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

September 10, 2004

MEMORANDUM TO: ACRS Members

FROM: Med El-Zeftawy, Senior Staff Engineer

SUBJECT: CERTIFICATION OF THE MINUTES OF THE ACRS SUBCOMMITTEE MEETING ON FUTURE PLANT DESIGNS, JUNE 25, 2004 -- ROCKVILLE, MARYLAND

The minutes of the subject meeting, issued on August 19, 2004, 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
H. Larson
R. Caruso

MEMORANDUM TO: Med El-Zeftawy, Senior Staff Engineer
ACRS

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

SUBJECT: CERTIFICATION OF THE MINUTES FOR THE MEETING OF THE ACRS SUBCOMMITTEE ON FUTURE PLANT DESIGNS, JUNE 25, 2004–ROCKVILLE, MARYLAND

I do hereby certify that, to the best of my knowledge and belief, the minutes of the subject meeting on June 25, 2004, are an accurate record of the proceeding for that meeting.

T. S. Kress 9/9/04

Thomas S. Kress Date
Subcommittee Chairman

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

August 19, 2004

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

FROM: Med El-Zeftawy, Senior Staff Engineer
ACRS

SUBJECT: WORKING COPY OF THE MINUTES FOR THE MEETING OF THE ACRS SUBCOMMITTEE ON FUTURE PLANT DESIGNS, JUNE 25, 2004–ROCKVILLE, MARYLAND

A working copy of the minutes of the subject meeting is attached for your review. Please review and comment on them at your earliest convenience. Copies are being provided to each ACRS Member who attended the meeting for information and/or review.

Attachment: Stated

cc: ACRS Members
J. Larkins
M. Sykes

ISSUED ON: 8/19/2004
CERTIFIED ON: 9/9/2004

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
FUTURE PLANT DESIGNS SUBCOMMITTEE MEETING MINUTES
JUNE 25, 2004
ROCKVILLE, MARYLAND

INTRODUCTION

The ACRS Subcommittee on Future Plant Designs met on June 25, 2004, at 11545 Rockville Pike, Rockville, Maryland, in Room T-2B3. The purpose of the meeting was to discuss with the NRC staff and Westinghouse representative the AP1000 Safety Evaluation Report and the resolution of any open items and ACRS concerns. The entire meeting was open to public attendance. Med El-Zeftawy was the cognizant ACRS staff engineer and the Designated Federal Official for this meeting. The Subcommittee has received no written comments, or requests for time to make oral statements from any members of the public regarding this meeting. The meeting was convened at 8:30 a.m and adjourned at 1:00 p.m

ATTENDEES

ACRS

T. Kress, Chairman
V. Ransom, Member
S. Rosen, Member
W. Shack, Member
J. Sieber, Member
G. Wallis, Member

NRC

J. Segala, NRR
D. Terao, NRR
G. Bagchi, NRR
T. Cheng, NRR
J. Colaccino, NRR
L. Quinones, NRR
J. Wilson, NRR
S. Bloom, NRR
P. Sekerak, NRR
R. Dening, NRR
L. Dudes, NRR
A. Drozd, NRR
M. Hart, NRR
M. Mitchell, NRR
B. Elliot, NRR
D. Solorio, NRR
G. Cheruvenki, NRR
D. Cullison, NRR

S. Sancaktar, RES

OTHERS

E. Cummins, Westinghouse
R. Vijuk, Westinghouse
C. Frepoli, Westinghouse
L. Sanjur, Westinghouse
T. Schulz, Westinghouse
J. Scobel, Westinghouse
J. Trotter, Framatome
P. Negus, GE
R. Gauntt, Sandia Nat. Lab.
M. Battaglia, Ionics, Inc.
R. Gonder, Ionics, Inc.

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 Kress, Future Plant Designs Subcommittee Chairman, convened the meeting at 8:30 a.m. He stated that the purpose of this meeting is to discuss with the NRC staff and Westinghouse representatives the AP1000 Safety Evaluation Report and the resolution of any open items and ACRS concerns and issues. The Subcommittee will gather information, analyze relevant issues and facts, formulate proposed positions and actions, as appropriate, for deliberation by the full Committee.

WESTINGHOUSE REPRESENTATION

Mr. Ron Vijuk stated that Westinghouse representatives will briefly describe the AP1000 design. Mr. Terry Schulz noted that on March 28, 2002, Westinghouse submitted its application to the NRC for final design approval of the AP1000 design in accordance with Appendix O to Part 52 of Title 10 CFR, and for standard design certification in accordance with Subpart B of 10 CFR Part 52. Accordingly, the NRC staff has reviewed the design certification application and has issued its draft safety evaluation report (DSER) on June 16, 2003.

The AP1000 design is a pressurized water reactor with a power rating of 3415 Mwt with an electrical output of at least 1000 Mwe. The AP1000 design contains features that are new in nature. The most significant improvement to the design is the use of safety systems that employ passive means such as gravity, natural circulation, condensation and evaporation, and stored energy for accident mitigation. These passive systems perform safety injection, residual heat removal, and containment cooling functions.

The AP1000 design is similar in concept to the AP600 design, but provides much higher power leels (1000 Mwe vs. 600 Mwe). Several components were increased in size over those of the AP600 design such as core length, height of reactor vessel, steam generators, canned motor

reactor coolant pumps, pressurizer, containment volume, capacity of passive safety system, automatic depressurization system (ADS) stage-4 squib valve, and turbine capacity.

The AP1000 design is intended to meet the safety requirements and goals defined for advanced reactors with passive safety features. The plant consists of five principal structure: the nuclear island, the turbine building, the annex building, the diesel generator building, and the radwaste building. The nuclear island includes all safety-related and seismic Category 1 structures and is designed to withstand the effects of natural phenomena and postulated events. It consists of a containment building, a concrete shield building, and an auxiliary building.

The AP1000 approach to safety is to use passive process only with no active pumps. There is one time alignment of valves, and no support systems are required after actuation. The design also greatly reduced dependency on operator actions. The design basis accidents will be mitigated without non-safety systems. The AP1000 design meets the NRC PRA safety goals without the use of non-safety systems.

The AP1000 design has a defense-in-depth provision for external flooding of the reactor vessel which is intended to provide for in-vessel retention of any accident induced core melt. The reactor vessel has no bottom head penetrations.

The regulatory treatment of non-safety systems process was used to impose special requirements on some non-safety systems to ensure, with high confidence, that they would be available when needed. These systems provide redundancy and diversity that contribute to achieving very low values for core damage frequency and large release frequency.

Mr. Vijuk stated that the AP1000 utilizes major reduction in bulk materials and field labor. The AP1000 improved construction maximize the use of modularization. It has a factory based manufacture and assembly of modules and a streamlined field installation. Currently, the construction schedule is 36 months. Mr. Vijuk added that the AP1000 comfortably meets the NRC and industry safety standards for future plant .

NRC STAFF PRESENTATION

Mr. John Segala, NRR, stated that the DSER originally contained 174 open items. Resolution of all open items has been completed. The security open items will be reviewed separately.

During the 510th meeting (March 3-6, 2004) of the ACRS, the Committee met with the NRC staff and Westinghouse representatives and discussed the status of the open items as well as issues previously raised by the ACRS. The Committee reviews have not addressed security matters and their impacts on the AP1000 design.

On March 17, 2004, the Committee issued its letter to Dr. William D. Travers, EDO. The Committee outlined seven technical issues in which the ACRS have comments related to the certification of the AP1000 design. These seven issues are:

Issue 1- Automatic Depressurization System (ADS)-4 Squib Valve Function.

Issue 2- Assurance of Long-Term Cooling (Strainer Blockage).

Issue 3- Code Deficiencies.

Issue 4- Range of Pi-Group Values.

Issue 5- In-Vessel Retention/Fuel-Coolant Interactions (FCI).

Issue 6- Organic Iodine Production.

Issue 7- Catastrophic Failure of a free-Standing Steel Containment.

On May 18, 2004, the NRC staff responded to the above 7 issues.

For Issue 1-- the staff indicates that the AP1000 design control document (DCD) will include an ITAAC that ensures the ADS-4 squib valves will perform their function. Tests or type tests will be performed to demonstrate the capability of the ADS-4 squib valves to operate under their design conditions. The staff concludes that the performance characteristics of the ADS-4 squib valves will be adequately verified.

For Issue 2-- the staff states that Westinghouse has revised the COL action item in the DCD to include the evaluation of chemical debris. This COL action item will capture any impact of chemical effects on the ability of the affected components to accommodate anticipated debris loadings identified during the resolution of GSI-191. The staff concludes that Westinghouse has resolved the concern related to additional debris that can be caused by chemical reactions in the containment.

For Issue 3-- the staff indicates that for the AP1000, an acceptable solution was to conservatively bound the calculations. The TRACE code was not used in the AP1000 review, but is currently being assessed using APEX-Ap1000 test data. Currently, there are two ongoing experimental studies to help correct code deficiencies. New thermal-hydraulic models will be developed and implemented into TRACE as deficiencies are identified. The experimental programs, along with the associated TRACE assessment will be described to the ACRS thermal-hydraulic Subcommittee at a future meeting.

For Issue 4-- the staff indicates that an acceptance range of 0.5 to 2.0 for various Pi-groups was selected. The staff notes that the appropriate Pi-groups is generic and does not represent an issue that is specific only to AP1000 designs. The staff, however, as a long term effort will work to develop and document a procedure to define an appropriate Pi-group range for scaling integral test facilities.

Materials-- the ACRS commented that ongoing and future studies may suggest material and environmental changes that will be addressed at the COL stage. The staff notes that any future and environmental changes that would be requested at the COL stage would have to follow the change processes set forth in paragraph VIII.C of the DCD.

Severe Accidents-- the ACRS commented and questioned the technical justification for the aerosol removal coefficient (lambda) for containment. The staff performed an independent dose analysis with the median aerosol removal coefficient values from the staff's uncertainty analysis

, along with other analysis parameters and the bounding hypothetical atmospheric dispersion factors provided by Westinghouse, and the results are within the dose criteria of 10 CFR 50.34 and General Design Criteria 19. The staff concludes that , while the staff and Westinghouse diverge on values for the intermediate steps in the dose calculations, the overall conclusion that the AP1000 dose results are acceptable.

For Issue 5-- the staff notes that it has performed a reasonably large number of sensitivity analysis and found that the ex-vessel FCI for the AP1000 is of no greater concern than that for the AP600 design. The ex-vessel FCI analysis for the AP600 indicated that the containment integrity would not challenged by the FCI load.

For Issue 6-- Westinghouse plans to provide the staff with additional information regarding the pH of the water film on the inside of the containment wall where acidification could produce organic iodine. The staff plans to provide such information to the ACRS along with the staff's evaluation.

For Issue 7-- the staff states that even if an event progressed to an intermediate or late release, it would likely involve a vented release rather than a catastrophic containment failure, since the AP1000 design will include the capability to vent the containment. The non-safety related containment spray function could also be effective in reducing re-suspended fission products following containment failure.

Expected Subcommittee's Action:

The Subcommittee will gather information, analyze relevant issues and facts, and formulate proposed positions and actions, as appropriate, for deliberation by the full Committee. The Committee is expected to write its final report regarding the AP1000 design approval during the July 7-9, 2004 ACRS meeting.

Documents provided to the Subcommittee prior to June 25, 2004:

- **Advanced Copy of the Final Safety Evaluation Report Related to Certification of the AP1000 Standard Design, Volumes 1 and 2, dated May 25, 2004.**

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NOTE : Additional details of this meeting can be obtained from a transcript of this meeting available for downloading or viewing on the Internet at "http://www.nrc.gov/ACRSACNW" or can be purchased from Neal R. Gross and Co.- 1323 Rhode Island Ave., NW,

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
MEETING OF THE SUBCOMMITTEE ON
FUTURE PLANT DESIGNS
ROOM T-2B3, 11545 ROCKVILLE PIKE, ROCKVILLE, MD
June 25, 2004

ACRS Contact: Medhat El-Zeftawy (301) 415-6889
E-mail: mme@nrc.gov

AP1000 Design Certification Review

- PROPOSED SCHEDULE -

Friday, June 25, 2004 - OPEN SESSION

Topic	Presenter	Time
1. Introduction	T. Kress (ACRS)	8:30 - 8:35 am
2. Design Overview	Westinghouse	8:35 - 9:35 am
3. NRC Review Status	NRC staff	9:35 - 9:50 am
** BREAK **		9:50 - 10:05 am
4. DSER Open Items	NRC staff	10:05 am - 12:00noon
A. General Open Items		
B. Post DSER Open Items		
C. COL Action Items		
D. ITAAC		
E. Squib Valve function		
F. Leak Before Break		
G. Sump Performance		
H. Structural/Seismic		
I. Thermal/Hydraulics		
J. PRA		
K. Aerosol Removal Coefficient		
** LUNCH **		12:00 - 1:00 pm
5. ACRS Interim Letter Issues	NRC staff	1:00 - 2:25 pm
A. Overview		
B. Organic Iodine Production Audit Results		
** BREAK **		2:25 - 2:40 pm
6. Summary	Westinghouse/NRC staff	2:40 - 3:00 pm
7. Agenda for 7/7/04 Full Committee Meeting		3:00 - 3:20 pm
8. Public Feedback		3:20 - 3:50 pm
9. Comments by the Subcommittee Members		3:50 - 4:50 pm
** ADJOURN **		4:50 pm

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

SUBCOMMITTEE MEETING ON FUTURE PLANT DESIGNS

June 25, 2004
Date

PLEASE PRINT

NAME	AFFILIATION
Jim Scobel	WESTINGHOUSE
John Trotter	Framatome
ED CUMMINS	WESTINGHOUSE
CESARE FREPOLI	WESTINGHOUSE
RON VIJUK	Westinghouse
LEE TUNON-SANJUR	Westinghouse
Paige Nagle	GE
R.O. Gauntt	Sandia Natl. Labs
MIKE BATTAGLIA	IONKS, INC
RANDOLPH GOUNDER	IONKS, INC.
TERRY SCHULZ	WESTINGHOUSE

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

SUBCOMMITTEE MEETING ON FUTURE PLANT DESIGNS

June 25, 2004
Date

NRC STAFF SIGN IN FOR ACRS MEETING

PLEASE PRINT

NAME	NRC ORGANIZATION
John Segala	NRR/DRIP
~~ED CUMMINS~~	
David Terao	NRR/DE/EMEB
GOUTAM BAGCHI	NRR/DE/EMEB
Thomas Cheng	NRR/DE/EMEB
JOSEPH COLACCINO	NRR/DRIP/RNRP
Lauren Quinones	NRR/DRIP/RNRP
JERRY WILSON	NRR/RNRP
Steven Bloom	NRR/RNRP
PATRICK SEKERAK	NRR/DE/EMEB
Bob Dennig	NRR/DSSA/SPSB
LAURA DUBES	NRR/DRIP/RNRP
Selim Sancaktar	RES/OERAB
ANDREA DONOZO	NRR/SPSB
Michelle Hart	NRR/SPSB
Matthew A. Mitchell	NRR/DE/EMCB
ROB ELLIOTT	NRR/DSSA
BARRY ELLIOT	NRR/DE/EMCB

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

SUBCOMMITTEE MEETING ON FUTURE PLANT DESIGNS

June 25, 2004
Date

NRC STAFF SIGN IN FOR ACRS MEETING

PLEASE PRINT

NAME	NRC ORGANIZATION
DAVID CULLISON	NRR/SPLB
David Solorio	NRR/~~SPLB~~ DSSA/SPLB
Ganesh Cheruvenki	NRR/DE/EMCB

AP1000 Review Status

June 25, 2004

ACRS Future Plant Subcommittee Briefing

John Segala, Senior Project Manager
Office of Nuclear Reactor Regulation

Introduction

- Purpose
  - Provide summary of the staff's review
  - Provide current status of the AP1000 project
  - Discuss major schedule milestones
- Success
  - Understand areas reviewed and resolution
  - Understand the current status
  - Understand the future milestones

Previous Review Milestones

- March 2002 - Completed pre-application review
- March 28, 2002 - Westinghouse (W) submitted DC application
- June 25, 2002 - NRC accepted the application for docketing
- June 16, 2003 - NRC issued DSER with 174 Open Items
- May 18, 2004 – NRC provided responses to the issues in the ACRS interim letter
- May 25, 2004 – Sent advanced copy of FSER to ACRS

Past ACRS Meetings

Subcommittee Meetings

Thermal Hydraulic Phenomena
5 Meetings
March 2001- February 2004

Future Plant Designs
3 Meetings
February 2002 - June 2004

Reliability and Probabilistic Risk Assessment
1 Meeting
January 2003

Full Committee Meetings

9 ACRS Meetings

August	2000 (Pre-Application)
April	2001 (Pre-Application)
March	2002 (Pre-Application)
November	2002
February	2003
April	2003
October	2003
March	2004
June	2004

Remaining Schedule Milestones

- July 7, 2004 – ACRS Full Committee Meeting
- July 17, 2004 – ACRS issues final approval letter
- August 6, 2004 – Division Directors Concurrence
- August 13, 2004 – OGC no legal objection
- August 30, 2004 - EDO memo to Commission w/FSER/FDA attached
- September 13, 2004 – FSER/FDA issued
- December 2005 – Final Design Certification Rule issued

Principal Contributors

Andruszkiewicz, Edward
Attard, Anthony
Bagchi, Goutam
Bajorek, Stephen
Barss, Daniel
Basu, Sudhamay
Bedi, Gurjendra
Behbahani, Ali
Bloom, Steve
Bongarra, James
Brown, Leta
Chang, Kenneth
Cheng, Thomas
Cheruvenki, Ganesh
Chiramal, Matthew
Colaccino, Joseph
Colpo, Sarah
Coyne, Kevin
Cubbage, Amy
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Duvigneaud, Dylanne
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Fields, Leslie
Forrest, Edwin
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Gill, Amritpal
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Lehning, John
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Li, Hulbert
Liang, Chu-Yu
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McIntyre, Richard
Medoff, James
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Mitchell, Matthew
Musico, Bruce
Palla, Robert
Parczewski, Krzysztof
Pettis, Robert
Pohida, Marie
Pulsipher, James
Quinones-Navarro, Lauren
Raval, Janak
Rivera-Varona, Aida
Rogers, Billy
Saltos, Nicholas
Scott, Michael
Scott, Wayne
Sebrosky, Joseph
Segala, John
Sekerak, Patrick
Sheng, Chia-Fu (Simon)
Shukla, Girija
Shum, David
Snodderly, Michael
Starefos, Joelle
Steingass, Timothy
Sullivan, Edmund
Sun, Summer
Talbot, Frank
Throm, Edward
Trehan, Narinder
Unikewicz, Steven
Walker, Harold
Ward, Leonard
Wilson, Jerry
Wu, Shih-Liang

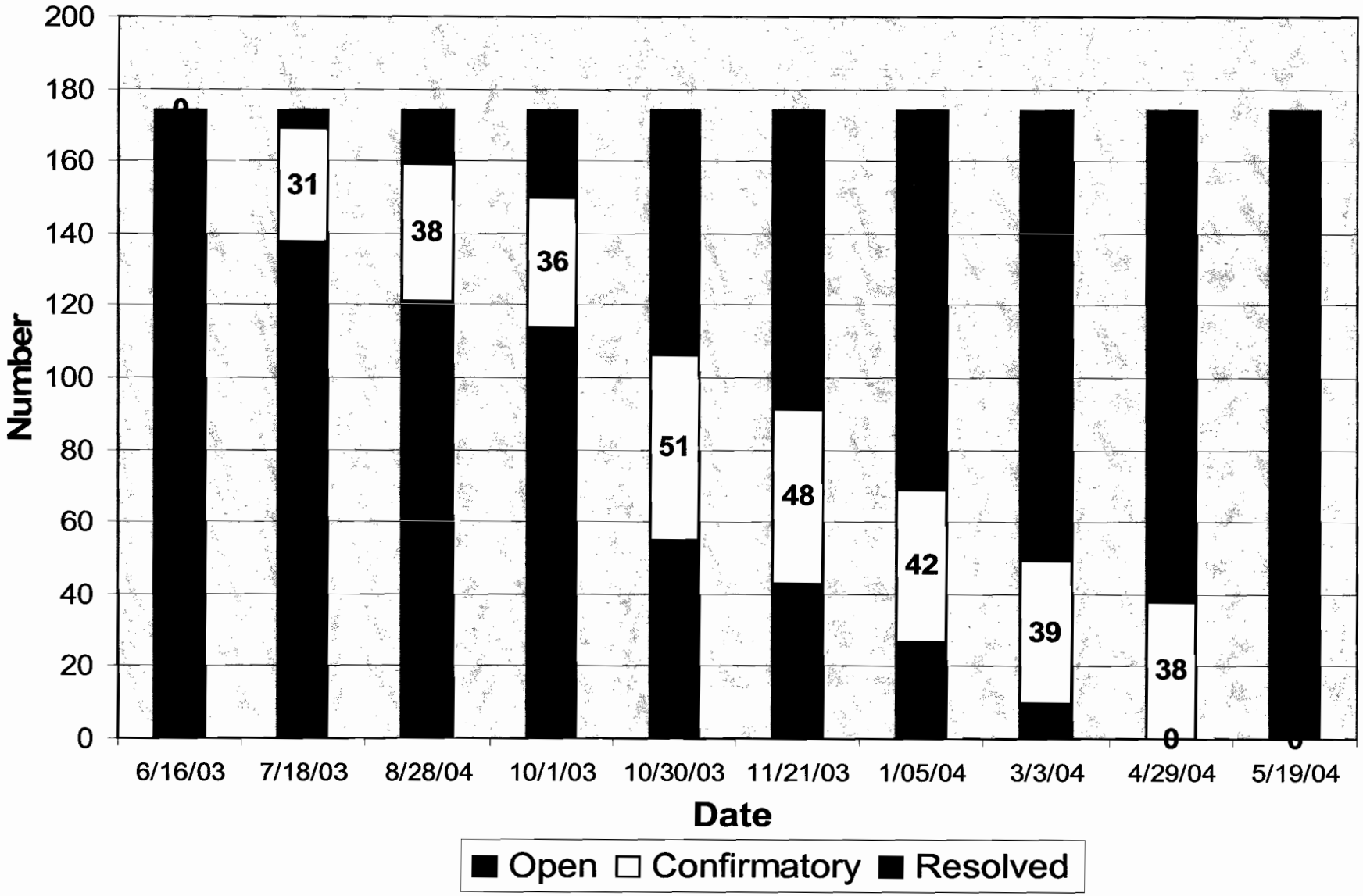
Contractors

- Brookhaven National Laboratory
  - Mechanical and Civil/Structural Engineering
- Information Systems Laboratories, Inc.
  - RELAP5 Input Development
- Energy Research, Inc.
  - Severe Accidents
- Carl J. Costantino Engineering Consultants
  - Structural and Earthquake Engineering
- Sandia National Laboratory
  - Aerosol Removal

RAIs

- We issued 742 RAIs
  - General - **3**
  - Mech. Eng - **70**
  - Structural Eng. - **19**
  - Seismology - **23**
  - Hydrol. and Meteor - **5**
  - Geotech. Eng. - **3**
  - Inservice Inspection - **3**
  - Component Integrity - **29**
  - Materials Application - **12**
  - QA and RAP - **8**
  - Emergency Preparedness - **3**
  - Containment Systems - **11**
  - Technical Specifications - **53**
  - ITAAC - **1**
  - Initial Test Program - **18**
  - Fire Protection - **11**
  - Chemical Technology - **4**
  - Auxiliary Systems - **22**
  - Instrumentation & Controls - **48**
  - Electric Power - **15**
  - Reactor Systems - **189**
  - Meteorology - **8**
  - Effluent Treatment - **11**
  - Radiological Impact - **13**
  - Radiation Protection - **11**
  - Human System Int. - **44**
  - PRA - **99**
  - Generic Issues - **6**

AP1000 DSER Open Item Status



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**

Design Acceptance Criteria

- Instrumentation and Controls
- Human Factors (Control Room)
- Piping

Exemptions

- Westinghouse requested 3 exemptions for the AP1000 (dated 12/3/02):
  - 50.34(f)(2)(iv) re: additional TMI-related requirements (requires SPDP)
    - Approved in FSER Chapter 18, "Human Factors Engineering"
  - 50.62(c)(1) re: ATWS requirements (requires diverse and auto initiation of AFW)
    - Approved in FSER Chapter 15, "Transient and Accident Analysis"
  - App. A, GDC 17 re: Electric power (requires 2 independent offsite power sources)
    - Approved in FSER Chapter 8, "Electric Power Systems"

AP1000 Status Summary

- All DSER open items resolved
- All ACRS issues addressed
- NRC staff still on schedule to issue FSER by September 13, 2004
- Questions or comments?

AP1000 Open Items

June 25, 2004

ACRS Future Plant Subcommittee Briefing

John Segala, Senior Project Manager
Office of Nuclear Reactor Regulation

Introduction

- Purpose
  - Provide summary of the staff's review
  - Provide summary of the staff's resolution of DSER open items
- Success
  - Understand areas reviewed and resolution
  - ACRS agreement with resolution of open items

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
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- 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

Supplemental DSER Sections

- The DSER identified 5 potential 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
- The staff concluded that it was not necessary to issue supplemental DSER sections since all issues were resolved

DSER Chapter 1 Open Items

- DSER based on Revision 3 of Design Control Document (DCD)
  - W submitted latest DCD revision on June 24, 2004
  - Expect no additional technical changes
- Tier 2* information
  - Staff has reviewed DCD Tier 2* information and finds acceptable
- Combined license (COL) action items
  - Staff has reviewed COL action items and finds acceptable

Post DSER Open Items

- Core Shroud Susceptibility to Stress Corrosion Cracking (SCC)
  - W stated that based on operational experience, no inspections required beyond ASME requirements.
  - Staff agrees
- Alloy 52/152 Weldment QA Criteria
  - W proposed to use 100% volumetric examination of all partial penetration J-groove welds in the vessel.
  - Staff found acceptable.

Post DSER Open Items (cont.)

- High-Chromium Nickel-based Alloys Susceptibility to low-temperature crack propagation

  W concluded four conditions are necessary for the occurrence of LTCP:

  - relatively high concentrations of hydrogen in the environment and in the metal
  - relatively low temperatures
  - the presence of a sharp crack tip
  - the presence of loads which rise at a moderate rate to levels great enough to fail the flawed material

  W concluded that the conditions necessary for the occurrence of LTCP cannot take place in the AP1000 design.

  The staff found acceptable.

Post DSER Open Items (cont.)

- ADS-4 Squib Valve Notch Susceptibility to SCC
  - W stated that shear section designed to ASME Code and environment is not susceptible to SCC.
  - Staff agrees
- Chemical Effects on Sump Screens
  - Related to issuance of RG 1.82, Revision 3
  - W added COL Action Item to consider generation of chemical debris
  - Staff found approach acceptable

Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC)

- 35 ITAAC related DSER open items
  - Some proposed new ITAACs or change to existing ITAACs
  - Some related to the resolution of open items in other chapters

- All open items related to ITAAC are resolved

Quality Assurance (QA)

- 5 QA related DSER open items
- QA Inspections
  - QA Test Control Implementation Inspection at Oregon State University
  - Inspection of the Implementation of the Project-Specific Quality Plan at Westinghouse
- All violations, non-conformances, and open items related to QA are resolved

Leak-Before-Break (LBB)

- 2 LBB related DSER open items
- Major open items include:
  - Alloy 690/52/152 susceptibility to PWSCC
    - Results from sensitivity study using SCC crack morphologies indicate that margin exists in LBB applications
  - Acceptability of W LBB approach
    - W used a combination of qualitative assessment and quantitative evaluation to evaluate all AP1000 candidate AP1000 LBB piping subsystems
      - 1 LBB analysis (DVI-A subsystem)
      - Assessment of AP1000 LBB subsystems using AP600 analyses and scaling factors for pipe diameters and response spectra against bounding analysis curves

Leak-Before-Break (LBB)

Acceptability of W LBB approach (cont.)

- W considered in LBB assessment statistically based material properties, more sensitive leakage detection capability and inclusion of pipe whip restraints
- Staff concluded approach acceptable

- All open items related to LBB are resolved

Sump Screen Performance

- 6 sump screen related DSER open items
- Open items related to:
  - debris loading of IRWST screens and recirculation screens
  - debris through reactor coolant system break
- All open items related to sump screen are resolved
- Staff concludes that the screen design is acceptable, based on the following:
  - Containment recirculation screens redesign
  - Screen design is robust to prevent screen blockage.
  - ITAAC verifies as-built screen design
  - COL Action Items
    - Cleanliness program
    - RG 1.82 evaluation

Structural/Seismic Design

- Review Methodology:
  - Decision based on review of critical sections selected by the staff
  - Similar critical sections reviewed for AP600
- Design Constraints:
  - Hard rock site
  - Fixed base model for seismic analyses

Structural/Seismic Design (cont.)

- 38 structural/seismic related DSER open items
- Major open items include:
  - Basemat uplift
  - Completion of containment design
- The staff performed several audits of specific W calculations throughout review
- All open items related to structural/seismic design are resolved
- The staff concludes that the AP1000 structural design is acceptable.

Thermal Hydraulics

- Five ACRS Thermal Hydraulic Phenomena Subcommittee Meetings
  - from March 2001- February 2004
- 4 thermal hydraulic related DSER open items
- Major open items include:
  - liquid entrainment
  - core swell
  - long term cooling
  - boron precipitation
- All open items related to thermal hydraulics are resolved
- The staff concludes that the AP1000 design meets 10 CFR 50.46, and is acceptable

Probabilistic Risk Assessment (PRA)

- January 23-24, 2003 - ACRS PRA Subcommittee Meeting
- 24 PRA related DSER open items
- Notable open item topics:
  - PRA input to design certification process
  - PRA input to RTNSS process
  - Impact of uncertainties on PRA results and conclusions
  - Success criteria and thermal-hydraulic uncertainty
  - SAMDA evaluation
  - Reactor vessel insulation design
  - Shutdown risk
- All open items are resolved

Evaluation of Aerosol Deposition

Michelle Hart
Dose Assessment Team
NRR/DSSA/SPSB

Westinghouse Analysis

- Westinghouse initially intended to use AP600 removal rates for AP1000 aerosol
- BE analysis using MAAP calculated T-H and aerosol mechanistic code STARNAUA
  - Credit was given for gravitational settling, diffusiophoresis and thermophoresis
- Staff accepted these phenomena as removal mechanisms, however questioned the Westinghouse calculated removal rate values

DBA LOCA T/H

- T-H scenario and aerosol models are not specified by NUREG-1465
- Westinghouse calculation based on a single T-H scenario and mechanistic aerosol model
- Adopted scenario (3BE- 1) is a double-ended DVI 4" line break with a failure to activate the intact train

LOCA T/H Scenario

- Scenario acceptance based on :
  - 3BE-1 representative of the "3BE" accident class, which is the dominant contributor to CDF for the AP1000
  - T- H conditions typical for majority of severe accident sequences (fully depressurized and reflooded)
  - Revised source term was intended to be representative of low pressure core- melt accidents

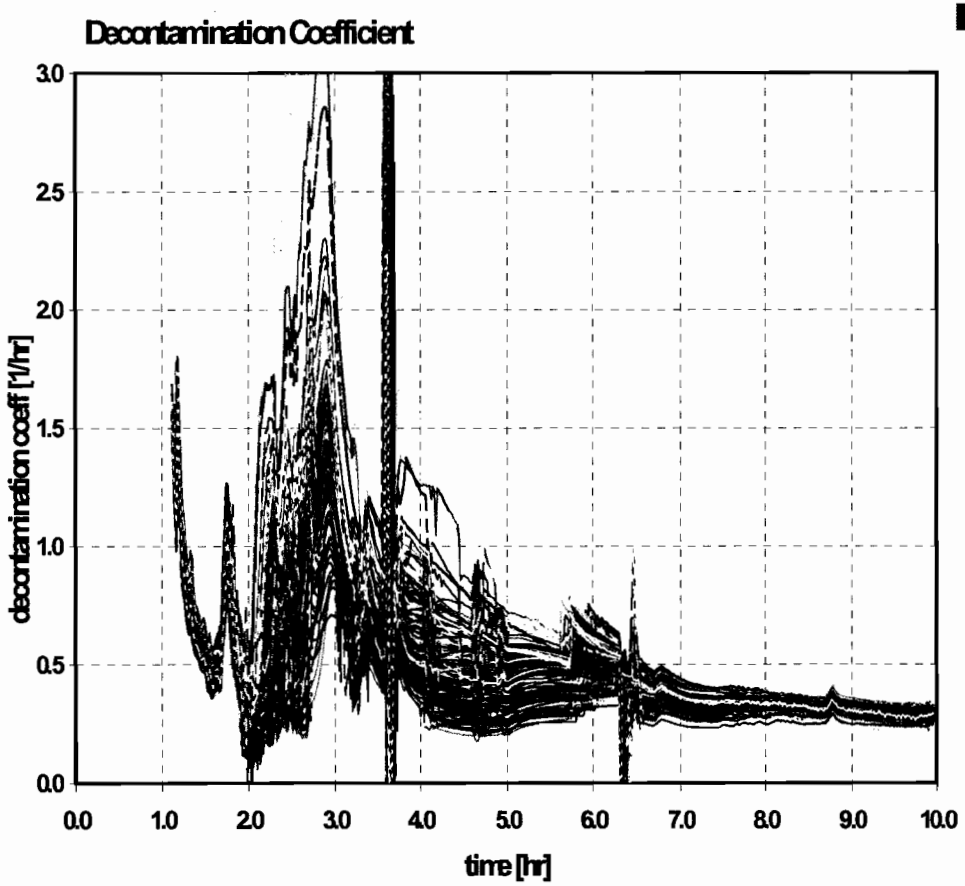
Staff Independent Analysis

- Monte Carlo sampling using MELCOR T/H and aerosol deposition modeling
- 13 parameters affecting aerosol behavior were sampled to achieve 95% confidence level
- Engineering judgment was used for the choice of parameters as well as for the range and distribution of their values

Sampled Parameters

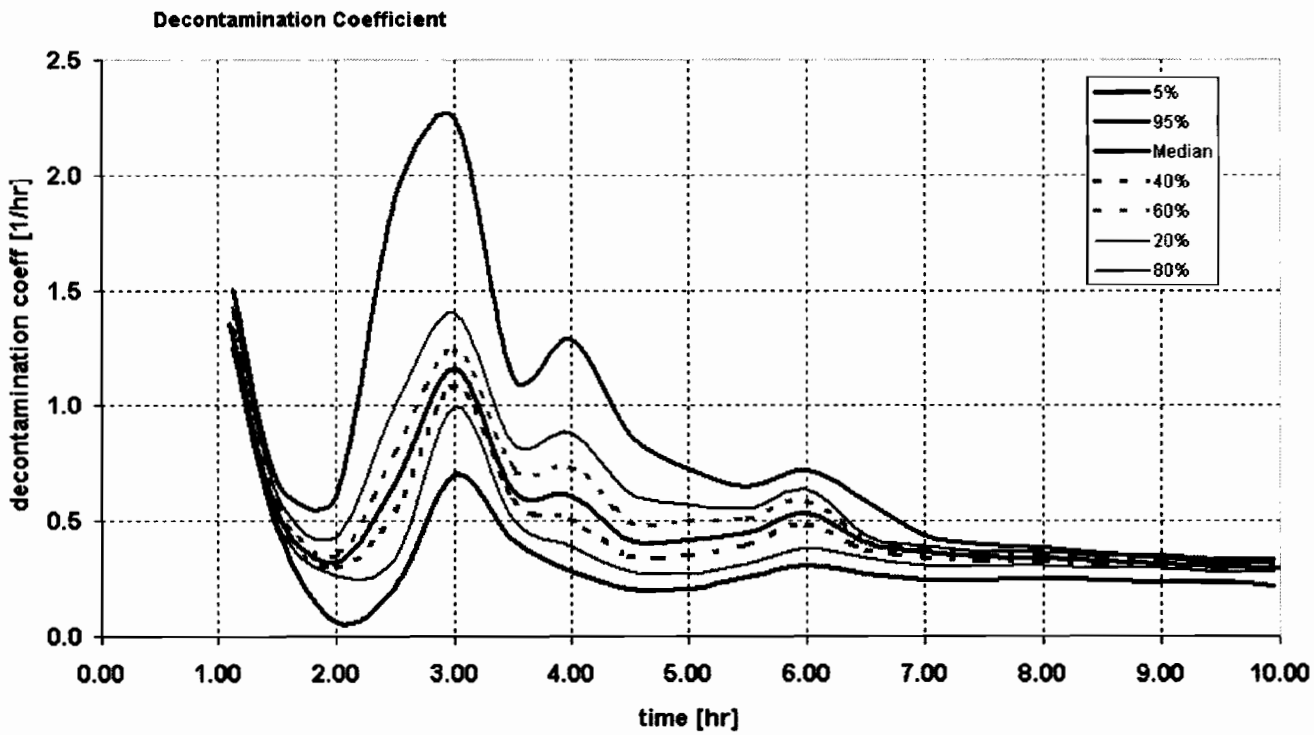
- aerosol mass mean diameter
- geometric standard deviation
- aerosol void fraction
- particle shape factors
- aerosol material density
- non- radioactive aerosol mass
- particle slip coefficient
- sticking probability for agglomeration
- boundary layer thickness for diffusion deposition
- thermal accommodation coefficient for thermophoresis
- ratio of thermal conductivity of particle to gas
- turbulent energy dissipation
- multipliers on heat and mass transfer to containment shell

Decontamination Coefficients



- Decontamination coefficients vary with time
  - Hydrogen burns cause transient spikes – increased thermophoresis
  - Late time values converge to around 0.3 hr^{-1}

Uncertainty of Aerosol Removal Coefficients (Lambdas)



Confirmatory Dose Calculation

- Staff did not base acceptance on Westinghouse values for aerosol removal coefficient
- Staff performed independent dose analysis of LOCA
  - Staff median aerosol removal coefficient
  - DCD X/Qs
  - DCD values for all other analysis inputs
- Met 10 CFR 50.34 and GDC-19 dose criteria

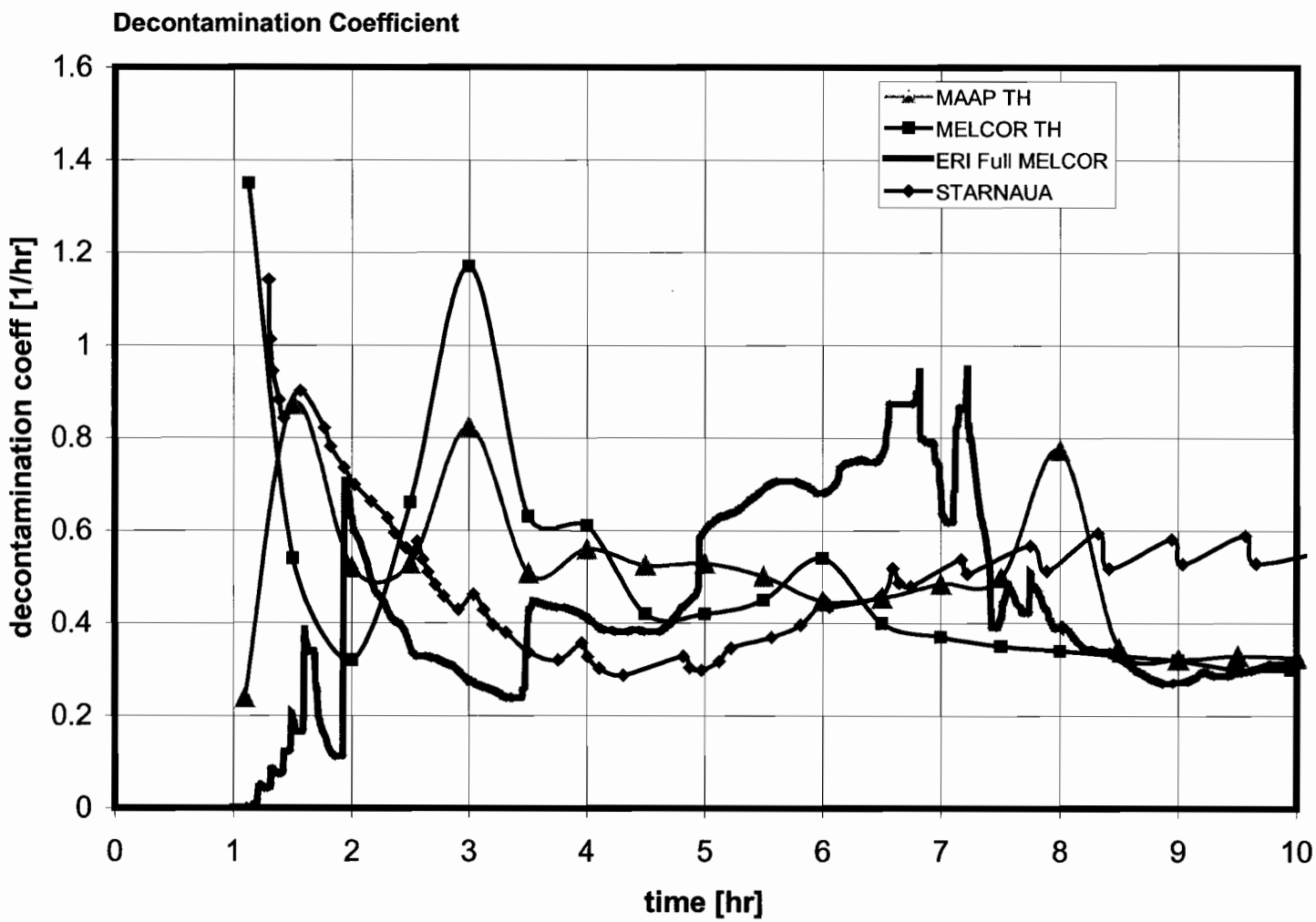
Use of Median Values

- Traditional regulatory approach is to accept a "bounding" value.
- Staff used median value for calculation
  - selected scenario belongs to a "worst case" category
  - median value is the least affected by the user's subjective judgment
  - staff introduced a "conservative bias" in choice of the initial ranges and distributions of the selected parameters
  - precedence in steam line deposition evaluation for Perry
  - staff's dose calculation code requires yet another "averaging" of the removal rates for the specified time periods
  - the fully integrated MELCOR calculated removal rates are mostly well above the 5 percentile

Additional Study

- Monte Carlo sampling using MAAP T/H and MELCOR aerosol deposition modeling
- Shows reasonable agreement with DCD values.
- Shows reasonable agreement with uncertainty analysis with MELCOR T/H

Decontamination Coefficients
Using MELCOR TH, MAAP TH, Full MELCOR ERI Analysis and MAAP-STARNAUA



AP1000 Open Item Summary

- All DSER open items resolved
- NRC staff still on schedule to issue FSER by September 13, 2004
- Questions or comments?

ACRS Interim Letter Issues

June 25, 2004

ACRS Future Plant Subcommittee Briefing

John Segala, Senior Project Manager
Office of Nuclear Reactor Regulation

Introduction

- Purpose
  - To discuss the staff's resolution of the ACRS interim letter issues
  - Provide update of recent staff activities
- Success
  - Understand the resolution of interim letter issues
  - ACRS agreement with resolution of interim letter issues

Issue 1 - Automatic Depressurization System (ADS)-4 Squib Valve Function

- ACRS Interim Letter Issue:
  - Agreed with the staff that ITAAC assures the valves meet the design basis specifications
- NRC Staff Resolution:
  - Simple design - ASME Code Section III Class 1
  - Redundant and Diverse Actuation
  - PRA Sensitivity Study
    - Increase in failure probability not change PRA conclusions
  - ITAAC (Squib Valve Type Test required)
    - Tests or type tests of squib valves will be performed that demonstrate the capability of the valve to operate under its design conditions.
    - A test report exists and concludes that each squib valve changes position under design conditions and that the as-installed squib valves are bounded by the tests or type tests.

Issue 2 - Assurance of Long-Term Cooling (Strainer Blockage)

- ACRS Interim Letter Issue:
  - AP1000 is a robust design to prevent screen blockage.
  - Recommended ITAAC to ensure compliance with GSI 191
- NRC Staff Resolution:
  - Containment recirculation screens redesign
  - Screen design is robust to prevent screen blockage.
  - ITAAC verifies as-built screen design
  - COL Action Items
    - Cleanliness program
    - RG 1.82 evaluation

Full Committee

Issue 3 - Code Deficiencies

- <u>ACRS Interim Letter Issue</u>:

  When deficiencies are identified in codes, the weaknesses should be corrected.

- <u>NRC Staff Resolution</u>:

  TRACE code is being assessed using APEX AP1000, ATLATS, and UPTF data.

Issue 4 - Range of Pi-Group Values

- <u>ACRS Interim Letter Issue</u>:
  - The staff should verify that a Pi group range of 0.5 to 2 is appropriate.
- <u>NRC Staff Resolution</u>:
  - This range has been used as a de facto standard in scaling analyses.
  - This issue is generic, not an issue specific only to AP1000.

Issue 5 - In-Vessel Retention/Fuel-Coolant Interactions

- ACRS Interim Letter Issue:
  - IVR assessment needs to consider the effects of exothermic intermetallic reactions.
  - Would like to review the FCI models and justification that intermetallic reactions will not result in energetic FCI that could fail the containment.
- NRC Staff Resolution:
  - Staff provided the ACRS a copy of their contractors IVR and FCI report for AP1000

Issue 6 - Organic Iodine Production

- ACRS Interim Letter Issue:
  - Water film pH determines iodine behavior
    - pH < 7 leads to production of elemental iodine some of which is subsequently converted into organic iodine
    - To prevent organic iodine production the film pH should be maintained above 7

Organic Iodine Production (cont.)

- W calculations determined:
  - Film pH is maintained above 7, assuming the amount of CsOH present in the DBA source term
  - A minimum of 270 g of CsOH (0.1% of available CsOH) is sufficient to keep pH above 7
  - The DBA dose criteria still met, even if assume no CsOH present
    - Increased amount of assumed organic iodine in containment from 0.15% to 0.33%
- Staff audited W calculations
  - Staff found the calculations to be acceptable
  - Staff agreed with W conclusions

Full Committee

Issue 7 - Catastrophic Failure of the Steel Containment

- ACRS Interim Letter Issue:
  - A free-standing steel containment can fail in a catastrophic manner when its failure pressure is exceeded. This failure mode can lead to rapid depressurization and resuspension of deposited fission products.
  - Like to see a sensitivity study on the fission product source term to assess the effect on the risk of latent fatalities as compared to the Safety Goal.
- NRC Staff Resolution:
  - Frequency of catastrophic containment failures are small
  - Resuspension would not have a noticeable impact on the Commission's safety goals.

AP1000 Summary

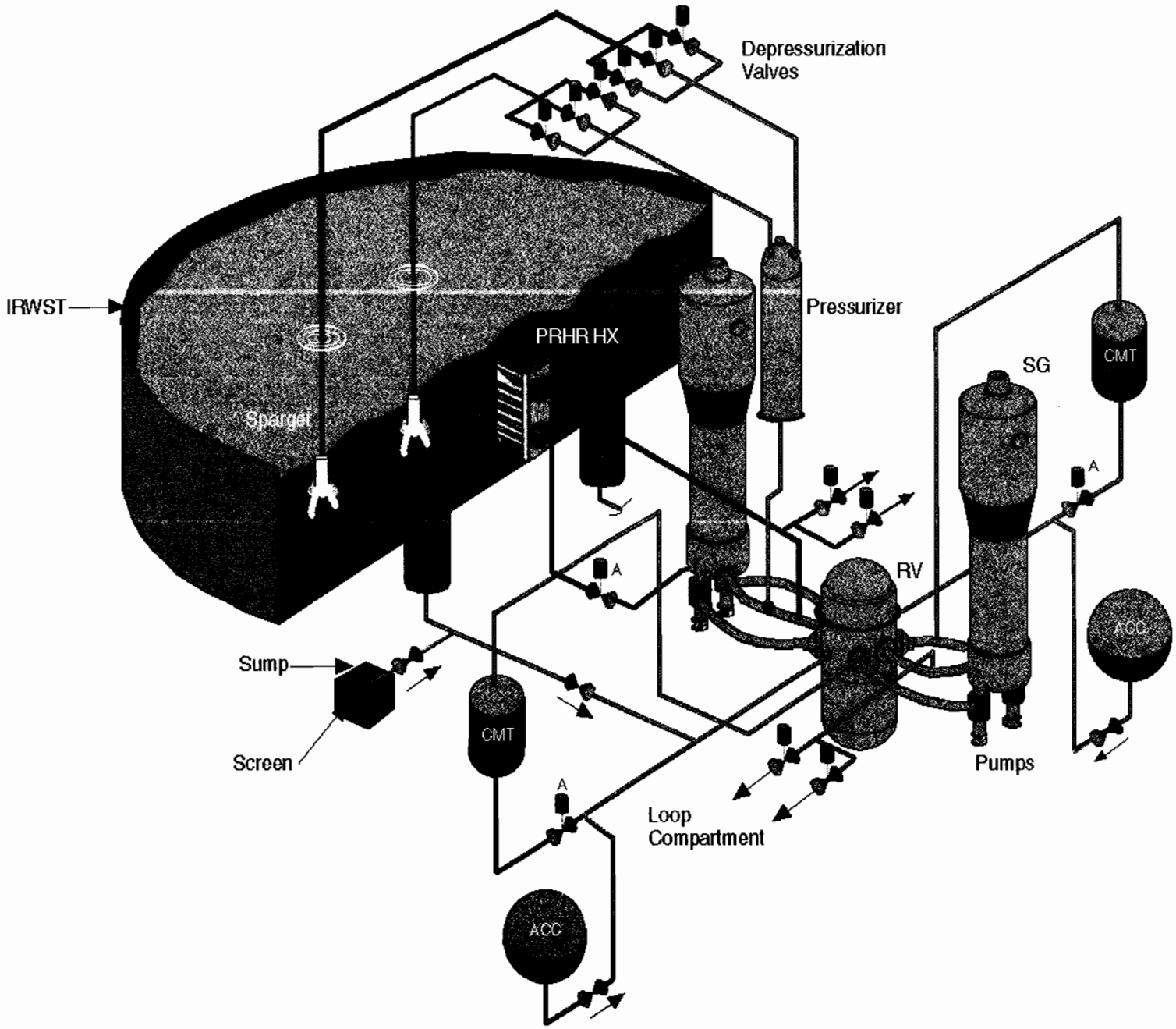
- All DSER open items resolved
- All ACRS issues addressed
- NRC staff still on schedule to issue FSER by September 13, 2004
- Questions or comments?

Background Slides

Leak-Before-Break

	Piping/Support	HELB	LBB	Benchmark Problem
ABWR	DAC	DAC	N/A	NUREG/CR-6049
System 80+	DAC	DAC	- DAC (bounding curves) - NRC reviewed 4 LBB calcs	NUREG/CR-6128
AP600	essentially complete (except support details)	essentially complete (except PW restraint details)	- DAC (bounding curves) - NRC reviewed 5 LBB calcs - LBB confirmatory analysis	NUREG/CR-6414
AP1000	DAC	essentially complete (except PW restraint details)	- DAC (bounding curves) - 1 LBB evaluation completed - Assessment of all other LBB piping	Same as AP600 (NUREG/CR-6414)

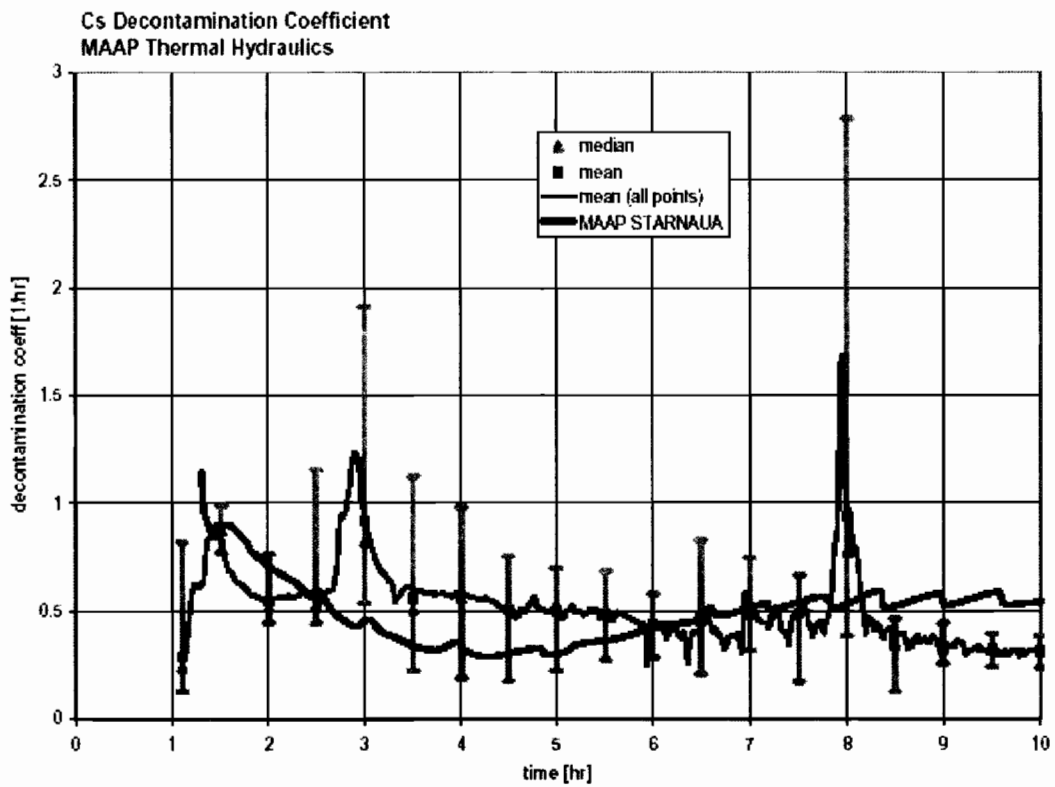
Aerosol Removal


Depressurization
Valves
IRWST
Pressurizer
PRHR HX
Sparger
SG
CMT
A
A
RV
ACC
Sump
CMT
Screen
A
Pumps
Loop
Compartment
ACC

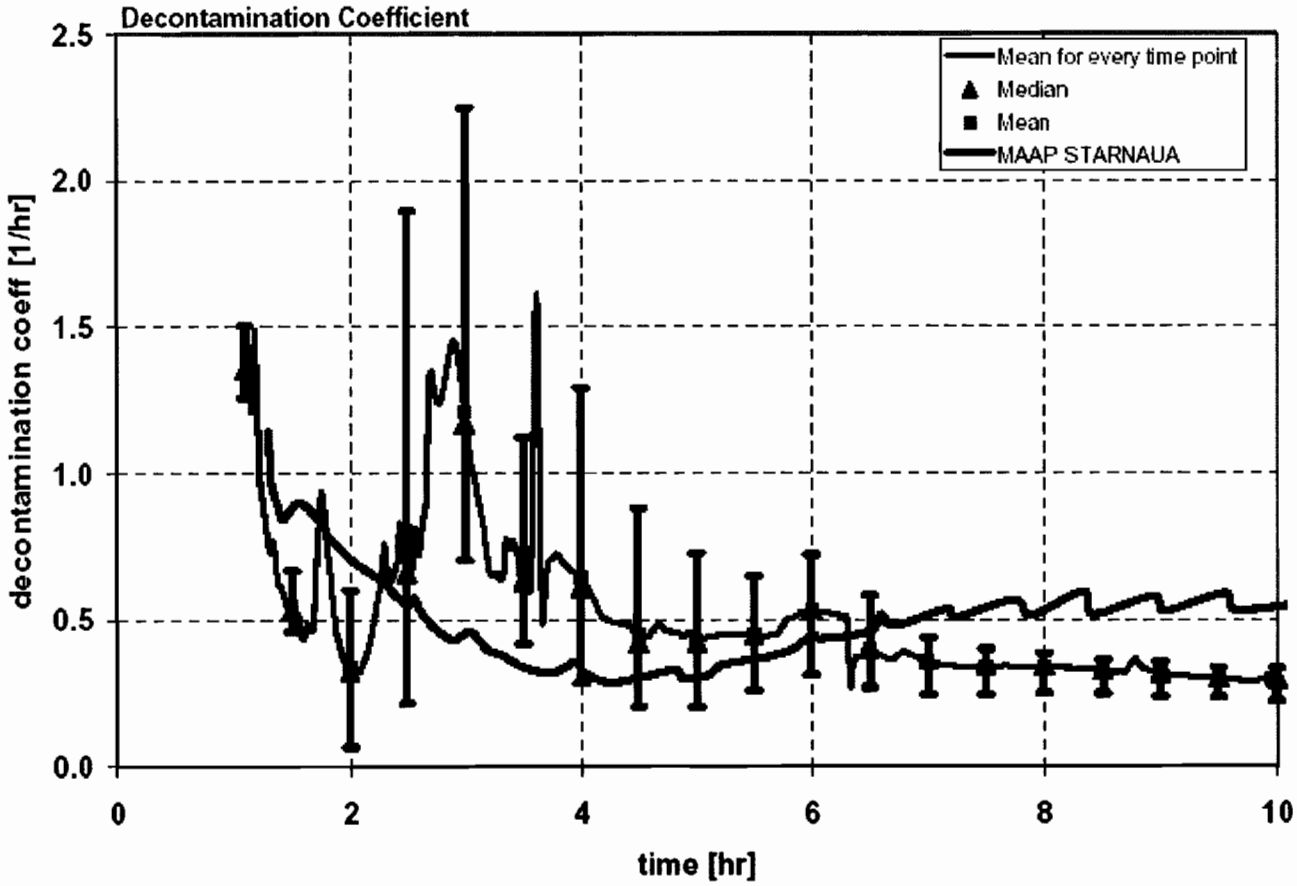
Parameter Distributions

Non-radioactive Mass	50 – 300 kg	uniform
Aerosol Mass Mean Diameter	1 – 4 μm	uniform
Geometric Standard Deviation	1.2 – 3	uniform
Aerosol Shape Factors	1 – 5	beta (bias to 1)
Particle Slip Coefficient	1.2 – 1.3	beta (normal)
Agglomeration Sticking Prob.	0.5 – 1	beta (bias to 1)
Boundary Layer Thickness	5 – 20 μm	uniform
Thermal Accommodation Coeff.	2.2 – 2.5	uniform
Thermal Conductivity Ratio	0.006 – 0.06	log-uniform
Turbulent Energy Dissipation	0.00075 – 0.00125	uniform
Aerosol Material Density	1000 – 5000 kg/m^3	beta (bias to 2000)
Heat and Mass Transfer Mult.	0.75 – 1.25	beta (a=1.5, b=1.5)

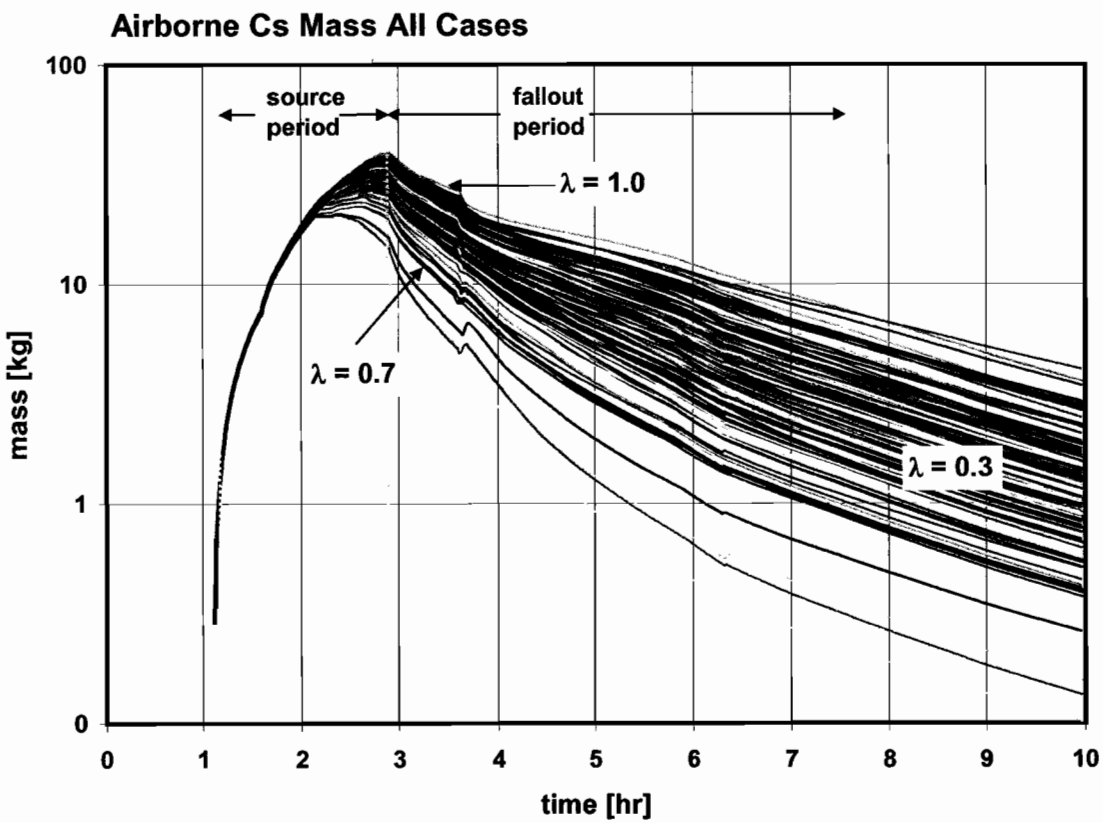
MAAP T/H with MELCOR Aerosol Calculation



MELCOR Analysis

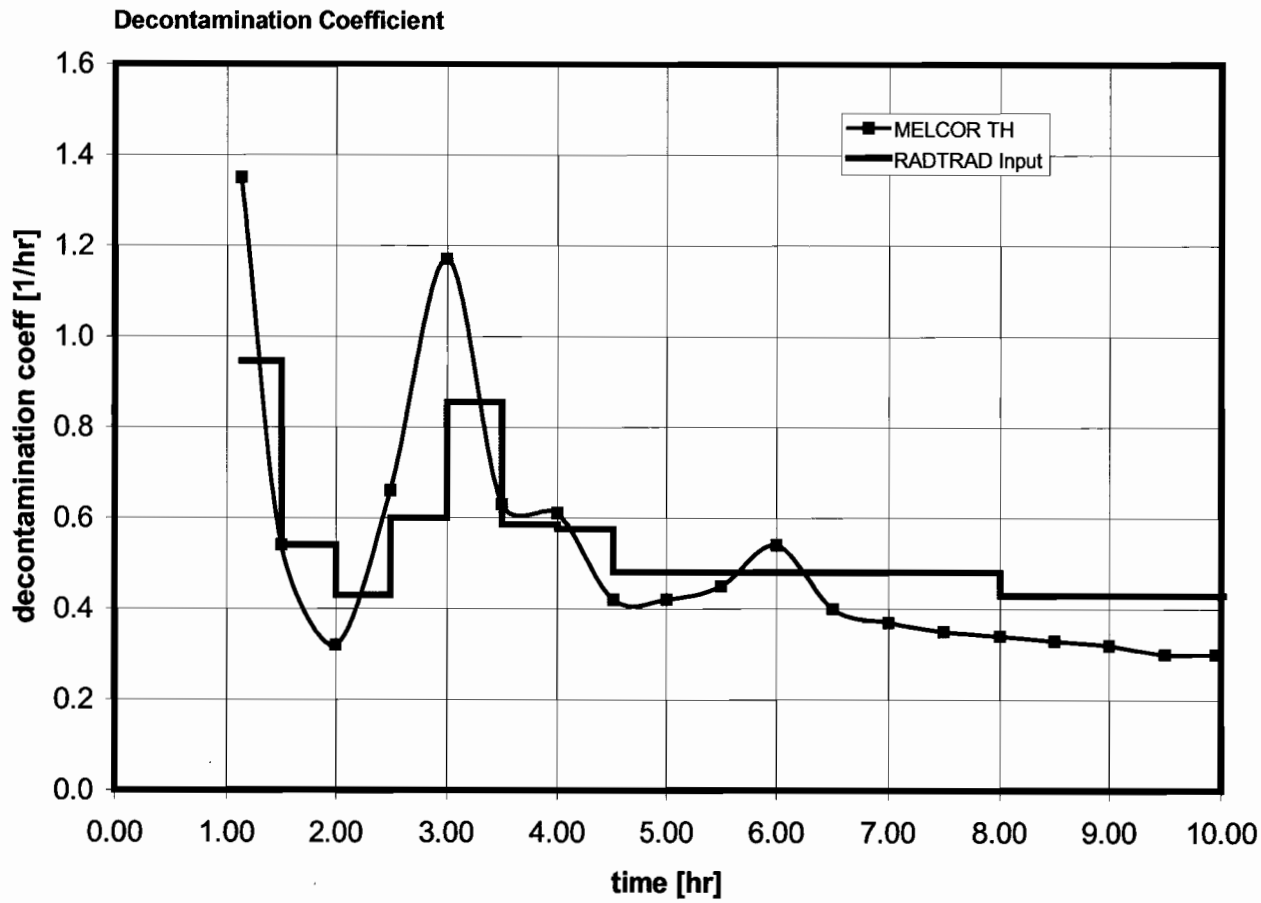


Behavior of Airborne Mass



- Suspended mass increases during aerosol sourcing period
- Initial depletion rate constant (lambda) during fallout period varies between 0.7 hr^{-1} and 1.0 hr^{-1}
- Late in time, all depletion constants in neighborhood of 0.3 hr^{-1}

MELCOR Thermal-Hydraulics and RADTRAD Input



AP1000

AP1000 Overview

Westinghouse Electric Company

Presentation to

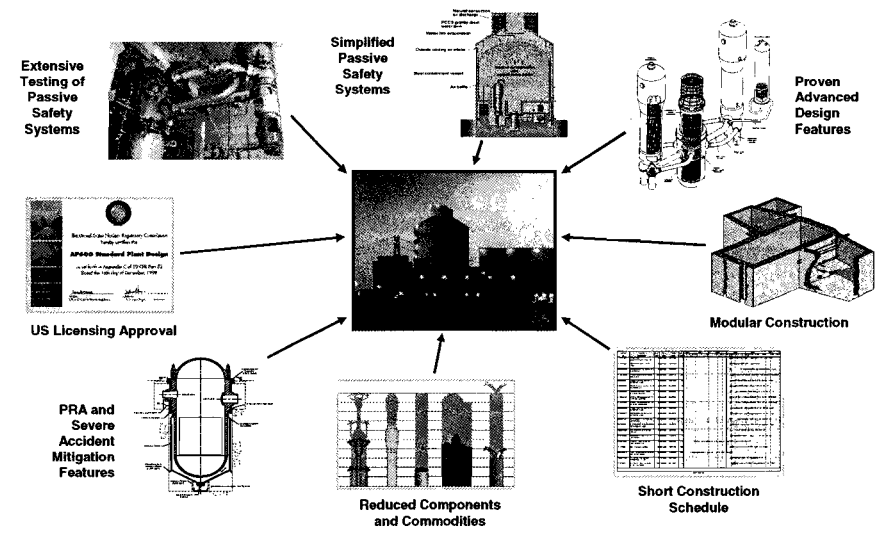
Advisory Committee on Reactor Safeguards

June 25, 2004

BNFL Westinghouse

AP1000

AP600/AP1000 Investment in Technology



BNFL Slide 2 Westinghouse

AP1000 Design Approach

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

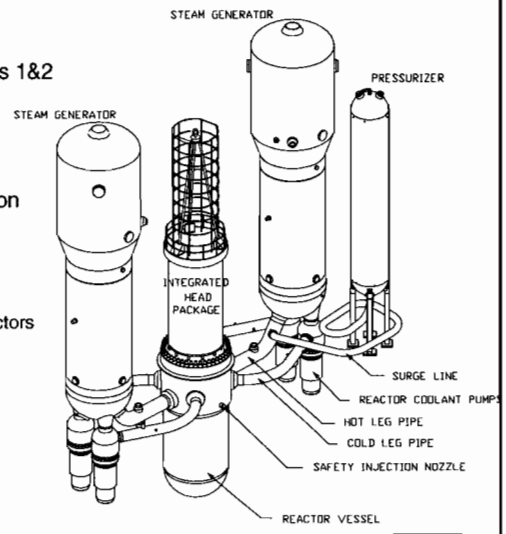
AP1000 Design Features .... Same as AP600

- **Integrated, Standard Power Plant Design**
- **Proven Power Producing Components (Reactor, Fuel, ...)**
  - No plant prototype required
- **Simplified RCS Loops with Canned Motor Pumps**
- **Simplified Passive Safety Systems**
  - Increase safety margins and address severe accidents
- **Simplified Nonsafety Defense-In-Depth Systems**
- **Microprocessor, Digital Technology Based I&C**
- **Compact Control Room, Electronic Operator Interface**
- **Optimized Plant Arrangement**
  - Construction, Operation, Maintenance, Safety, Cost
- **Extensive Use of Modular Construction**

AP1000

Proven AP1000 Major Components

- **Fuel, Internals, Reactor Vessel**
  - Similar to Doel 4, Tihange 3, S. Texas 1&2
  - No bottom-mounted instrumentation
  - Improved materials - 60 yr life
- **Steam Generators, Δ125**
  - Similar to large W/CE SGs in operation
    - System 80, ANO RSG
- **Reactor Coolant Pumps**
  - Canned motor pumps, no shaft seals
    - Naval reactors, early commercial reactors (Shippingport, Yankee Rowe)
- **Simplified Main Loop**
  - Reduces welds 50%, supports 80%
- **Pressurizer**
  - Typical West. design
  - 50% larger than operating plants



BNFL Slide 5 Westinghouse

AP1000

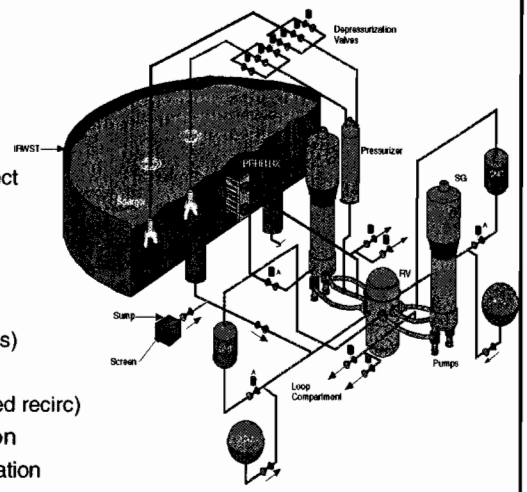
AP1000 Approach to Safety

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

BNFL Westinghouse

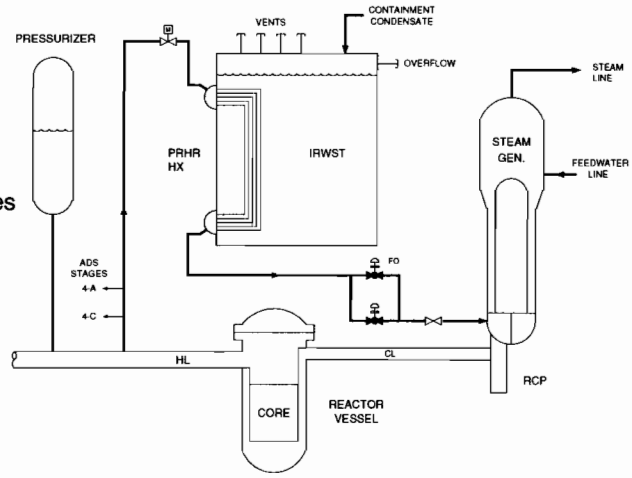
AP1000 Passive Core Cooling System

- **PRHR HX**
  - Natural circ. heat removal
- **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



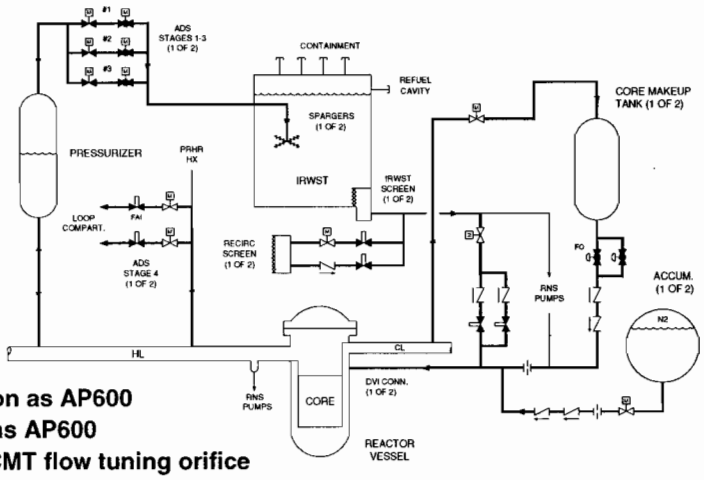
Passive Decay Heat Removal

- Same configuration as AP600
- Same elevations as AP600
- Larger pipe / valve sizes
- Increased HX surface
  - More tubes
  - Longer horizontal section



AP1000

Passive Safety Injection



- **Same configuration as AP600**
- **Same elevations as AP600**
- **Larger CMT and CMT flow tuning orifice**
- **Larger IRWST, Recirc, ADS 4 lines**

AP1000

ADS 4 DCD Valve Qualification

- **Design Verification (5.4.6.3)**
  - Includes valve qualification and pre-operational, in-service tests
  - Valve qualification includes
    - Testing to verify flow capability (5.4.6.3)
    - Type testing(1) and/or analysis to verify opening capability (5.4.8.1.3)
      - Min/max DP, limiting plant conditions, ...
      - Structural loads from seismic, flow actuation, ...
      - Environmental aging from radiation, heat, steam, ...

Note 1. Testing, type testing based on ASME QME-1 and IEEE 627.

AP1000

ADS 4 ITAAC Requirements

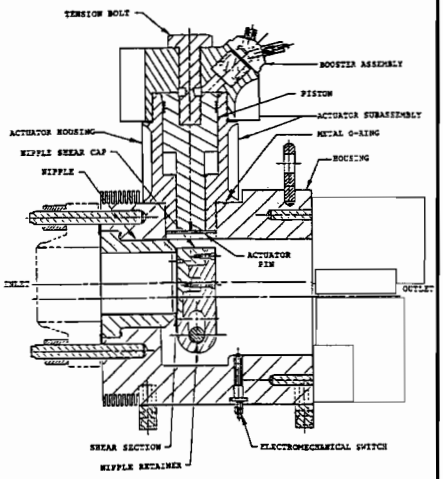
- **Seismic Capability (5.a.ii)**
  - Type test(1) and/or analysis
- **Harsh Environment Qualification (7.a)**
  - Type test(1) and/or analysis
- **Change Position (12.a.iv)**
  - Tests or type tests(1)
- **RCS Depressurization**
  - Stage 4 lines, HL to cont (8.d.ii)
    - Inspection of as-installed lines and analysis (R $\leq$ xxx ft/gpm^2)
  - Stage 4 valve minimum flow area (8.d.iii)
    - Inspection of as-installed valves (min area $\geq$ xx in^2)
  - Stage 4 valve discharge elevation (8.d.vi)
    - Inspection of as-installed valves

Note 1. Testing, type testing based on ASME QME-1and IEEE 627.

AP1000

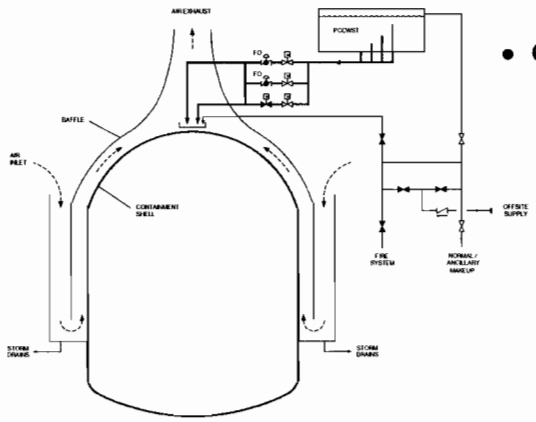
Anticipated ADS 4 Valve Qualification

- **Hydro, Leakage, Thermal (hot inlet)**
- **Valve Operability**
  - Design max / min inlet pressure
  - Degraded booster
- **Flow Capability**
  - Water flow capability (L/D)
  - Saturated steam flow capacity
    - Also provides opening loads
- **Environmental Conditions**
  - Accelerated radiation, heat, steam aging
    - Storage, normal, post accident conditions
    - Aged boosters actuated
  - Seismic and other dynamic loads
    - Vibration testing to cover seismic and valve actuation loads



AP1000

Passive Containment Cooling



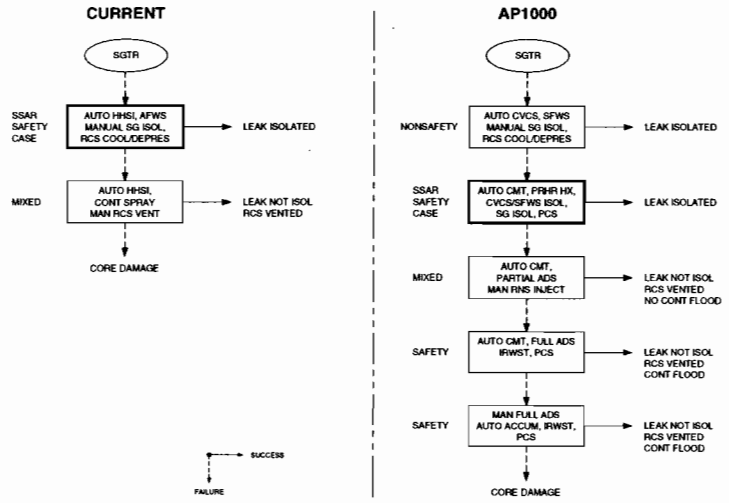
- **Same Configuration as AP600**
  - Except added 3rd diverse valve
    - Adds PRA margin
- **Capacities Increased**
  - Containment volume & design pres
  - PCS Water Storage Tank
    - Provides 72 hr drain
      - Afterwards use on/offsite water
      - Air only cooling prevents/delays failure in PRA sequences
    - Flow decreases with time
      - Uses 4 standpipes
  - PCS Flow Rates
    - Same high initial flow
      - Rapidly forms water film
      - Effectively reduces cont pressure
    - Later flows increased
      - Match decay heat

AP1000

AP1000 Safety Margins

	Typical Plant	AP600	AP1000
• Loss Flow Margin to DNBR Limit	~ 1 - 5%	~16%	~19%
• Feedline Break (°F) Subcooling Margin	>0	~170	~140
• SG Tube Rupture	Operator actions required in 10 min	Operator actions **NOT** required	Operator actions **NOT** required
• Small LOCA	3" LOCA core uncovers PCT ~1500°F	< 8" LOCA **NO** core uncovery	< 8" LOCA **NO** core uncovery
• Large LOCA (°F)	2000 - 2200	1676	2124
• ATWS, Pres (psig)	3200	3200	2800
UET (% core life)	5-10%	10%	0%

AP1000 Has More Levels of Defense (SG Tube Rupture Example)



BNFL Westinghouse

AP1000 CDF and LRF Summary

- **Meets US NRC Safety Goals with High Margin & Low Uncertainty**
  - Demonstrates effectiveness of passive safety features
    - Reduced dependency on operator actions and nonsafety features
    - Low safety risk from floods and fires
  - Severe accidents addressed by design

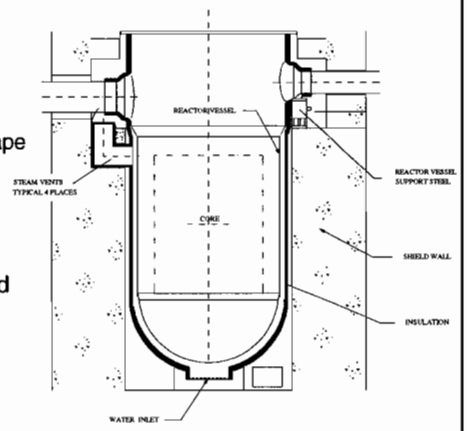
	Core Damage Frequency		Large Release Frequency	
	At-Power	Shutdown	At-Power	Shutdown
Internal Events	2.41E-07 /yr	1.23E-07 /yr	1.95E-08 /yr	2.05E-08 /yr
Internal Floods	8.80E-10 /yr	3.22E-09 /yr	7.10E-11 /yr	5.40E-10 /yr
Internal Fires	5.61E-08 /yr	8.52E-08 /yr	4.54E-09 /yr	1.40E-08 /yr
Sub-Totals	2.98E-07 /yr	2.11E-07 /yr	2.41E-08 /yr	3.50E-08 /yr
Grand-Totals	5.09E-07		5.92E-08	
NRC Safety Goals	1 E-4		1 E-6	

BNFL Slide 16 Westinghouse

AP1000

AP1000 Addresses Severe Accidents

- **Core-Concrete Interaction**
  - Invessel Retention / ex-vessel cooling
  - Means of cooling damaged core
  - Tests and analysis of IVR
    - Performed for AP600 and AP1000
    - AP1000 uses improved insulation shape
- **High Pressure Core Melt**
  - Eliminated by redundant, diverse ADS
- **Hydrogen Burn, Detonation**
  - Hydrogen vent paths from RCS located away from containment shell
  - Redundant, diverse igniters / PARs
- **Ex-Vessel Steam Explosions**
  - Prevented by IVR
  - Containment integrity even if IVR fails

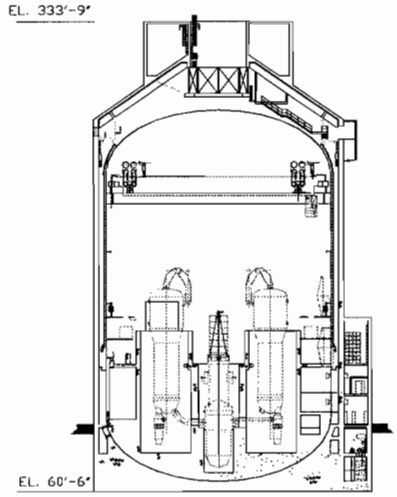


BNFL Westinghouse

AP1000

AP1000 vs AP600 Structures

- **Containment vessel, shield building raised 25'6"**
- **PCS tank capacity increased ~50% to 800,000 gal.**
- **PCS air inlets reconfigured to 12' x 6.5'**
- **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"**

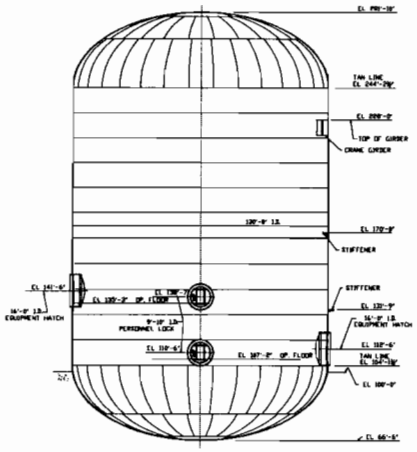


BNFL Slide 18 Westinghouse

AP1000

Containment Vessel General Outline

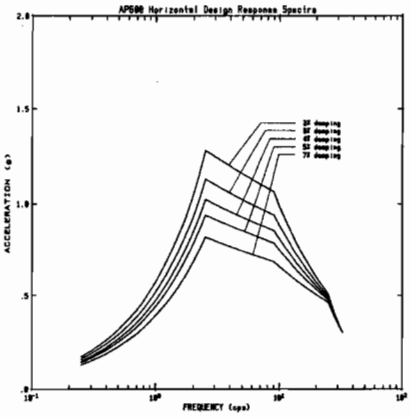
- Diameter: 130 feet
- Height: 215 feet 4 inches
- Design Code: ASME III, Div. 1
- Material: SA738, Grade B
- Design Pressure: 59 psig
- Design Temperature: 300°F
- Design External Pres: 2.9 psid



AP1000

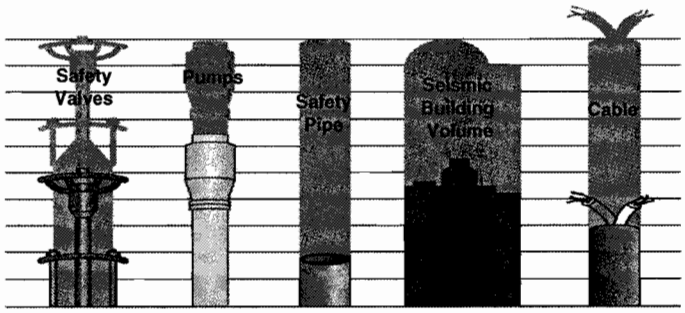
Seismic Design Basis

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



AP1000 Simplifications Drive Economics and Construction Schedule

AP1000

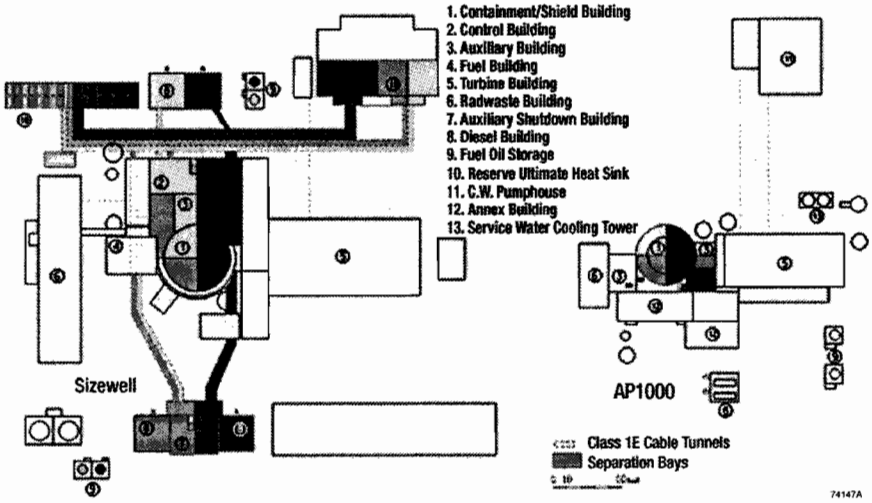


- **Reduced Number of Components:**

	1000 MW Reference	AP1000	Reduction
- Safety Valves	2844	1400	51%
- Pumps	280	184	34%
- Safety Piping	11.0×10^4 feet	1.9×10^4 feet	83%
- Cable	9.1 mil. feet	1.2 mil. feet	87%
- Seismic Building Volume	12.7 mil. ft^3	5.6 mil. ft^3	56%

AP1000 is Smaller and Dramatically Simpler than Evolutionary Plants

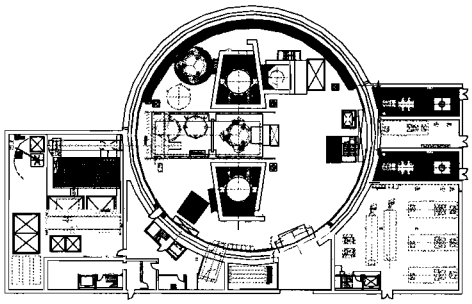
AP1000



AP1000

Improved General Arrangement

- **Same as AP600**
- **Improved Separation**
  - Radioactive vs nonradioactive
  - Fire areas, especially inside containment
  - Safety vs nonsafety
- **Improved Maintenance / Inspection**
  - Increased laydown area inside containment
  - Access platforms provided for equipment maintenance / inspection
- **Improved Access to Containment**
  - Equipment hatches access from auxiliary building
  - Equipment hatches and personnel airlocks at both grade and operating deck levels



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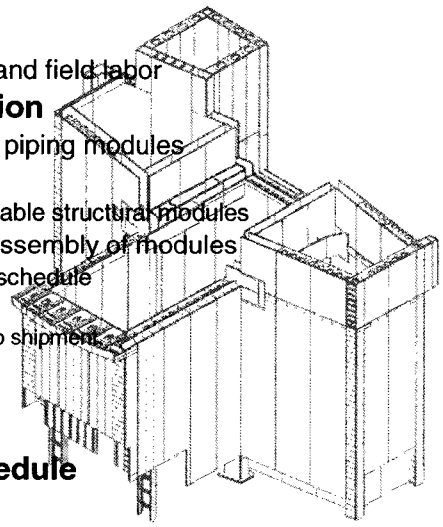
AP1000 Improved Construction

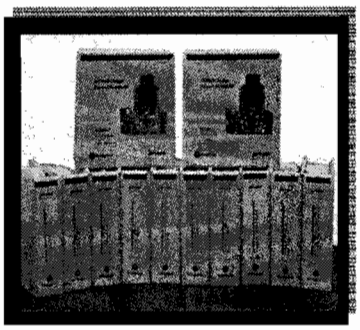
- **Simplification of Systems**
  - Major reduction in bulk materials and field labor
- **Maximize Use of Modularization**
  - 300 rail-shippable equipment and piping modules
  - 50 large structural modules
    - Assembled on-site from rail-shippable structural modules
  - Factory based manufacture and assembly of modules
    - Predictable, short manufacturing schedule
    - Improved quality control
      - Pre-testing and inspection prior to shipment
  - Streamlined field installation
    - Modules reduce field labor
    - Use of detailed work sequencing
- **36 Months Construction Schedule**
  - Confirmed by independent review



AP1000 Design Certification Nearing Completion

AP1000

Past Milestones	Target Schedule	
1. W Submits Application	3/28/02	☑
2. Staff Issues Requests for Additional Info (RAI)	9/30/02	☑
3. W Provides Responses to All RAI's	12/2/02	☑
4. NRC Identifies Potential Open Items	2/28/03	☑
5. W Addresses Potential Open Items	4/15/03	☑
6. NRC Issues Draft Safety Evaluation Report	6/16/03	☑
Future Milestones	**NRC Target**	
7. NRC Issues Final SER	September 2004	
8. NRC Issues Final Design Approval	September 2004	
9. NRC issues Design Certification	August 2005	



AP1000

AP1000 Safety

- AP1000 Comfortably Meets NRC and Industry Safety Standards for Future Plants
  - **Both deterministic and probabilistic**
- AP1000 Final Design Approval Will Enable Next Steps to Realizing New Plant Construction
  - **As proposed by Nu-Start Energy Consortium**