 have been widely deployed
- Alloy 52 is used for gas-tungsten-arc [GTA] or gas-metal-arc [GMA] welding; Alloy 152 is the stick electrode composition used for shielded metal-arc welding [SMAW]
- Alloys 52 and 152 contain the same nominal concentrations of Cr and Fe, with slightly less Ni - relative to Alloy 690

A52 and A152 - Applications in PWRs

- The earliest application of these weld metal alloys was in CE pressurizers in which partial penetration welds were used to complete the repairs; these applications extend as far back as early 1989
- Westinghouse replacement SGs at N. Anna 1 and V. C. Summer were the first units to employ large-scale use of A52 and A152
- These SG applications included safe end-to-nozzle welds, and welding of the divider plate and stub runner to the channel head
- The initial SG applications went into service in late 1993, accruing nearly ten years of service since that time

Alloy 52 & Alloy 152 - SCC Resistance

- Owing primarily to high Cr content, Alloys 690, 52, and 152 exhibit apparent immunity to primary water stress corrosion cracking (PWSCC)
- Service experience with Alloy 690 in SG heat transfer tubing applications, and Alloys 52/152 as buttering, cladding and weld filler materials has been exemplary, with no reported degradation
- Laboratory testing of each of these materials endorse the exceptional corrosion resistance - no known incidence of crack initiation or crack propagation in primary water environments in any of these materials

Alloys 52 & 152 - SCC Resistance (Cont'd.)

- The laboratory tests of these weld metals continues to support the concept of “immunity” to PWSCC
- Even specimens precracked in fatigue will not propagate; details of these tests have been provided in the Revision 1 response to RAI 251.004
- Alloys 52 & 152 have been used in operating PWRs for RV nozzle repairs at V.C. Summer and Ringhals 3 & 4
- The use of Alloy 52 for an embedded flaw weld repair of CRDM pipe degradation at N. Anna Unit 1 was approved in late 1992, and generically approved in July 2003.

Open Item 3.6.3.4-1 [PWSCC] - Conclusions

- Alloys 52 and 152 welds have been shown to exhibit excellent resistance to PWSCC, both in lab and field experience
- However, Westinghouse recognizes the reservations expressed by the NRC with respect to the limited [ca. 9.5 years] field experience
- Additional field experience - and laboratory evaluations currently underway - will accrue prior to final operation of AP1000
- Westinghouse remains confident this experience will validate the decision to extensively deploy these materials in the AP1000 primary system

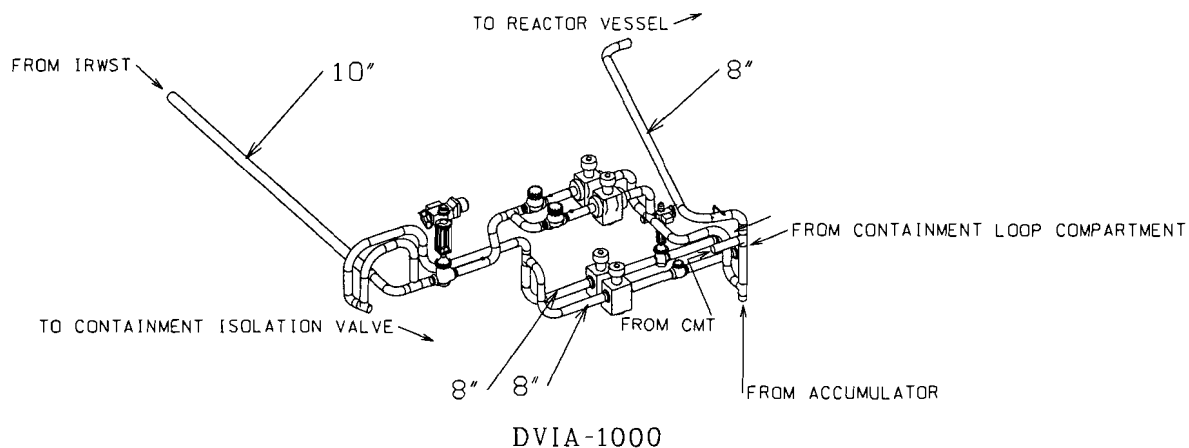
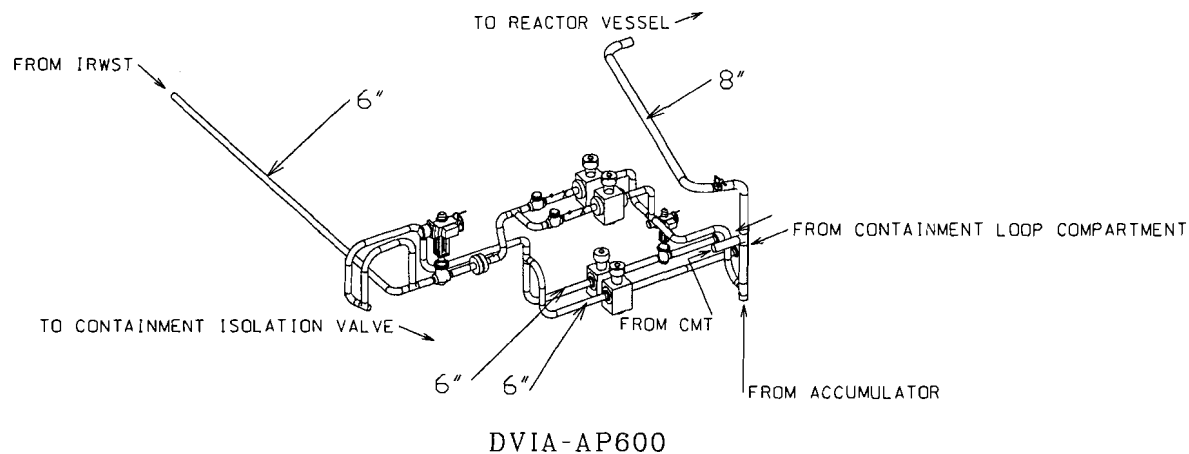
AP1000 Piping Systems Designed for LBB

- **AP1000 LBB Piping Systems are the same as those designed for AP600**
 - Some line sizes increased
 - Line routings the same
 - Stress analyses completed for AP600 demonstrate the feasibility that the AP1000 piping systems can be designed to meet bounding analysis curves

AP1000 Piping Systems Designed for LBB

- **Reactor Coolant Loop**
- **Pressurizer Surge Line**
- **Direct Vessel Injection Lines A & B**
- **Core Makeup Tank Inlet Lines A & B**
- **Passive RHR HX Return Lines**
- **ADS-1/2/3 Piping**
- **ADS-4 Piping A & B**
- **Normal RHR Piping**
- **Main Steam Lines A & B**

Comparison of IRWST Injection/DVI Line



AP1000 LBB Analysis Method

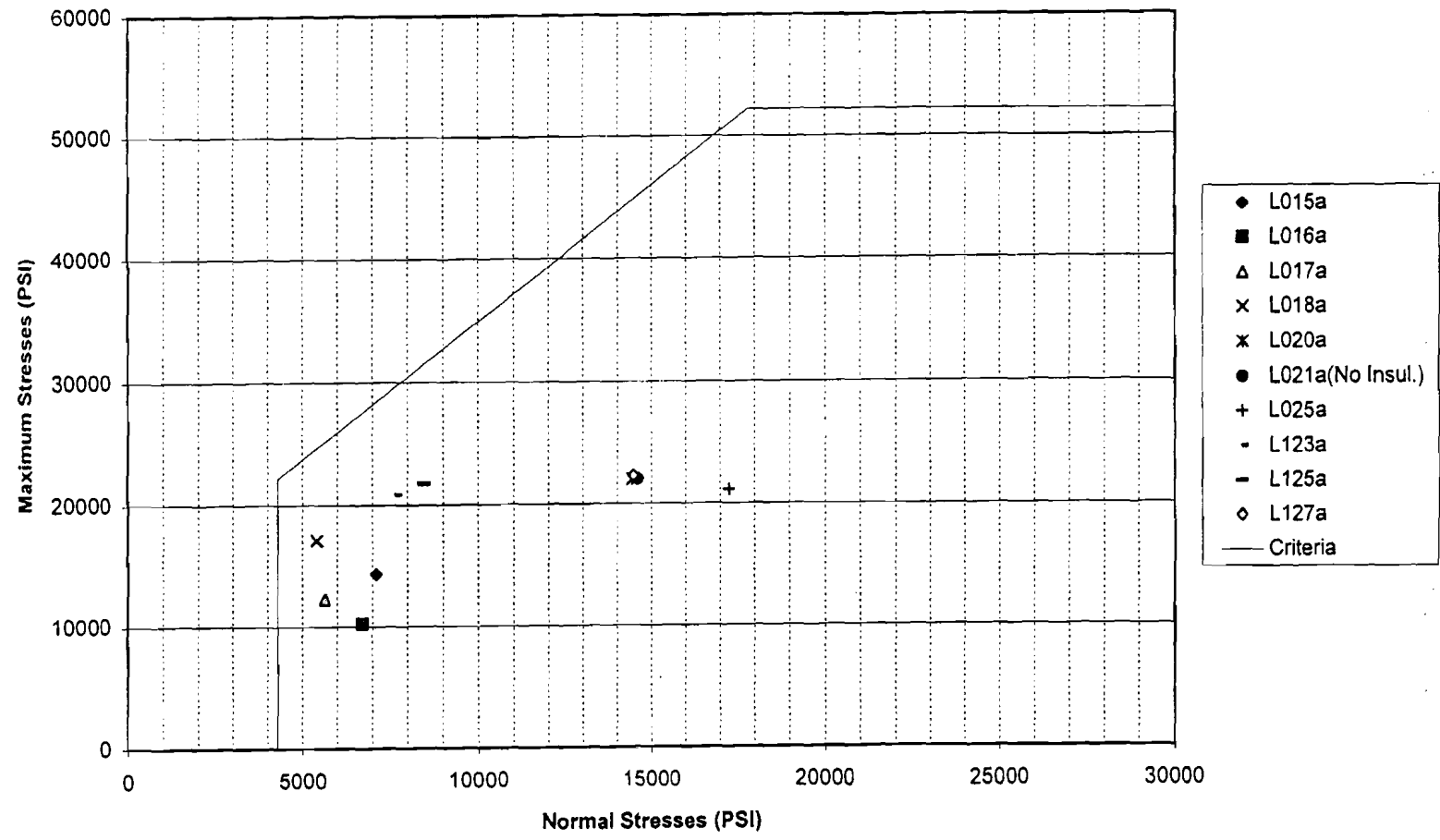
- **Leak-Before-Break Bounding Analysis Methods**
 - Develop Bounding Analysis Curves based on pipe material and pipe size
 - Bounding Analysis Curve Margins
 - Margin of 10 on leak detection capability
 - Margin of 2 on flaw size
 - Margin of 1 on load by using absolute summation method of maximum loads combination
 - Consistent with AP1000 DCD Appendix 3B
- **These methods and criteria were reviewed by NRC staff during an audit in Westinghouse in September 2002**

AP1000 Preliminary Stress Analysis

- **To resolve this issue, Westinghouse has proposed to complete a Preliminary piping stress analysis for NRC review**
- **DVI-A Piping Analysis Package**
 - Selected based on our experience with AP600
 - Difficult to qualify
 - Complicated piping system
 - Some piping sizes were changed
 - Contains smallest piping line qualified for LBB
 - Subcompartment pressurization impacts if line would not meet LBB criteria

Preliminary Results: An Example

Figure 8.6.1 AP1000 Bounding Analysis Curve for 8" CMT, DVI,IRWST
 (Line Numbers: L015A,016A,018A,020A,021A-No Insul.,025A,125A,127A)



Status of Discussions with NRC on LBB

- AP1000 Piping systems are very similar to AP600, which has been approved
- Evaluation of one AP1000 LBB piping system is currently in progress
- Discussions continue on the best way to ensure that Alloy 690, and Alloy 52 and 152 welds will be immune from SCC throughout the service lifetime of an AP1000

DSER Open Items Planned Resolution Paths

Mike Corletti

Deputy Project Manager

(412)374-5355;corletmm@westinghouse.com

DSER Open Items Already Discussed with ACRS

- T&H Issues
 - PRA
 - Seismic and Structural Design
 - Leak-Before Break
 - Sump Performance Issues
- Security
- Control Room X/Q
- 10 CFR 50.44
- ITAAC

DSER Open Item: Security

- **Security is largely the responsibility of the COL applicant**
- **Westinghouse has submitted a Security Assessment report to NRC**
 - Assess design features of AP1000 and their compliance to design-related requirements contained in the revised Design Basis Threat and Interim Compensatory Measures
 - AP1000 complies with applicable requirements
- **Review was delayed due to NRC staff resource issues in this area but review is now underway**

DSEER Open Item: Control Room X/Q

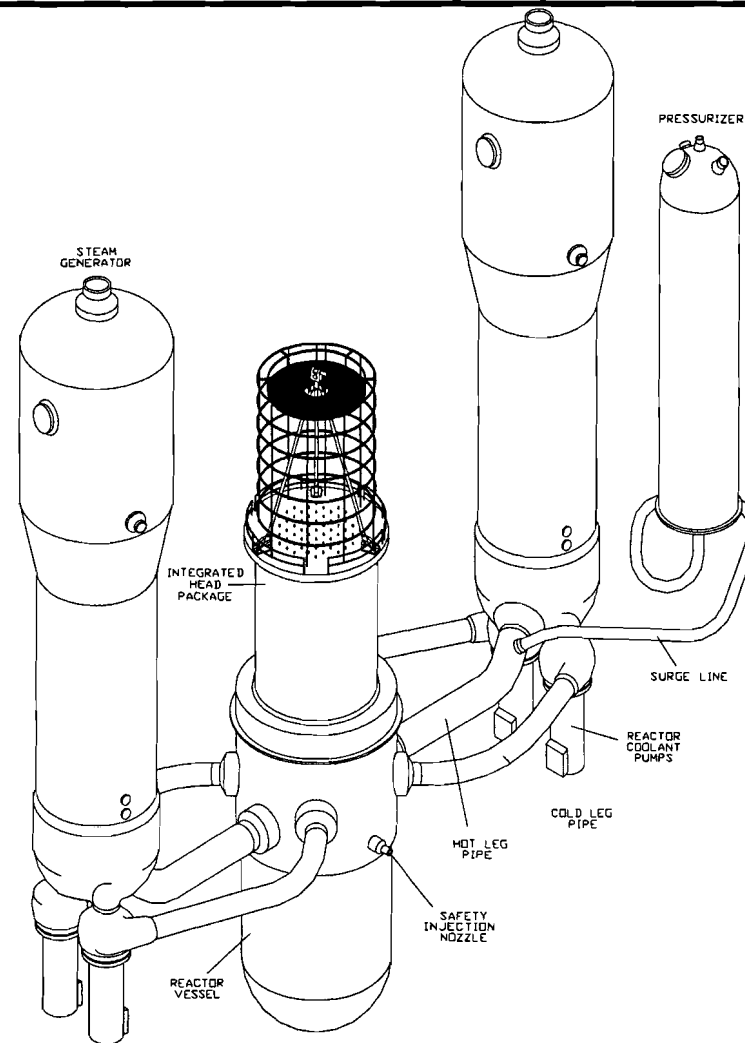
- **Main control room doses are calculated for design basis accidents**
 - Atmospheric dispersion factor influences dose rate
- **AP1000 control room X/Q was performed using same methodology as applied to AP600**
- **NRC Reg Guide 1.193 issued in June 2003**
- **DSEER item requests a comparison of AP1000 calculation to the Reg Guide**
- **Westinghouse is preparing a compliance summary to Reg Guide 1.193**

DSER Open Item: 10 CFR 50.44

- **10 CFR 50.44 - Standards for combustible gas control system in light-water-cooled power reactors**
 - Regulation is undergoing revision
 - Relaxation of the requirements regarding hydrogen recombiners
 - Expected to be issued this year
- **AP1000 has been designed considering new requirements**
 - Passive H₂ recombiners are still provided
 - Not required based on new regulation
 - Provided for defense in depth
 - Same design as AP600; downgraded safety classification

AP1000 Proven Components

- **Core**
 - 14 ft / 157 Fuel Assemblies
 - Doel 4; Tihange
- **Reactor Vessel - 3XL**
 - Same OD as AP600
 - 60 year design life
 - No welds in high flux region
 - No bottom-mounted instrumentation
- **Δ125 Steam Generators**
 - ANO RSG / CE System 80
- **Reactor Coolant Pump**
 - Canned motor pumps
 - No seals / high reliability
 - Naval reactors; AP600
- **Simplified Main Loop**
 - Same as AP600
 - Reduced welds / supports
- **Pressurizer**
 - 50% larger than similar units



IN-VESSEL RETENTION AND EX-VESSEL FUEL COOLANT INTERACTIONS FOR AP1000

Presentation to
Advisory Committee on Reactor Safeguards
Subcommittee on Future Plant Designs

July 17, 2003

by:

H. Esmaili and M. Khatib-Rahbar
Energy Research, Inc.
6167 Executive Blvd.
Rockville, Maryland 20852



ERI

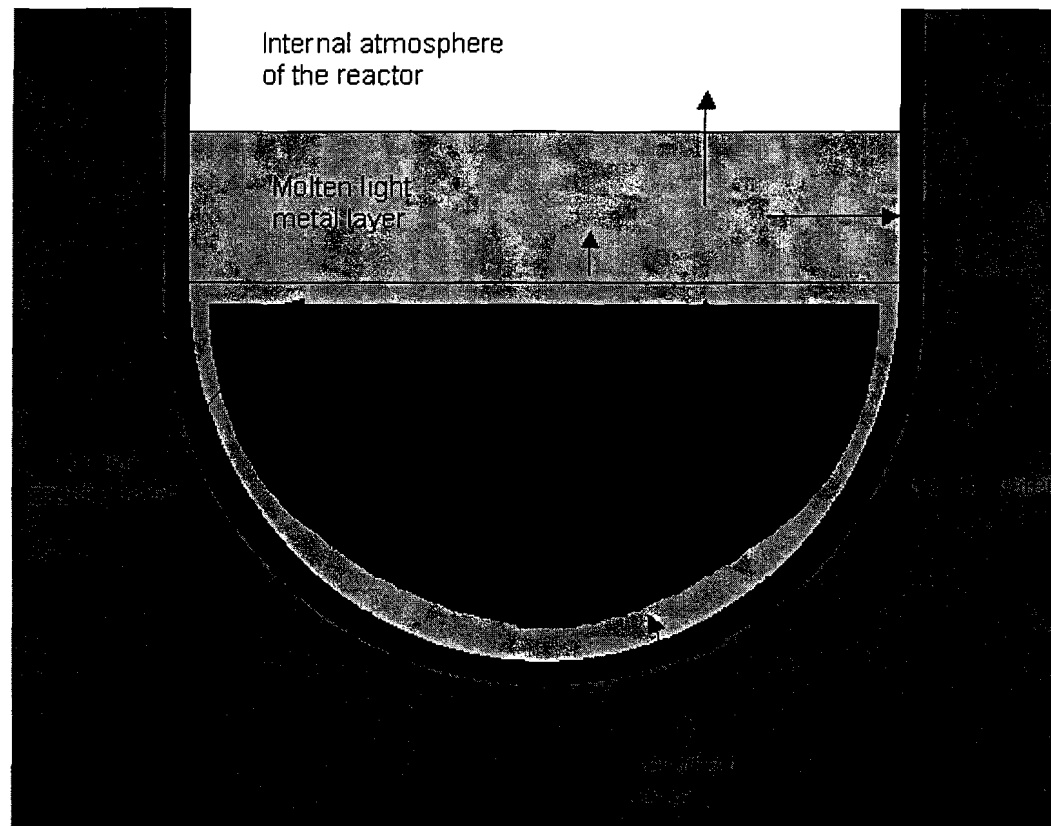
Energy Research, Inc.

OBJECTIVES

- Assessment of In-Vessel Retention (IVR) to Determine the Likelihood and Location of Vessel Breach
- Formulation of FCI Scenarios and Quantification of FCI Impulse Loads Using an Approach Similar to that Used by ERI/NRC for AP600

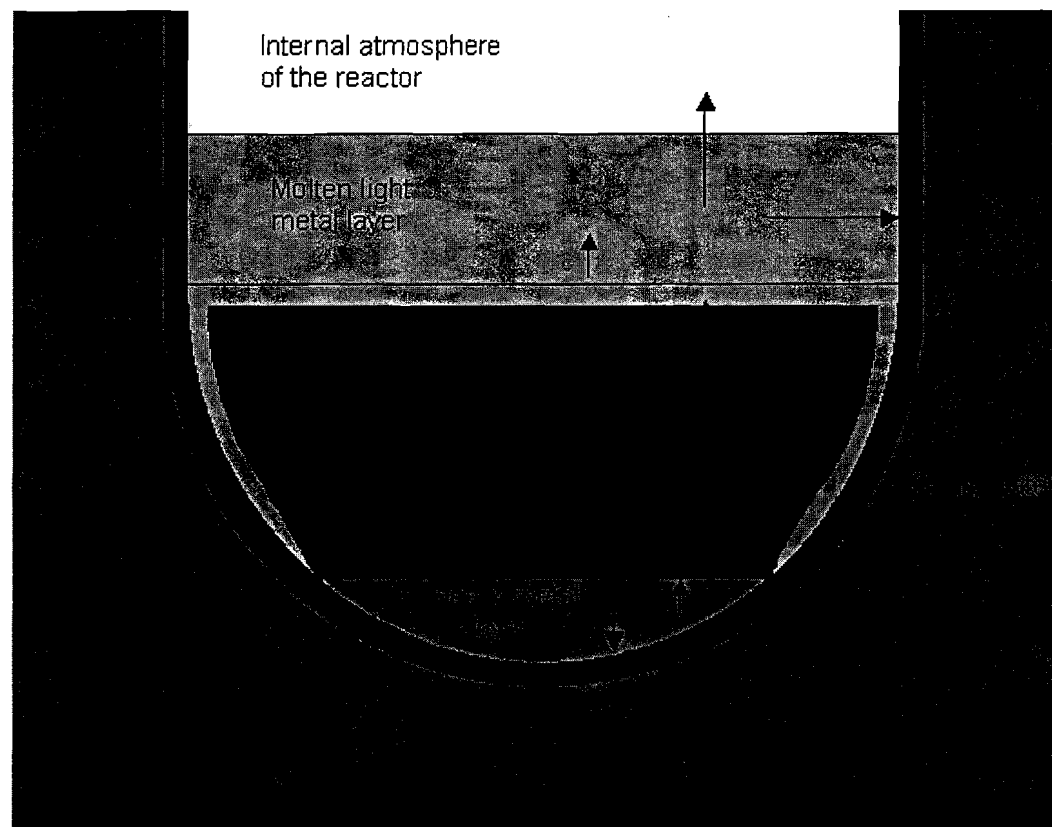
AP1000 IN-VESSEL RETENTION ANALYSIS

- 1-D Mathematical Model Based on ERI 2-D Model for AP600 (Modified for Comparison to DOE and INEEL models)



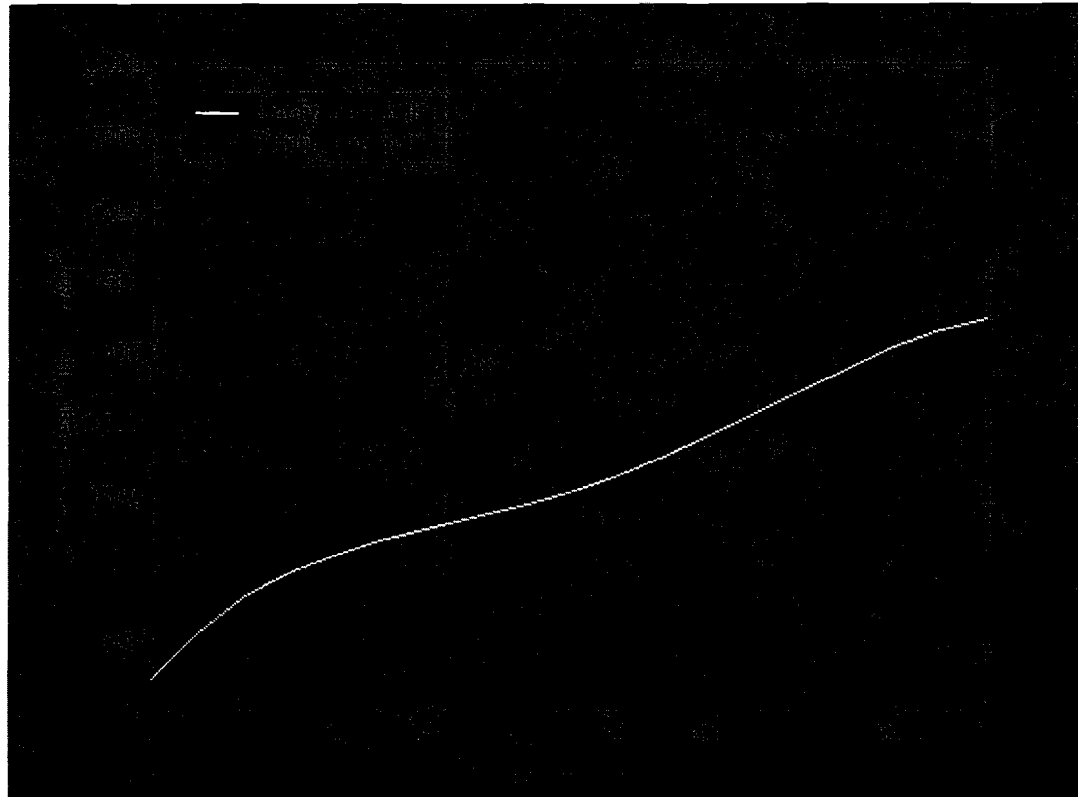
AP1000 IN-VESSEL RETENTION ANALYSIS

- 1-D Mathematical Model Based on ERI 2-D Model for AP600 (Modified for Comparison to DOE and INEEL models)



AP1000 IN-VESSEL RETENTION ANALYSIS

- Critical Heat Flux (Lower Head Configuration V) – Data and Correlation not Available; However, Assumed that CHF is Higher by Factor of 1.44 as Compared to Configuration III (Reported by Westinghouse)



AP1000 IN-VESSEL RETENTION ANALYSIS

- Heat Transfer Correlations (in molten pool regions)

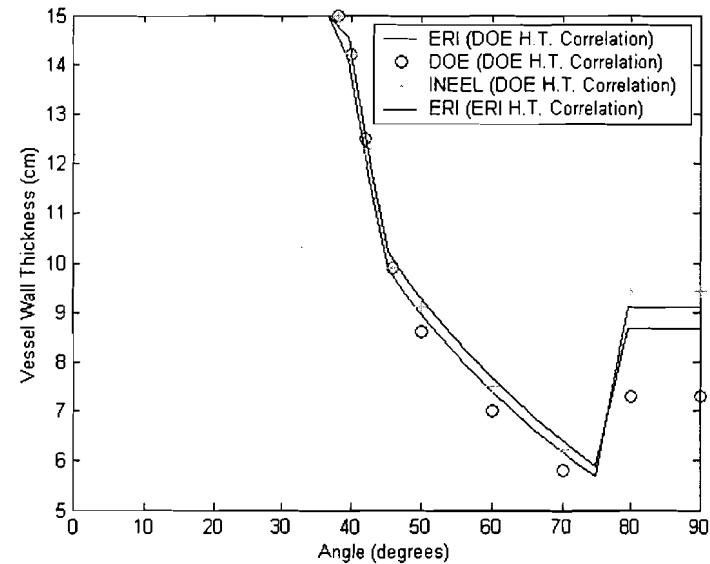
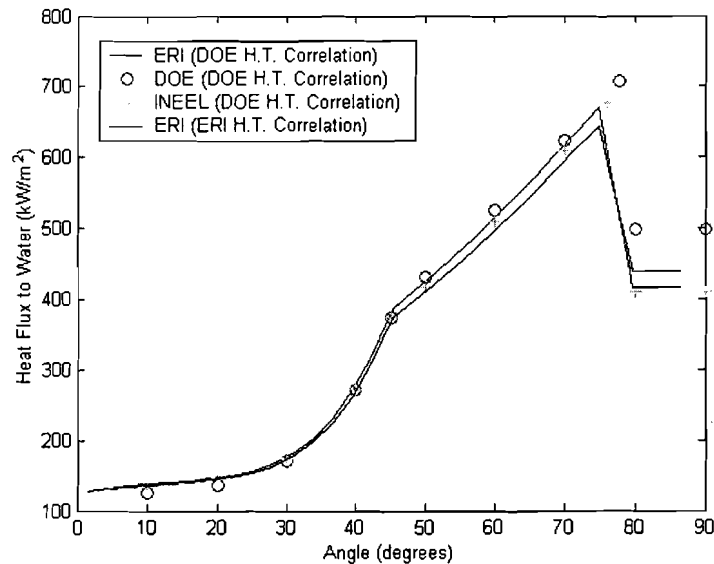
Model		Heat Transfer Correlation	Range of applicability	
			Ra	Pr
ERI	Ceramic Pool	Mayinger	$7 \times 10^6 - 5 \times 10^{14}$	0.5
		Kulacki-Emara	$2 \times 10^4 - 4.4 \times 10^{12}$	7
	Top Metal Layer	Globe-Dropkin	$3 \times 10^5 - 7 \times 10^9$	0.02-8750
		Churchill-Chu	$< 10^{12}$	Any
DOE	Ceramic Pool	Mini-ACOPO	$10^{12} - 7 \times 10^{14}$	2.6-10.8
	Top Metal Layer	Globe-Dropkin	$3 \times 10^5 - 7 \times 10^9$	0.02-8750
		Churchill-Chu	$< 10^{12}$	Any
INEL	Ceramic Pool	ACOPO		
	Top Metal Layer	Globe-Dropkin	$3 \times 10^5 - 7 \times 10^9$	0.02-8750
		Churchill-Chu	$< 10^{12}$	Any

AP1000 IN-VESSEL RETENTION ANALYSIS

- Solution Method Based on Non-Linear Newton-Raphson Method and Allows for Temperature Dependence of Viscosity
- Material Properties Based on the INEEL Report for AP600
- Decay Heat Partitioning Based on the Amount of U in the Bottom Layer

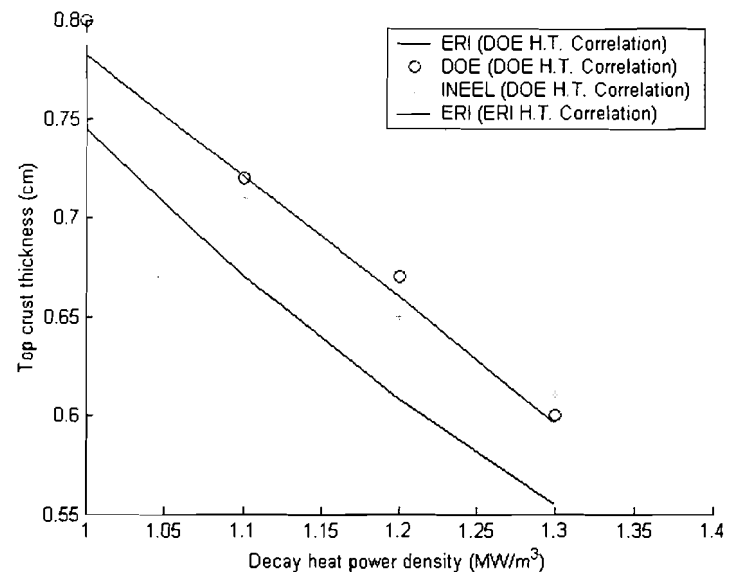
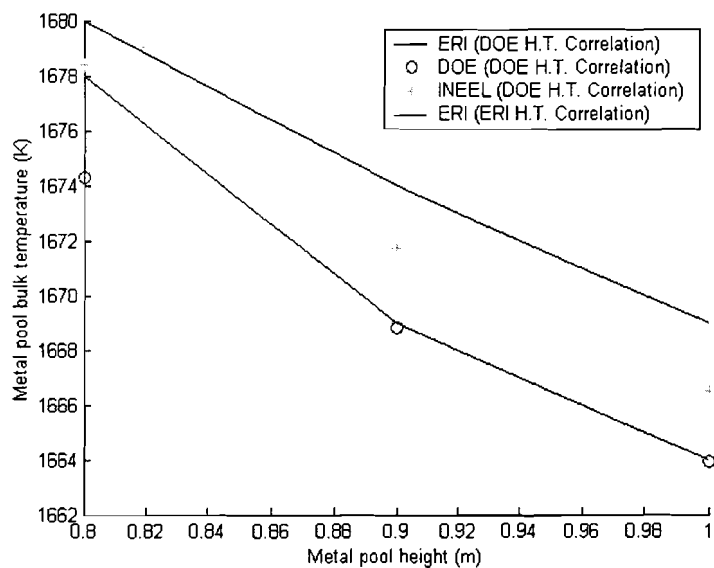
AP1000 IN-VESSEL RETENTION ANALYSIS

- Model Verification by Comparison to DOE and INEEL Results for AP600



AP1000 IN-VESSEL RETENTION ANALYSIS

- Model Verification by Comparison to DOE and INEEL Results for AP600



AP1000 IN-VESSEL RETENTION ANALYSIS

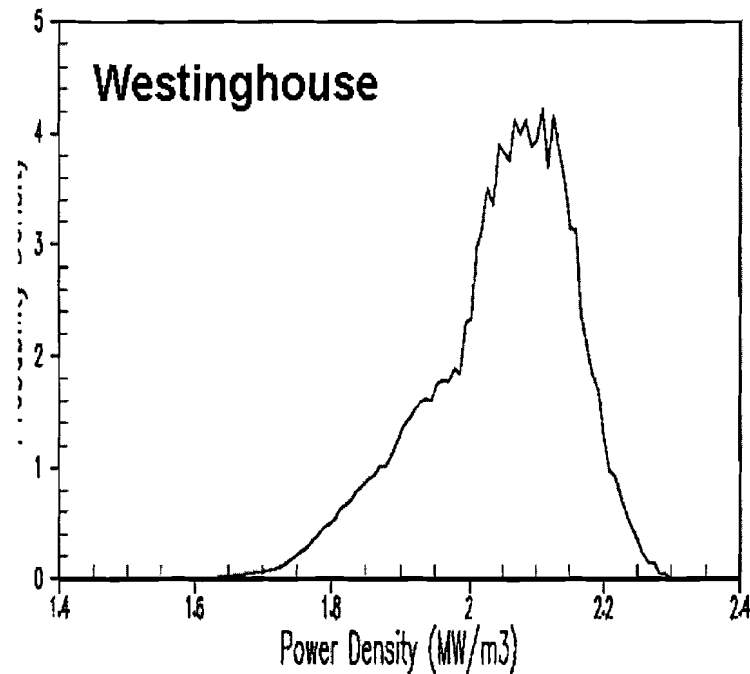
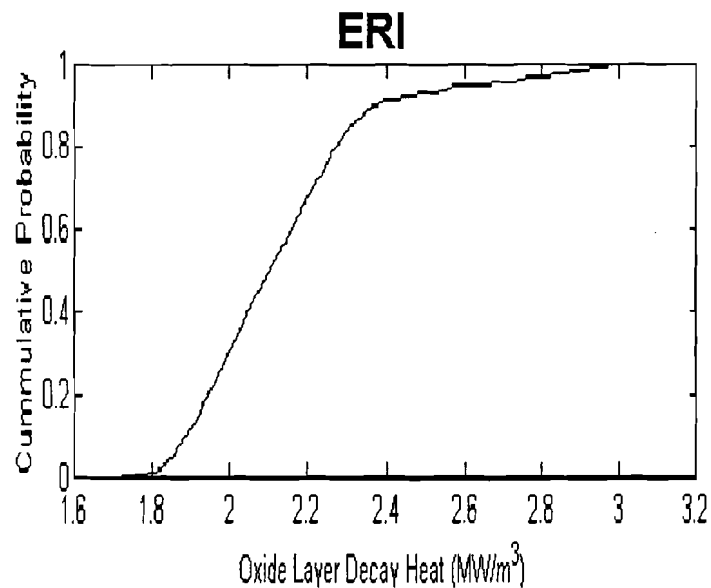
- Differences in Design from AP600
 - Power Increase to 3400 MWt
 - Core Shroud in AP1000 vs. Reflector in AP600
 - Thicker (~ 1") Lower Core Support Plate

- Two Bounding Melt Configurations
 - Melt Configuration I - Molten Ceramic Pool with an Overlaying Molten Light Metallic Layer
 - Melt Configuration II - Molten Ceramic Pool Between a Bottom Heavy Metallic Layer and a Top Light Metallic Layer

- ERI Model Only Accounts for Thermal Interactions. Chemical Reactions with Vessel Wall Not Considered (i.e., in the absence of a crust, the potential for chemical reactions and dissolution could be important).

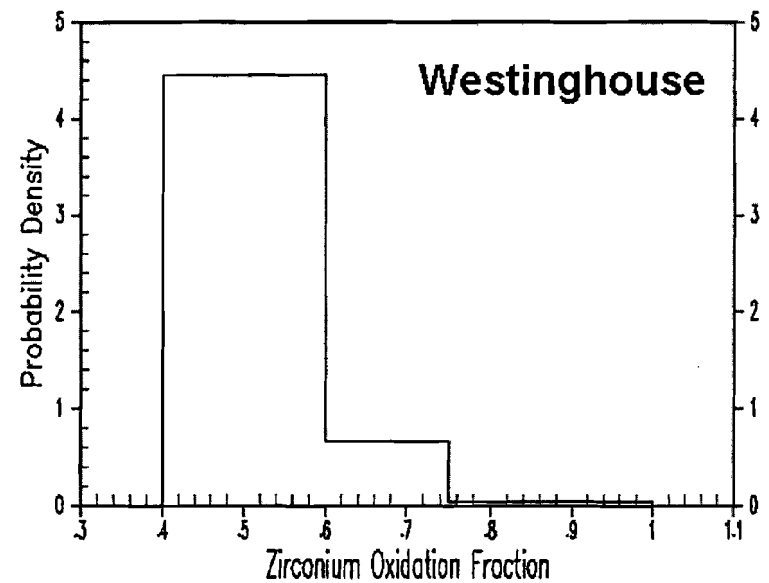
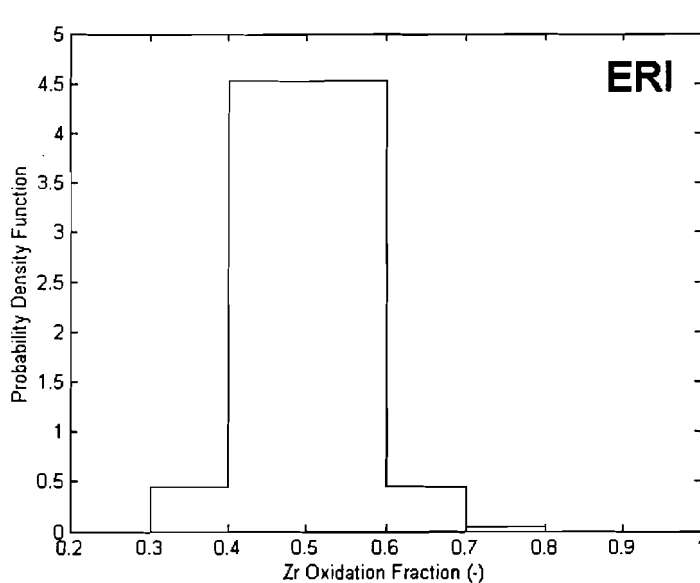
AP1000 IN-VESSEL RETENTION ANALYSIS

- Initial and Boundary Conditions for ERI Melt Configuration I
 - Decay Heat Specified Based on Plant-Specific Timing of Core Relocation



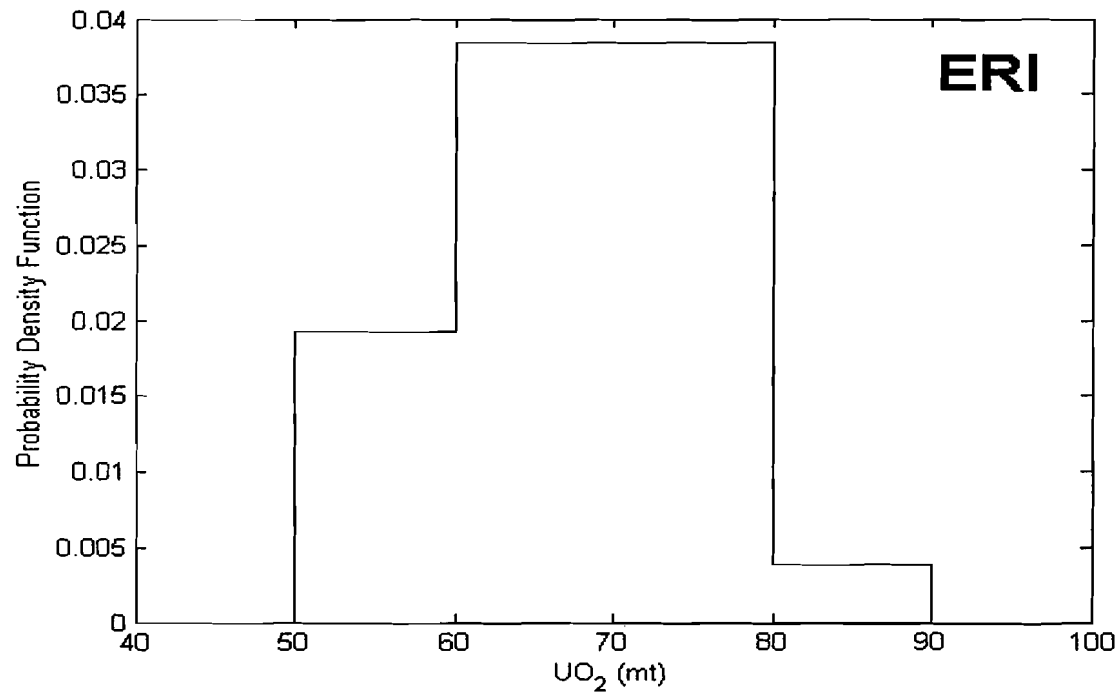
AP1000 IN-VESSEL RETENTION ANALYSIS

- Initial and Boundary Conditions for ERI Melt Configuration I
 - Zr-Oxidation Fraction (mainly based on plant-specific MELCOR calculations)



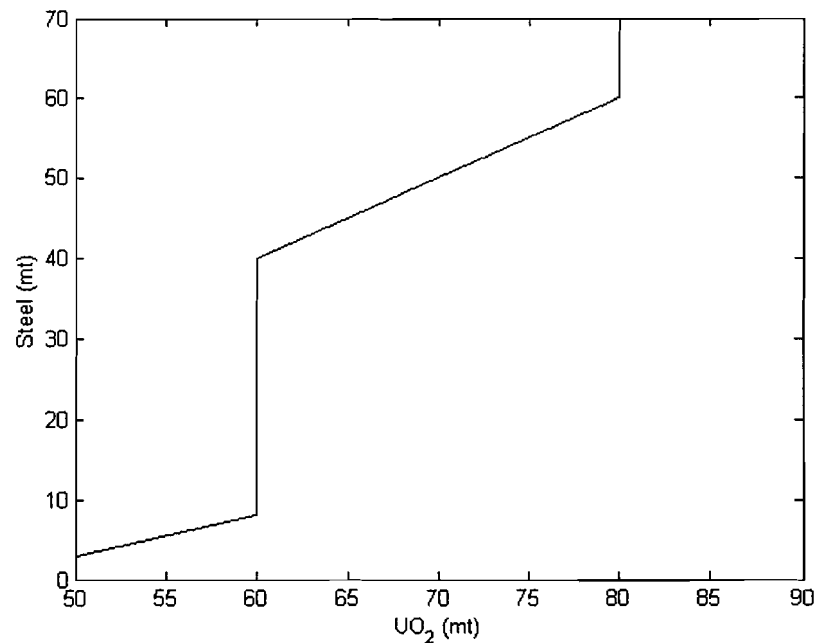
AP1000 IN-VESSEL RETENTION ANALYSIS

- Initial and Boundary Conditions for ERI Melt Configuration I
 - Core Ceramic Material Relocation Based on Plant-Specific MELCOR Calculation for Scenarios 3BE/3BR & Insights from SCDAP/RALAP5 Calculation for AP600 and Westinghouse MAAP Calculations for AP1000



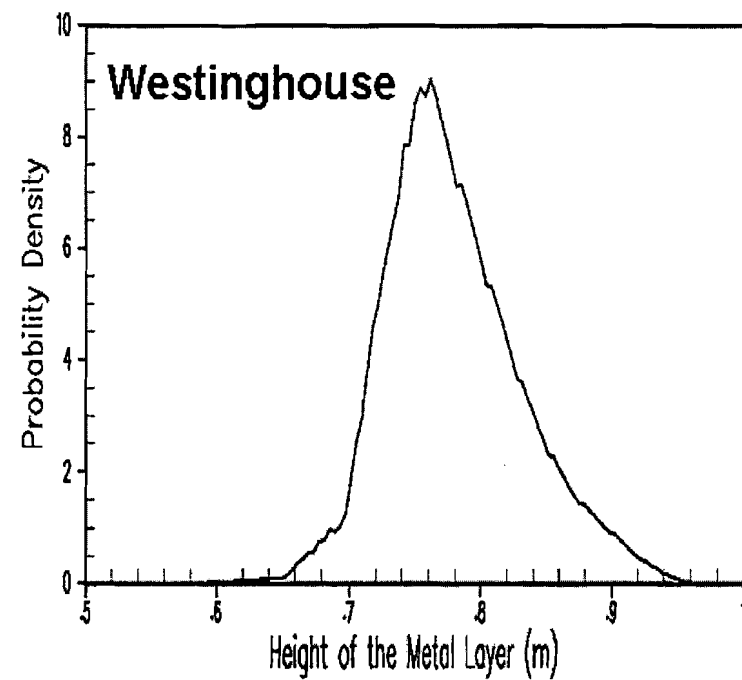
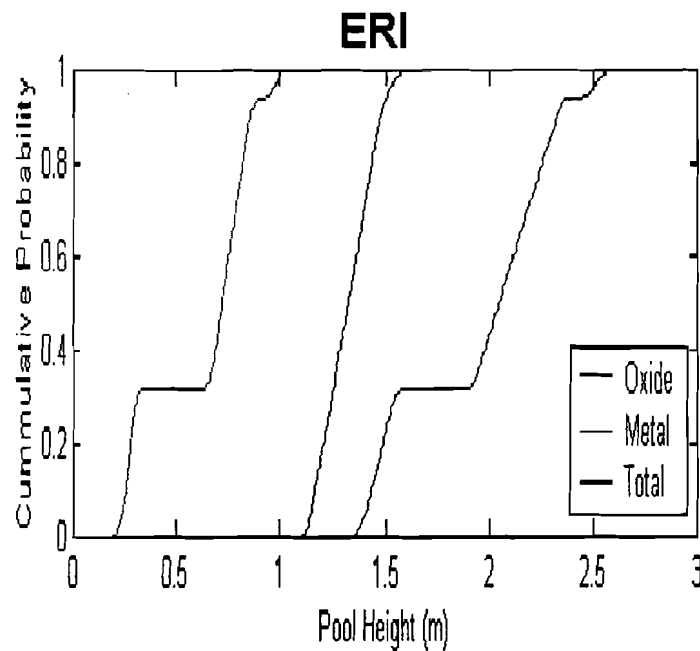
AP1000 IN-VESSEL RETENTION ANALYSIS

- Initial and Boundary Conditions for ERI Melt Configuration I
 - Metal Relocation Dependent on Extent of Core Relocation



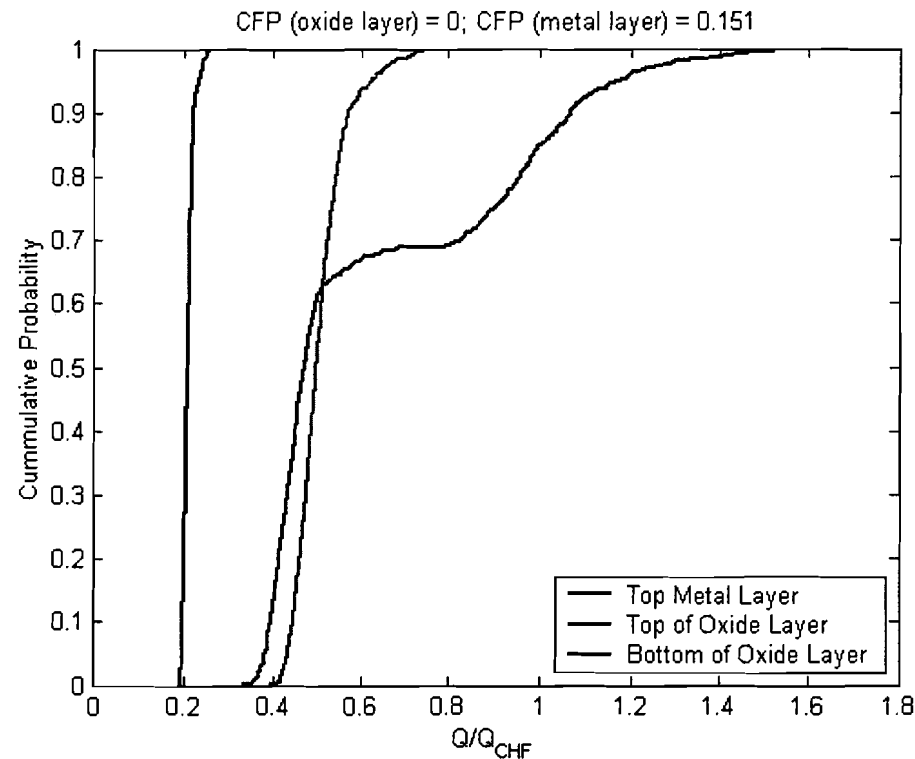
AP1000 IN-VESSEL RETENTION ANALYSIS

- Initial and Boundary Conditions for ERI Melt Configuration I
 - Metal Relocation Dependent on Extent of Core Relocation



AP1000 IN-VESSEL RETENTION ANALYSIS

- Results of ERI Melt Configuration I for Base Case (using ERI default heat transfer correlations). Probabilistic Framework Based on Stratified Monte Carlo (LHS) with 1000 Random Samples



AP1000 IN-VESSEL RETENTION ANALYSIS

- Results of Melt Configuration I – Sensitivity Calculations
 - Decay heat in top metallic layer (0.1 to 0.2)

Case	Description	Ceramic Layer CFP	Metal Layer CFP
1	Base Case	0	0.16
2	DOE heat transfer correlations	0	0.20
3	INEEL heat transfer correlations	0	0.30
4	Material Properties	0	0.16
5	Reduce probability level of low UO ₂ range from 0.5 to 0.1	0	0.04
6	Base Case + Decay heat in the top metal layer	0	0.27
7	DOE heat transfer correlations + Decay heat in the top metal layer	0	0.30
8	INEEL heat transfer correlations + Decay heat in the top metal layer	0	0.31
9	Reduce probability level of low UO ₂ range from 0.5 to 0.1 + Decay heat in the top metal layer	0	0.07

AP1000 IN-VESSEL RETENTION ANALYSIS

- Parametric Results of Melt Configuration II
 - Density Ratio > 1 (i.e., bottom layer heavier)
 - Mass fraction of U in Bottom Layer Fixed at 0.4
 - Variation of Fraction of U that is in Oxide Form (i.e. in the oxide layer)
- Partitioning of the Decay Heat in Bottom Layer
 - Decay heat ratio proportional to mass ratio of U

AP1000 IN-VESSEL RETENTION ANALYSIS

- Parametric Results of Melt Configuration II (Failure of Lower Head at the Bottom Location Not Likely)

Fraction of U in Oxide Form	0.95	0.9	0.85
m_U (kg) [bottom layer]	2,921	5,841	8,762
m_{UO_2} (kg) [oxide layer]	62,953	59,639	56,326
Decay Heat			
Q_h (MW/m ³) [bottom layer]	1.126	1.084	1.071
Q_o (MW/m ³) [oxide layer]	2.127	2.112	2.096
Q/Q_{CHF}	0.22	0.30	0.36

AP1000 EX-VESSEL FCI

- Initial and Boundary Conditions for Ex-Vessel FCI Analyses (using plant-specific MELCOR calculations and IVR analysis)
 - Melt Initial Conditions in the Lower Plenum at Vessel Breach
 - Cavity Condition at Vessel Breach (deep water, fully submerged lower head)
 - Location and Size of Vessel Breach (side-failure, 0.4 m)
 - Containment Pressure and Temperature at Time of Vessel Breach

Parameter	Value
Pour Composition	Metallic
Lower head failure hole diameter	0.4 m
Pour Temperature	2060 K
Pour velocity	1.7 m/s
Containment pressure	2 bar
Cavity water temperature	343 K



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AP1000 EX-VESSEL FCI

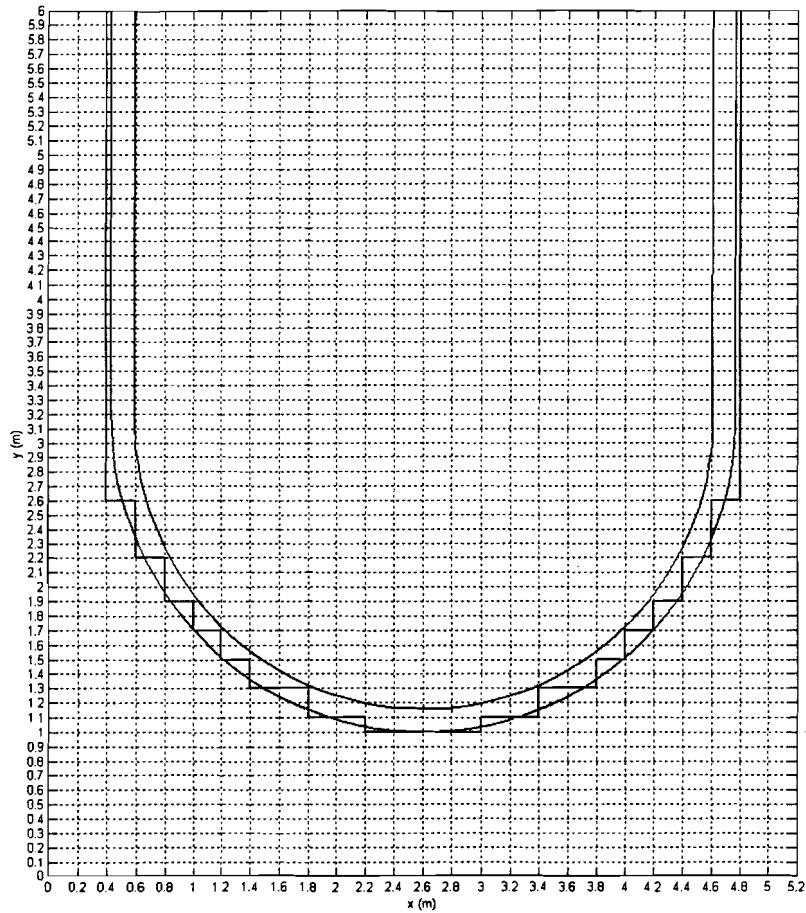
- Calculation Matrix

- Variability in Melt Progression (melt pour composition and the RPV failure size)
- Variability in Modeling of Fuel Coolant Interactions (PM-ALPHA/ESPROSE [2D version] Computer Code Modeling)

Case	Variation from the base case	Comments
1	Base case scenario	Metallic pour at 2060 K, lower head failure size of 0.4 m, melt particle diameter of 0.01 m, and the maximum fragmentation rate per particle of 4 kg/s.
2	Ceramic composition at 3150K	Pour involves ceramic material
3	Failure size of 0.6 m	Larger hole size
4	Particle diameter of 0.10 m and maximum fragmentation rate per particle of 400 kg/s	Larger particle diameter and fragmentation rate
5	Bottom failure of the lower head	Metallic pour (U+Fe+Zr) at 2300 K, lower head failure size of 0.4 m, melt particle diameter of 0.01 m, and the maximum fragmentation rate per particle of 4 kg/s.

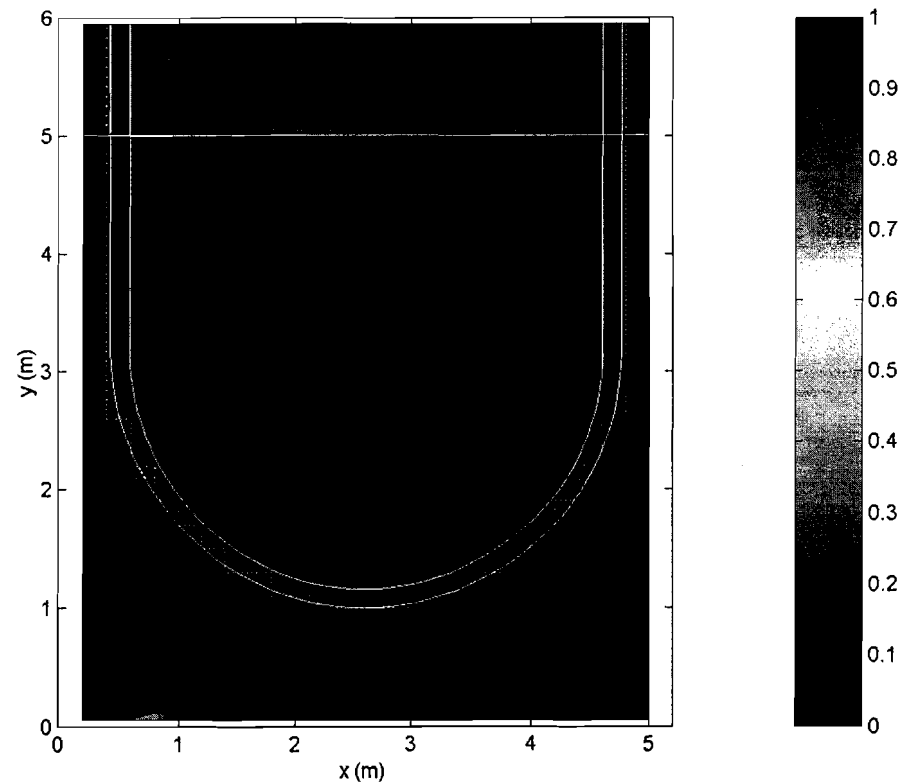
AP1000 EX-VESSEL FCI

- Problem Nodalization



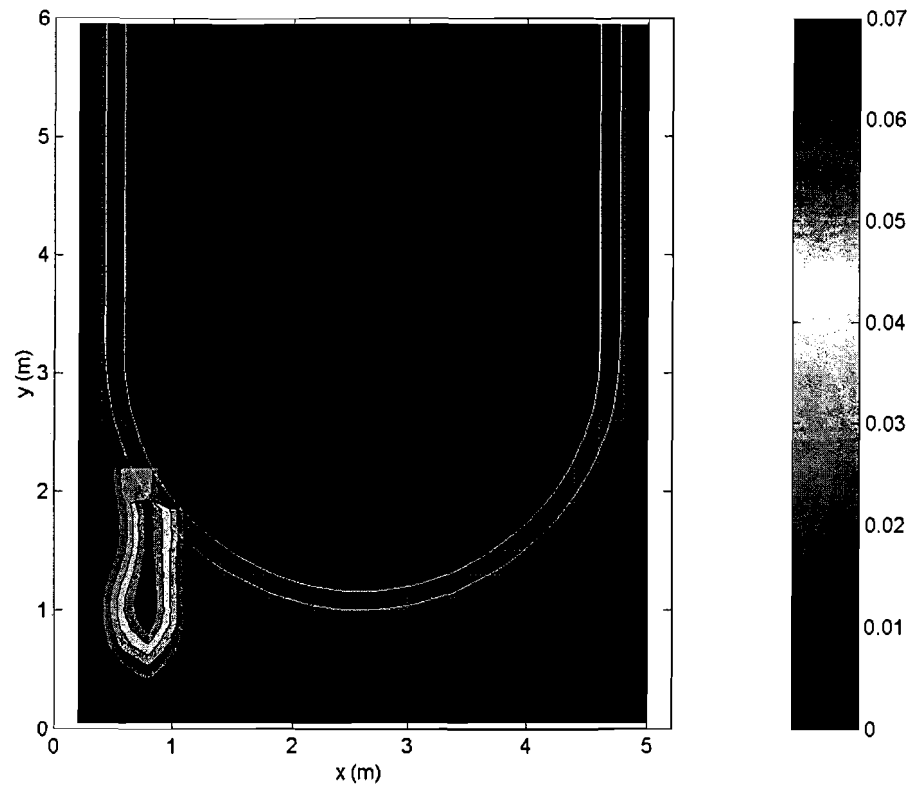
AP1000 EX-VESSEL FCI

- Pre-Mixing Conditions for the Base Case



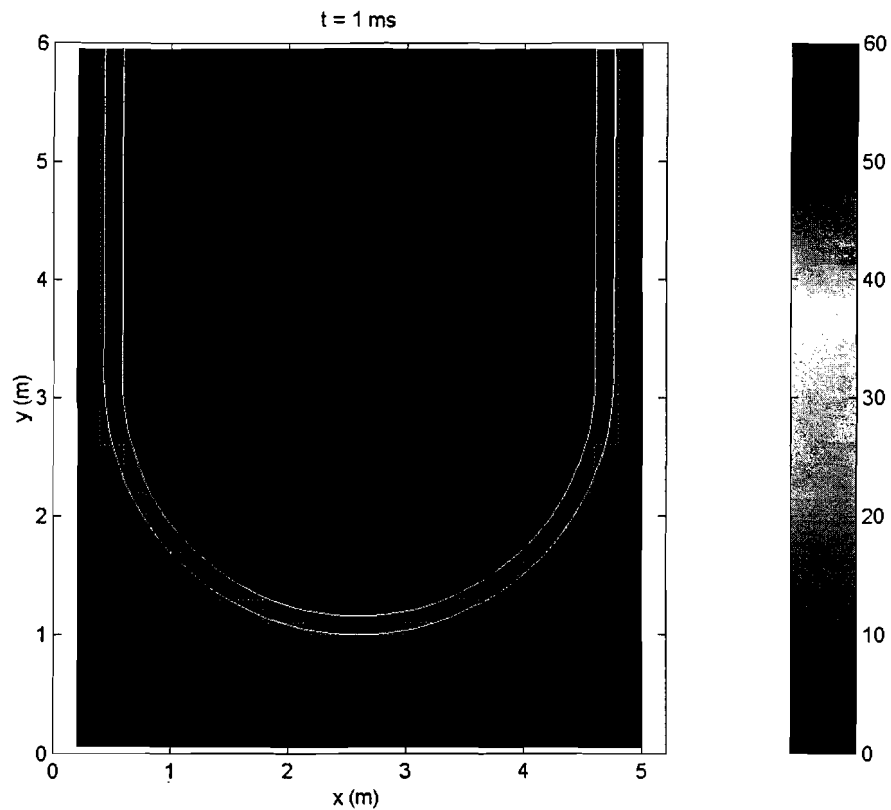
AP1000 EX-VESSEL FCI

- Pre-Mixing Conditions for the Base Case



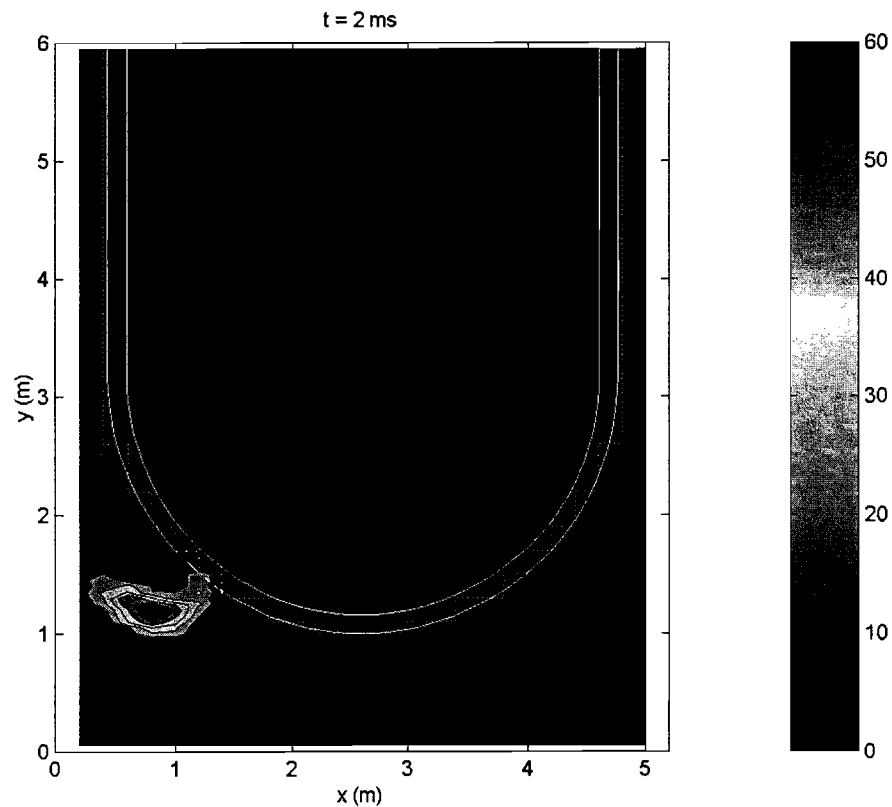
AP1000 EX-VESSEL FCI

- Results of Explosion Propagation for Base Case



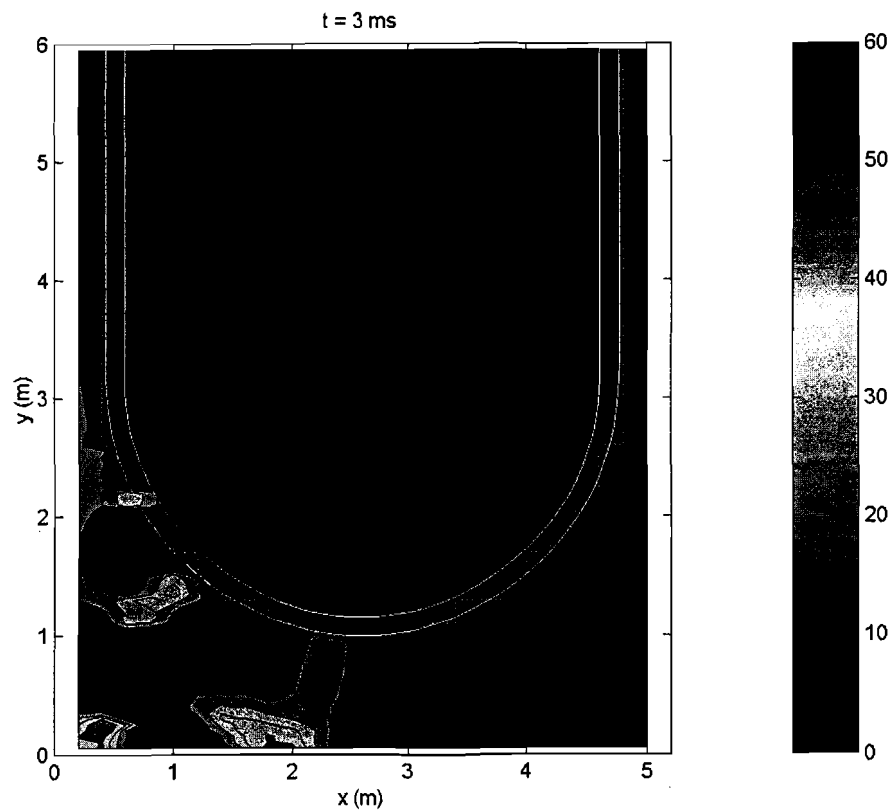
AP1000 EX-VESSEL FCI

- Results of Explosion Propagation for Base Case



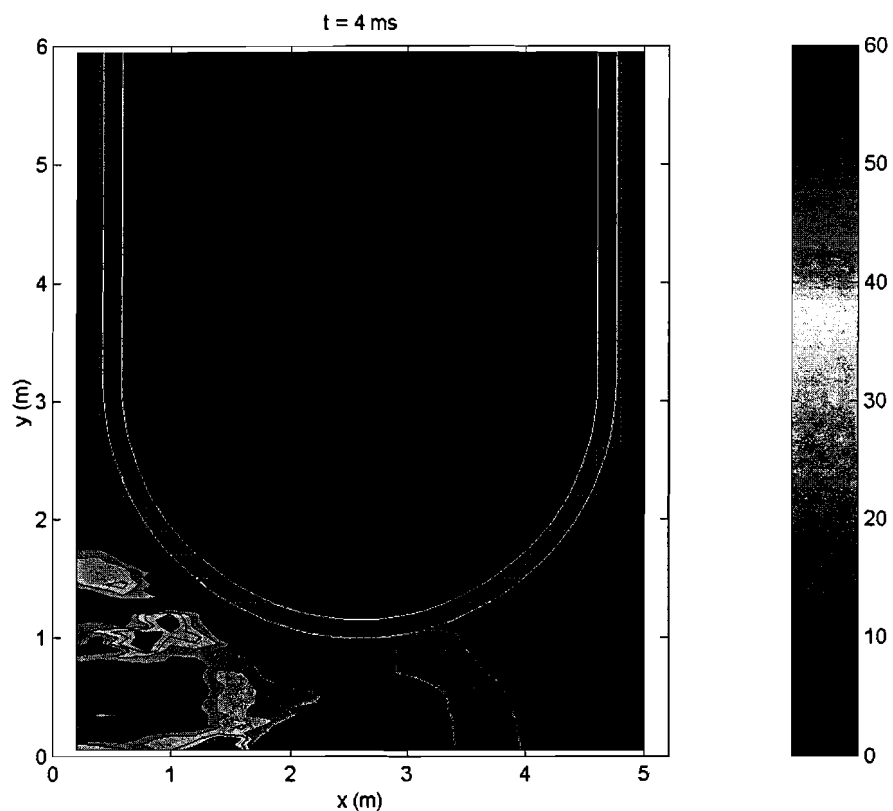
AP1000 EX-VESSEL FCI

- Results of Explosion Propagation for Base Case



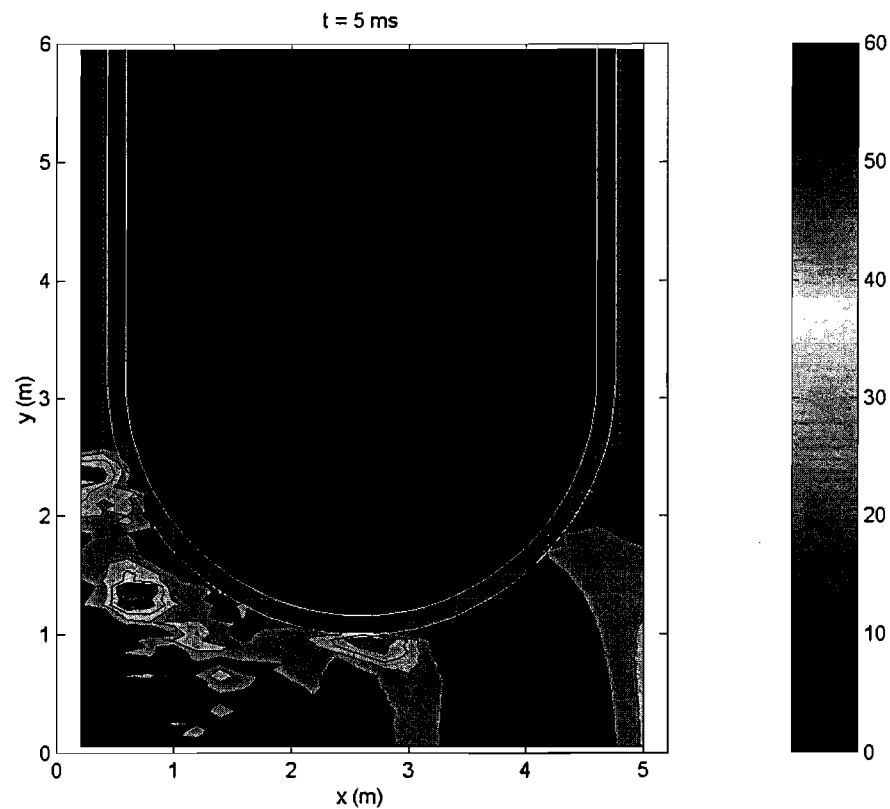
AP1000 EX-VESSEL FCI

- Results of Explosion Propagation for Base Case



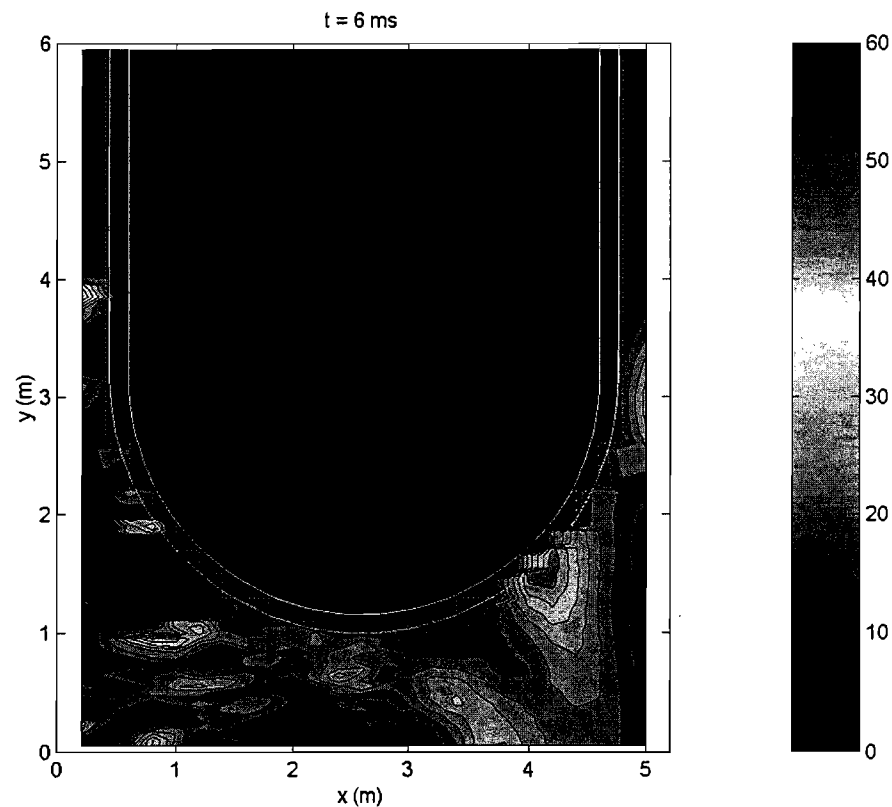
AP1000 EX-VESSEL FCI

- Results of Explosion Propagation for Base Case



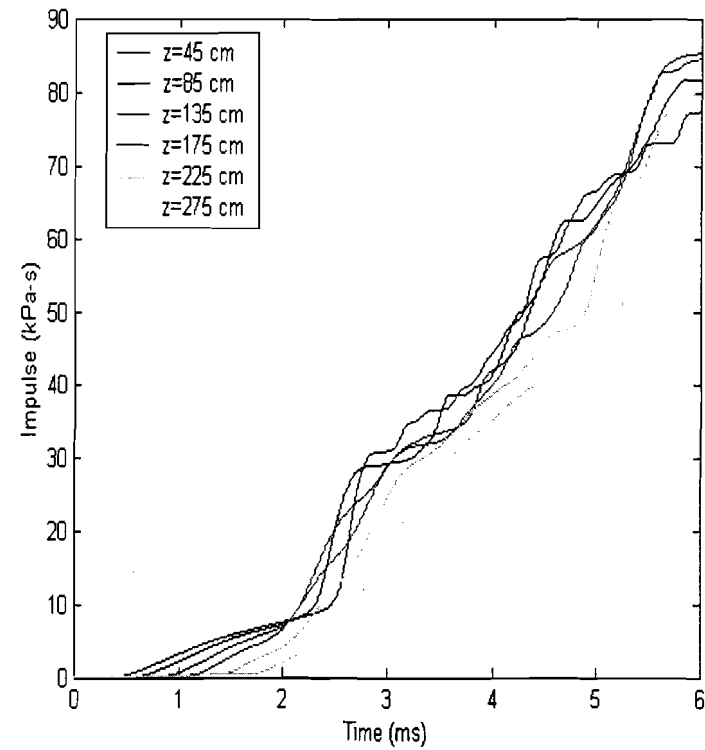
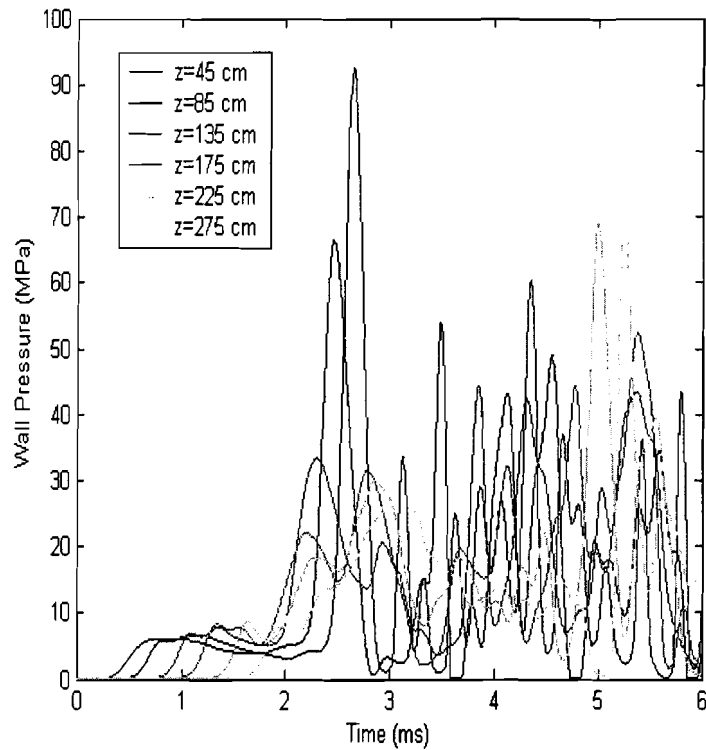
AP1000 EX-VESSEL FCI

- Results of Explosion Propagation for Base Case



AP1000 EX-VESSEL FCI

● FCI Pressures and Impulse Loads for Base Case



COMPARISON of AP1000 to AP600 EX-VESSEL FCI LOADS

Case	Impulse Load (kPa-s)	Wall Pressure (MPa)	Pool Pressure (MPa)
Base case	85	90	220
Ceramic Melt	305	290	1000
Hole diameter of 0.6 m	145	135	425
Particle diameter of 0.1 m and maximum fragmentation rate of 400 kg/s per particle	12	8	10
Bottom failure of the lower head	9	8	60

COMPARISON of AP1000 to AP600 EX-VESSEL FCI LOADS

Description of Calculations for AP600	Maximum Impulse Load (kPa-s)		
	ESPROSE.m		TEXAS
Scenario I (Unsubmerged RPV)	Cavity Wall	RPV	Cavity Wall
Base Case, Saturated Pool	85	-	-
Subcooled Pool	160	-	-
Scenario II (Partially Submerged RPV)	Cavity Wall	RPV	Cavity Wall
Base Case, Saturated Pool	150	-	205
Melt Superheat	147	-	215
Subcooled pool	300	-	335
Metallic melt	27 (128 ^a)	-	153
Hole diameter of 0.2 m	68	-	-
Hole diameter of 0.8 m	383 ^b	-	644
Impact of RPV lower head	190	192	-
Dp=0.1 m, Maximum Fragmentation Rate = 4 kg/s	15	-	-
Dp=0.1 m, Maximum Fragmentation Rate = 400 kg/s	86	-	-
Fragmentation Constant (increase to 0.0125)	-	-	457
Scenario III (Fully Submerged RPV)	Cavity Wall	RPV	Cavity Wall
Base case, subcooled water	300	-	-
Saturated pool, RPV modeled	288	320	-
Subcooled pool, RPV modeled	625	670	-

^a with a water pool subcooling of 20 K and initial melt temperature of 3100 K

^b calculation failed after 3 ms due to numerical problems

CONCLUDING REMARKS

- There is the Potential for the Side Failure of the Lower Head as a Result of the Focusing Effect for Melt Configuration I (Estimated Failure Likelihood Ranges from 0.04 to 0.30).
- Failure of Lower Head at the Bottom Location Not Likely (Based on Results of ERI Melt Configuration II Parametric Calculations).
- Ex-Vessel FCI Loads for Side Failure of the Fully Submerged Lower Head in AP1000 Remain Lower than Those of AP600. A Sensitivity Calculation Assuming the Bottom Failure of the Lower Head, Results in Lower Impulse Loads as Compared to Cases Involving Side Failure.

MELCOR ANALYSIS OF SELECTED SEVERE ACCIDENT SEQUENCES

Presentation to

Advisory Committee on Reactor Safeguards

Subcommittee on Future Plant Designs

July 17, 2003

by:

M. Zavisca and M. Khatib-Rahbar

Energy Research, Inc.

6167 Executive Blvd.

Rockville, Maryland 20852



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Objectives

- 1) Derive initial and boundary conditions for analysis of IVR and ex-vessel FCI issues
- 2) Calculate extent and consequences of molten core-concrete interactions (MCCI)
- 3) Provide information about potential containment challenges from hydrogen combustion
- 4) Verify expected changes in overall accident progression relative to AP600



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Sequences Selected for Analysis

Sequence	Description	Basis for Selection
3BE	<ul style="list-style-type: none"> •DVI Line Break •ADS successful •Cavity Flooding successful 	<ul style="list-style-type: none"> •Provides ICs and BCs for IVR/FCI analyses •Similar to <u>W</u> Sequence #1 (29% of CDF)
3BE'	Sensitivities to 3BE: Containment shell coverage efficiency by PCCS	•Establishes bounds of performance for PCCS
3BR	<ul style="list-style-type: none"> •Large LOCA •Cavity Flooding not successful in time for IVR 	•Similar to <u>W</u> Sequence #2 (18% of CDF)
3D	<ul style="list-style-type: none"> •Spurious ADS •Full RCS depressurization unsuccessful •IRWST injection unsuccessful 	<ul style="list-style-type: none"> •MCCI analysis •Similar to <u>W</u> Sequence #3 (9% of CDF)
3D'	Sensitivities to 3D: Mode of vessel breach, Concrete composition, Reactor cavity conditions	•MCCI analysis
1A	<ul style="list-style-type: none"> •Loss of feed water •ADS unsuccessful •IRWST injection unsuccessful 	<ul style="list-style-type: none"> •Similar to W Sequence #20 (0.6% of CDF), largest contributor without RCS depressurization. •Potential to end in induced SG tube rupture (high risk consequence)



MELCOR Model

- 44 control volumes, 75 flow paths
- Core has 10 axial, 5 radial nodes
- All safety systems relevant to severe accidents modeled, including CMTs, PCS, PCCS, ADS, PRHR, Cavity Flooding
- Melting of core shroud not included
- Mixing of molten debris in lower plenum modeled with enhanced conductivity

Results – 3BE (IVR)

	MELCOR	MAAP4
Core Melt Time (hr)	1.3 - 2.1	0.8 – 1.3
Core Plate Failure Time (hr)	2.6 – 3.7	1.4 – 2.1
Core Plate Melt Time (hr)	3.0 – 4.1	unav.
LP Dryout Time (hr)	3.1 – 4.2	1.9 – 2.5
Initial Relocation Size (% UO ₂)	80%	50%
Molten Steel Mass (t)	27	unav.
(Unoxid.) Molten Zr Mass (t)	11 - 12	unav.

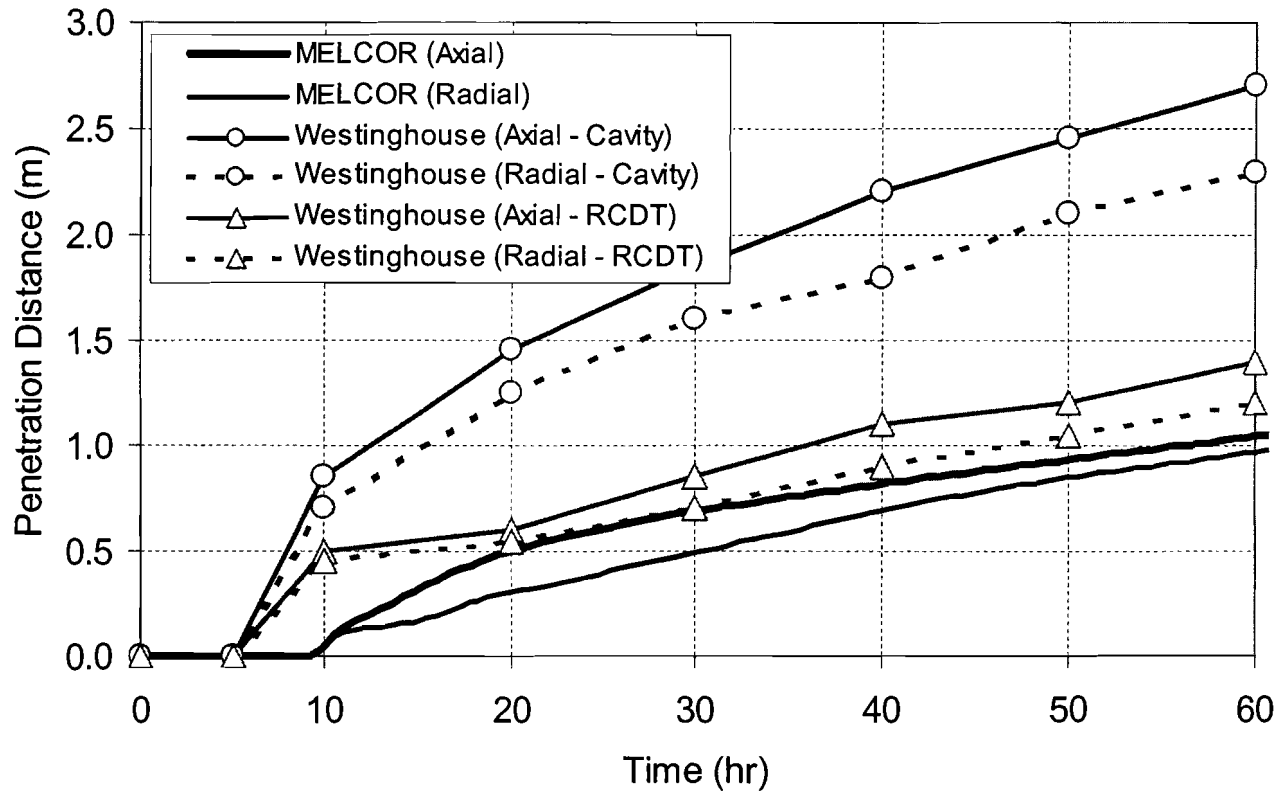
Results – MCCI

- Scenarios included limestone, basalt, and limestone/sand aggregate concrete types, flooded and dry cavity (all based on 3D/3D' – spurious ADS sequences)

Results – MCCI (cont.'d)

	MELCOR	MAAP4
Ablation, Limestone (m @ 60h)	1.0	2.7
Ablation, Basaltic (m @ 60h)	2.2	3.1
Hydrogen generated (t @ 60h)	0.7 – 2.8	unav.
CO generated (t @ 60h)	4.3 – 43.3	unav.
CO ₂ generated (t @ 60h)	0.1 – 19.5	unav.
Cont. Pressure (bar @ 24h)	2.4	3.1

Results – MCCI (cont.'d)



Concrete ablation results: Limestone/sand concrete, localized vessel failure, ~2m depth of water in cavity

Results – Hydrogen

- Multiple small deflagrations occur in confined compartments near point of release (e.g., IRWST, SG cubicles) until O₂ is depleted; afterwards, some burns in upper containment occasionally observed
- With MCCI, one or more deflagrations of H₂ and CO occur in late time frame
- Most calculations conservatively assumed igniters not operating (in MELCOR by default, combustion occurs at 10% H₂ concentration)

Results – Hydrogen (cont.'d)

	MELCOR	<u>W</u>
In-Vessel Zirconium Oxidation	44 – 65% (420 – 650 kg)	28 – 83% (MAAP4)
Early Containment Loads due to H ₂ Deflagration	< 3.5 bar	< 4.3 bar (MAAP4/AICC, no reflood)
Gases from MCCI	H ₂ : 660–2830 kg CO: 4,300 – 43,000 kg	unav.
Late Containment Loads due to H ₂ /CO Deflagration (with MCCI)	4.5 – 7.4 bar	unav.



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Results – General Accident Progression

	MELCOR	MAAP4
Time to Core Damage (hr)	1.3 – 2.1	0.8 – 1.3
Time of Debris Relocation to LP (hr)	2.6 – 3.7	1.4 – 2.1
Time of Induced SG Tube (Creep) Rupture (hr)	7.4	3.9
Peak Containment Pressure from Large LOCA (bar)	3.6	3.6
Late Quasi-static Containment Pressure, no MCCI (bar)	1.4 – 1.8	1.3 – 1.5



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Conclusions

- Global deflagrations do not challenge the AP1000 containment integrity
- PCCS successful in preventing overpressure (with or without MCCI)
- Concrete basemat penetration not expected within 3 days
- Accident timing and containment response generally similar to AP600, scaled to changes in power, core mass, containment volume, etc.



Post-LOCA Design Basis Aerosol Deposition in AP1000 Containment

Dr. Jun Li

Dr. David Leaver

Polestar Applied Technology, Inc.

July 17, 2003



Introduction

- As part of AP600 design certification, Polestar performed a QA calculation of radiological design basis fission product aerosol removal rates (λ) in containment by natural processes.
- Similar to AP600, the AP1000 containment has a large steel shell cooled on the outside, leading to higher heat transfer rate and higher natural aerosol removal rate for fission product aerosols than would exist from sedimentation alone.
- Since AP1000 and AP600 have a similar design, this calculation is a repetition of the AP600 calculation with AP1000 parameters and thermal hydraulics. The AP600 sensitivity study was also referenced to assess possible variation of AP1000 λ s.

Introduction (cont.)

- Perspective
 - + Tightly sealed containment (0.1 vol% per day).
 - + Large steel shell cooled on the outside by more than 80% sub-cool water coverage.
 - + Wet inner containment surface where significant condensation and heat transfer take place.
 - + Steamy and turbulent containment atmosphere.
 - + Favorable environment for natural removal of aerosols from the containment atmosphere.
 - + Containment aerosols cannot bypass these removal mechanisms to leak out directly.

AP1000 and AP600 Comparison

- AP1000 has, compared to AP600:
 - + 75% higher thermal power.
 - + 20% larger containment by volume.
 - + 75% more aerosol mass.
- Expected results, compared to AP600:
 - + Higher diffusio- and thermophoresis due to higher heat transfer to containment shell.
 - + Similar sedimentation due to similar concentration and well-mixed assumption (conservative).
 - + As a result, higher containment lambda.

Calculation Procedure

- Select the AP1000 sequence that has relatively high probability and has timing that is similar to the NRC specified RG 1.183 timing for PWR fission product release.
- Calculate containment thermal hydraulic conditions for the selected sequence using the MAAP4 code.
- Calculate containment aerosol removal rates for MAAP4 thermal hydraulic conditions and aerosol assumptions, using the Polestar QA code STARNAUA.
- STARNAUA has been documented and benchmarked against experiment, was applied as part of AP600 design certification, and has recently been applied to numerous operating plant DBA alternate source term aerosol calculations

Aerosol Assumptions

Containment atmosphere: Well-mixed

Particle size: $r_g=0.22\mu\text{m}$, $\sigma_g=1.81$
-> $\text{mmd}=1.3\mu\text{m}$

Inert to FP ratio: 1.5:1

Hygroscopicity: Neglect hygroscopicity.

Packing fraction: 0.8

Release fraction and timing: NRC Regulatory Guide 1.183

Removal Mechanisms

- Sedimentation

$$v_{sed} = \frac{2\rho_p g r_p^2}{\phi 9\mu_v} Cn(Kn)$$

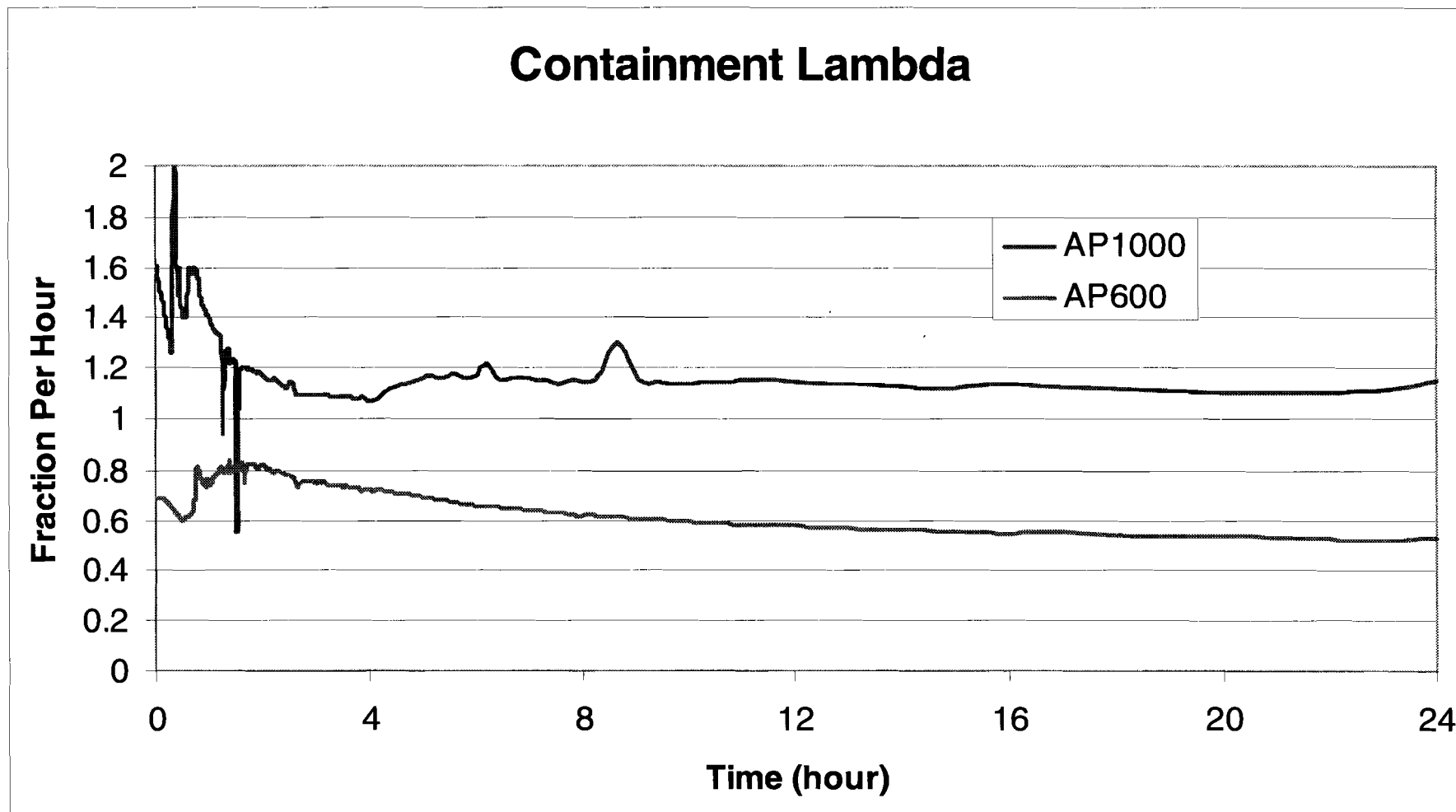
- Diffusiophoresis

$$v_{dif} = \frac{x_v \sqrt{M_v}}{x_v \sqrt{M_v} + x_a \sqrt{M_a}} \frac{\dot{m}''}{\rho_v}$$

- Thermophoresis

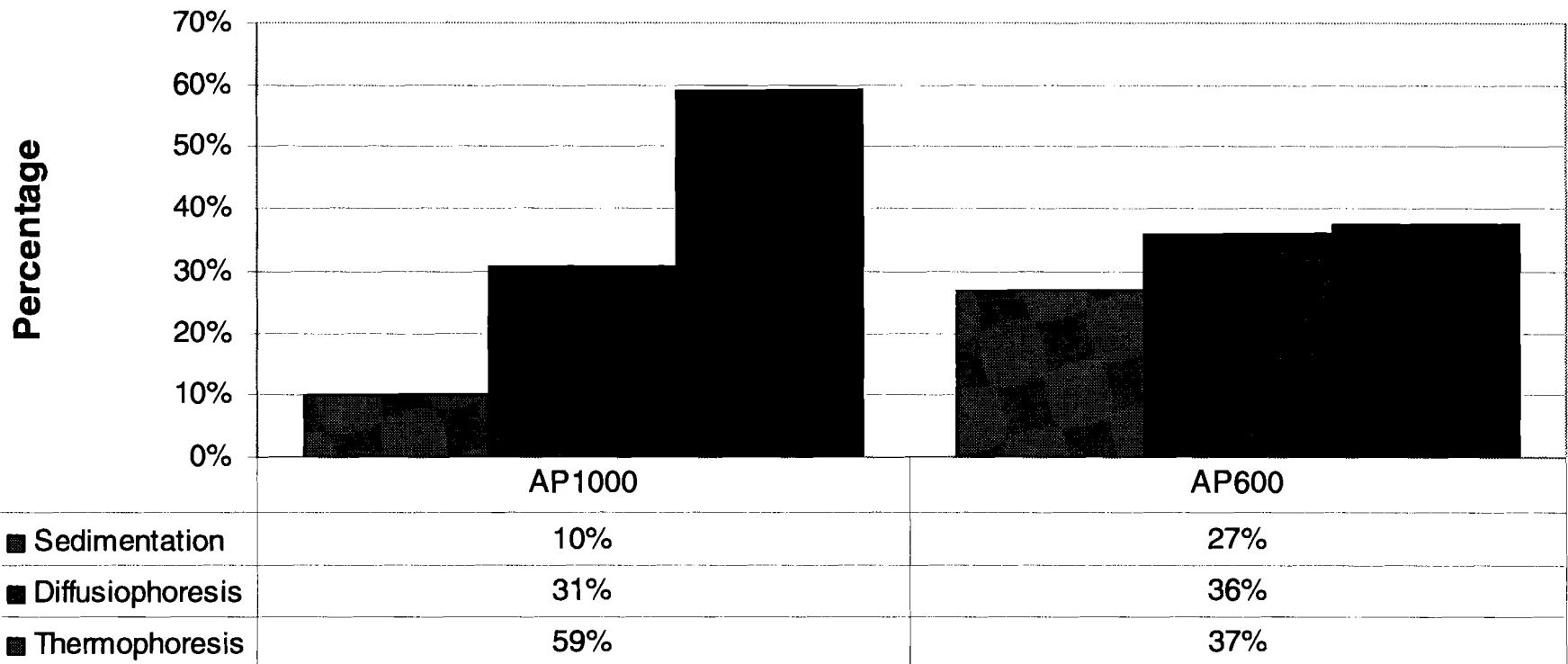
$$v_{th} = \frac{2C_s Cn(Kn)(\mu_v / \rho_v)(\alpha + C_T Kn)}{(1 + 2(\alpha + C_T Kn))(1 + 3C_M Kn)} \frac{1}{T} \frac{\dot{q}''}{k_g}$$

Calculation Results



Calculation Results (cont.)

Aerosol Removal by Different Mechanisms



Discussion

- According to AP600 sensitivity study
 - + Changing FP to inert mass ratio from 0.5 to 3 caused a 5 – 6% change in the leakage.
 - + Cutting the sedimentation area in half increased the leakage by 13 – 14%.
 - + Reducing the packing fraction from 0.8 to 0.1 also increased the leakage by 14%.
 - + When smaller values of r_g and s_g were used, the increases in leakage were only of the order of 5%.
- AP1000 lambda variation will be lower than the percentages above since sedimentation is relatively less important in AP1000 than in AP600.

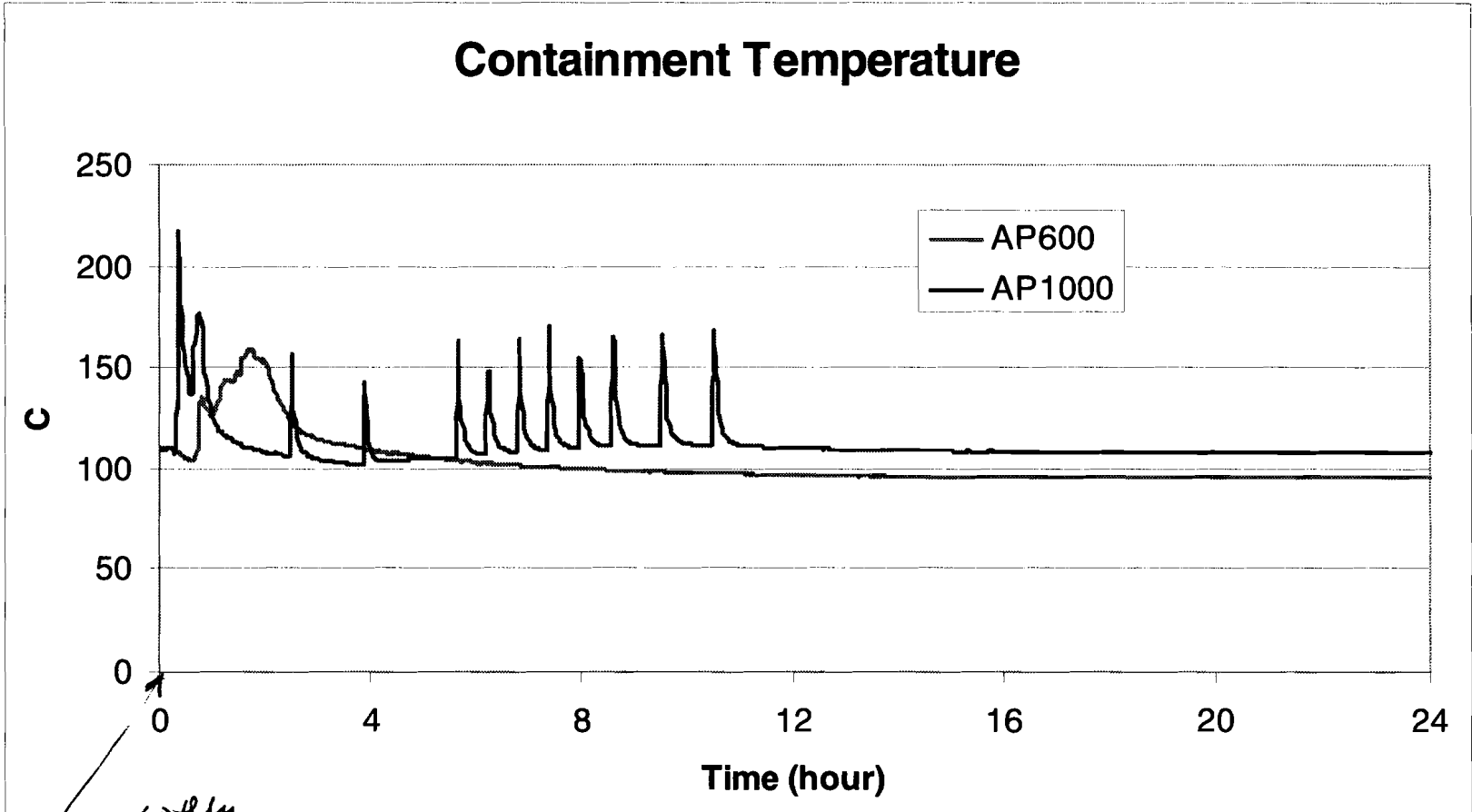
Discussion (cont.)

- Conservatism
 - + Hygroscopicity neglected.
 - Hygroscopic particles grow bigger and settle faster.
 - + Inertial impaction on wet surface neglected.
 - + Retention of aerosols in leak paths neglected.
 - Experiment (e.g., LACE) shows that aerosols tend to form sticky material that either be retained in narrow path or fall quickly to the ground, if they resuspended, after leaving the path.
 - + Smaller than usual particle size (mmd=1.3 μ m) used.
 - Mass mean diameters used in NUREG/CR-5966 for the analysis of natural aerosol removal rates range from 1.5 to 5.5 μ m.

Conclusions

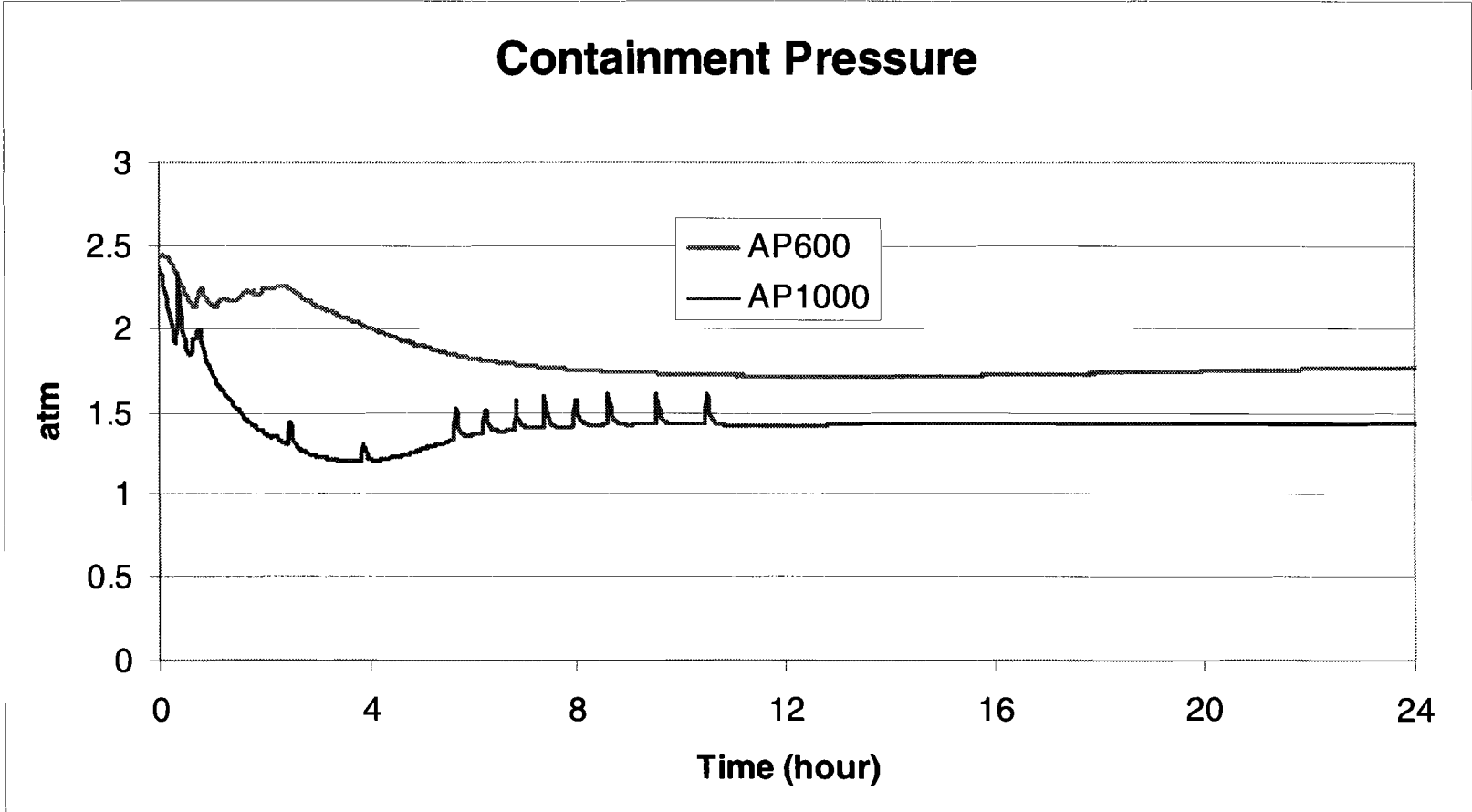
- Average lambda is 1.1 per hour.
- Result is quite robust due to combination of removal mechanisms ($\pm 25\%$ variation on “exaggerated” sensitivities).
- The results are conservative.

Containment Temperature

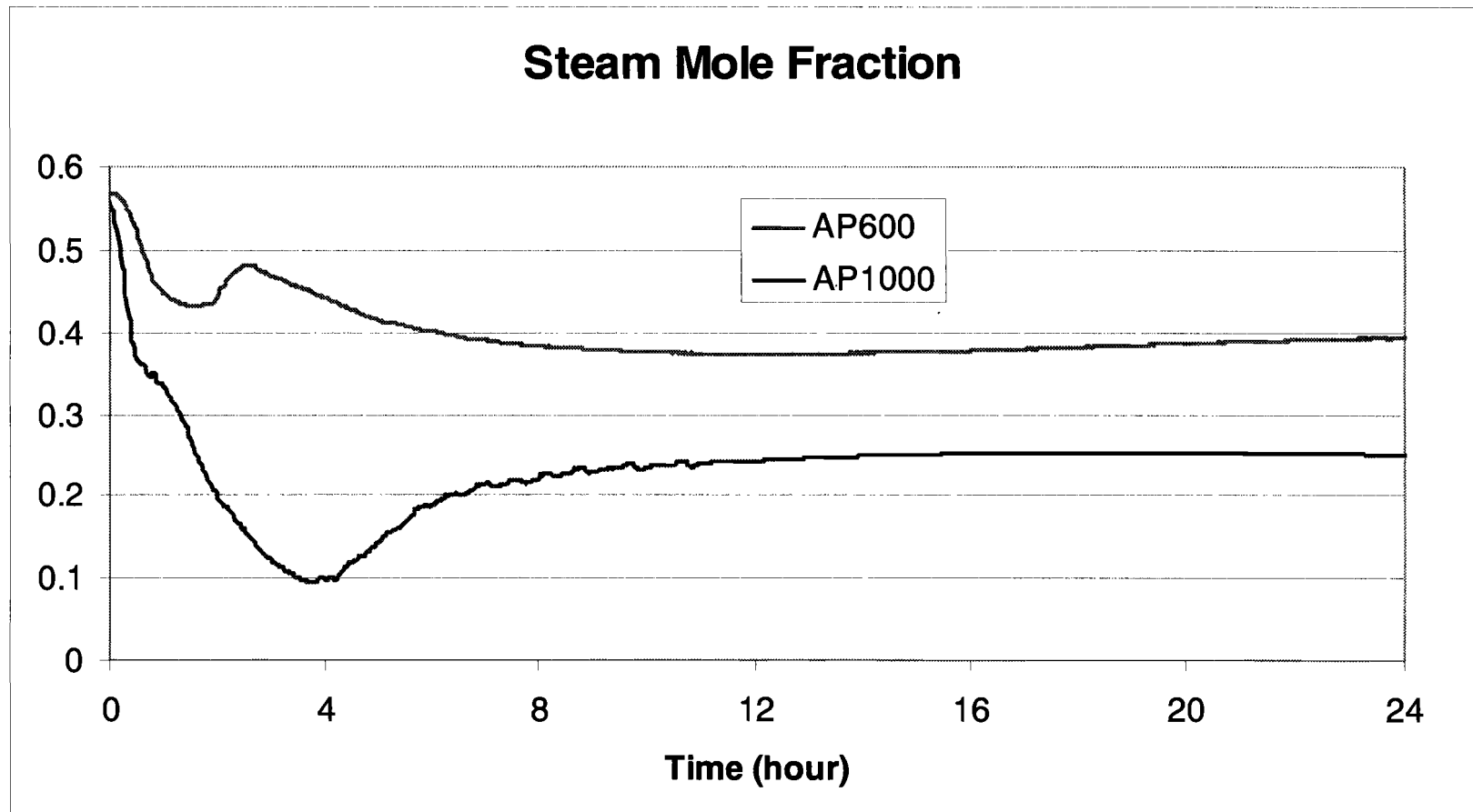


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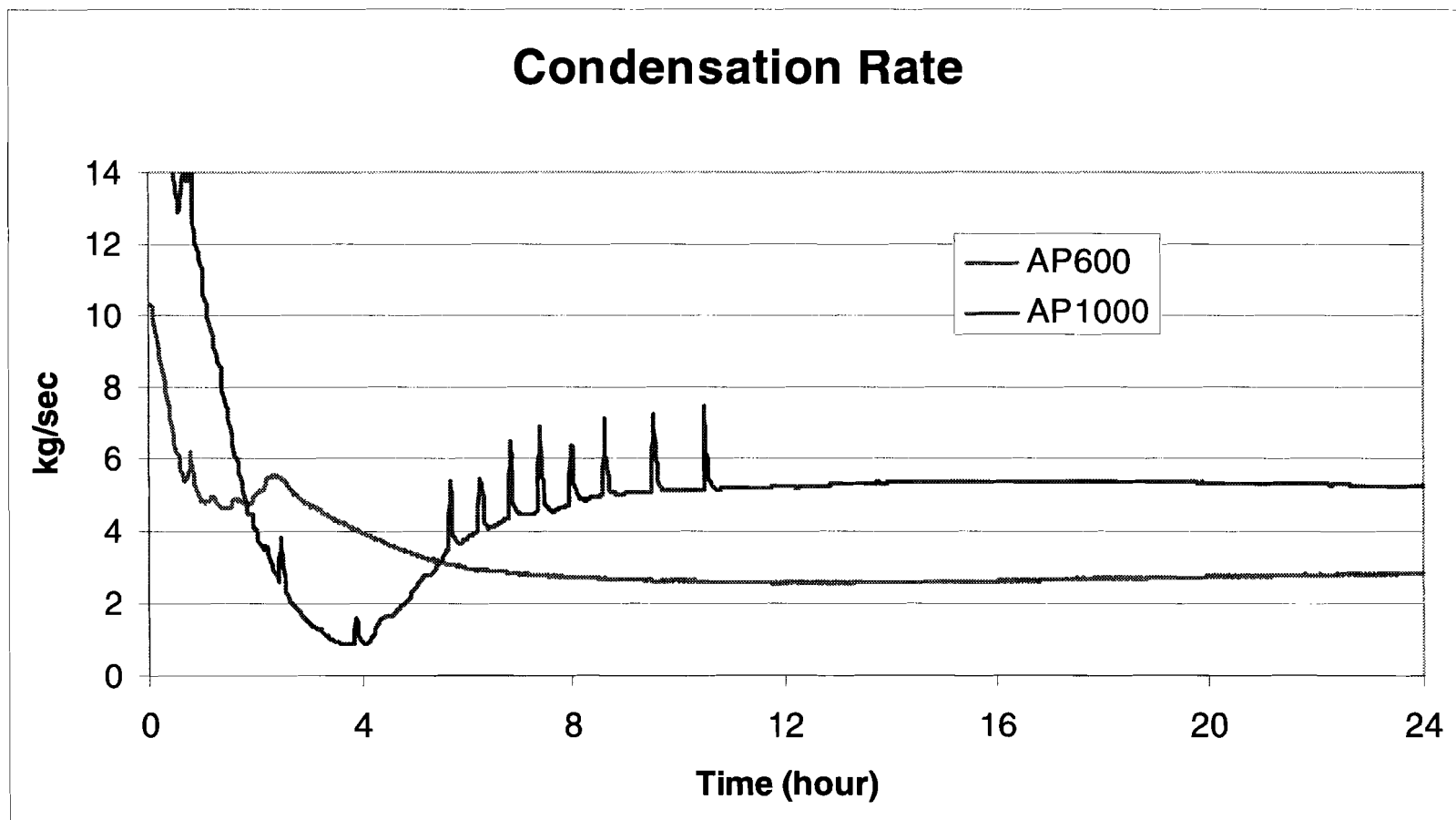
Containment Pressure



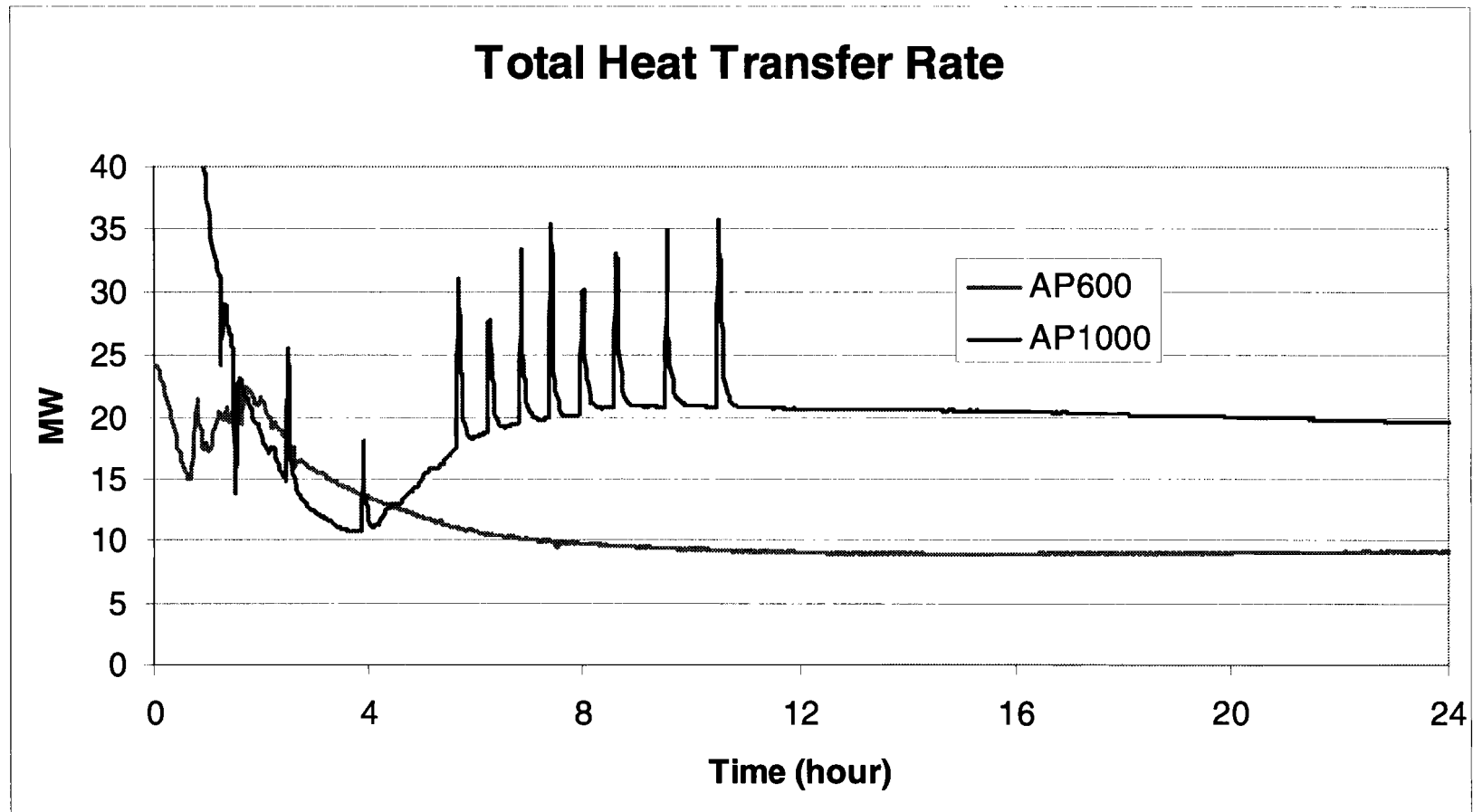
Steam Mole Fraction



Condensation Rate



Total Heat Transfer Rate



STARNAUA 1.03

- Purpose
 - + STARNAUA 1.03 was developed to perform containment aerosol removal calculation in support of work to develop and apply a physically-based DBA source term for advanced and operating LWRs.
- History
 - + NAUA Mod 4 code was developed in early 1980s by Kernforschungszentrum Karlsruhe (German) to assess dry aerosol behaviors (sedimentation and diffusion only) in a containment.
 - + NAUAHYGROS was developed by EPRI in late 1980s as an enhanced version of NAUA Mod 4 with added treatment of hygroscopic aerosols and diffusiophoresis, and was benchmarked against LACE tests with good agreement.
 - + STARNAUA 1.03 was developed in 1990s by Polestar as an enhanced version of NAUAHYGROS which added aerosol removal by thermophoresis and spray.

STARNAUA Validation

- STARNAUA 1.03 was benchmarked against NAUA Mod 4 and NAUAHYGROS using the sample problems in the manuals. The results matched perfectly.
- STARNAUA 1.03 was benchmarked against LACE test with a good agreement as shown below:

	Settled (g)	Plated (g)	Leaked (g)
LA-4, test data	4490±450	532±110	108±33
LA-4, calculated	4317	686	138.5

STARNAUA Application

- STARNAUA was used for the design basis radiological calculation for AP600 design certification.
- STARNAUA has been or is being used in a number of U.S. operating plant design basis source term applications including Browns Ferry, Oyster Creek, Columbia Generating Stations, Perry, and Vermont Yankee.



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July 30, 2003

To: ACRS Subcommittee on Future Plant Designs

Subject: AP1000 Fluid Systems Design & QA Procedures

1. Purpose

At the July 18th Meeting of the ACRS Subcommittee Meeting on Future Plant Designs held in Monroeville, Pennsylvania, I took advantage of the opportunity afforded members of the public to remark on a topic discussed at that meeting: the NRC's review of QA control of processes used in the AP1000 design currently under licensing review. At that meeting, the NRC staff (Ms. Joelle Starefos) responded by saying that the NRC staff would reply in a letter.

As I did not know which open items were going to be discussed, my remarks were impromptu and I did not have a prepared text. The purpose of this letter is to provide a written statement of the concerns I expressed at that meeting, which made reference to concerns I had expressed earlier, at the 501st ACRS meeting. (References 6 and 7) For completeness, I also include a chronology of the questions and responses already received from other members of the NRC staff in sections 2.1 and 2.2 below. The statement incorporating the concerns I raised at the July 18th, 2003 ACRS meeting appears in section 2.3 below.

According to the policy on Advisory Committee Meetings (10CFR7.12 (b)), " Any member of the public who wishes to do so shall be permitted to file a written statement with an NRC advisory committee regarding any matter discussed at a meeting of the committee." I am filing this letter as such a written statement, as a member of the public, unaffiliated with any organization.

I am currently a professor of philosophy at Duke University in Durham, North Carolina. Prior to my academic career, I worked in the nuclear power industry, including a few years in the mid-nineties on the AP600 fluid systems design as a consultant to Westinghouse. My involvement with the nuclear power industry ended in early 1998 when I began my academic career in philosophy full-time.

I began following the NRC licensing review of the AP1000 in mid-2002 by reading the information publicly available via the NRC's electronic reading room. My knowledge about the AP1000 design and licensing review comes from reading these publicly available documents. I decided to make use of the provisions for public participation in the AP1000 licensing process (References 8, 9) in part because, according to the 10CFR52 licensing process under which the AP1000 is being licensed, opportunities for public participation are extremely limited once design certification is granted. Thus, as a member of the public, providing this input about the AP1000 design and licensing review is a "now-or-never" situation.

2. Chronology of Questions and Statements

2.1 Two Issues Raised with NRC Staff in July 2002 -- Systems Design & AP1000 QA

In mid-2002 (July 10), after the AP1000 design certification submittal, I asked questions about the general 10CFR52 process and the AP1000 licensing review in particular in an email exchange with Jerry Wilson of the NRC. (Reference 3) One question was: what ensures that, by the close of the licensing process, the design process for some components was not still at the stage wherein only preliminary sizing of components had been performed.? In particular, I asked:

"(i) Are there supposed to be signed-off, proof-of-design calculations, (using the actual piping sizes, equipment parameters, and layout) for the flows reported for all the systems in the AP1000 DCD submitted? Or, performance analyses for the more complex pieces of equipment such as the pressurizer, the steam generator, large control and relief valves, etc.?"

(ii) Does the submittal of the DCD imply that the things in (i) are done?

(iii) Does the NRC verify or ask for proof that the things above are in fact completed and signed off by the appropriate functional groups, and that they justify the design details in the DCD? If so, when does this occur?

[Reference 3]

In reply, Jerry Wilson cited 10 CFR 52.47(a)(2), and explained that the level of detail required for a DCD (design control document) submittal was sufficient information to support a safety finding in any technical area, and that this level of information corresponds to the level that, under the previous two-step 10CFR50 process, was available at the operating license stage. However, he qualified this by saying that, since design acceptance criteria were to be used in the "piping design area", that "we [NRC staff] didn't expect that signed-off, proof-of-design calculations will need to be completed to support construction." [Reference 10]

This reply made me wonder whether the NRC was in practice approving delaying performing the proof-of-design calculations for system flows, temperatures, and pressures to later stages as well, without explicitly meaning to do so. The rationale for accepting the (DAC) approach for "the piping design area", which was articulated in SECY-02-0059 [Reference 2], was based on the ability to specify piping stress and piping structural analysis acceptance criteria; that rationale does not support delaying the fluid system design to the later COL stage. It is in fact important that the finalized fluid system design be performed prior to or in conjunction with specifying pipe sizes and valve characteristics to be used in the final design. It is always possible to use preliminary calculations to size piping, valves and equipment in order to obtain values to be placed in a design certification application. Proof-of-design calculations differ from preliminary sizing calculations in that they are a set of calculations chosen to take into account all the system criteria that must be met in order for the system to perform the capabilities that are claimed for it. As explained in followup emails, in lieu of using complete piping layout information as input to "proof-of-design" calculations, L/D criteria can be specified based upon "proof-of-design" calculations; these can then be used in piping layout to ensure that the considerations underlying the "proof-of-design" calculations are met. This kind of criteria would be the fluid systems design analogue of piping DAC. My worry was that unless some attention was paid to ensuring that the "proof-of-design" kinds of analyses are done, whether in the form of calculations using "as-built" data or in the form of L/D layout criteria, that the NRC would actually be certifying a design that was based on preliminary sizing considerations rather than on proof-of-design calculations that document that the various fluid systems have actually been designed to provide the system capabilities claimed for them. Since such fundamental things as the classification of initiating events assumes that even many non-safety systems actually do provide the capabilities attributed to them by the design documents, the issue is related to the safety basis of the plant even for the design of non-safety systems.

The problem is particularly acute on the AP1000 because much of the AP1000 makes reference to AP600 documentation. This makes it especially difficult to discern whether a particular pipe size and equipment parameter is merely inherited from the AP600 design or whether final "proof-of-design" kinds of calculations specific to the AP1000 have been performed to support it. Further, there is the danger of making the false assumption that if a system configuration has not changed, the fluid system performance has not changed either. This is not always true; a system temperature or pressure in one system can affect the fluid system performance in another. Thus reasoning about the similarity to AP600 layout that applies for piping stresses and loads does not necessarily extend to fluid systems performance. A comprehensive review of the AP1000 fluid systems designs is called for, similar to the kind of review appropriate when reviewing an extended power uprating.

In further email exchanges with the NRC (Jerry Wilson and Larry Burkhart), I tried to clarify my first question about the fluid system design. These emails are references 11 and 12 and are attached to this letter.

The second question I asked in my July 10, 2002 email to Jerry Wilson concerned the QA program covering the engineering design processes. I wrote there:

The AP1000 design processes cannot be exactly the same as for the AP600, simply in virtue of the fact that the AP1000 refers to so many design documents for the previously certified, yet different, AP600 design. If the quality assurance program covers the engineering design processes, it seems it needs to be looked at (and maybe revised or supplemented) to ensure that it appropriately covers the case of producing a new design that references another, different, certified design, and to explicitly state what is required in such a case. Here's why I think it is a very important issue:

The AP1000 DCD claims that many of the AP600 documents are applicable to the AP1000. The crucial question is, who (in Westinghouse) makes the determination that a particular AP600 document does in fact apply for the AP1000? It seems to me crucial that the same engineering functional group (preferably the same individual engineer) that was responsible for producing and signing off the document for the AP600 pass judgement on its applicability to the AP1000. Is there a guarantee of this? If not, I suggest that there be such a requirement and that it be made explicit.

Otherwise, there is a gigantic loophole that can be used to circumvent the whole intent of the quality assurance provisions covering the engineering design process -- i.e., otherwise, individuals in other functional groups such as marketing, licensing, or project management, can circumvent the engineering process by simply stating that a certain AP600 engineering report or design document applies to the AP1000. (I don't think I need to explain the conflict of interest involved were this to be permitted.)
[Sterrett to Wilson July 10, 2002 Reference 3]

Jerry Wilson replied to this question as well [Reference 10]. He referred me to the NRC's letter on the AP1000 Design Certification Review Schedule [Reference 4], and explained that the NRC staff did plan to inspect Westinghouse's implementation of its design control program for the AP1000 design "in the future." Mr. Lyons's letter of July 12, 2002 stated that the NRC planned to perform these inspections "as necessary", adding that "These inspections will be coordinated with Westinghouse to support the design certification schedule." [Reference 4 , p. 4]

2.2 Clarification & Discussion of Issues with NRC Staff -- December 2002

In December 2002 Larry Burkhart, who was then the NRC's AP1000 Project Manager, held a telecon to discuss my questions. Jerry Wilson, Dave Terao, and other members of the NRC technical staff were present. In this telephone conference call, I clarified my question about fluid system design. Nothing was resolved other than the clarification of the question. However, it was agreed that we should get in contact again to revisit the issues closer to the time the DSER was about to be issued.

Subsequently, after unsuccessful attempts to reach Larry Burkhart in March 2003, I learned that there had been a change in management of the NRC's AP1000 Licensing team. The entire team had been replaced with the current team (John Segala, Joelle Starefos and Joseph Colaccino).

2.3 Concerns Raised at ACRS Meetings (April & July 2003)

Soon thereafter, I requested time to speak at the 501st ACRS meeting held on April 11th, where I read a statement presenting the first question I had raised in the original July 10th email. My oral presentation followed the draft text of my comments fairly closely [Reference 7 , included as Attachment II to this letter] and was included in the summary report for the 501st ACRS meeting [Reference 6].

The second question raised in my original email (regarding quality control procedures governing the design processes used in the AP1000) was brought up at an ACRS Subcommittee on Future Plant Designs held on July 18th, 2003, shortly after the NRC issued the Draft Safety Evaluation Report (DSER), and almost a year after I sent the original email expressing concerns about the QA process on the AP1000.

The list of AP1000 DSER Open Items included Open Item 17.3.2-2, which reads in part:

Westinghouse stated that a project-specific quality control plan was used to implement the requirements of the Westinghouse QMS program. The staff plans to conduct an inspection of the implementation of the project-specific quality plan to verify that design activities conducted for the AP1000 project complied with the Westinghouse QMS and the requirements of 10CFR Part 50, Appendix B. [Reference 5]

However, the "project-specific quality control plan" Westinghouse refers to is just the AP600 plan. Although Open Item 17.3.2-2 indicates "N/A" for the original RAI corresponding to the open item, there was an RAI about the AP1000-specific quality assurance plan [RAI 260.008-1 dated May 13, 2003]. Westinghouse's response to that RAI had been to claim that the AP600 document applied to the AP1000. The rationale given in Westinghouse's response to RAI 260.008-1 was:

As the DCD identifies: " The plan ... is applicable to work performed for the AP1000 design." Westinghouse considers that it has identified a project specific quality plan (i. e., WCAP- 12600) for the AP1000 design.

There is also a discussion of the use of the AP600 project quality plan in Chapter 17 of the DSER, which states:

A project-specific quality plan was issued to supplement the quality management system document and the topical reports for design activities affecting the quality of structures, systems, and components for the AP600 project . . . This plan addresses the NQA-1-1989 edition through NQA-1b-1991 addenda and is applicable to work performed for the AP1000 design. [Reference 1, page 17-1]

These statements raise concern, for the reasons mentioned in my original July 10, 2002 email and excerpted in section 2.1 above. When I attended the ACRS Subcommittee Meeting on Future Plant Designs held on July 17th and 18th, I did not anticipate that the subject open item would be mentioned, and did not request time to speak beforehand. However, when I saw that the NRC's presentation included mention of the issue of an inspection of Westinghouse's QA plan during the meeting, I asked to make some impromptu remarks along the lines of the concern raised in my email. There was not time to gather the previous correspondence, relevant Open Items, RAIs, and RAI responses at that time. Therefore, I provide a more complete statement of the situation and my concerns about it here.

My concerns regarding QA of the AP1000 design process are:

A. Integrity of design process for the singular kind of project that the AP1000 is

The kind of process by which the AP1000 design was produced resembles an uprating in some ways, in spite of the fact that it is not regarded as an uprating. That is, one constraint was to use the AP600 design details insofar as possible. An uprating involves activities and considerations not addressed by the kind of design control procedures intended to address design of a plant where the design process starts with the specification of plant parameters and detail is filled in as the design progresses from functional specifications to detailed equipment specifications. Thus I would not expect the AP600 design control procedures to cover all the design processes on the AP1000.

Of special concern is QA control of the overall plant parameters, both in terms of the design process by which they were obtained, and the design processes that use them as input. (Perhaps this question was dealt with in the pre-application phase, but in case not, I raise it here.) I believe the generation of overall plant

The issue here is the QA control on information that is in the DCD: was there design control guaranteeing that the generation and implementation of the basic plant parameters for the AP1000, as well as the fluid systems design details (e.g., equipment parameters, piping size, valve specifications) were the result of design work of the appropriate kind (i.e., not merely preliminary sizing calculations), performed in a context where there was proper control of design information input into the design process, and where there were the appropriate checks and balances that provide assurance of the integrity of the design process? If it turns out there were areas where it was not, it seems there is not a lot of time to allow review and comment on the required design changes if the design certification schedule is to be adhered to.

3. Additional Remarks -- Schedule for Resolution of DSER Open Items and Role of Public Review and Participation

In general, the AP1000 design certification schedule seems to permit a number of potentially significant open items at the DSER stage. This limits the time available for review and comment by the public after the open item is resolved. Considering the finality of a design certification, it seems that the time available for public review and comment should not be abbreviated in the only stage provided for it.

Respectfully submitted,



Susan G. Sterrett
Assistant Professor of
Philosophy
Duke University, Durham, NC

Attachment I Email correspondence Sterrett to NRC dated September 15, 2003.

Attachment II Draft Text of Comments Read at 501st ACRS Meeting --Dr. S. G. Sterrett

References:

1. AP1000 Draft Safety Evaluation Report (DSER)
Chapter 17 "Quality Assurance"
2. SECY-02-059 April 1, 2002
"Use of Design Acceptance Criteria for the AP1000 Standard Plant Design"
William D. Travers to The Commissioners
3. Email S. G. Sterrett to J. N. Wilson,
"AP1000 Review/ 10CFR Part 52 Process"
Wednesday, July 10, 2002
4. Letter July 12, 2002 James E. Lyons to W. E. Cummins
"AP1000 Design Certification Review Schedule (TAC NO. MB4682)"
5. Letter May 27, 2003 James E. Lyons to W. E. Cummins
"Westinghouse AP1000 Draft Safety Evaluation Report
Potential Open Items Chapter 17 Quality Assurance"
6. Letter May 7, 2003 M. Bonaca to Nils J. Diaz
"Summary Report of 501st Meeting of the Advisory Committee
on Reactor Safeguards, April 10 - 12, 2003"
7. Draft text of comments by S. G. Sterrett at 501st ACRS Meeting
(Attachment II to this letter. No transcript of April 11th meeting was made; oral
statement followed written draft closely)
8. Nuclear Regulatory Commission Policy Statement
"Enhancing Public Participation in NRC Meetings; Policy Statement"
Federal Register May 28, 2002 (Volume 67, Number 102; Page 36920-36924)
9. 10 CFR7.12 Public Participation in and public notice of advisory committee
meetings.
10. Email J. N. Wilson to S. G. Sterrett
"Re: Followup on Questions: AP1000 Review/ 10CFR Part 52 Process"
August 13, 2002.
11. Email S. G. Sterrett to L. J. Burkhart
"Thanks for RAls"
September 15, 2002
(included in Attachment I to this letter)

12. Email S. G. Sterrett to J. N. Wilson
"Piping Layout L/D Criteria for Fluid System Performance"
September 15, 2002
(included in Attachment I to this letter)

cc:

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New Reactor Licensing Project Office
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Mr. Joseph Colaccino, Project Manager, AP1000 Licensing
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Ms. Joelle Starefos, Project Manager, AP1000 Licensing
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ATTACHMENT I

Emails Sterrett to NRC (L. J. Burkhart; J. N. Wilson) dated September 15, 2003

This first email clarifies a question sent earlier to Jerry Wilson and discussed by telephone with Larry Burkhart . In it, I explain why the question is not addressed by the considerations provided in the rationale used in accepting DAC for the AP1000, nor covered by the RAIs sent to Westinghouse as of that date. The email below is followed by a longer one addressed to Jerry Wilson and cc'd to Larry Burkhart and Marsha Gamberoni.

Date: Sun, 15 Sep 2002 16:21:36 -0400 (EDT)
From: sterrett@duke.edu
To: Lawrence Burkhart <LJB@nrc.gov>
Subject: Thanks for RAIs

Dear Larry,

I have looked over the RAIs, and don't see any that address the question I asked Jerry Wilson about paying attention to fluid system performance in doing the piping layout. The RAIs do mention thermal-hydraulic loads, but that isn't what I meant; thermal-hydraulic loads are still related to the mechanical loads on the piping and concern the piping structural-mechanical analysis.

What I meant is the fluid system performance -- flowrates, pressures and temperatures that are achieved by the combination of driving head and fluid piping resistance. The fluid piping resistance is affected by the piping layout. In an email to Jerry Wilson, which I put you on cc for, and which I will send immediately after this one, there is more explanation. The bottom line is that even though the piping layout isn't final, the piping resistance criteria ("L/D criteria") for the AP1000 should be computed and provided at this point. In that email, following this one, there is also an explanation as to why the L/D criteria for the AP1000 will be different in many cases from the AP600.

In our conversation, you mentioned that the AP1000 is so similar to the AP600. That may be, but the question is, should the piping layout really be so similar? It is the fluid system's performance that sets the requirements of the design, and the layout has to meet those criteria. That's the point. One has to check, not just assume it will all turn out okay.

In spite of the length of this email, the two points are simple; I am just including the text of the things I reference to avoid any possible ambiguity.

(a) Clarification of Question Re: Calculations Supporting Fluid System Performance

To recapitulate, the question I asked (July 10) was:

`1. What point of maturity is the design supposed to have at the stage the AP1000 application is presently at? I take it that by the time a design is certified, it is not supposed to be one for which only preliminary sizing calculations have been performed to size the equipment. What ensures this doesn't happen?

(i) Are there supposed to be signed-off, proof-of-design calculations, (using the actual piping sizes, equipment parameters, and layout) for the flows reported for all the systems in the AP1000 DCD submitted? Or, performance analyses for the more complex pieces of equipment such as the pressurizer, the steam generator, large control and relief valves, etc.?

(ii) Does the submittal of the DCD imply that the things in (i) are done?

(iii) Does the NRC verify or ask for proof that the things above are in fact completed and signed off by the appropriate functional groups, and that they justify the design details in the DCD? If so, when does this occur?" [excerpt from email of July 10, 2002 Sterrett to Wilson]

In your response (August 13) you explained why proof-of-design calculations for fluid system performance were _not_ expected to have been performed at the time of DCD submittal:

`With regard to question #1, the Commission expects that when submitted, the design maturity is equivalent to the level of design information available at the operating license stage under the old 2-step process in Part 50 (Final Safety Analysis Report). The NRC's requirement for the level of detail of design information supporting an application for design certification is set forth in 10 CFR 52.47(a)(2). Specifically, it is sufficient information to support a safety finding in any technical review area. However, with regard to piping design, Westinghouse is proposing to use design acceptance criteria in lieu of detailed design information for design

certification. The Commission found that approach acceptable for the ABWR and System 80+ designs. Therefore, for questions #1(i) and (ii), we didn't expect that signed-off, proof-of-design calculations were complete when the DCD was submitted. However, piping design calculations will need to be completed to support construction and the NRC will do verification inspections of the design and construction activities [#1(iii)]. `` [excerpt from email of August 13, 2002 Wilson to Sterrett]

I would like here to clarify my earlier question: by ``proof-of-design calculations'', I was referring to proof-of-design calculations for fluid system performance, rather than to piping design calculations. By ``piping design calculations'', I assume you are referring to calculations concerning things such as piping stress, fatigue and mechanical loads. But, of course, the proper flow performance of fluid systems sets another kind of criterion: that is, in addition to the criteria that aim to ensure that the structural/mechanical behavior of the piping is acceptable, piping layout activities also have to take into account criteria that ensure that the piping flow resistances will result in the flows through the system called for by the fluid system design (and for which the design of numerous interfacing systems may take credit). In addition, pressures (and, sometimes, temperatures) in the system at various key points, such as at heat exchangers and control valves, are influenced by the piping layout. And here I am including normal system operation. Your response to the question of whether there have been proof-of-design calculations for fluid flow performance was that you did not expect them to be done, because the piping layout wasn't final.

However, if the piping layout isn't far enough along to permit proof-of-design calculations to be performed, the calculations related to fluid system performance should still be done -- the only difference is that they would result in piping fluid flow resistance criteria, or ``L/D criteria."

From your response, I wasn't sure if ``L/D criteria'', or piping fluid resistance criteria were included in the DAC. After looking at various meeting transcripts and the RAIs regarding DAC attached to the meeting notice for September 9, 2002 (Reference 3), it doesn't appear to me that the ``L/D criteria" are addressed in these places.

So, the question is whether L/D criteria have been provided for the AP1000 fluid systems. Even if the piping layout for the AP1000 were exactly the same as the AP600 layout, new L/D criteria would need to be calculated for the AP1000. For, anytime the design flowrate for a system changes, the

L/D criteria need to be re-calculated, since piping flow resistances vary with flowrate. Even for those systems, if any, where the fluid flowrate of the system is exactly the same for the AP1000 as it was for the AP600, there is still the question whether there are differences in the inlet or outlet pressures -- i.e., in the pressure in the system or piece of equipment to which it connects and from which the fluid enters the fluid system or to where it discharges. Hence the fluid flow performance would be different for the same layout. Thus, the layout criteria would differ between the AP1000 and the AP600 for cases where a system's inlet or discharge pressures differ. (An example here of such a difference in the AP1000 is the significant change in main steam pressure: obviously L/D criteria will be different between the AP600 and the AP1000 for the inlet piping to the steam relief valves, for example.)

Thus, to rephrase the question in my July email:

``(i) Are there supposed to be signed-off, L/D criteria and supporting calculations, (using the AP1000 fluid system functional requirements and equipment parameters) for the system flows and pressures reported for all the systems in the AP1000 DCD submitted? Or, L/D criteria for the piping associated with the more complex pieces of equipment such as the pressurizer, the steam generator, large control and relief valves, etc.?

(ii) Does the submittal of the DCD imply that the things in (i) are done?

(iii) Does the NRC verify or ask for proof that the things above are in fact completed and signed off by the appropriate functional groups, and that they justify the design details in the DCD? If so, when does this occur?"

This is the question I have now, given your response that you did not expect ``proof-of-design calculations" to be performed due to the fact that the piping layout is not final at the DCD application stage.

(b) Previous process versus new 10CFR52 process

It is simply good common sense to provide L/D criteria for the preliminary piping layout, in order to have confidence that when the final piping layout is in fact completed, the design will be such that the fluid performance functional requirements of the system are in fact met, avoiding major changes to the preliminary layout. As you may be aware, this is the process that was followed on the Westinghouse standard plants.

As I see it, requiring that L/D criteria for performance of fluid system functional requirements be provided at the DCD submittal stage in the AP1000 design process is also a policy issue. Here is why: under the older process, L/D criteria were provided to the architect-engineer for use in laying out piping, that is, in the preliminary layout. Thus they were performed PRIOR to the application for an operating license under the old process. L/D criteria can be provided now, as they do not depend upon the piping layout, much less on the piping layout being final. (They are criteria calculated for use in laying out piping such that the fluid system functional requirements (which should be final at the DCD submittal stage) are met.) The L/D criteria are criteria that apply for preliminary layout as well as final layout.

Certainly the ITAACs and other operational tests are going to provide a checkpoint where deficiencies in system performance are found, but, I trust, it certainly isn't the intent of the new 10CFR52 process to increase the surprises encountered during operational testing! I assume that everyone agrees that the intent is to have confidence that the certified design results in fluid systems that meet their functional requirements in terms of flowrates, pressures, and temperatures, even if the piping layout for the certified design may not be final in every detail.

Thus, it seems clear that the L/D criteria should be provided at the DCD submittal stage in the 10CFR52 process. It's an issue of policy because, otherwise, the 10CFR52 process would result in the NRC certifying a design for which there was less confidence in the design than existed under the old process at a comparable stage.

It would be great to hear the answer that L/D criteria for all the AP1000 systems have in fact been calculated and provided, but, in any case, I look forward to your reply. As with my previous inquiry, I am asking these questions as an individual member of the public, unaffiliated with any organization.

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ATTACHMENT II

Draft of Remarks by Dr. S. G. Sterrett - 501st ACRS meeting, April 11th, 2003, Rockville, MD

I'm Susan G. Sterrett. I am currently a professor at Duke University in Durham, North Carolina. I should perhaps mention that, prior to my academic career, I worked as a design engineer in the commercial nuclear power plant industry, including on fluid system design of the AP600 and EPP plants in the mid-nineties. I am making these remarks as a member of the public, unaffiliated with any organization.

I'm here today because I have some questions about the NRC's review of the AP1000. Put briefly, my question is whether the NRC verifies or asks for proof that the system parameters reported in the AP1000 design certification application (and used in the analyses) are actually justified by a detailed design, as opposed to the AP1000 system designs being at the stage of conceptual system design or justified only by preliminary equipment sizing calculations. I'd like a few minutes to explain the relevance and the significance of the question.

According to the rules under which the AP1000 is being licensed by the NRC, the level of design information required in a design certification application is, with a few explicit exceptions, the level of information that was required at the operating license stage under the previous two-step licensing process. I think this requirement makes sense, too, inasmuch as what the NRC is licensing in approving the AP1000 is an actual plant design that is certified to be constructed and operated.

In following some of the AP1000 licensing activities via the NRC's website, I have noticed that much is often made of the similarities between the AP1000 systems and the AP600 systems. This can be misleading: the performance of the various fluid systems in the plant -- that is, the flows, temperatures, and pressures that obtain at various points within a system are affected by many kinds of differences in a plant design. As I am sure everyone here realizes:

--- Anytime a system flowrate changes, pressure drops in the system will change.

--- Likewise, anytime the pressure at some point in a system changes, flowrates in it or some other system can be affected.

--- Thus, even for those systems that are exactly the same physically speaking (i.e., same pipe size and layout) for the AP1000 as for the AP600, there is still the question of whether there are differences in the inlet or outlet pressures in a system or piece of equipment to which it connects. Different inlet or outlet pressures will result in differences in fluid system performance.

For example, suppose the main steam system pressure is different on the AP1000; then, on the AP1000, there would be a different driving head for lines connected to it than there was on the AP600. So, even if the system hardware and layout of a system connected to the main steam system, say, is exactly the same on the AP1000 as it was for the AP600, the resulting values of major fluid system parameters -- e.g., the mass and volume flowrates and the pressures that result -- could be quite different. Obviously the effects on things like the flow capability of relief valve piping and valve arrangements would need to be looked at. Accommodating these changes could require resizing piping or control valves in order to achieve the flowrate claimed for the system.

I've given the main steam system as an example, but the general point holds for every system in the plant. To infer from the fact that the hardware and layout on an AP1000 system is exactly the same as on the AP600, to the conclusion that the performance is the same, is incorrect. The various AP1000 analyses now under review are only as valid as the assumptions made in them about the performance of the plant systems.

What does this point mean for the review of the AP1000 design, which makes frequent appeal to the certified AP600 design? In many aspects of the safety analyses, the NRC has been very alert to the differences between the AP1000 and the AP600. The point of my examples is that this awareness ought to be extended to plant fluid system performance, specifically, that some reassurances should be sought that the fluid system design details for all the plant systems have been properly attended to, and that, given that the level of detail required at this stage is supposed to be the same as that at the operating license stage, these should not be just preliminary sizing calculations. I worry about the complacency with which the AP600 design is referenced in justifying the AP1000 system designs.

The AP1000 is sometimes referred to as an uprating of the AP600 design. Of course this would be significantly larger than any uprating that the NRC has licensed so far, and of course it differs from most upratings in that there is no AP600 operating experience to draw upon. To the extent that thinking of the AP1000 as an uprating of the AP600 is appropriate, however, it would make sense to require that all the plant system reviews that would be required for an extended power uprating be performed for the AP1000. As there is now a draft review standard for extended power uprates that could be used to guide such a review of the AP1000 (RS-001, dated December 2002), this seems a natural thing to do. I wonder whether there has in fact been a review of this sort for the AP1000. So let me ask: has there?

For those systems whose layout is finalized at this stage of the AP1000 design certification application, there should be formally signed-off engineering calculations justifying the claims that the AP1000 system flow, temperature, and pressure parameters will actually be achieved using the AP1000 equipment and layout. These are often referred to as fluid system "proof-of-design" calculations. I gather from the NRC's approval of the use of DAC (design acceptance criteria) for structural piping analysis on the AP1000 that

there may be some systems for which the layout details will not be completed until after design certification. For those systems, what is needed as far as ensuring proper fluid system performance is to provide layout criteria related to the piping flow resistance, so that the fluid flowrates claimed for the system will actually be achieved. Such criteria are commonly called "L/D criteria" and are considered part of the fluid system design. In fact, for the Westinghouse standard plant designs licensed under the previous two-step process, L/D criteria were provided for various fluid systems prior to construction so that the architect engineer could properly perform the piping layout. As I see it, at least this level of design detail is required at the time of the DCD submittal.

Why not just rely on the ITAACs (Inspections, Tests, Analysis, and Acceptance Criteria) to provide such reassurance? Certainly the ITAACs and other operational tests provide a checkpoint where some deficiencies in the plant design would show up. However, I trust that it isn't the intent of ITAACs to relieve the designer of the responsibility of the engineering design work of designing the plant systems so that the system parameters crucial to safety are achieved. Certainly increasing the number of surprises encountered during plant testing is not part of the intent of the new one-step licensing process! I assume that everyone agrees that the intent of design certification is to provide confidence that the certified design will result in fluid systems that meet their stated functional requirements in terms of flowrates, pressures, and temperatures, even if the piping layout for the certified design may not be final in every detail.

In conclusion, I am asking whether the review of the AP1000 design has included ensuring that the design details upon which the analyses that the ACRS has been reviewing depend, have in fact been attended to. In particular, I think it is clear that L/D criteria should be provided at this stage for systems whose layout is to be finalized at a later date, and "proof-of-design" calculations be provided for those whose layout is determined at this stage. Otherwise, there is no assurance that the analyses you are reviewing so carefully and thoughtfully apply to the plant design you are certifying.

Thank you for listening.

Respectfully submitted,

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July 31, 2003

To: ACRS Subcommittee on Future Plant Designs

Subject: Heat of Solar Radiation and AP1000 Ultimate Heat Sink

Although I did not make an oral statement on the subject topic at the ACRS Subcommittee on Future Plant Designs held on July 17th and 18th, 2003, I am taking the opportunity afforded members of the public to file a statement on subjects associated with the topics discussed at ACRS meetings. This statement is related to the AP1000 safety systems and the recently-issued AP1000 Draft Safety Evaluation Report (DSER).

The AP1000, unlike operating PWRs, uses the outside air as the ultimate heat sink. The Passive Containment Cooling System is responsible for transferring heat to the ultimate heat sink in the event of a design basis accident. The question I have is: whether (and if so, how) the heat of solar radiation is taken into account in the design of the AP1000 Passive Containment Cooling System.

As described in the DSER, heat removal from the containment after a design basis accident is to be accomplished by the Passive Containment Cooling System (PCS). The PCS uses the water in the PCS water storage tank located atop the containment, along with the flow of air through the spaces between the primary steel containment and the surrounding concrete building, to cool and depressurize the containment. It is the means by which heat is transferred from the reactor to the ultimate heat sink (the outside air) in the event of a design basis accident.

Thus, the temperature of the water in the PCS water storage tank and the temperature of the concrete walls affect the heat removal capabilities of the PCS. Since the heat of solar radiation can cause the temperature of objects to exceed that of the surrounding air, it seems to me that its effect on:

- (i) the temperature of the concrete building, whose walls form the air passages relied upon for the efficacy of cooling by the PCS, and
- (ii) the temperature in the PCS water storage tank,

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ought to be addressed by the AP1000 design. The effect will vary with geographical location (i.e., one of the coefficients involved is a function of geographical latitude) and will also depend upon the surface geometry, the properties of the concrete and/or the surface coatings used, and the humidity of the outside air.

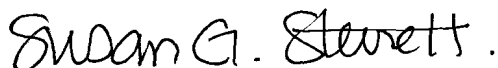
The site parameters do not include geographical latitude, so I am wondering whether the heat of solar radiation was considered or quantified. I do not see the effect of the heat of solar radiation accounted for explicitly in the DSER. However, it is clear that, unless the heat of solar radiation is shown to make only a negligible contribution, this heat source is relevant to the design of the safety features of the plant. The question does not arise for operating PWR plant designs, since those designs do not use the method of containment cooling employed on the AP1000. It appears to me that some of the regulations and criteria related to ultimate heat sink assume that the ultimate heat sink is a body of water; thus I would not expect them to have specifically addressed the effect of heat of solar radiation on the temperature distribution in concrete walls.

Perhaps this was already addressed at earlier stages of the project. However, even if this is so, there should be some discussion in the DSER of the rationale and assumptions used in making the determination that the effect of the heat of solar radiation on the structures used by the PCS for containment cooling could be neglected.

If in fact the effect of the heat of solar radiation on PCS performance is not determined to be negligible, the assumptions regarding PCS water storage tank temperature and PCS efficacy in heat removal used in the AP1000 PRA (Probabilistic Risk Assessment) Report should also be examined to see if the heat of solar radiation might need to be taken into account in the rationales employed there.

I have raised this question with the NRC staff. I do not know what the response will be. However, due to the late point in the licensing process (the DSER is already issued), the safety significance of the ultimate heat sink, and the finality of design certification which limits opportunities to raise the issue later, I am raising it in a statement to the ACRS now.

Respectfully submitted,



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