UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, DC 20555 - 0001

August 19, 2010

MEMORANDUM TO: ACRS Members

FROM: Sherry Meador **/RA/**
Technical Secretary, ACRS

SUBJECT: CERTIFICATION OF THE MEETING MINUTES FROM THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS 553rd FULL COMMITTEE MEETING HELD ON JUNE 4-6, 2008 IN ROCKVILLE, MARYLAND

The minutes of the subject meeting were certified on July 16, 2008 as the official record of the proceedings of that meeting. A copy of the certified minutes is attached.

Attachment:
As stated

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, DC 20555 - 0001

July 16, 2008

MEMORANDUM TO: Sherry Meador, Technical Secretary
Advisory Committee on Reactor Safeguards

FROM: Cayetano Santos, Chief **/RA/**
Reactor Safety Branch
Advisory Committee on Reactor Safeguards

SUBJECT: MINUTES OF THE 553rd MEETING OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS (ACRS), June 4-6, 2008

I certify that based on my review of the minutes from the 553rd ACRS Full Committee meeting, and to the best of my knowledge and belief, I have observed no substantive errors or omissions in the record of this proceeding subject to the comments noted below.

OFFICE	ACRS	ACRS:RSB
NAME	SMeador	CSantos/sam
DATE	05/ 22 /08	05/22/08

OFFICIAL RECORD COPY

CERTIFIED

Date Issued: 7/1/2008
Date Certified: 7/16/2008

TABLE OF CONTENTS
MINUTES OF THE 553rd ACRS MEETING

June 4-6, 2008

APPENDICES

Minutes of the 553rd ACRS Meeting
June 4-6, 2008

During its 553rd meeting, June 4-6, 2008, the Advisory Committee on Reactor Safeguards (ACRS) discussed several matters and completed the following report and memoranda.

REPORT

Report to Dale E. Klein, Chairman, NRC, from William J. Shack, Chairman, ACRS:

- ARTIST Test Program, dated June 13, 2008

MEMORANDA

Memoranda to R. W. Borchardt, Executive Director for Operations, NRC, from Frank P. Gillespie, Executive Director, ACRS:

- Draft Final Revision to 10 CFR 50.55a, "Codes and Standards," dated June 9, 2008
- Draft Final Regulatory Guide 4.21, "Minimization of Contamination and Radioactive Waste Generation: Life-Cycle Planning," dated June 9, 2008
- Withdrawal of Regulatory Guide 1.139, "Guidance for Residual Heat Removal," dated June 9, 2008
- Draft Regulatory Guides 1186, 4013, and 3034, dated June 9, 2008

MINUTES OF THE 552nd MEETING OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS May 8-10, 2008 ROCKVILLE, MARYLAND

The 553rd meeting of the Advisory Committee on Reactor Safeguards (ACRS) was held in Conference Room 2B3, Two White Flint North Building, Rockville, Maryland, on June 4-6, 2008. Notice of this meeting was published in the *Federal Register* on May 20, 2008 (72 FR 29169-29170) (Appendix I). The purpose of this meeting was to discuss and take appropriate action on the items listed in the meeting schedule and outline (Appendix II). The meeting was open to public attendance.

A transcript of selected portions of the meeting is available in the NRC's Public Document Room at One White Flint North, Room 1F-19, 11555 Rockville Pike, Rockville, Maryland. Copies of the transcript are available for purchase from Neal R. Gross and Co., Inc., 1323 Rhode Island Avenue, NW, Washington, DC 20005. Transcripts are also available at no cost to download from, or review on, the Internet at http://www.nrc.gov/ACRS/ACNW.

ATTENDEES

ACRS Members: Dr. William J. Shack (Chairman), Dr. Mario V. Bonaca (Vice-Chairman), Dr. Said Abdel-Khalik (Member-at-Large), Dr. Joseph Armijo, Dr. Dennis Bley, Dr. George E. Apostolakis, Mr. Charles Brown, Dr. Michael Corradini, Mr. Otto L. Maynard, Dr. Dana A. Powers, Mr. John Sieber, and Mr. John Stetkar. Mr. Charles Brown was unable to attend this meeting. For a list of other attendees, see Appendix III.

I. <u>Chairman's Report</u> (Open)

[Note: Mr. Sam Duraiswamy was the Designated Federal Official for this portion of the meeting.]

Dr. William J. Shack, Committee Chairman, convened the meeting at 8:30 a.m. In his opening remarks he announced that the meeting was being conducted in accordance with the provisions of the Federal Advisory Committee Act. He reviewed the agenda items for discussion and noted that no written comments or requests for time to make oral statements from members of the public had been received. Dr. Shack also noted that a transcript of the open portions of the meeting was being kept and speakers were requested to identify themselves and speak with clarity and volume. Various administrative announcements were made.

II. <u>ARTIST Test Program</u>
[Note: Mr. David Bessette was the Designated Federal Official for this portion of the meeting.]

The Committee met with representatives of the NRC staff to review the results of the tests conducted by Paul Scherrer Institut (PSI) at the Aerosol Trapping in a Steam Generator Test (ARTIST) facility in Switzerland. The objective was to investigate the retention of aerosols representative of those produced during several core damage accidents as they pass through a ruptured steam generator tube and transported through the secondary side of the steam generator. The ARTIST test program was sponsored by an international consortium, including NRC. This was a four-year program and completed testing in 2007. During this four-year program, a number of separate effects tests as well as large-scale integral tests were performed. It was found that there was limited deposition of aerosols in the steam generators. The staff plans to use the test data obtained from the ARTIST test program to refine the MELCOR code. The Committee issued a report to the NRC Chairman on this matter dated June 13, 2008, concurring with the staff that the ARTIST test program has provided sufficient experimental data to closeout Item 3.3a, “Development of Experimental Information on Aerosol Source Term Attenuation on the Secondary Side of Steam Generators,” of the NRC Steam Generator Action Plan.

III. Risk Assessment Standardization Project

[Note: Mr. Harold J. VanderMolen was the Designated Federal Officer for this portion of the meeting.]

The Committee met with representatives of the NRC staff to discuss the Risk Assessment Standardization Project (RASP) and related matters. This Project is intended to provide consistent methods between the Accident Sequence Precursor (ASP) calculations, the Significance Determination Process (SDP) Phase 3 programs, and incident investigation programs. Thus, the focus of RASP is to standardize the event assessment programs.

Based on the user need requests, the RASP project was divided into four tasks:

Task 1: Develop guides for the analysis of internal events during power operations.
Task 2: Develop new methods and guides for the analysis of external events, internal events during low power and shutdown operations
Task 3: Enhance the Standardized Plant Analysis Risk (SPAR) models and SAPHIRE/GEM code package
Task 4: Provide on-going technical support

The RASP handbook was issued in January 2008. This handbook provides guidelines for performing internal and external event analysis to address the first two tasks. To address Task 3, the SPAR models are under active development. There are now internal events models for essentially all operating plants. In addition, there are 15 external events models and five shutdown events models. Also, two Level II (LERF) models are under development. Finally, a new version of the SAPHIRE/GEM code package is being developed, including a new user interface and improved features and capabilities. RES continues to provide technical support to address Task 4. The staff briefly discussed the future activities that it plans to perform in this area. This was an information briefing. No Committee action was necessary at this time. The Committee plans to discuss this matter with the staff during future meetings.

IV. Overview of the U.S. Evolutionary Power Reactor (EPR) Design

[Note: Mr. Derek Widmayer was the Designated Federal Official for this portion of the meeting.]

The Committee was briefed by representatives of the NRC staff and AREVA NP Inc. regarding the design concepts, major safety systems, and components of the U.S. EPR design, as well as the main differences in design between the U.S. EPR and a standard Pressurized Water Reactor (PWR) now in operation in the U.S.

The staff discussed the schedule for reviewing the Design Certification Application for the US EPR, which was accepted for review on March 26, 2008. The target date for completing the Safety Evaluation Report (SER) with Open Items is March 5, 2010. The staff plans to group several chapters of the SER with Open Items and submit to the ACRS for review. The staff also stated that the Reference Combined License Application (RCOLA) for the US EPR at the Calvert Cliffs site in Lusby, MD is under review.

AREVA provided information on the design objectives for the US EPR, the general plant layout, the core design, digital instrumentation and control systems, severe accident mitigation, steam generator tube rupture and small-break loss-of-coolant accident mitigation, and the probabilistic risk assessment prepared for the EPR design. Major design differences between the U.S. EPR and an uprated 4-loop PWR were presented. The major components and systems for safety were discussed, including the radial design of the safeguards buildings, the four trains of safety systems, aircraft impact protection of the reactor and safeguards buildings, and the main safety systems for the primary and secondary sides of the reactor. This was an information briefing. No Committee action was necessary at this time. The ACRS Subcommittee on EPR plans to review parts of the Safety Analysis Report submitted by AREVA and the associated Chapters of NRC staff's SER.

V. <u>Status of the Development of Rules and Regulatory Guidance in the Areas of Safeguards and Security</u>

[Note: Ms. Maitri Banerjee was the Designated Federal Official for this portion of the meeting.]

The Committee was briefed by the staff regarding the status of the ongoing rulemaking and regulatory guidance development activities in the area of Safeguards and Security. The staff was previously informed that consistent with the Commission direction in the October 31, 2003

Staff Requirements Memorandum, the Committee is reviewing primarily four parts of the rule. These are: safety and security interface; cyber security; mitigative strategies for large area fire/explosion and aircraft threat; and aircraft impact assessment. The last subject was not included in the current briefing. The staff had already provided the draft final rulemaking package on aircraft impact assessment to the Committee for review and this part of the rulemaking is scheduled to be submitted to the Commission in September while the other parts will be delivered in July 2008.

The staff discussed the major changes made to the rulemaking package in response to the public comments, and the logic behind restructuring the rule in various parts of the 10 CFR, including moving parts of the rule outside 10 CFR Part 73. The mitigative strategies and response procedure for potential and actual aircraft attacks duplicate what was imposed on the operating rectors via orders, and are now in 10 CFR 50.54, as they will be imposed as license conditions.

Draft regulatory guides have been developed for safety/security interface and protection of digital computers and communication system (cyber security). While these draft guides have been made available to the ACRS, the draft guidance on mitigative strategies will not be available before July 2008. The staff requested ACRS comments on the draft final rule prior to the Committee's review of the draft guides. ACRS members commented that given the very general wordings in some draft rules, having the draft guidance along with the rule package will help in reviewing the draft final rule. The Chairman of the ACRS Subcommittee on Safeguards and Security noted the difficulty in preparing a report during the July 9-11, 2008 meeting, given the anticipated submittal of the rule package to the ACRS at the end of June.

This was an information briefing. No Committee action was necessary at this time. The Committee plans to review the draft final rulemaking package during the July 9-11, 2008 ACRS meeting.

VI. Status of the Quality Assessment of Selected Research Projects
[Note: Dr. Hossein Nourbakhsh was the Designated Federal Official for this portion of the meeting.]

The Committee discussed the status of the quality assessment of the research projects selected for FY 2008. The Committee agreed that the panel review of research project on FRAPCON/FRAPTRAN code work at the Pacific Northwest National Laboratory (PNNL) should be focused on the revised draft NUREG/CR report entitled, "Assessment of Predictive Bias and the Influence of Manufacturing, Model, and Power Uncertainties in NRC Fuel Performance Code Predictions." The Committee plans to discuss the draft report on quality assessment of the selected research projects during September 4- 6, 2008 ACRS meeting.

VI. Overview of the US-Advanced Pressurized Water Reactor (US-APWER) Design
[Note: Mr. David Bessette was the Designated Federal Official for this portion of the meeting.]

The Committee met with representatives of the NRC staff and Mitsubishi Heavy Industries, LTD (MHI) to discuss the design features and preliminary design certification review schedule for the US-APWR design. MHI submitted the application for US-APWR standard design certification on December 31, 2007. The staff acceptance review was completed and the application was docketed on February 29, 2008. The staff's review of the design certification application is currently under way along with the preparation of the SER with Open Items. The staff's proposed dates for ACRS review of the SER with Open Items and the final SER are June 2010 and August 2011, respectively. Luminant Generation Company, LLC has selected the US-APWR design for proposed new units at the Comanche Peak Nuclear Power Plant site. A COL application is expected to be submitted in September 2008.

Representatives of MHI provided an overview of the US-APWR design features. The US-APWR design is similar to the Japanese APWR that is currently undergoing licensing review in Japan. The MHI presentation included information about the fuel and core design, details of the system design and safety features, instrumentation and control (I&C) systems and architecture, and a discussion of MHI experience with digital I&C applications and reliability. MHI has developed a simulation facility near Pittsburgh, PA and stated that members of the ACRS were welcome to tour this facility.

This was an information briefing. No Committee action was necessary. During upcoming meetings the Committee will discuss and prioritize which US-APWR reports will be reviewed in detail. The ACRS staff will follow-up with NRC staff to schedule future briefings. The Committee plans to send representatives to visit MHI's Pittsburgh simulation facility later this year.

VII. Status of NRC Staff Activities Associated with the Resolution of Generic Safety Issue (GSI)- 191, "Assessment of Debris Accumulation on Pressurized-Water Reactor (PWR) Sump Performance"

[Note: Mr. David Bessette was the Designated Federal Official for this portion of the meeting.]

The Committee met with the NRC staff to discuss the status of staff activities associated with the resolution of GSI-191. GSI-191 addresses the impact and consequence of debris generated during design basis-LOCA on the capability and performance of the emergency core cooling and containment spray systems in the recirculation mode. The staff discussed the areas that still remain challenging as the licensees work towards completion of the actions related to PWR sump performance.

The staff acknowledged the substantial work done by licensees so far, including the installation of new strainers with larger surface areas and better pressure drop performance under strainer clogging conditions. The licensees changed the buffers and removed insulations in the zone of influence (ZOI) that adversely contribute to the debris fibers and chemical effects. The staff had by large reviewed and accepted or commented on most of the testing protocols that are intended to demonstrate adequate strainer functions under conditions representative of the plant-specific characteristics. In reference to the December 31, 2007 implementation deadline, the staff stated that most licensees requested and were granted extension for completion of certain corrective actions such as downstream effects analyses, integrated head loss testing, and plant modifications.

The staff approved the topical reports related to chemical effects and ex-vessel downstream effects. Although the draft safety evaluation of the in-vessel (core blockage) topical report was issued, the staff and the PWR owners group are currently addressing the concerns raised during the March 19, 2008 ACRS Subcommittee meeting.

The staff discussed some of the still pending challenges in resolution of GSI-191 and projected that the GSI-191 issue will be resolved by 2009. However, the staff also pointed out that the actions implemented so far reduced the risk of strainer clogging significantly. This was an information briefing. No Committee action was necessary at this time. The Committee plans to review the proposed resolution of GSI-191 during future meetings.

VI. Executive Session

[Note: Mr. Frank Gillespie was the Designated Federal Official for this portion of the meeting.]

A. Reconciliation of ACRS Comments and Recommendations/EDO Commitments

- The Committee considered the EDO's response of May 27, 2008 to comments and recommendations included in the May 2, 2008 ACRS report on the, Hope Creek Generating Station Extended Power Uprate Application. The Committee decided that it was satisfied with the EDO's response.

- The Committee considered the EDO's response of May 28, 2008 to comments and recommendations included in the April 29, 2008 ACRS report on the review of Digital Instrumentation and Control Systems Interim Staff Guidance. The Committee decided that it was satisfied with the EDO's response.

- The Committee considered the EDO's response of May 20, 2008 to comments and recommendations included in the April 21, 2008 ACRS report concerning State-of-the-Art Reactor Consequences Analyses (SOARCA) Project. The staff has restated its justifications for the current SOARCA approach. The staff plans to clarify the SOARCA methodology. The staff's intent is not clear. The Committee plans to discuss this matter during future meetings.

B. Report of the Planning and Procedures Subcommittee Meeting

Review of the Member Assignments and Priorities for ACRS Reports and Letters for the June ACRS Meeting

Member assignments and priorities for ACRS reports and letters for the June ACRS meeting were discussed. Reports and letters that would benefit from additional consideration at a future ACRS meeting were also discussed.

Anticipated Workload for ACRS Members

The anticipated workloads for ACRS members through September 2008 were discussed. The objectives were:

- Review the reasons for the scheduling of each activity and the expected work product and to make changes, as appropriate
- Manage the members' workload for these meetings
- Plan and schedule items for ACRS discussion of topical and emerging issues

Visit to the Braidwood Nuclear Plant and Meeting with the Region III Administrator

During its May 2008 meeting, the members decided to visit the Braidwood Nuclear Plant, and meet with the Region III Administrator to discuss items of mutual interest. A proposed schedule is as follows:

- Tuesday, July 22, 2008 ─ travel to Braidwood
- Wednesday, July 23, 2008 ─ plant visit
- Thursday, July 24, 2008 ─ meet with the Regional Administrator

The Committee requested that Maitri Banerjee provide detailed arrangements for the trip and Mr. Sieber propose a list of topics for meeting with the Regional Administrator. Accordingly, arrangements for this trip and proposed topics for meeting with the licensee and Region III Administrator were discussed.

Staff Requirements Memorandum

In a Staff Requirements Memorandum (SRM) dated May 8, 2008 resulting from the Commission meeting with an industry Panel and the staff regarding the status of new reactor issues, the Commission stated the following:

- The ACRS should advise the staff and the Commission on the adequacy of the design basis long-term core cooling approach for each new reactor design based, as appropriate, on either its review of the design certification or the first license application referencing the reactor design.

Draft Final Revision to 10 CFR 50.55a, “Codes and Standards”

The staff provided a copy of the draft final revision to 10 CFR 50.55a to the ACRS requesting that the Committee decide whether it wants to hear a briefing on this rule. The current version of this rule reflects incorporation of public comments, as appropriate. In the revision, the staff requires the use of:

- ASME Code Case N-722, “Additional Inspections for PWR Pressure Retaining Welds in Class 1 Pressure Boundary Components Fabricated with Alloy 60/82/182 Materials, Section XI, Division 2.”

- ASME Code Case N-729-1, “Alternative Examination Requirements for PWR Reactor Vessel Upper heads with Nozzles Having Pressure-Retaining Partial- Penetration Welds, Section XI, Division 1.”

Based on his review of the draft final revision of this rule, Dr. Shack recommended that the Committee not hear a briefing on this rule.

Regulatory Guide (RG) 1.139, “Guidance for Residual Heat Removal”

During its May 2008 meeting, the Committee discussed the staff’s proposal to withdraw RG 1.139. The Committee deferred action pending receipt of additional information from the staff. In May, John Flack sent information to the members regarding this subject. He also sent a list of questions to the staff. The staff’s response to these questions was discussed.

This Guide describes an overly conservative and prescriptive method for complying with the regulations. A draft version of this Guide was issued for public comment in 1975. However, it has never been issued final. Existing plant licensees have developed alternatives, without reliance on this Guide, for complying with the regulations; these alternatives were approved by the staff on a case-by-case basis. Since alternatives, acceptable to the staff, have been developed by the existing plant licensees without relying on this Guide and guidance for the staff reviewers is provided in the SRP, the staff has decided that there is no further use for this Guide.

Draft Final Regulatory Guide

The staff plans to issue RG 4.21, “Minimization of Contamination and Radioactive Waste Generation: Life-Cycle Planning,” as final. RG 4.21 provides guidance to licensees in meeting the requirements of 10 CFR 20.1406, “Minimization of Contamination.”

The Advisory Committee on Nuclear Waste & Materials (ACNW&M) was briefed on the draft version of this Guide (DG-4012) at its June 2007 meeting. The ACNW&M wrote a report to the Chairman dated June 28, 2007, with its recommendations to improve the draft Guide. Based on the ACNW&M report, the staff modified DG-4012 before it was issued for public comment, as identified in the Executive Director for Operation’s (EDO) August 7, 2007, response.

ACNW&M reviewed the EDO response and the revised DG-4012 and wrote another report to the Chairman dated November 27, 2007, primarily because the Committee thought the modifications made to DG-4012 were insufficient. The staff considered the additional comments, but disagreed with the major changes recommended by the ACNW&M. The EDO replied with a letter dated January 14, 2008. DG-4012 was issued for public comment in July 2007. This final version of RG 4.21 addresses the comments received from the public.

The two ACNW&M recommendations made in its November 27, 2007 report were that (1) DG-4012 should be modified so that it is only applicable to reactors, and (2) additional and better guidance should be issued in a separate guide for other radioactive material licensees. The staff responded to these comments in its resolution of comments document available with the final RG 4.21.

The staff responded that RG 4.21 was useful to other licensees because it included a graded approach for other radioactive material licensees that were favorably reviewed by the State of Washington (an Agreement State that regulates many radioactive materials users). The staff responded that if it revised RG 4.21 to apply only to reactors that other licensees would be left with no guidance. The staff further stated that, despite a great deal of effort to obtain comments on the merits of the approach in this Guide for other licensees, only the State of Washington replied, indicating there was not a great deal of need for improvement of that part of the guidance. The vast majority of the comments on DG-4012 addressed reactor-specific guidance, and most of the revisions in the final RG 4.21 improve this Guide for new reactor licensees.

Proposed Regulatory Guides

The staff plans to issue the following Regulatory Guides for public comment:

- Proposed Revision 2 to RG 1.21 (DG-1186), “Measuring, Evaluating, and Reporting Radioactive Materials in Liquid and Gaseous Effluents and Solid Wastes”

- Proposed Revision 2 to RG 4.1 (DG-4013), “Environmental Monitoring for Nuclear Power Plants”

On March 10, 2006, the EDO established the Liquid Radioactive Release Lessons Learned Task Force in response to incidents at some nuclear power plants related to unplanned and unmonitored releases of radioactive liquids into the environment. The Task Force issued a final report, “Liquid Radioactive Release Lessons Learned Task Force Final Report,” that recommended the revision of effluent and environmental monitoring program requirements and guidance and the provision of additional guidance on detecting, evaluating, and monitoring unplanned and unmonitored releases of radioactive liquids into the environment.

Proposed Revision 2 to RG 1.21 updates the Guide to describe the improved methods of measuring, evaluating, and reporting radioactivity in solid waste and radioactivity in liquid and gaseous effluents, and incorporates other editorial corrections and revisions to enhance clarity.

Proposed Revision 2 to RG 4.1 updates the Guide to describe the improved methods of environmental monitoring, and incorporates other editorial corrections and revisions to enhance clarity.

- Proposed Revision 1 to RG 3.12 (DG-3034), "General Design Guide for Ventilation Systems of Plutonium Processing and Fuel Fabrication Plants."

Regulatory Guide 3.12 was first issued in August 1973 to provide guidance for facilities processing plutonium. Since that time, there have been few commercial facility applications for plutonium processing and fuel fabrication. At this time, the NRC is licensing a mixed oxide fuel fabrication facility for use in processing surplus weapons materials, and it is expected that, in the future, additional facilities may be proposed for licensing. These future facilities may be either commercial facilities or facilities licensed through the U.S. Department of Energy's Global Nuclear Energy Partnership. In 2000, the NRC made significant regulatory changes to 10 CFR Part 70 that require applicants to prepare Integrated Safety Assessments (ISAs), which are systematic evaluations of nuclear facility hazards using risk-informed approaches. In addition, new industry consensus standards are available that update previous guidance reflecting new experiences and state-of-the-art equipment.

Regulatory Guide 3.12 is being revised in its entirety to address changes in Part 70 regarding ISAs.

Proposed ACRS Meeting Dates for CY 2009 – CY 2012

In March 2008, the staff provided the ACRS a description of the Committee's anticipated workload and a proposed schedule for Subcommittee and full committee meetings. The proposed ACRS meeting dates from CY 2009 through CY 2012 were discussed and summarized below. These meeting dates were provided to the Committee during the May ACRS meeting. Since May, the July 2009 meeting dates have been changed from July 15-17 to July 8-10, 2009.

Meeting Number	Dates	Days
---	January 2009	(No Meeting)
559	February 5-7, 2009	Thursday-Saturday
560	March 5-7, 2009	Thursday-Saturday
561	April 2-4, 2009	Thursday-Saturday
562	May 7-9, 2009	Thursday-Saturday
563	June 3-5, 2009	Wednesday-Friday
564	July 8-10, 2009	Wednesday-Friday
---	August 2009	(No Meeting)
565	September 10-12, 2009	Thursday-Saturday
566	October 8-10, 2009	Thursday-Saturday

567	November 5-7, 2009	Thursday-Saturday
568	December 3-5, 2009	Thursday-Saturday
---	January 2010	(No Meeting)
569	February 4-6, 2010	Thursday-Saturday
570	March 4-6, 2010	Thursday-Saturday
571	April 8-10, 2010	Thursday-Saturday
572	May 6-8, 2010	Thursday-Saturday
573	June 9-11, 2010	Wednesday-Friday
574	July 14-16, 2010	Wednesday-Friday
---	August 2010	(No Meeting)
575	September 9-11, 2010	Thursday-Saturday
576	October 7-9, 2010	Thursday-Saturday
577	November 4-6, 2010	Thursday-Saturday
578	December 2-4, 2010	Thursday-Saturday
---	January 2011	(No Meeting)
579	February 10-12, 2011	Thursday-Saturday
580	March 10-12, 2011	Thursday-Saturday
581	April 7-9, 2011	Thursday-Saturday
582	May 12-14, 2011	Thursday-Saturday
583	June 8-10, 2011	Wednesday-Friday
584	July 13-15, 2011	Wednesday-Friday
---	August 2011	(No Meeting)
585	September 8-10, 2011	Thursday-Saturday
586	October 6-8, 2011	Thursday-Saturday
587	November 3-5, 2011	Thursday-Saturday
588	December 1-3, 2011	Thursday-Saturday
---	January 2012	(No Meeting)
589	February 9-11, 2012	Thursday-Saturday
590	March 8-10, 2012	Thursday-Saturday
591	April 12-14, 2012	Thursday-Saturday
592	May 10-12, 2012	Thursday-Saturday
593	June 6-8, 2012	Wednesday-Friday
594	July 11-13, 2012	Wednesday-Friday
---	August 2012	(No Meeting)
595	September 6-8, 2012	Thursday-Saturday
596	October 4-6, 2012	Thursday-Saturday
597	November 1-3, 2012	Thursday-Saturday
598	December 6-8, 2012	Thursday-Saturday

The proposed dates for Subcommittee meetings would be the following:

- two days before a full committee meeting,
- the second Thursday/Friday after a full committee meeting, or
- the Thursday/Friday during the third week of a month with no full committee meeting.

The meeting was adjourned at 1:00 p.m. on June 6, 2008.

NUCLEAR REGULATORY COMMISSION

Advisory Committee on Reactor Safeguards (ACRS); Subcommittee Meeting on Planning and Procedures; Notice of Meeting

The ACRS Subcommittee on Planning and Procedures will hold a meeting on June 3, 2008, Room T–2B1, 11545 Rockville Pike, Rockville, Maryland.

The entire meeting will be open to public attendance, with the exception of a portion that may be closed pursuant to 5 U.S.C. 552b (c) (2) and (6) to discuss organizational and personnel matters that relate solely to the internal personnel rules and practices of the ACRS, and information the release of which would constitute a clearly unwarranted invasion of personal privacy.

The agenda for the subject meeting shall be as follows:

Tuesday, June 3, 2008, 8 a.m. Until 9 a.m.

The Subcommittee will discuss proposed ACRS activities and related matters. The Subcommittee will gather information, analyze relevant issues and facts, and formulate proposed positions and actions, as appropriate, for deliberation by the full Committee.

Members of the public desiring to provide oral statements and/or written comments should notify the Designated Federal Officer, Mr. Sam Duraiswamy (telephone: 301–415–7364) between 7:30 a.m. and 4 p.m. (ET) five days prior to the meeting, if possible, so that appropriate arrangements can be made. Electronic recordings will be permitted only during those portions of the meeting that are open to the public. Detailed procedures for the conduct of and participation in ACRS meetings were published in the **Federal Register** on September 26, 2007 (72 FR 54695).

Further information regarding this meeting can be obtained by contacting the Designated Federal Officer between 7:30 a.m. and 4 p.m. (ET). Persons planning to attend this meeting are urged to contact the above named individual at least two working days prior to the meeting to be advised of any potential changes in the agenda.

Dated: May 13, 2008.

Cayetano Santos,

Chief, Reactor Safety Branch.

[FR Doc. E8–11230 Filed 5–19–08; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Advisory Committee on Reactor Safeguards; Meeting Notice

In accordance with the purposes of Sections 29 and 182b. of the Atomic Energy Act (42 U.S.C. 2039, 2232b), the Advisory Committee on Reactor Safeguards (ACRS) will hold a meeting on June 4–6, 2008, 11545 Rockville Pike, Rockville, Maryland. The date of this meeting was previously published in the **Federal Register** on Monday, October 22, 2007 (72 FR 59574).

Wednesday, June 4, 2008, Conference Room T–2B3, Two White Flint North, Rockville, Maryland

8:30 a.m.–8:35 a.m.: Opening Remarks by the ACRS Chairman (Open)—The ACRS Chairman will make opening remarks regarding the conduct of the meeting.

8:35 a.m.–10 a.m.: ARTIST Test Program (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff regarding the findings from the ARTIST Tests on aerosol retention in the secondary side of a steam generator, and related matters.

10:15 a.m.–11:45 a.m.: Risk Assessment Standardization Project (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff regarding the Risk Assessment Standardization Project (RASP) and related matters.

1:45 p.m.–3:45 p.m.: Overview of the U.S. Evolutionary Power Reactor (EPR) Design (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff and AREVA Nuclear Power Inc., regarding design features of the EPR and related matters.

4 p.m.–5 p.m.: Status of the Development of Rules and Regulatory Guidance in the areas of Safeguards and Security (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff regarding the status of activities associated with the development of rules and regulatory guidance in the safeguards and security areas.

5 p.m.–5:30 p.m.: Status of the Quality Assessment of Selected Research Projects (Open)—The Committee will hold discussions with the Chairmen of the ACRS Panels regarding the status of the quality assessment of the research projects on: FRAPCON/FRAPTRAN Code work at the Pacific Northwest National Laboratory; and NUREG–6943, ''Study of Remote Visual Methods to Detect Cracking in Reactor Components.''

5:45 p.m.–7 p.m.: Preparation of ACRS Report (Open)—The Committee will prepare and discuss the proposed ACRS report on the ARTIST Test Program.

Thursday, June 5, 2008, Conference Room T–2B3, Two White Flint North, Rockville, Maryland

8:30 a.m.–8:35 a.m.: Opening Remarks by the ACRS Chairman (Open)—The ACRS Chairman will make opening remarks regarding the conduct of the meeting.

8:35 a.m.–9:30 a.m.: Future ACRS Activities/Report of the Planning and Procedures Subcommittee (Open)—The Committee will discuss the recommendations of the Planning and Procedures Subcommittee regarding items proposed for consideration by the full Committee during future ACRS meetings. It will hear a report of the Planning and Procedures Subcommittee on matters related to the conduct of ACRS business, including anticipated workload and member assignments.

9:30 a.m.–9:45 a.m.: Reconciliation of ACRS Comments and Recommendations (Open)—The Committee will discuss the responses from the NRC Executive Director for Operations to comments and recommendations included in recent ACRS reports and letters.

10 a.m.–11:15 a.m.: Preparation for Meeting with the Commission (Open)—The Committee will hold discussions in preparation for their meeting with the Commission on the following topics: Safety Research Program Report, Digital I&C Matters, State-of-the-Art Reactor Consequence Analysis Program, ESBWR Design Certification, and Extended Power Uprates and related Technical Issues.

1:30 p.m.–3:30 p.m.: Meeting with the Commission (Open)—The Committee will meet with the Commission to discuss topics noted above.

3:45 p.m.–6 p.m.: Preparation of ACRS Report (Open)—The Committee will continue its discussion of a proposed ACRS report on the ARTIST Test Program.

Friday June 6, 2008, Conference Room T–2B3, Two White Flint North, Rockville, Maryland

8:30 a.m.–8:35 a.m.: Opening Remarks by the ACRS Chairman (Open)—The ACRS Chairman will make opening remarks regarding the conduct of the meeting.

8:35 a.m.–10:30 a.m.: Overview of the US-Advanced Pressurized Water Reactor (US-APWR) Design (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff and Mitsubishi Heavy Industries, Ltd., regarding design features of the US-APWR and related matters.

10:45 a.m.–11:45 a.m.: Status of NRC Staff Activities Associated with the Resolution of Generic Safety Issue (GSI)-191, ''Assessment of Debris Accumulation on Pressurized-Water Reactor (PWR) Sump Performance'' (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff regarding the status of NRC staff activities associated with the resolution of GSI–191.

1:15 p.m.–1:30 p.m.: Miscellaneous (Open)—The Committee will discuss matters related to the conduct of Committee activities and matters and specific issues that were not completed during previous meetings, as time and availability of information permit.

Procedures for the conduct of and participation in ACRS meetings were published in the **Federal Register** on September 26, 2007 (72 FR 54695). In accordance with those procedures, oral or written views may be presented by members of the public, including representatives of the nuclear industry. Electronic recordings will be permitted only during the open portions of the meeting. Persons desiring to make oral statements should notify the Cognizant ACRS staff named below five days before the meeting, if possible, so that appropriate arrangements can be made to allow necessary time during the meeting for such statements. Use of still, motion picture, and television cameras during the meeting may be limited to selected portions of the meeting as determined by the Chairman. Information regarding the time to be set aside for this purpose may be obtained by contacting the Cognizant ACRS staff prior to the meeting. In view of the possibility that the schedule for ACRS meetings may be adjusted by the Chairman as necessary to facilitate the conduct of the meeting, persons planning to attend should check with the Cognizant ACRS staff if such rescheduling would result in major inconvenience.

Further information regarding topics to be discussed, whether the meeting has been canceled or rescheduled, as well as the Chairman's ruling on requests for the opportunity to present oral statements and the time allotted therefor can be obtained by contacting Mr. Girija S. Shukla, Cognizant ACRS staff (301–415–6855), between 7:30 a.m. and 4 p.m., (ET). ACRS meeting agenda, meeting transcripts, and letter reports are available through the NRC Public Document Room at *pdr@nrc.gov*, or by calling the PDR at 1–800–397–4209, or from the Publicly Available Records System (PARS) component of NRC's document system (ADAMS) which is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html or http://www.nrc.gov/reading-rm/doc-collections/ACRS/.

Video teleconferencing service is available for observing open sessions of ACRS meetings. Those wishing to use this service for observing ACRS meetings should contact Mr. Theron Brown, ACRS Audio Visual Technician (301–415–8066), between 7:30 a.m. and 3:45 p.m., (ET), at least 10 days before the meeting to ensure the availability of this service. Individuals or organizations requesting this service will be responsible for telephone line charges and for providing the equipment and facilities that they use to establish the video teleconferencing link. The availability of video teleconferencing services is not guaranteed.

Dated: May 14, 2008.

Andrew L. Bates,

Advisory Committee Management Officer.

[FR Doc. E8–11232 Filed 5–19–08; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Sunshine Act Notice

AGENCY HOLDING THE MEETINGS: Nuclear Regulatory Commission.

DATES: Weeks of May 19, 26, June 2, 9, 16, 23, 2008.

PLACE: Commissioners' Conference Room, 11555 Rockville Pike, Rockville, Maryland.

STATUS: Public and Closed.

Week of May 19, 2008

There are no meetings scheduled for the Week of May 19, 2008.

Week of May 26, 2008—Tentative

Tuesday, May 27, 2008

1:30 p.m.—NRC All Hands Meeting (Public Meeting), Marriott Bethesda North Hotel, 5701 Marinelli Road, Rockville, MD 20852.

Wednesday, May 28, 2008

9:30 a.m.—Briefing on Equal Employment Opportunity (EEO) and Workforce Planning (Public Meeting) (Contact: Kristin Davis, 301–492–2266).

This meeting will be webcast live at the Web address—*http://www.nrc.gov.*

Week of June 2, 2008—Tentative

Wednesday, June 4, 2008

9 a.m.—Briefing on Results of the Agency Action Review Meeting (AARM) (Public Meeting) (Contact: Shaun Anderson, 301–415–2039).

This meeting will be webcast live at the Web address—*http://www.nrc.gov.*

Thursday, June 5, 2008

1:30 p.m.—Meeting with Advisory Committee on Reactor Safeguards (ACRS) (Public Meeting) (Contact: Tanny Santos, 301–415–7270).

This meeting will be webcast live at the Web address—*http://www.nrc.gov.*

Week of June 9, 2008—Tentative

There are no meetings scheduled for the Week of June 9, 2008.

Week of June 16, 2008—Tentative

There are no meetings scheduled for the Week of June 16, 2008.

Week of June 23, 2008—Tentative

Friday, June 27, 2008

9:30 a.m.—Periodic Briefing on New Reactor Issues (Public Meeting) (Contact: Donna Williams, 301–415–1322).

This meeting will be webcast live at the Web address—*http://www.nrc.gov.*

* * * * *

* The schedule for Commission meetings is subject to change on short notice. To verify the status of meetings, call (recording)—301–415–1292. Contact person for more information: Michelle Schroll, 301–415–1662.

* * * * *

Additional Information

The start time for the Briefing on Results of the Agency Action Review Meeting (AARM) (Public Meeting) on Wednesday, June 4, 2008, has been changed from 9:30 a.m. to 9 a.m.

* * * * *

The NRC Commission Meeting Schedule can be found on the Internet

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, DC 20555 - 0001

May 13, 2008

Appendix II

SCHEDULE AND OUTLINE FOR DISCUSSION
553rd ACRS MEETING
JUNE 4-6, 2008

WEDNESDAY, JUNE 4, 2008, CONFERENCE ROOM T-2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

1)	8:30 - 8:35 A.M.	Opening Remarks by the ACRS Chairman (Open) (WJS/CS/SD) 1.1) Opening statement 1.2) Items of current interest
2)	8:35 - 10:00 A.M.	ARTIST Test Program (Open) (JSA/DEB) 2.1) Remarks by the Subcommittee Chairman 2.2) Briefing by and discussions with representatives of the NRC staff regarding the synthesis on the findings from the ARTIST tests on aerosol retention in the secondary side of a steam generator, and related matters. Representatives of the nuclear industry and members of the public may provide their views, as appropriate.
	~~10:00 - 10:15~~ A.M. 10:05-10:20	*****BREAK*****
3)	~~10:15 - 11:45~~ A.M. 10:20-11:20	Risk Assessment Standardization Project (Open) (GEA/HJV) Briefing by and discussions with representatives of the NRC staff regarding the Risk Assessment Standardization Project (RASP) and related matters. Representatives of the nuclear industry and members of the public may provide their views, as appropriate.
	~~11:45~~ - 1:45 P.M. 11:20	*****LUNCH*****
4)	1:45 - ~~3:45~~ P.M. 3:50	Overview of the U.S. Evolutionary Power Reactor (EPR) Design (Open) (DAP/DAW) 4.1) Remarks by the Subcommittee Chairman 4.2) Briefing by and discussions with representatives of the NRC staff and AREVA Nuclear Power Inc. regarding design features of the EPR and related matters. Members of the public may provide their views, as appropriate.
	~~3:45 - 4:00~~ P.M. 3:50-4:05	*****BREAK*****

5) ~~4:00~~ - 5:00 P.M. Status of the Development of Rules and Regulatory Guidance in
6) 4:05 the areas of Safeguards and Security (Open) (MVB/MB)

5.1) Briefing by and discussions with representatives of the NRC staff regarding the status of activities associated with the development of rules and regulatory guidance in the safeguards and security areas.

6) 5:00 - 5:30 P.M. Status of the Quality Assessment of Selected Research Projects (Open) (DAP/HPN)

6.1) Report by and discussions with the Chairmen of the ACRS Panels regarding the status of the quality assessment of the research projects on: FRAPCON / FRAPTRAN Code work at the Pacific Northwest National Laboratory; and NUREG-6943, "Study of Remote Visual Methods to Detect Cracking in Reactor Components."

ADJOURNED AT 5:30 P.M.

~~**5:30 - 5:45 P.M. ***BREAK*****~~

~~7) 5:45 - 7:00 P.M. Preparation of ACRS Report (Open)~~
~~Discussion of proposed ACRS report on:~~
~~7.1) ARTIST Test Program (JSA/DEB)~~

THURSDAY, JUNE 5, 2008, CONFERENCE ROOM T-2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

8) 8:30 - 8:35 A.M. Opening Remarks by the ACRS Chairman (Open) (WJS/CS/SD)

9) 8:35 - 9:30 A.M. Future ACRS Activities/Report of the Planning and Procedures Subcommittee (Open) (WJS/FPG/SD)

9.1) Discussion of the recommendations of the Planning and Procedures Subcommittee regarding items proposed for consideration by the full Committee during future ACRS meetings.

9.2) Report of the Planning and Procedures Subcommittee on matters related to the conduct of ACRS business, including anticipated workload and member assignments.

10) 9:30 - 9:45 A.M. Reconciliation of ACRS Comments and Recommendations (Open) (WJS, et al. /CS, et al.)

Discussion of the responses from the NRC Executive Director for Operations to comments and recommendations included in recent ACRS reports and letters.

9:45 – 10:00 A.M. ***BREAK***

11) 10:00 – 11:15 A.M. Preparation for Meeting with the Commission (Open) (WJS, et al. /FPG, et al.)
Discussion of the following topics for meeting with the Commission:

- Overview (WJS/SD)
- Safety Research Program Report(DAP/HPN)
- Digital I&C Matters (GEA/CEA)
- State-of-the-Art Reactor Consequence Analysis (SOARCA) Program (WJS/HPN)
- ESBWR Design Certification (MLC/DEB)
- Extended Power Uprates and related Technical Issues (MVB/ZA)

11:15 - 1:30 P.M. ***LUNCH***

12) 1:30 – 3:30 P.M. Meeting with the Commission (Open) (WJS, et al. /FPG, et al.)
Meeting with the Commission, Commissioners' Conference Room, One White Flint North, to discuss topics listed under item 11.

~~**3:30 – 3:45 P.M. ***BREAK*****~~

13) ~~3:45 – 6:00~~ 4:00 P.M. Preparation of ACRS Report (Open)
Discussion of proposed ACRS report on:
13.1) ARTIST Test Program (JSA/DEB)

MEETING ADJOURNED AT 5:45 P.M.

FRIDAY, JUNE 6, 2008, CONFERENCE ROOM T-2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

14) 8:30 - 8:35 A.M. Opening Remarks by the ACRS Chairman (Open) (WJS/CS/SD)

15) 8:35 - 10:30 A.M. Overview of the US-Advanced Pressurized Water Reactor (US-APWR) Design (Open) (OLM/NMC/DEB)
15.1) Remarks by the Subcommittee Chairman
15.2) Briefing by and discussions with representatives of the NRC staff and Mitsubishi Heavy Industries, Ltd. regarding design features of the US-APWR and related matters.

Members of the public may provide their views, as appropriate.

10:30 – 10:45 A.M. ***BREAK***

16) 10:45 - 11:45 P.M. Status of NRC Staff Activities Associated with the Resolution of Generic Safety Issue (GSI)-191, "Assessment of Debris Accumulation on Pressurized-Water Reactor (PWR) Sump Performance" (Open) (SB/DEB)

16.1) Briefing by and discussions with representatives of the NRC staff regarding the status of NRC staff activities associated with the Resolution of GSI-191.

Representatives of the nuclear industry and members of the public may provide their views, as appropriate.

11:45 - 1:15 P.M. ***LUNCH***

17) 1:15 - 1:30 P.M. Miscellaneous (Open) (WJS/FPG)
Discussion of matters related to the conduct of Committee activities and matters and specific issues that were not completed during previous meetings, as time and availability of information permit.

NOTE:

Presentation time should not exceed 50 percent of the total time allocated for a specific item. The remaining 50 percent of the time is reserved for discussion.

One (1) electronic copy and thirty-five (35) hard copies of the presentation materials should be provided to the ACRS.

Appendix III

June 4, 2008

	NAME	NRC ORGANIZATION
1	H. Esmaili	RES
2	S. Wong	NRR
3	J. Foster	RES
4	T. Kolb	NRR
5	C. Hinten	RES
6	E. Goldfeiz	RES
7	R. Jenkins	RES
8	B. Wagner	RES
9	S. Lai	RES
10	D. O'Neal	RES
11	M. Stutzke	RES
12	P. Appignani	RES
13	J. Monninger	RES
14	D. Marksberry	RES
15	S. Sancaletar	RES
16	A. Kurtzky	RES
17	M. Fravonovich	NRR
18	D. Helton	RES
19	C. Lui	RES
20	G. Testaye	NRO
21	T. Roy	NRO
22	S. Arora	NRO
23	P. Hearn	NRO
24	S. Lu	NRO
25	J. Rycyna	NRO
26	M. Canova	NRO
27	J. Donohue	NRO
28	F. Forty	NRO
29	D. Dube	NRO
30	H. Phen	NRO

June 4, 2008

29	J. Colaccino	NRO
30	B. Schnetzler	NSIR
31	T. Reed	NRR
	D. Rahn	NMSS
33	N. Gilles	NRO
34	P. Madden	NRO
35	P. Holahan	NSIR
36	J. Zimmerman	NRR
37	S. Ali	RES
38	B. Richter	NRR
39	R. Lois	NRR

June 5, 2008

	NAME	NRC ORGANIZATION
1	E. ODonnell	RES
2	E. Roach	NRO
3	J. Shaperow	RES
4	J. Mitchell	RES

June 6, 2008

	NAME	NRC ORGANIZATION
1	J. Thompson	NRP
2	W. Ward	NRO
3	N. Otto	NRO
4	M. Takacs	NRO
5	J. Perez	RES
6	J. Schmidt	NRR
7	J. Honcharik	NRO
8	E. Reichelt	NRO
9	R. Clement	NRO
10	L. Monica	NRO
11	D. Dube	NRO
12	L. Burkhart	NRO
13	H. Boltman	NRO

14	S. Schroer	NRO
15	R. Chazell	NRO
16	Y. Wong	NRO
17	P. Hearn	NRO
18	G. Hammer	NRO
19	Y. G.	NRO
20	A. Drozd	NRO
21	S. Monagene	NRO
22	R. Landry	NRO
23	M. Yoder	NRR
24	P. Klein	NRR
25	D. McGovern	NRO
26	J. Barr	RES
27	B. Ruland	NRR
28	A. Hiser	NRR
29	J. Burke	RES
30	M. Galloway	NRR
31	R. Artchizel	NRR
32	T. Koshy	RES

June 4, 2008

	NAME	OUTSIDE ORGANIZATION
1	D. Algama	Mitsubishi Nuclear Energy Systems
2	R. Pederson	AREVA NP
3	V. Fregunese	AREVA NP
4	M. Parece	AREVA NP
5	S. Sloan	AREVA NP
6	R. Salm	AREVA NP
7	M. Carpenter	AREVA NP
8	P. Baker	AREVA NP
9	J. Tucker	AREVA NP
10	J. Mihalak	Uniston Nuclear Energy
11	C. Tally	AREVA
12	M. Owens	AREVA
13	T. Oswald	AREVA
14	R. Sgarro	PPL Nuclear Development
15	J. McLella	NARP
16	G. Zyby	Alion Science & Technology
17	D. Fischer	NUMARK Associates

June 5, 2008

	NAME	OUTSIDE ORGANIZATION
1	M. Purnell	TVA Licensing
2	J. D. Wolcott	TVA Licensing

June 6, 2008

	NAME	OUTSIDE ORGANIZATION
1	M. Kanzda	MNES
2	M. Onozuica	MNES
3	M. Hoshi	MHI
4	H. Arikawa	MHI
5	H. Teshima	MHI
6	M. Takashima	MHI
7	H. Hamamoto	MHI
8	M. Kikuta	MHI
9	Y. Ogata	MHI
10	M. Ishida	MNES
11	S. Watanabe	MNES
12	D. Wood	MHI
13	K. Kawai	MNES
14	S. Kaawanago	MNES
15	T. Hafesa	Worley Parsons
16	S. Unkewicg	Alion
17	T. Shiraishi	MHI
18	K. Yamauchi	MHI
19	D. Fischer	NUMARK Associates
20	M. Lucas	Luminant
21	K. Paulson	MNES
22	D. Lange	MNES
23	J. Butler	NEI

Appendix IV

LIST OF DOCUMENTS FROM THE
553rd ACRS MEETING JUNE 4-6, 2008

Agenda Item 2:
ARTIST Test Program

1. Proposed Schedule
2. Status Report
3. References

Agenda Item 3:
Risk Assessment Standardization Project

4. Proposed Schedule
5. Status Report
6. Attachments

Agenda Item 4:
Overview of the U.S. Evolutionary Power Reactor (EPR) Design

7. Proposed Schedule
8. Status Report
9. References
10. Enclosure

Agenda Item 5:
Status of the Development of Rules and Regulatory Guidance in the Areas of Safeguards and Security

11. Table of Contents
12. Proposed Meeting Agenda
13. Status Report

Agenda Item 15:
Overview of the US-Advanced Pressurized Water Reactor (US-APWR) Design

14. Proposed Schedule
15. Status Report
16. References
17. Attachments
18.

Agenda Item 16:
Status of NRC Staff Activities Associated with the Resolution of Generic Safety Issue (GSI)-191, "Assessment of Debris Accumulation on Pressurized-Water Reactor (PWR) Sump Performance"

19. Proposed Schedule
20. Status Report
21. References

Synthesis on the findings from the ARTIST tests on aerosol retention in the secondary side of steam generators

Presented to the ACRS
June 4, 2008

M. Salay
U.S. Nuclear Regulatory Commission
Washington, D.C., USA

Overview

- Steam Generator Tube Ruptures (SGTR) background and NRC interest-SGAP
- ARTIST test program pertaining to SGAP
- Major Observations
- MELCOR modifications
- Conclusions

Steam generator tube rupture accidents

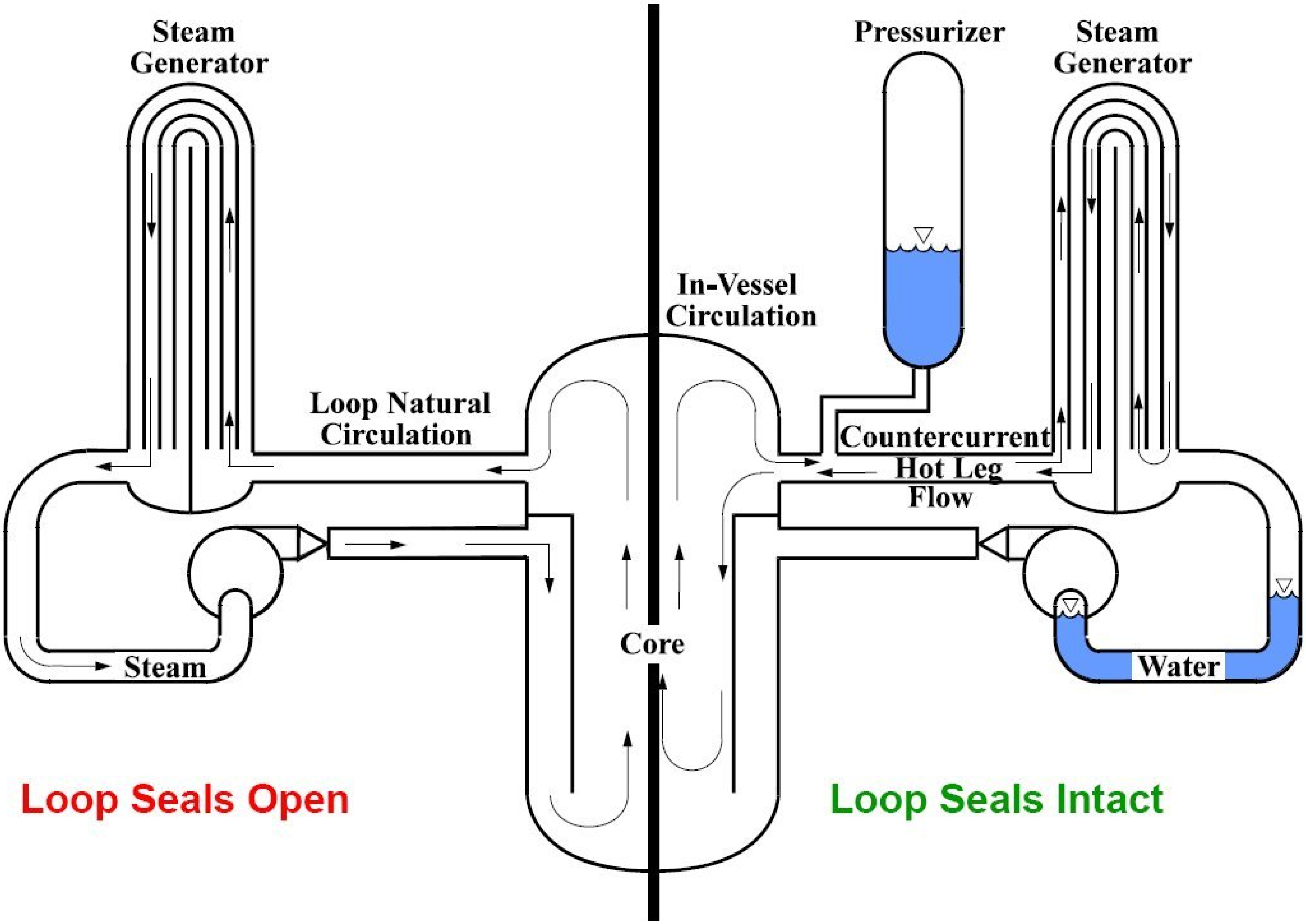
- **Design basis event**
  - Plants designed to cope
  - Have for all events to date
- **Progresses to severe accident only if something else happens**
  - Operator error

Induced steam generator tube rupture

- **Induced rupture greater concern**
  - Plants operate with detectable flaws in tubes
  - Limit on flaw size
  - Stress corrosion cracking is the cause of most flaws
  - Crevice corrosion at tube support plates of concern

Induced steam generator tube rupture

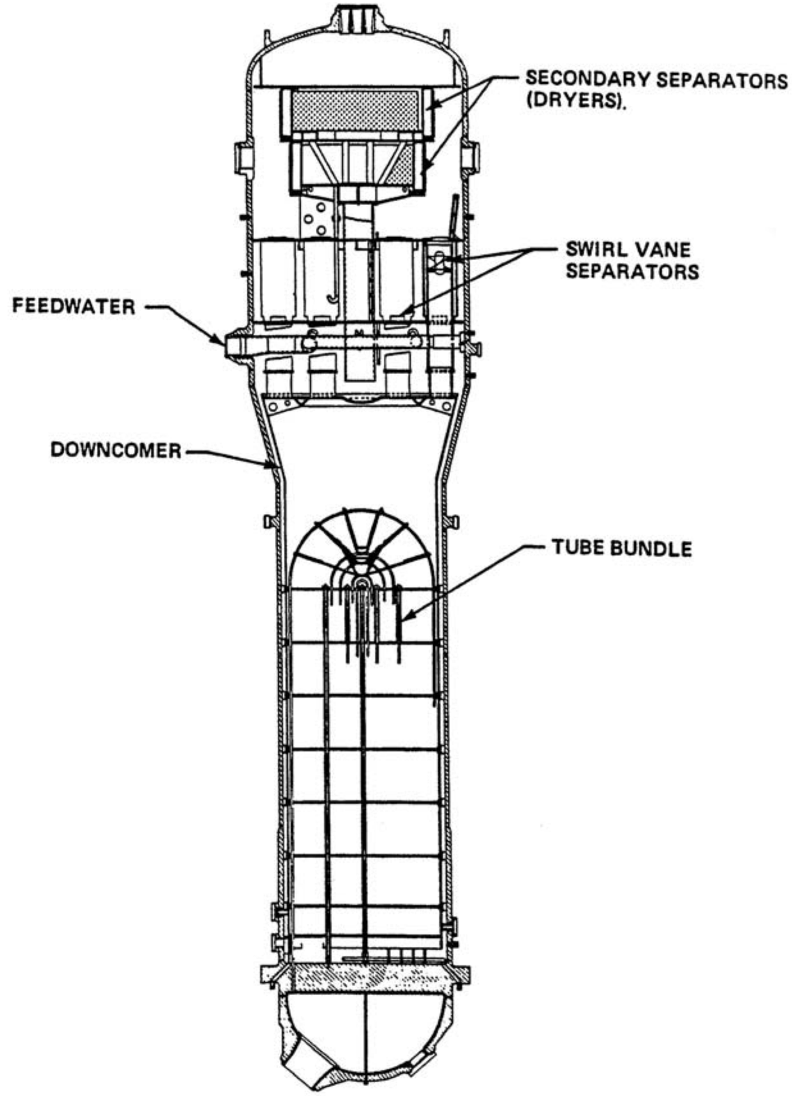
- **Heat transfer from core to primary pressure boundary weakens structures**
- **Vulnerable locations**
  - **Hot leg nozzle**
  - **Surge line to pressurizer**
  - **Steam generator tubes**
- **Codes do not reliably predict failure location and depressurization timing**



Severe accident natural circulation flows

Aerosol retention in SGTR SA

- **at tube inlet from steam generator plenum (inlet efficiency)**
- **in the steam generator tube prior to reaching the tube rupture**
- **in the immediate vicinity of the break where particles could impact on adjacent tubes**
- **in tubes between one tube support plate and another**
- **on top of tube support plates**
- **on envelope by thermophoretic deposition**
- **in the steam separators and steam dryers at the top of the steam generator.**
- **at steam generator safety relief valve (inlet efficiency)**



Westinghouse Model F Steam Generator

Aerosol retention processes

- **Removal mechanisms particle size dependent**
  - **Laminar**
    - **large – impaction, settling, interception**
    - **small – diffusion**
  - **Turbulent**
    - **turbulent deposition**
      - bounce
      - flow resuspension
      - saltation
- **Removal of particles alters particle size distribution**
  - **maximum penetration size**
  - **retention of individual sections can not be simply combined to obtain overall retention**
    - **integral tests**
  - **SETs obtain individual section retention as function of size**

Aerosol size

- A recommendation of prototypic aerosol size based on an IRSN survey of AECL, PBF-SFD and PHÉBUS experiments:
  - "size distribution at SG: near-lognormal, AMMD ~1μm or less, σ ~ 2; larger particles comprise agglomerates of small (~0.1 μm) highly coordinated clusters"
    - Sizes in two of the facilities were in the maximum penetration size range
    - Larger size range in third facility

Consequences of tube rupture

- **Radionuclides vent directly to environment or to auxiliary building without any attenuation from engineered safety features in containment**
- **Accidents have sufficiently high consequences that they are risk dominant despite low probability**

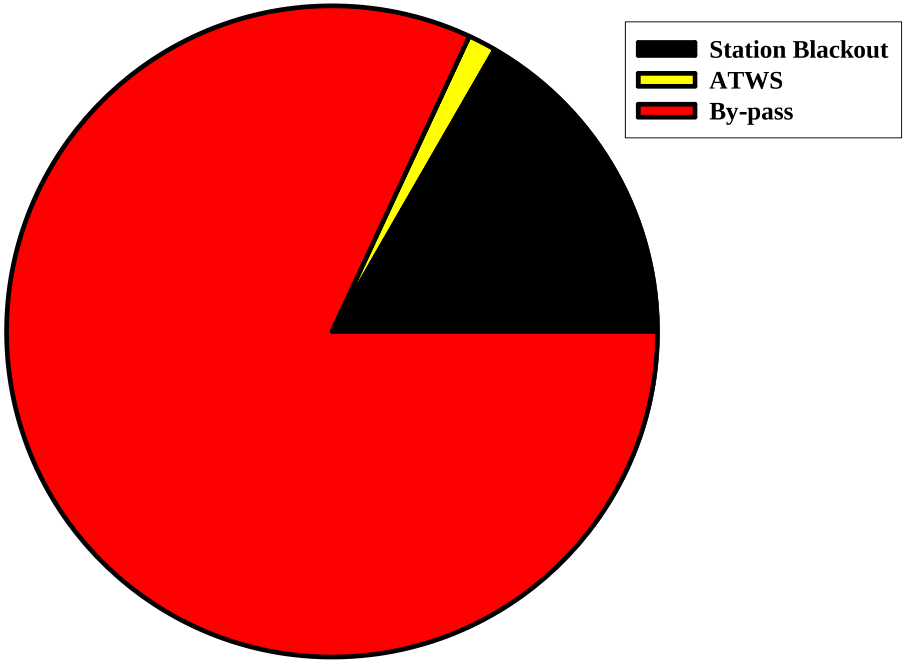
NUREG-1150

- Risk analysis of five US plants
  - Two PWRs had significant probabilities of steam generator tube rupture
  - All three PWRs could suffer induced steam generator tube rupture
- Limited modeling of aerosol behavior on secondary side of steam generators
  - None in the Source Term Code Package
  - Data unavailable

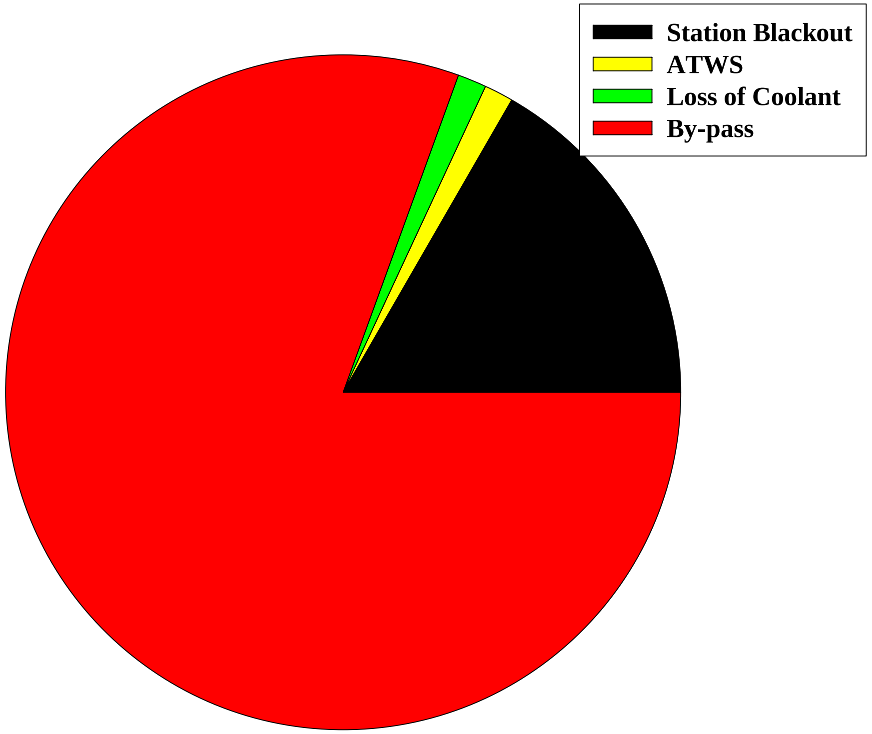
NUREG-1150 expert opinion elicitation

- Inlet efficiency from steam generator plenum to ruptured tubes – DF (mass in/mass out) ~2
- Retention in tubes - DF <~10 - no credit given
  - resuspension
  - revaporization
  - agglomerate breakup
- Retention in secondary side - DF ~4 to 6
  - deposition on outside of tubes resisted by thermophoresis
- No credit for steam dryer/separators
  - proprietary design information
- Large uncertainty in estimates

Surry Early Fatalities



Surry Latent Cancer Fatalities



Alternate retention analysis

- Industry analyses provided far different estimates of retention in the secondary side of steam generator
  - Calculated steam generator DF on the order of 10,000
    - >100 in tube, depending on break location
    - 10s secondary near break
    - 2-3 far from break

Focus on SGTR bypass accident

- **attention to SGTR bypass accidents justified by risk**
- **Direct connection between risk and source term attenuation**
- **“are safety resources being misdirected to an unneeded attention on containment bypass accidents because we underestimate attenuation”**

SGAP ITEM 3.3a

- **STEAM GENERATOR ACTION PLAN (SGAP) ITEM 3.3a – DEVELOP EXPERIMENTAL INFORMATION ON SOURCE TERM ATTENUATION ON THE SECONDARY SIDE OF STEAM GENERATORS**

ARTIST Project

- **AeRosol Trapping In a STeam generator**
  - International project conducted by the Paul Scherrer Institut (PSI)
  - seven phase project (NRC participated in 5)
  - separate and integral tests (38)
- **retention measured:**
  - in the steam generator tube prior to reaching the tube rupture (15)
  - in the immediate vicinity of the break where particles could impact on adjacent tubes (9)
  - in tubes between one tube support plate and another and on top of tube support plates (6) (1 stage,2 stage)
  - in the steam separators and steam dryers at the top of the steam generator. (5)
  - overall with all steam generator components (3)
- **Other phases (not NRC)**
  - retention in flooded bundle
  - droplets in dryers and separators

ARTIST facilities

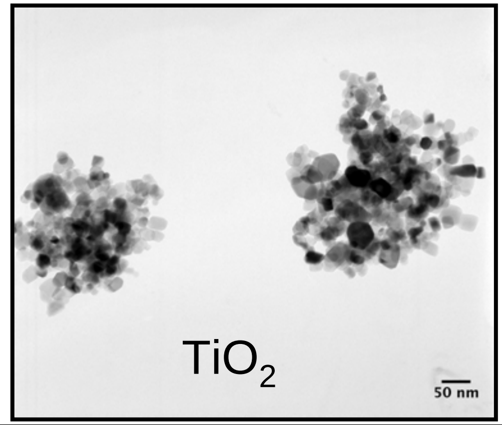
- ARTIST
  - based on Beznau plant: 365 MWe Westinghouse 2 loop PWR (1969,1972)
  - scaled for SGTR
  - 19.08 mm tube diameter
  - approx 1:20 flow area and number of tubes
- Main facility
  - shortened and narrowed bundle with U-bend tube section
  - a tube sheet
  - 3 support plates
  - full scale separator and dryer
- SET facilities
  - in tube
  - at break
  - rods far from break and support plates
  - separator and dryer

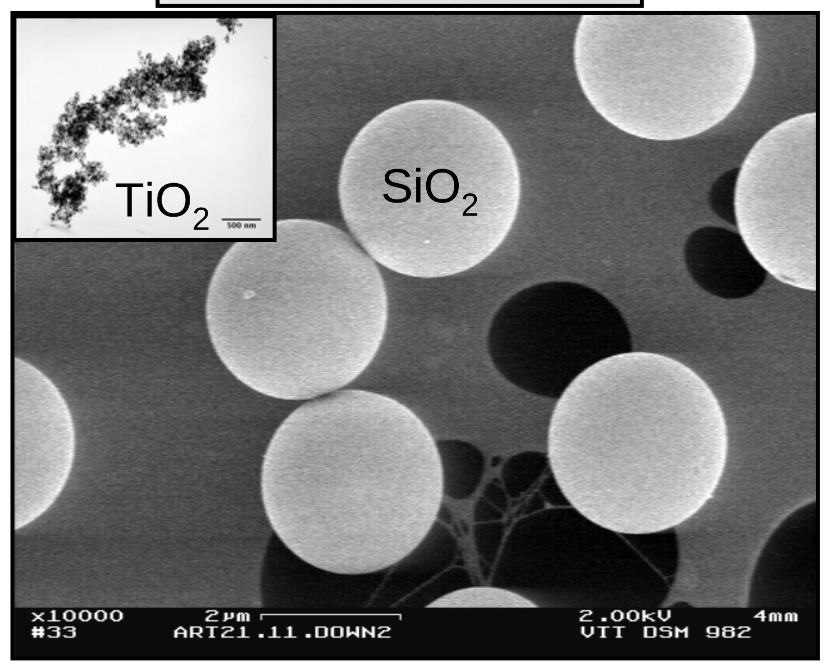
	Beznau	ARTIST
Number of tubes	3238	270 (89)*
Dryers	12	1
Separators	12	1
Bundle dia. (m)	2.68	0.57
Max tube height (m)	9	3.8 (9)**
Flow area (m^2)	3.79	0.185
Sup. plate flow area (m^2)	1.288	0.052
Bundle D_h (cm)	3.1	3.1
Total height (m)	17	10.5

*separate test section for assessing retention far from break
**in tube retention tests

Test Parameters

- **Guillotine break**
- **Aerosol particles (composition/size)**
  - **TiO_2 agglomerates (AMMD 1-5 μm)**
    - **Degussa**
    - **Nanophase**
  - **SiO_2 spheres, D_{ae} = 0.7, 1.4, 3.7 μm**
  - **Latex spheres, D_{ae} = 0.4 μm**
- **Concentrations**
  - **0.01 to 100s of mg/m^3**
- **Flow rate:**
  - **nitrogen (steam)**
  - **few 10s – several 100s kg/h**
- **scoping tests to determine suitable parameters precede experiments**
- **tests to determine experimental uncertainty**





TEM micrographs: Dr. Jerry Egeland / PSI

SEM micrograph: Dr. Unto Tapper / VTT

Primary Measurement Methods

- **Size distribution, concentration, retained mass, and DF**
  - sampling at inlet, outlet, and other locations
- **Size distribution:**
  - Berner Impactor
  - Electrical Low Pressure Impactor
  - Optical Particle Counter
- **Concentration:**
  - Filter
  - Photometer
  - Optical Particle Counter
- **Mass collection, concentrations with flow used to determine DF**
- **Flow rates at inlet and outlet and at all sampling devices, gauge pressures at inlet and outlet, gas T**

Major observations

- Two forms of aerosol deposition:
  - Always a fairly uniform layer of fine aerosol on surfaces exposed to the aerosol-laden flow. “tenacious”
  - A second form of deposit noticed in some tests consists of ‘clumps’ of deposited material.
- Widely varying retention in tubes
  - from test to test
  - high retention over short periods of time
- Resuspension can occur for deposits in tubes
  - bounce and break-up of aerosol important
- Large agglomerates did not survive transport at high flows
  - uniform size distribution leaving tube
  - particles smaller than ~1 μm don’t break up but larger particles do
- No major retention at rupture site
  - Expected based on studies of rupture propagation

Major observations

- **Away from break, most of deposited mass on support plate**
  - **May be flow recirculation at broached holes for steam generator tubes**
  - **May not occur for US plants with drilled tube support plates**
    - **Flow occurs through larger holes; jets**
    - **Gaps around tubes usually filled with "crud"**
- **Dryer/Separator not a major source of aerosol retention even for relatively coarse aerosols**
  - **Fin spacing large and little aerosol diffusion**

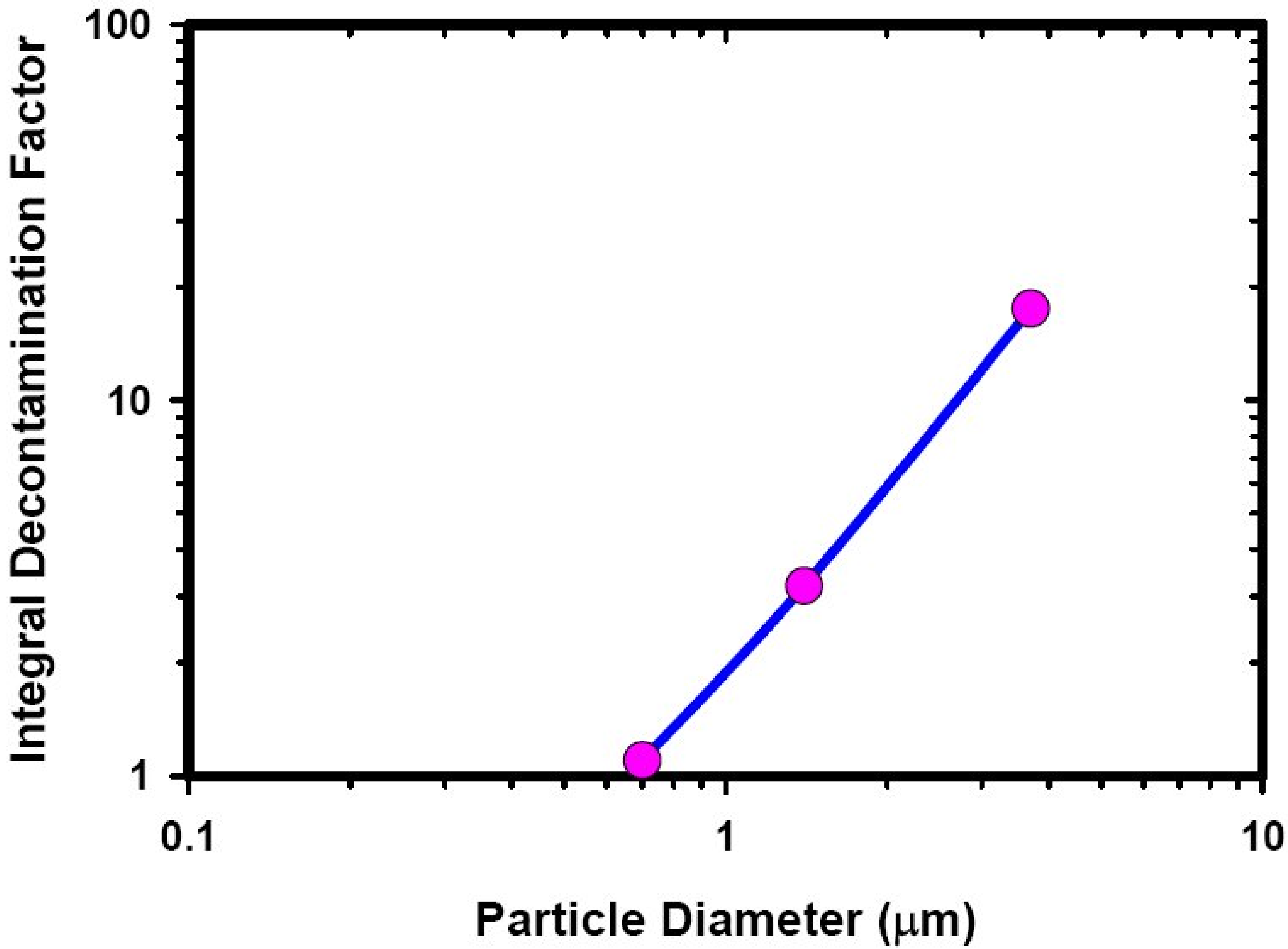
Outstanding issues

- **Understanding "bounce"**
- **Understanding breakup**
  - specific to test aerosol?
- **Understanding resuspension**
  - effect of vibrations
- **Features of steam generator**
  - Thermophoretic deposition on envelope
- **Shapes and sizes of particles coming from the degrading reactor core reaching SG**

Changes to MELCOR

- include a “lambda” factor based directly on the ARTIST results
  - based on particle size
  - insufficient risk change incentive to do more in the face of other pressing work
- monitoring 1D model being developed at Ciemat in Spain

ARTIST integral test results



Conclusions

- **Expert panel recommendations made for NUREG 1150 risk analyses by and large confirmed**
- **MELCOR predicts decontamination factors similar to those obtained from ARTIST data.**
- **Modifications made to MELCOR based on ARTIST data**
- **ARTIST provides experimental data on source term attenuation on the secondary side of steam generators**
  - **Steam Generator Action Plan (SGAP) item 3.3a complete**

ARTIST II
- proposal for a follow-up project -

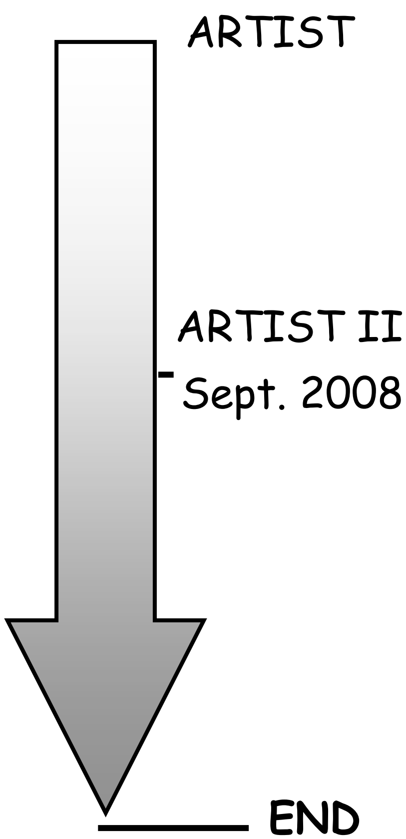
ARTIST team:

Salih Güntay, Abdel Dehbi, Steffen Danner, Ralf Kapulla,
Yehong Liao, Terttaliisa Lind, Hauke Schütt, Detlef Suckow

Paul Scherrer Institut, Switzerland

Aims of the ARTIST II project

- **Produce high quality data for:**
  - Data needs established in ARTIST I
  - Other conditions due to AM necessitate description of the retention where the application of current data beyond its validity range may be misleading
  - Development of models; fundamental and detailed to simplified and application oriented
- **Develop methodology for SGTR Risk Assessment**
  - Re-assessment of SGTR induced environmental risk
  - Producing international consensus about the risk significance of SGTR events during DBA and SA
- **Continue fundamental investigations including model development in form of PhDs/Masters**

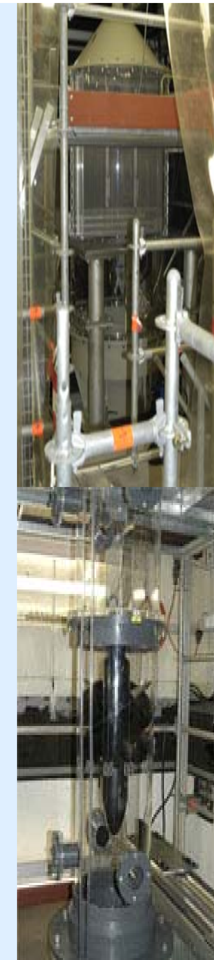


ARTIST II Facilities

Single tube

Break stage

Flooded separator



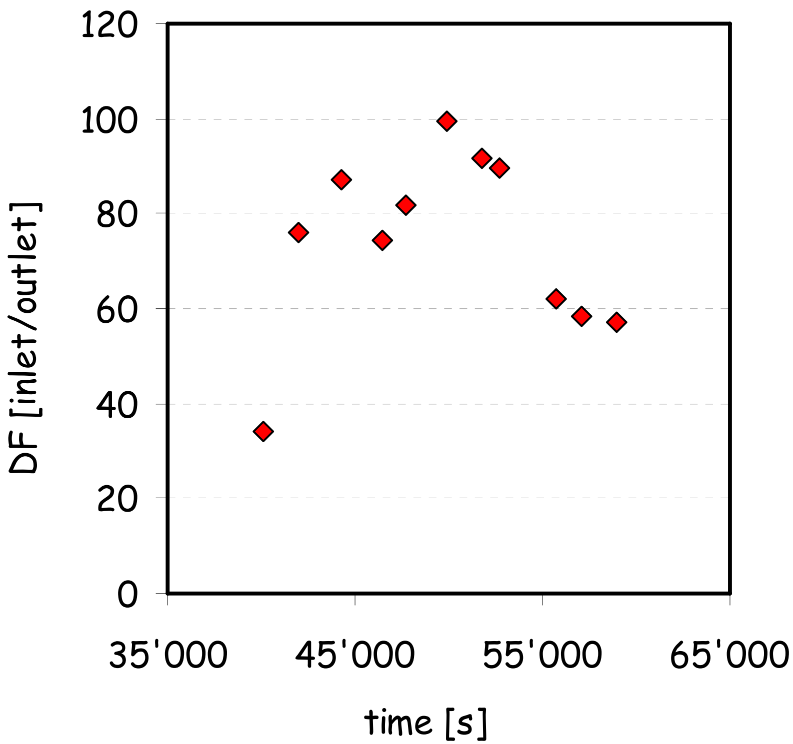
Droplet retention

Phase I, In-tube retention - no bounce

- Straight tube, 9 m
- Spherical SiO_2, AMMD = 1.4µm, or equivalent particles
- 2 tests
  1. With „normal bounce" conditions
  2. No bounce, or bounce minimized with particle surface coating

=> **Effect of bounce and resuspension on deposition at** very high velocity

Spherical SiO_2, AMMD = 0.7 µm

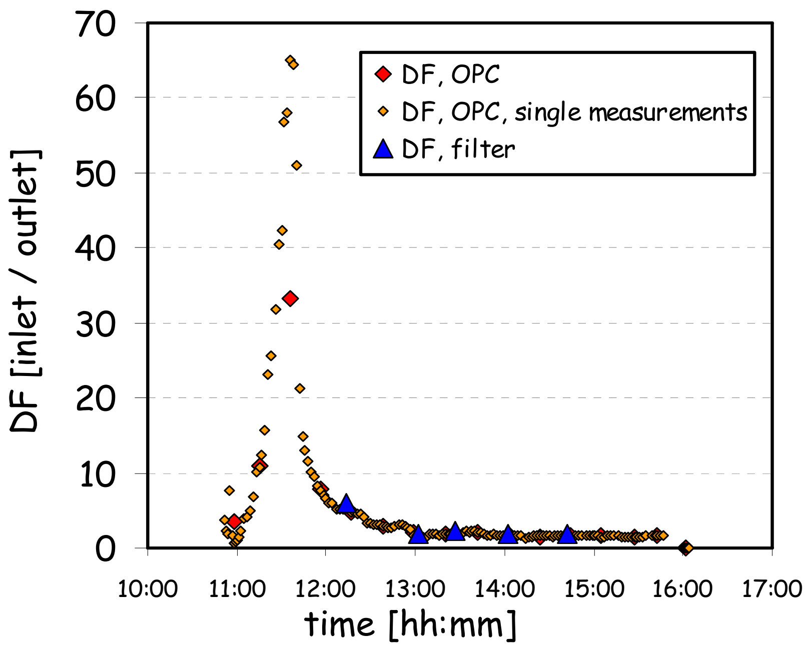


Phase I, In-tube retention - high concentration

- U-tube, 18 m, intermediate bend (384 mm)
- Spherical SiO_2, AMMD = 1.4 µm, or equivalent particles
- 2 tests
  1. Particle concentration ~ 50-100 mg/Nm^3
  2. Particle concentration ~ 500-1000 mg/Nm^3

=> Does high retention phase repeat itself?

1.4 µm SiO_2, high concentration



Phase II, Break stage - no bounce

- Break stage test facility
- Spherical SiO_2, AMMD = 1.4 µm, or equivalent particles
- 2 tests
  1. With „normal bounce" conditions (only if particle material other than SiO_2)
  2. No bounce, or bounce minimized with particle/surface treatment

=> Effect of bounce and resuspension on deposition

TiO_2 agglomerates

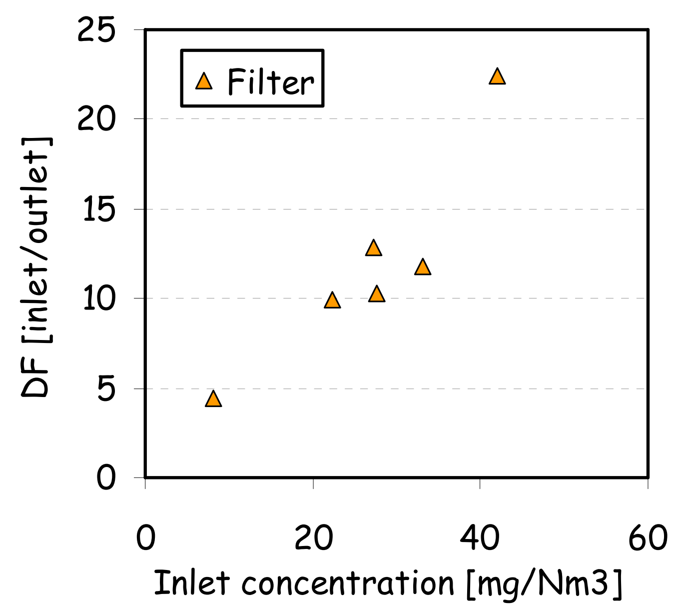
SiO_2 spherical, D_{ae} = 3.7 µm

Break stage: higher retention with higher concentration

Possible theory:

- bounce is significant on bare tubes, so DF low with low concentration
- With deposition, surface roughness increases and bounce subsequently reduced
- With higher concentration, reduction in bounce quickly reached, so DF higher
- Question: with ever increasing concentration, when does resuspension start to be effective?
- Need to experiment with much higher concentration to answer questions

SiO_2 spherical,
D_{ae} = 3.7 µm



Phase II, Break stage - high concentration

- Break stage test facility
- Spherical SiO_2, AMMD = 1.4 μm, or equivalent particles
- 2 tests
  1. Particle concentration ~ 50-100 mg/Nm^3 (only if not SiO_2)
  2. Particle concentration ~ 500-1000 mg/Nm^3

=> Net effect of resuspension?

=> High retention with realistic particle concentration?

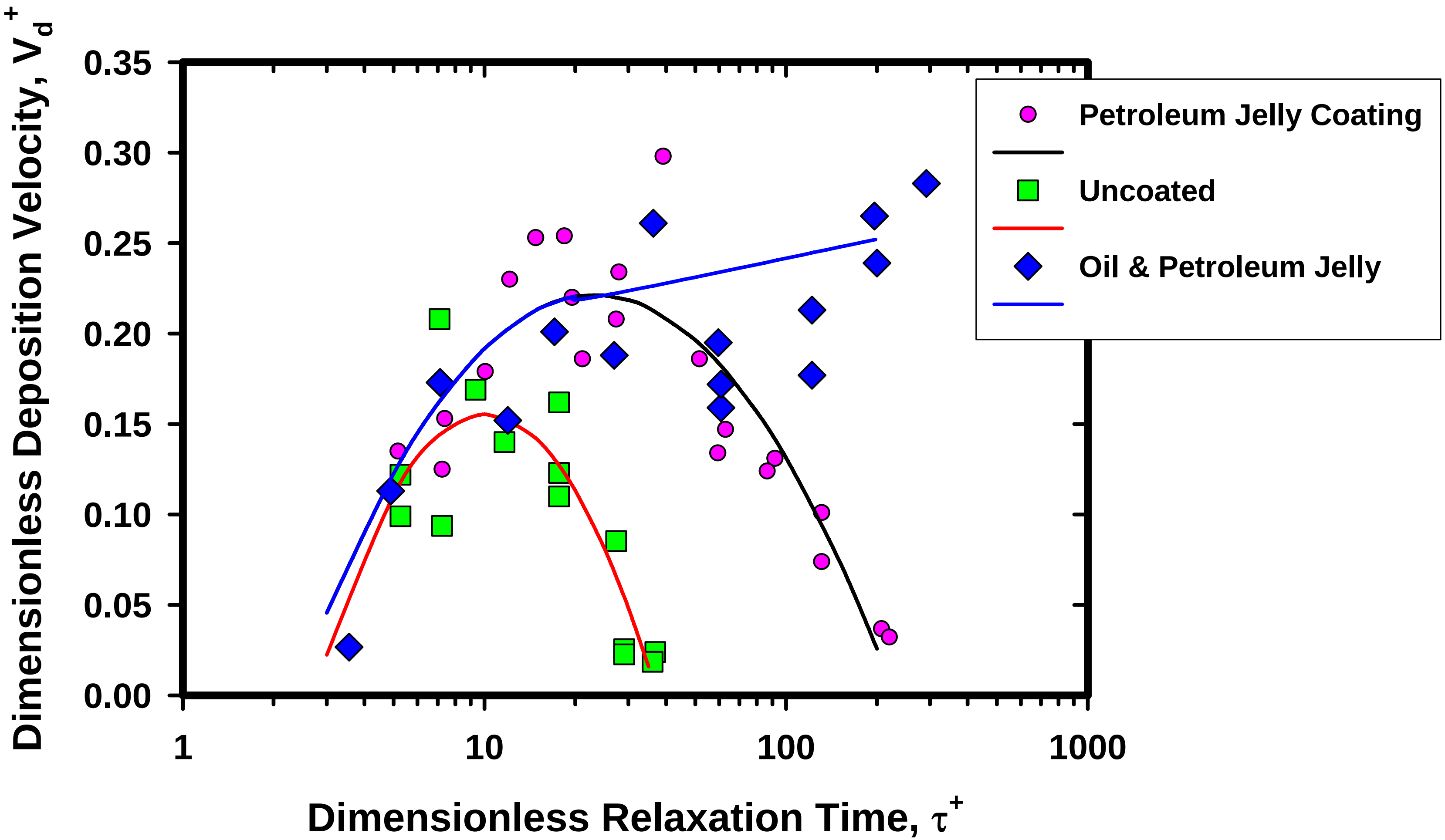
Phase II, Break stage - break geometry

- Break stage test facility with fish mouth break
- Spherical SiO_2, AMMD = 1.4 µm, or equivalent particles
- 1 test

=> Radially directed jet with max momentum and deepest radial penetration

=> Retention dependence of break geometry

Effects of "bounce" on Particle Deposition



Information provided by D. Powers, April 2008

Feasibility of achieving low bounce particles

Calculations show that inside SG tube, we have τ^+ of order 10 to 300 (AMMD 1-3.7 μm, u* of order 10 m/s). Hence bounce would be strong if particles uncoated.

•It is shown in tests that bounce is strongly dependant on particle type:

- Strongest for uncoated particles
- Somewhat mitigated with petroleum jelly coated particles
- Strongly mitigated with mix of oil-petroleum jelly coating

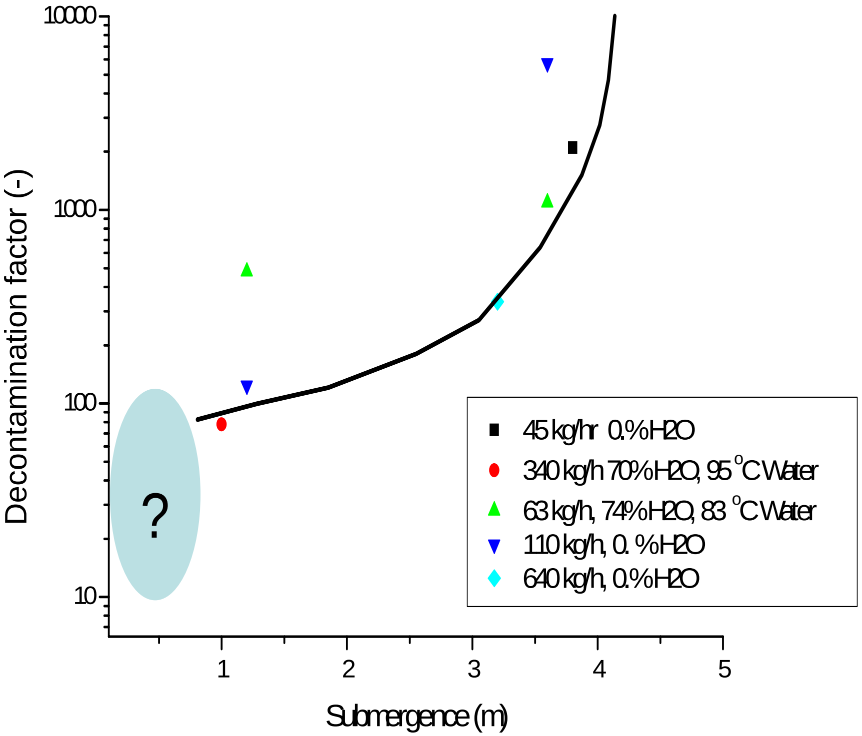
•Hence: in principle, it is feasible to conduct tests in ARTIST-II with particle bounce very much reduced.

Thanks to Dana Powers for bringing this information to our attention.

Phase V, Flooded bundle – small submergence

- ARTIST bundle, guillotine break, 0.3 m submergence
- Spherical SiO_2, AMMD = 1.4 and 3.7 µm, two sizes simultaneously (or equivalent particles)
- 2 tests
  1. 45 kg/h
  2. 650 kg/h

=> Retention due to jet impingement on neighboring tubes

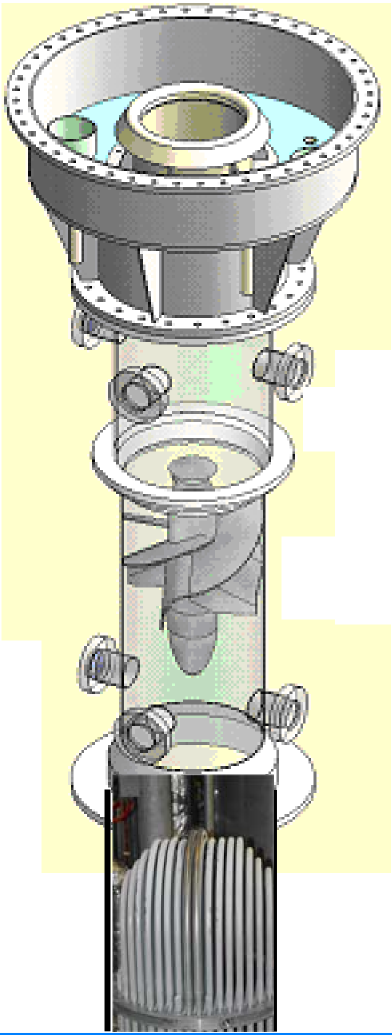


Phase VIII, Flooded separator – new phase

- ARTIST separator, flooded to separator outlet
- Spherical SiO_2, AMMD = 1.4 and 3.7 µm, or equivalent particles
- 4 tests
  1. 50 kg/h, two particle sizes
  2. 200 kg/h, two particle sizes

  =>Extent of retention by removal by flow-swirling and by cyclones

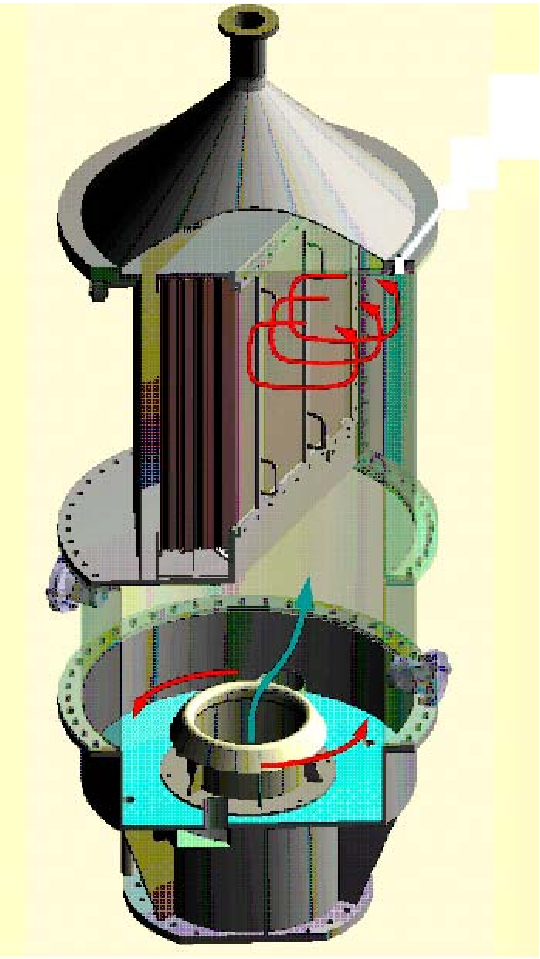
  Quantification of DF: Important if the break is at the top of tube bundle and operators able to flood according to SAMG and long range dP-transducer



Phase VI, Droplet retention in Dryer

- Measurements:
  - Velocity fields
  - Droplet size distribution and retention in several locations in the dryer unit
- 4 flow rates
  - 50 kg/h
  - 200 kg/h
  - 400 kg/h
  - 800 kg/h

=> Droplet size and flow rate dependent retention



ARTIST and ARTIST II experimental program

BDBA source term quantification		ARTIST	ARTIST II
Phase I:	In tube	15	4
Phase II:	Break stage	9(+2)	5
Phase III:	Far field	8(+2)	
Phase IV:	Separator&dryer	5	
Phase V:	Flooded bundle	2(+3)	2
Phase VII:	Integral mock-up	3	
Phase VIII:	Flooded separator, **new**		4
	Total	**42(+7)**	**15**

DBA source term quantification

Phase VI:	Droplets (in separator & dryer)	yes	yes

(x): EU-SGTR

ARTIST II Consortium (anticipated funding organizations)

- CSN/CIEMAT (Spain)
- IRSN (France)
- JNES (Japan)
- HSK (Switzerland)
- NPP Gösgen-Däniken (Switzerland)
- NPP Beznau (Switzerland)
- VTT/STUK/TVO/FORTUM (Finland)
- NRG/Borssele NPP/KFD (the Netherland)
- US Nuclear Regulatory Commission (USA)
- AREVA (Germany)
- SKI/Ringhals NPP (Sweden)
- Nuclear Safety Directorate (UK)

ARTIST II experimental and proposed analytical program

Start: Sept.1, 2008

Duration: 3 years

CFD flow simulations: bundle, separator and dryer

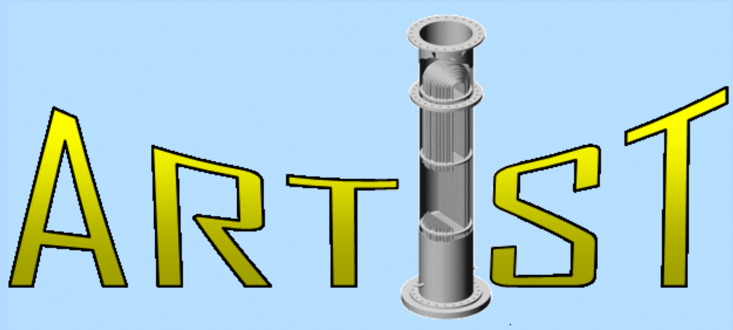
full size SG (NRG, JNES, SKI)

Aerosol removal model: break stage (dry and wet) (CIEMAT)

Incorporation of ARTIST data base in MELCOR (PSI)

Droplet retention in Separator and dryer (?)

???



Aerosol Trapping in Steam Generator ARTIST: Findings and Potential Effects on SGTR Risk Profile

ARTIST team:

Salih Güntay, Abdel Dehbi, Steffen Danner, Ralf Kapulla,

Terttaliisa Lind, Hauke Schütt, Detlef Suckow

Paul Scherrer Institut, Switzerland

Outline

- History
- Aims of ARTIST
- ARTIST International Consortium Project
- Facility and scaling
- Model aerosol particles
- Experimental Program and results
- Conclusions
- A new SGTR risk assessment methodology and use of ARTIST data
- Final remarks

History

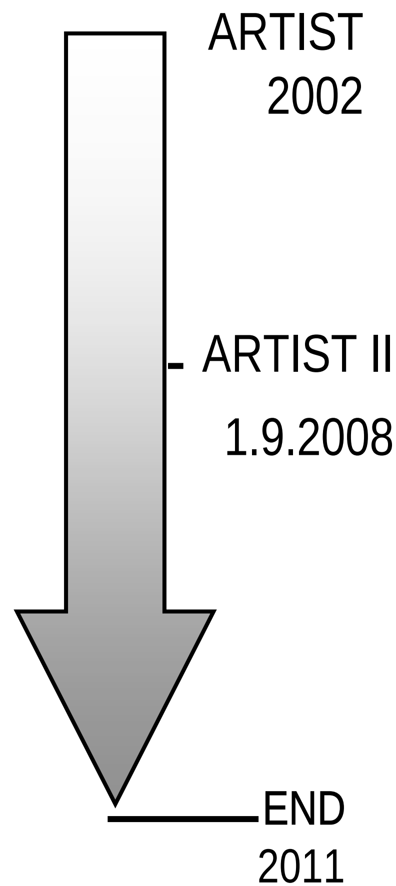
- Motivation and support from Utility: Large contribution of SGTR in CDF and Risk in NPP-Beznau due to excessive tube problems in 1997
- Design and Procurement: 1998-2000
- EU 5. Framework Project SGTR: 2000-2002: PSI (Vertical SG without Dryer/separator), VTT (Exp: horizontal SG), NRG , Rez, CIEMAT
- ARTIST International Consortium Project

  Phase I: 2002-2007

  Phase II: 2008-2011
- Potential continuation >2011: in form of Fundamental Studies (PhD), model development efforts at PSI

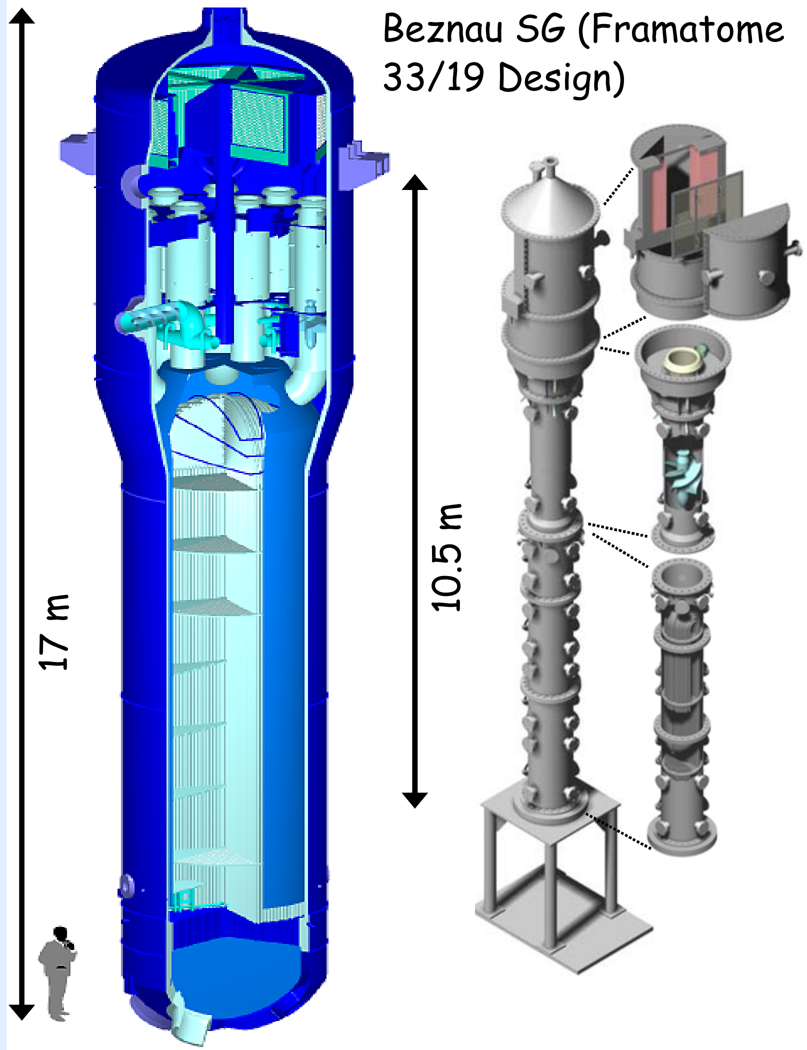
Aims of the ARTIST International Consortium project

- **Provide an international forum to develop new information and share among partners**
- **Produce high quality data for:**
  - Development of fundamental and detailed to simplified and application oriented models
  - Facilitate evaluation of effectiveness of SAMG
- **Develop methodology for SGTR Risk Assessment**
  - Re-assessment of SGTR induced environmental risk
  - Provoke international consensus about the risk significance of SGTR events during DBA and SA
- **Initiate fundamental investigations in form of PhDs/Masters**



ARTIST Consortium (in alphabetical order)

- AVN (Belgium)
- Ciemat (Spain)
- CSN (Spain)
- HSK (Switzerland)
- IRSN (France)
- JNES (Japan)
- KK Gösgen-Däniken (Switzerland)
- NOK, KK Beznau (Switzerland)
- Nuclear Safety Directorate (UK)
- Ringhals NPP (Sweden)
- Universidad Politecnica de Madrid (Spain)
- University of Newcastle (UK)
- US Nuclear Regulatory Commission (USA)
- VTT (Finland)



Beznau SG (Framatome 33/19 Design)

ARTIST Facilities

Break stage

Larger scale-bundle

Droplet retention

Integral mock-up facility

Scaling

Design basis: Framatome 33/19 Design

- Separator: 1:1 (steal or mostly transparent)
- Dryer: 1:1 (with actual Chevron panels) (all steel or inlet transparent)
- Bundle: 264 straight tubes, height: 1:0.42, with 1:1 layout
  - Broached support plates with 1:1 layout
- Single tube length: 1:1 with smallest and medium curvatures
- Tube dimensions: 1:1

Flow rates: 40 kg/h to 800 kg/h (fully representative)

Pressure: < 5 bar in primary, ~ 1 bar secondary

Dry conditions (except 1 in-tube test with slight steam condensation)

Model Aerosol Particles

- Evaporation and Condensation generated single/multi component Particles (SnO/CsI/CsOH, etc) (not used for ARTIST due to high costs)
- Fluidization of mono/polydisperse powders (TiO_2 (two types), SiO_2)
- Dispersion of suspended material (Latex, SiO2 in solution) and drying droplets

. Monodisperse particles (SiO_2/Latex): well known size

. Polydisperse particles (TiO_2): lots of problems due to

unknown surface finish characteristics affecting deposition and no size control due to de agglomeration at high velocity/sonic front

Particle Morphology and Size in PWR Hot leg

- Working group: M. Kissane (IRSN), D. Powers (SNL), M. Reeks (NC)
- Very complicated and not resolved issue since many parameters (pressure, core degradation, etc) influence
- Hot leg conditions based on Phébus and other tests
- Phébus:
  - 15-40 % control rod metals, similar amount of oxides, and rest FPs
  - implies an "onion-skin" type of structure where the kernel rich in highly refractory materials and on top condensed species of more volatile species containing cesium and rubidium and perhaps migrated into and interact chemically with the substrate
  - **For practical purpose AMMD at SG inlet or in SG based on impactor data**
    - **3 µm (gsd 2) at 150 °C, 1.7 µm (gsd 2) at 730 °C, 0.1 µm at 930 °C following an exponential increase along inverse temperature**

ARTIST experimental program

BDBA source term quantification		ARTIST
Phase I:	In tube	15
Phase II:	Break stage	9(+2)
Phase III:	Far field	8(+2)
Phase IV:	Separator&dryer	5
Phase V:	Flooded bundle	2(+3)
Phase VII:	Integral mock-up	3
	Total	**42(+7)**

DBA source term quantification

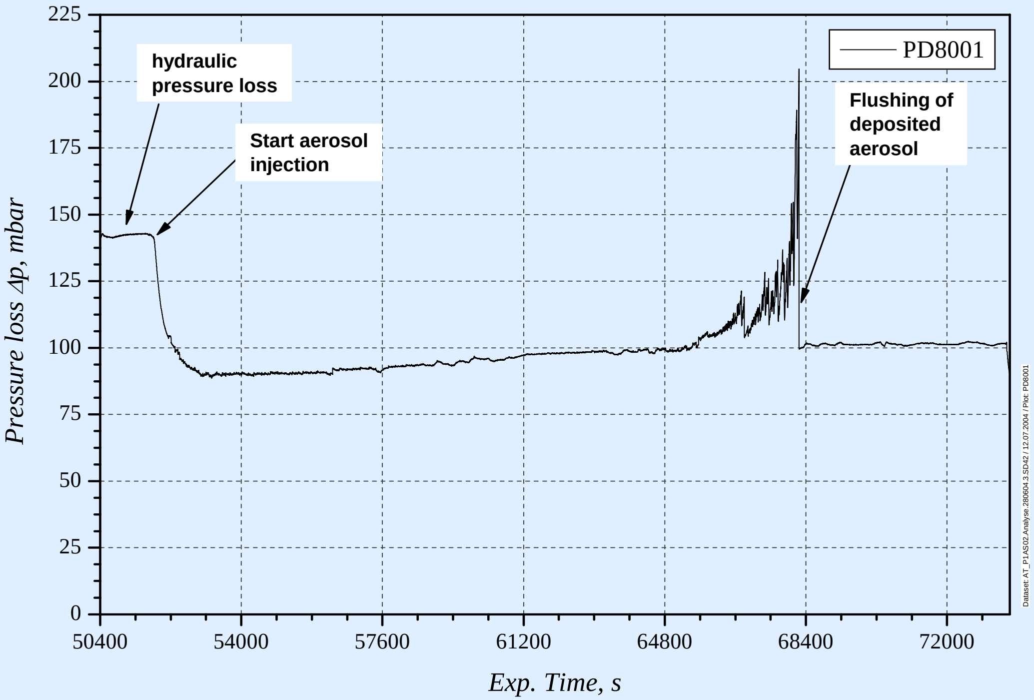
Phase VI:	Droplets (in separator & dryer)	yes

(x): EU-SGTR

Phase I, In-tube retention (1:3)

- **15 tests**
  - 225 – 364 kg/h, with pressure ratio of 3.5:1
  - Straight tube and
  - U-tube with two bend diameters (83 and 384 mm)
  - Dry conditions, except 1 test with slight steam condensation
  - Mono/Polydisperse particles
  - Very low to modest concentrations

Phase I, In-tube retention (2:3)

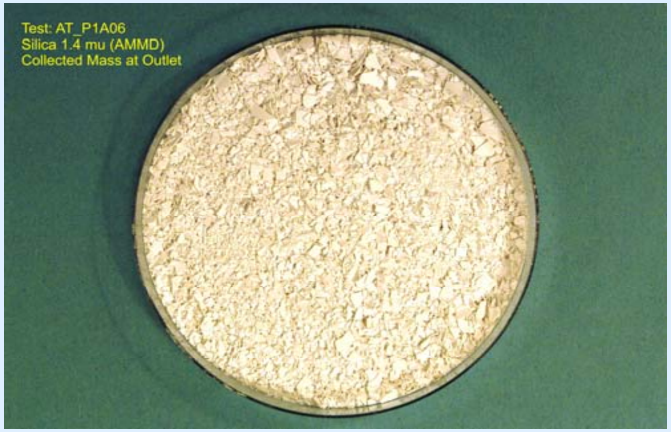


- 2*9 m with 83.2 mm curvature
- 70 -240 m/s velocity in Tube
- Dry TiO2 (2-3 μm inlet/<1 μm outlet)

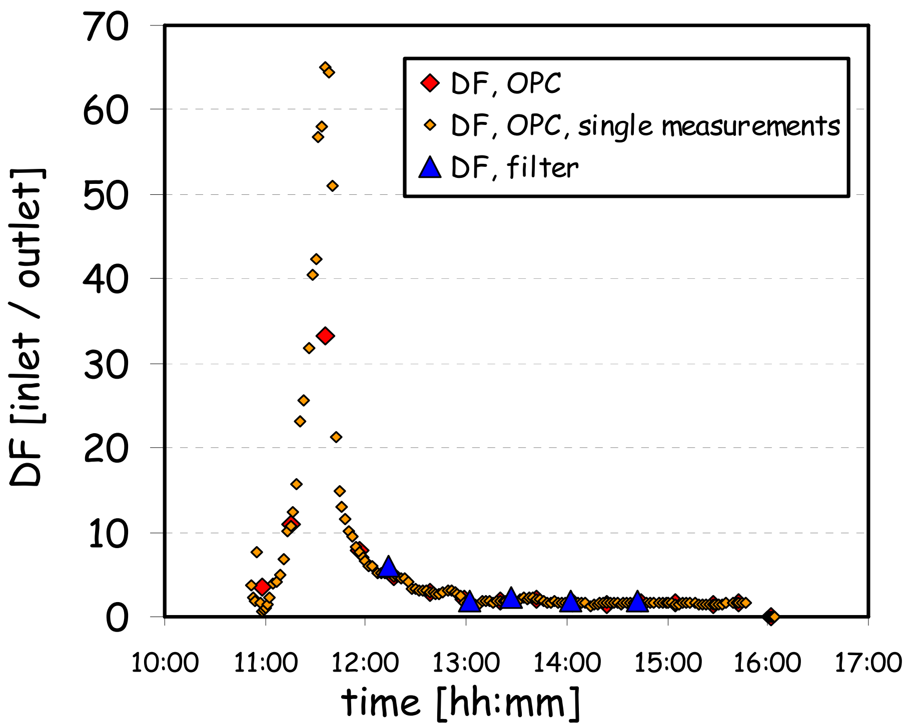
- Very dynamic aerosol processes (turbulent deposition/resuspension, de-agglomeration of TiO_2)
- Challenge for modeling (PhD Pamela Longmire/SNL)
- Effect on flow re-distribution among intact tubes in inlet plenum

Phase I, In-tube retention (2:3)

DF	Conc.	Particles
< 65	medium	SiO_2
1.0 - 2.2	medium	TiO_2
8.2	Slight steam cond.	TiO_2
< 100	very low	SiO_2, latex

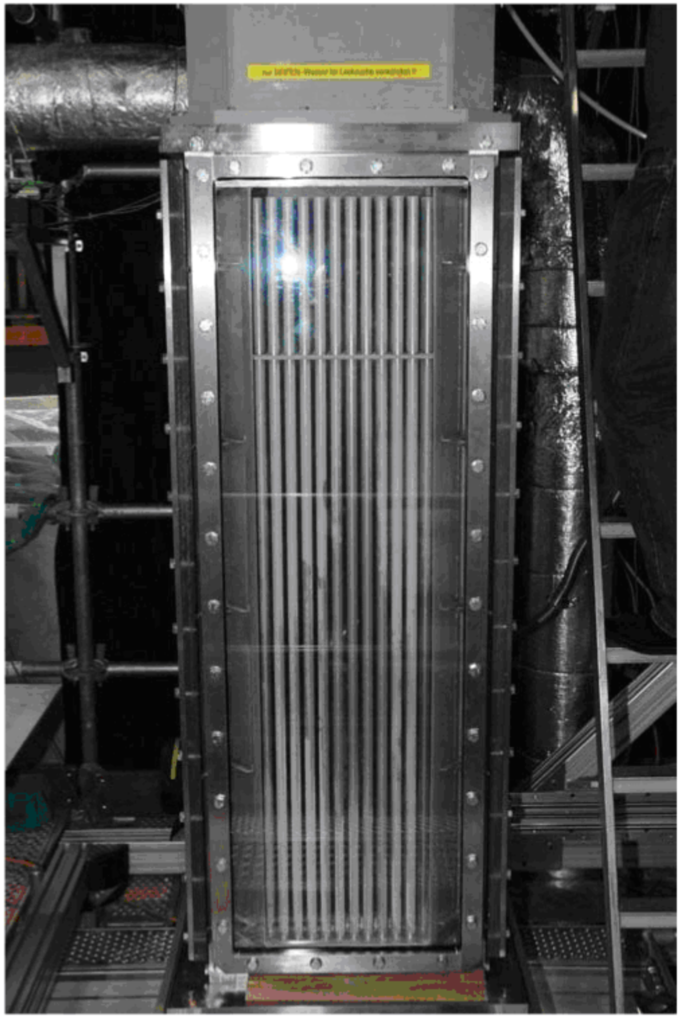


Aerosol (SiO_2) fragments collected in the outlet plenum



1.4 µm SiO_2, high concentration

Phase II: Break-Stage Retention: Dry conditions (1:6)



- **Chocked flow at the break**
- **Guillotine Break**
- **Dry conditions**

9 tests

- 360 kg/h,
- Monodisperse SiO_2 particles
- AMMD: 1.4 to 3.8 μm

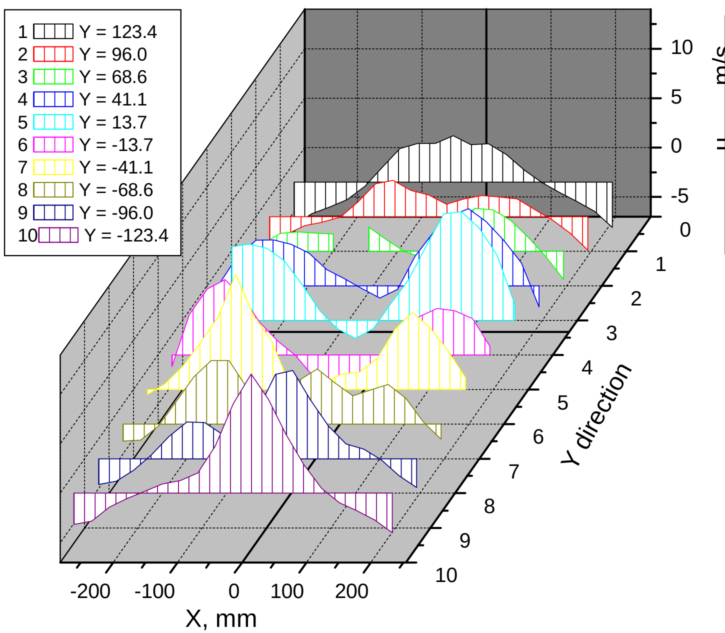
2 tests with full bundle

- 600 kg/h
- **Polydisperse TiO_2 particles**
- AMMD: 2.3 μm before break

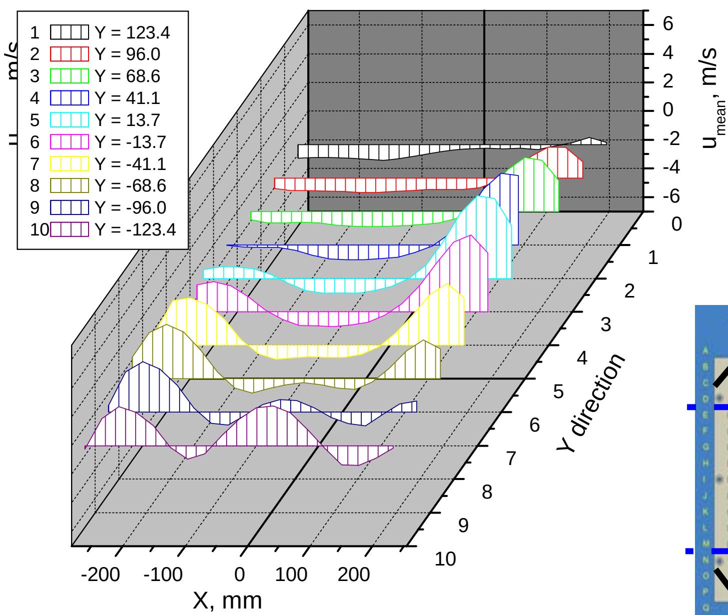
Phase II: Break-Stage Retention: Velocity profiles (2:6)

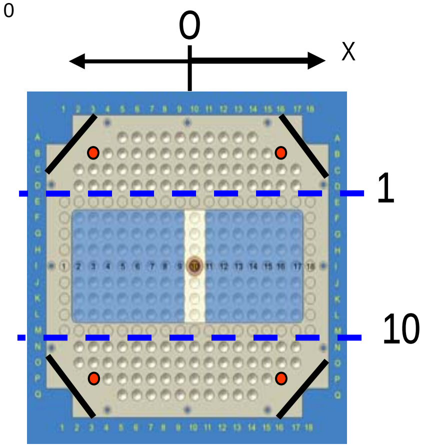
Measured velocity profile: **Guillotine Break, 360 kg/h**

Z=600 mm



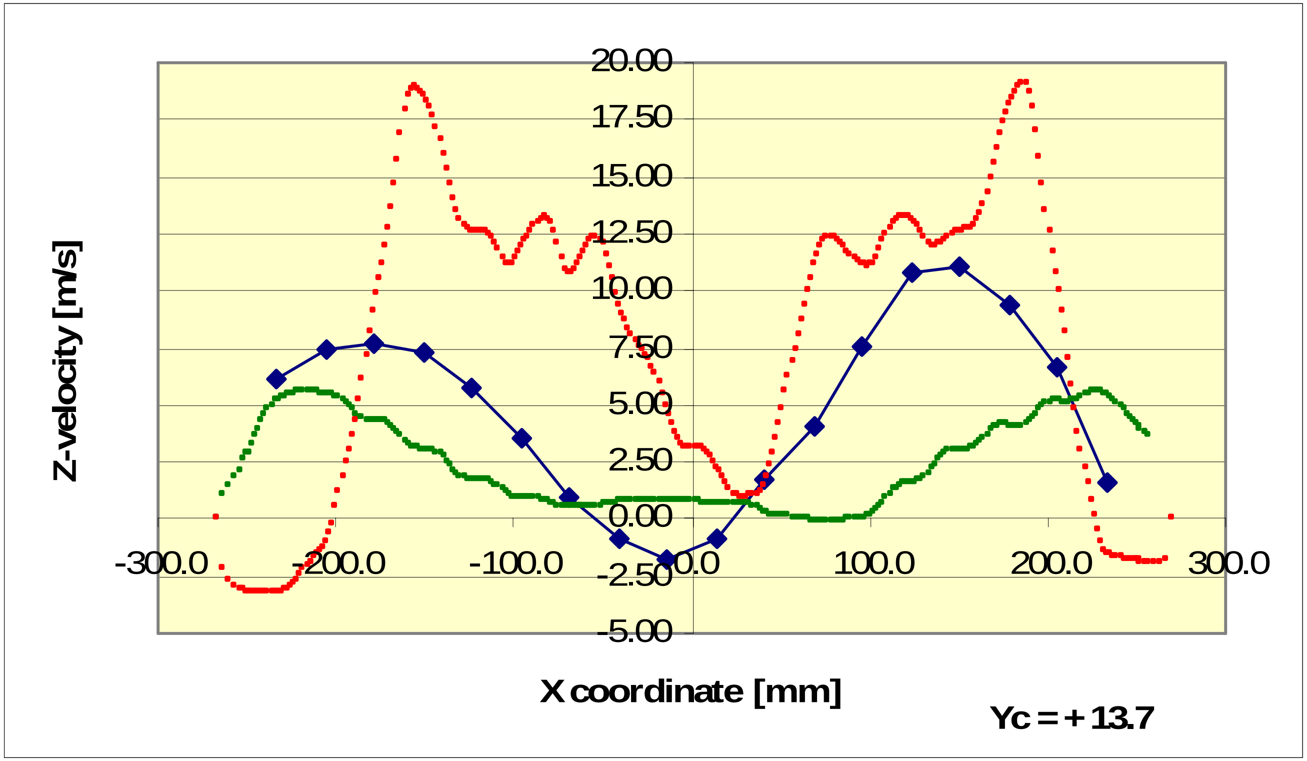
Z=1000 mm





- Very 3D flow

Phase II: Break-Stage Retention: Velocity profiles (3:6)



- Measured velocity profile

FLUENT Simulations by Ringhals/EPSILON

- with k-ε
- with Reynolds Stress Model (RSM)

Phase II, Break stage (4:6): Aerosol material type dependent local deposition pattern

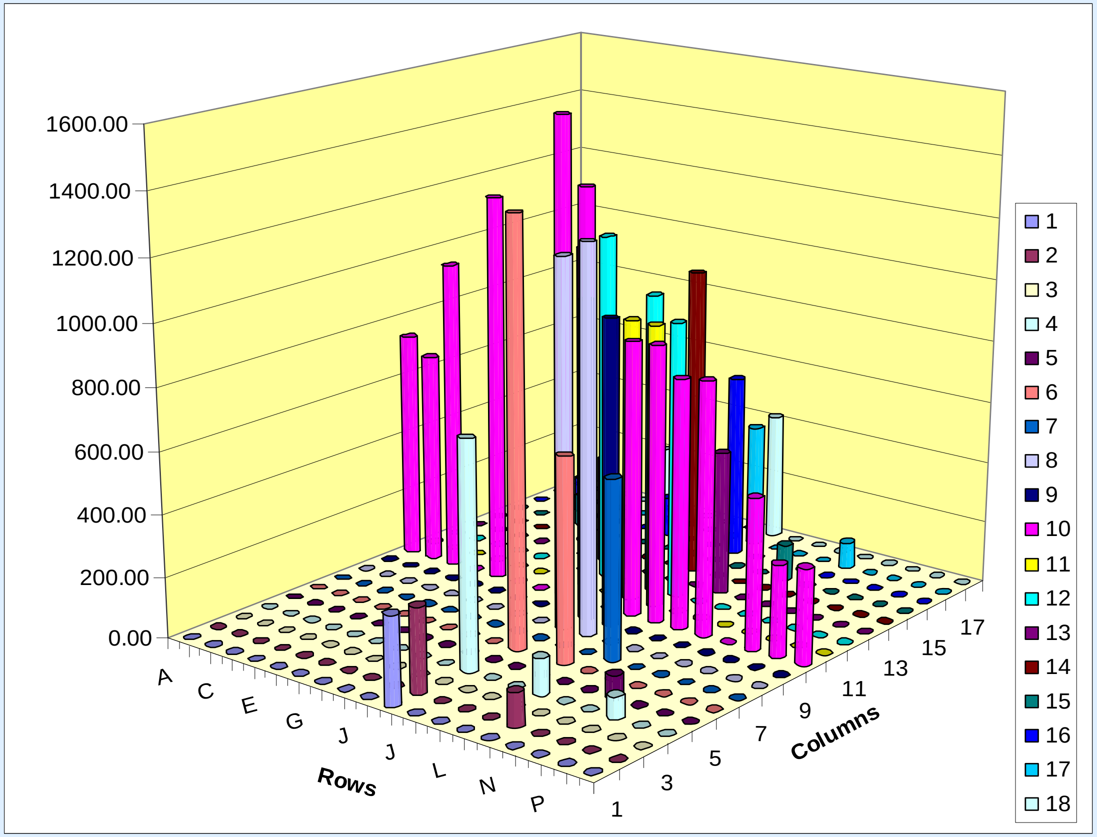
TiO_2, Dae = 2.3 µm

SiO_2, Dae = 1.4 µm

SiO_2, Dae = 3.7 µm

- Flow rate: 600 kg/h for TiO_2, 360 kg/h for SiO_2 tests

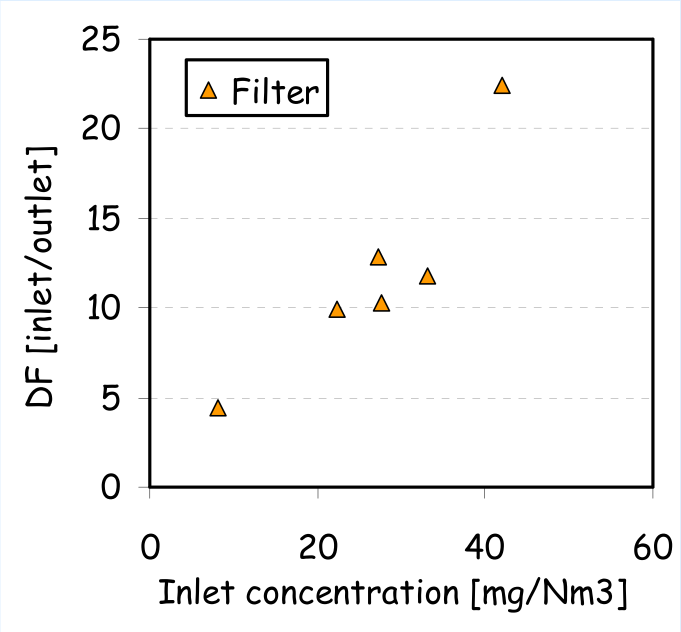
Phase II, Break stage (5:6): Deposition pattern

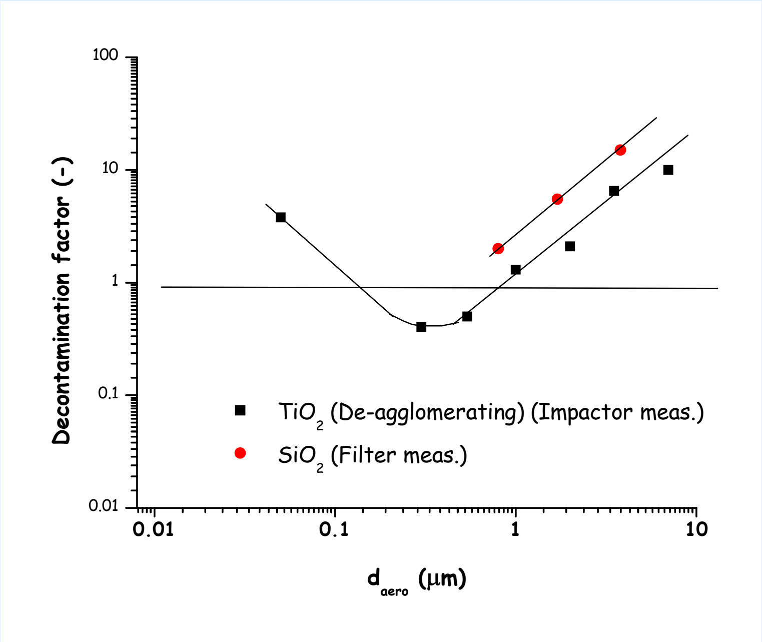


Tube to tube aerosol deposition profile (SiO2, 3.8 μm)

Phase II, Break stage (6:6): Retention

- Highest retention potential among other retention stages
- Decontamination Factor =
  - increases with increasing inlet concentration
  - increases with increasing D_p





Phase III, Far field stage (1:1)

- 8 (+2 EU-SGTR) tests
- Mass flow rate 33 & 105 kg/h
- TiO2: deposition everywhere
- Collected mass on certain tubes indicates roughly constant DF per stage
- SiO2: mostly on support plates
- SiO2 (d_{aero} 3.7 μm) DF: ~1.07
- DF might be higher at higher inlet concentration

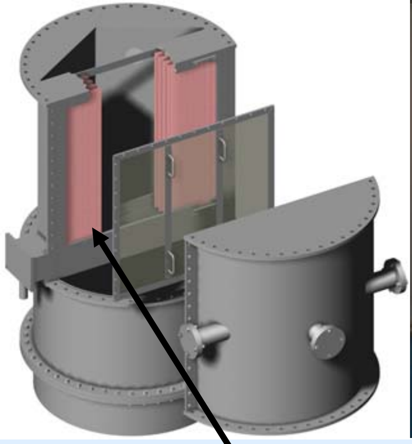
TiO_2 Bundle test

SiO_2 Far field stage test

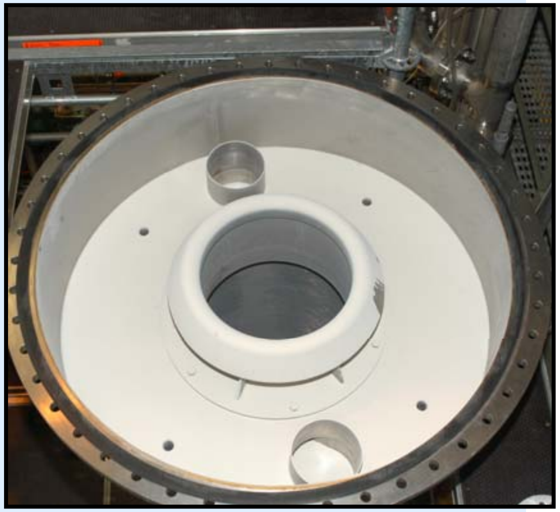
Phase IV: Separator & Dryer (1:2)

- 5 tests (2 only separator)
- Mass flow rate 100, 360 and 650 kg/h
- Local turbulence initiated agglomeration and hence sedimentation
- Decontamination Factor

DF	Particles	D_{ae}
1.2 – 1.4	TiO_2	3 μm, aggl.
1.5 – 1.6	SiO_2	integral mock-up



Aerosol collected in Condensate collector below the panels



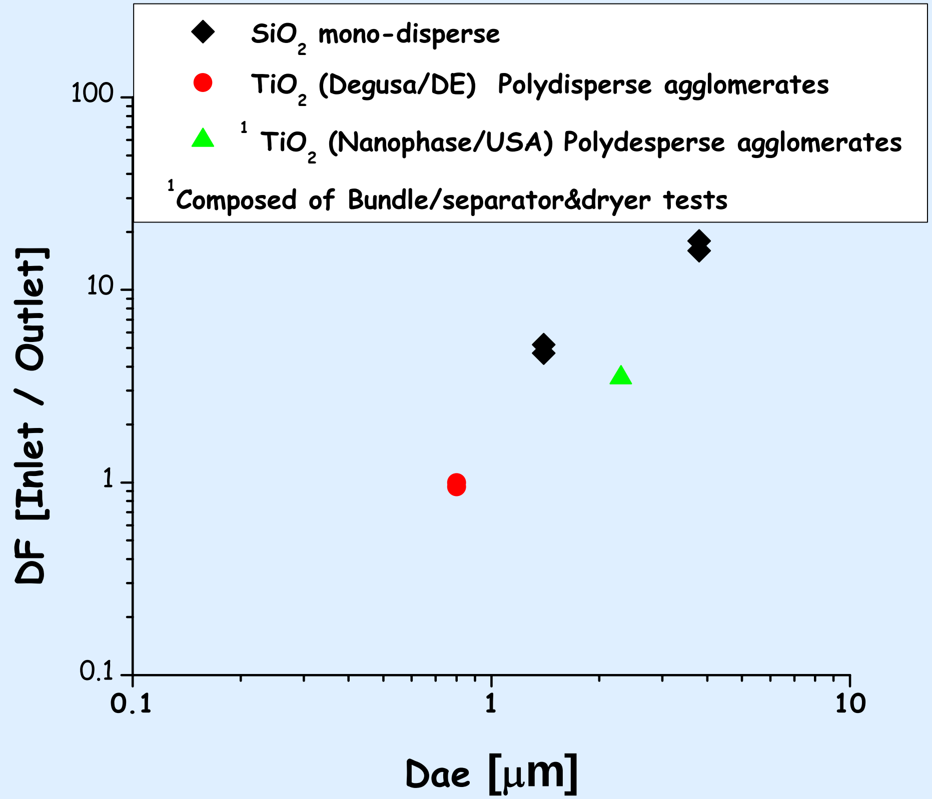
Phase VII: Integral mock-up tests

Aim: verify consistency of separate effect data at certain conditions

Decontamination Factor =

- Consistent with Break Stage Tests
- DF increases with concentration
- DF increases with particle size

Effect of model aerosol particle material/surface treatment



Conclusions #1, aerosol tests

- In-tube retention
  - Dynamic, depends on particle size and concentration
  - Steam condensation increases DF significantly
  - => the effect of particle concentration?
  - => the effect of bounce/resuspension?
- Retention largest in the break stage
  - Depends on particle size and concentration
  - => the effect of particle concentration?
  - => fish-mouth break leading to higher gas/particle momentum and deeper penetration in Bundle?
  - => data with minimized bounce/resuspension needed for modeling

Conclusions #2, aerosol tests

- **Retention in the far field**
  - **=> the effect of particle concentration?**
  - **=> Effect of aerosol composition?**
- **Retention in the flooded bundle**
  - => High DF (50 – 2000) with submersion 1.2 – 3.8 m
  - **=> retention close to the break (?) with smaller submersion**
- **Retention in Separator & Dryer**
  - => ~ 30-40 % of incoming mass retained independent of Flow Rate
- **Retention in the integral mock-up facility**
  - Dominated by retention in the break stage
  - Consistency of separate effect data demonstrated

Transport/Removal of Activity in Steam Generator

- SGTR concurrent with core damage involves:
  - Major activity in vapour form at SG inlet
  - Rest of activity and inactive material in aerosol form
- Transformation of activity in vapour form by vapour condensation dependent on local temperature
- Removal of some fraction of vapour by condensation on structure surface
- Transport/removal of Rest of vapour of condensed on particles or form new particles dependent on aerosol removal/transport process

ARTIST addresses only aerosol removal/transport process in SG

Motivation for a new SGTR risk assessment methodology

- MELCOR contains models for vapor/aerosol behavior but lacks specific aerosol transport/removal in SG complex structures at relevant thermal-hydraulic conditions
- For risk assessment with many hundred variations to consider uncertainties: MELCOR is too expensive
- A fast running lump parameter model including Monte-Carlo sampling for uncertainties under development
- Preliminary sample analysis demonstrates the strength and provides feasibility of SGTR risk reduction

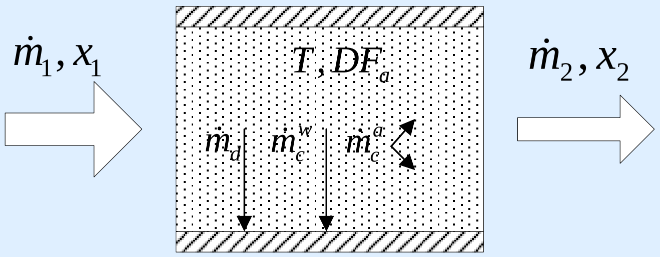
A new SGTR risk assessment methodology

- Lump Parameter Model tracking vapor/aerosol phases in each release path in SG secondary side with:
  - T/H and Vapor/aerosol boundary conditions and uncertainties from SA code predictions
  - Temperature dependent ultimate particle size based on Phébus tests
  - Temperature dependent vapor fractions of released classes including all species from SOPHAEROS code (IRSN/FR) analysis
  - Release path dependent ARTIST DFs (d_p, c)
- Monte-Carlo sampling for all uncertainties
- APET for all SGTR sequences
- Running Model for each APET branches for determination of risk

Lump Parameter Model: Key Aspects

- Accounts for aerosol behavior in complex structures of SG at hydrodynamic conditions by use of ARTIST data for each SG retention stage
- Accounts for vapor conversation using temperature dependent vapor fraction data base generated from SOPHAEROS code runs
- Accounts for vapor fraction condensed on structure and converted to particles by user input including its uncertainty
- Accounts for temperature dependent aerosol size determined by measured sizes in hot leg in all Phébus tests with AgInCd
- Neglects other processes playing a secondary role: thermophoresis, diffusiophoresis,…

Lump Parameter Model Description



$$\dot{m}_2 = \frac{1 - x_1(1-\alpha)}{DF_a - x_2(DF_a - \alpha)}\dot{m}_1$$

α: Vapour split fraction on walls/ particles = 0.5 (0.1-0.9)

DFa: ARTIST DF

m: mass flow of release class (I, Cs, ..)

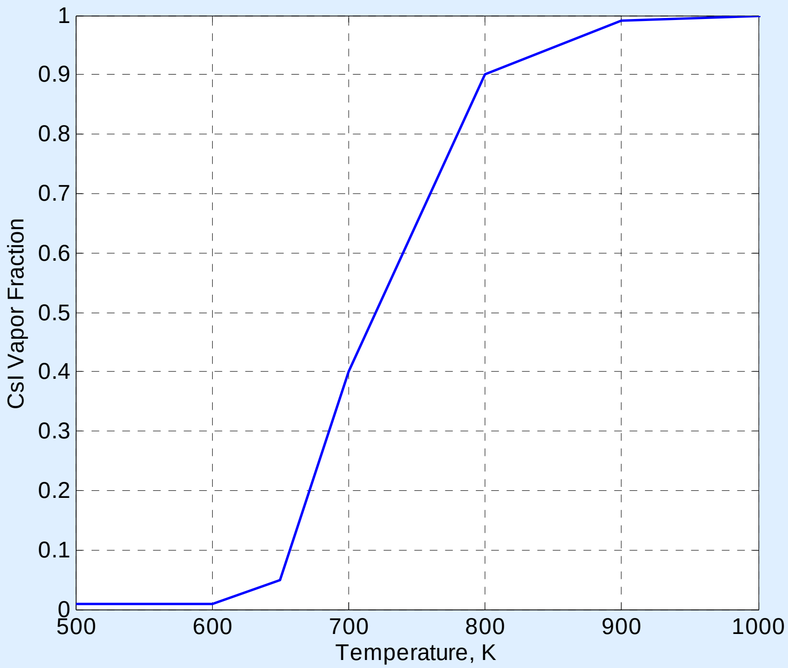
X: vapor fraction of the mass flow

T: Gas temperature

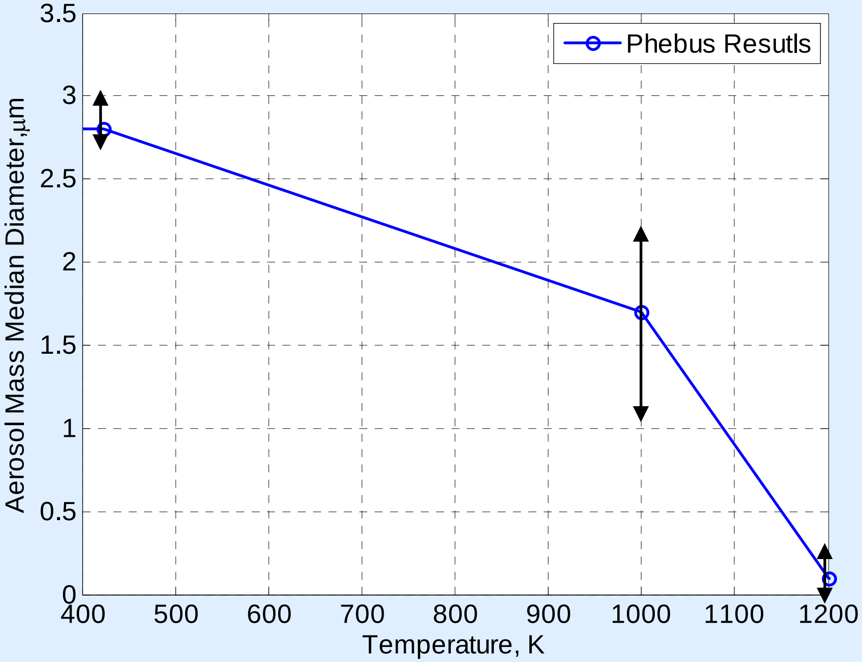
1: donor volume

2: current volume

Lump Parameter Model Data Base (1:3)

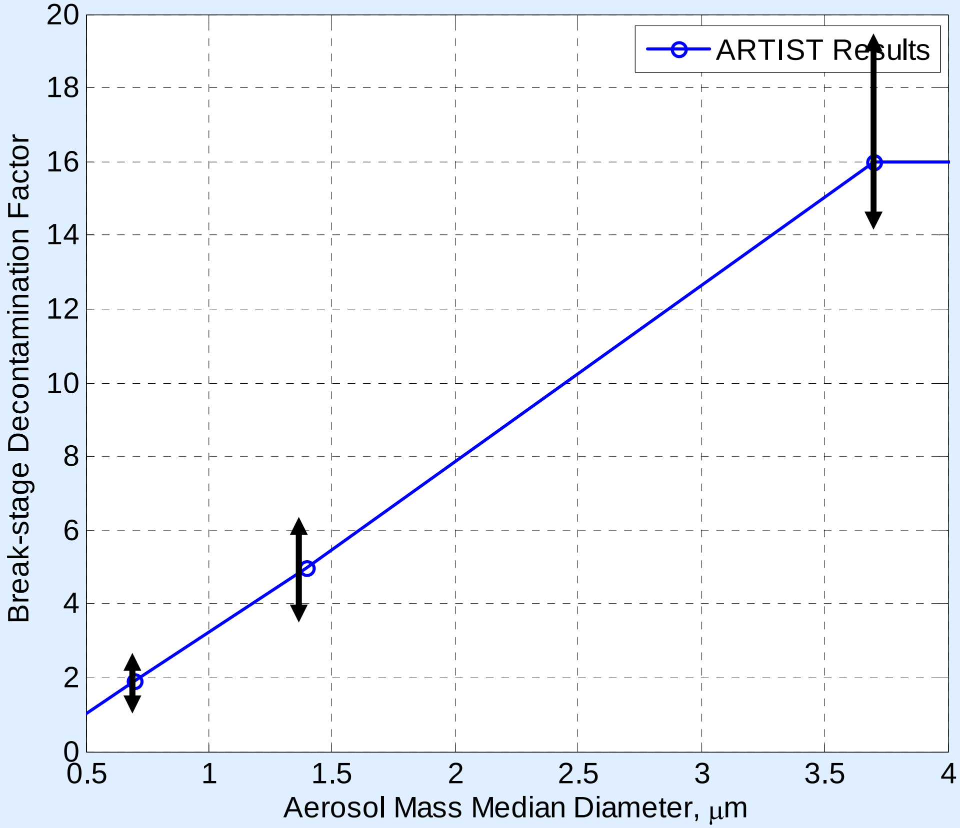


Vapor fraction data base generated from SOPHAEROS code runs



Particle size as measured in all Phébus tests with AgInCd

Lump Parameter Model Data Base (2:3)



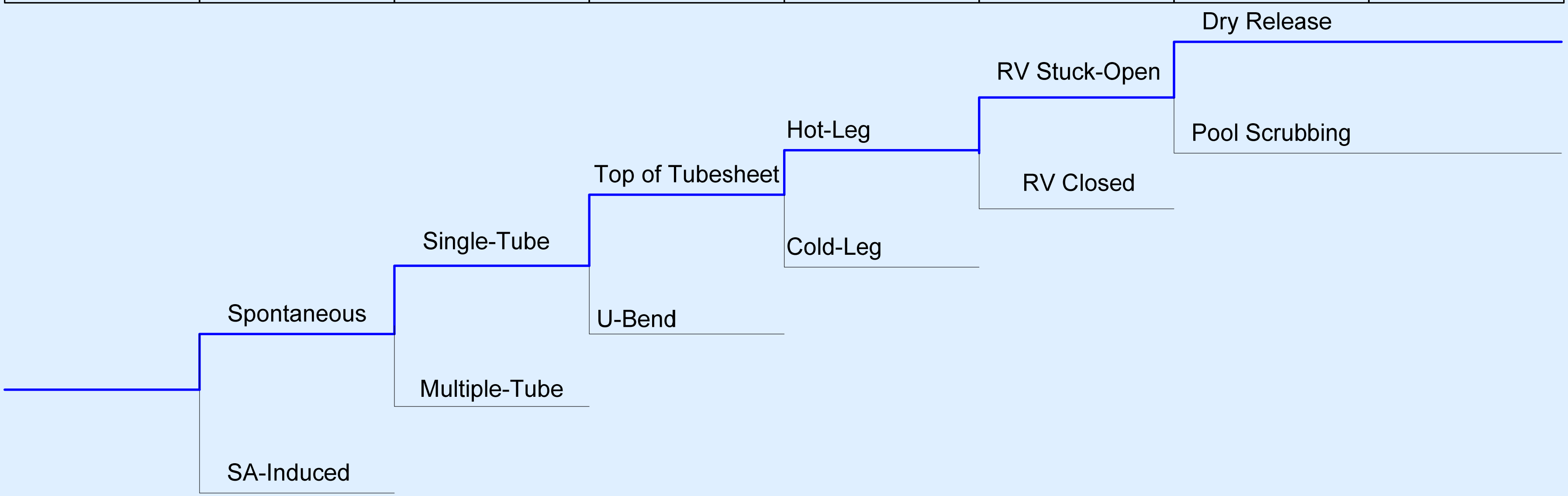
ARTIST Break Stage Particle Size Dependent DF

Lump Parameter Model Data Base (3:3)

Retention Stage	DF	Error Factor	Source
Reactor vessel	1.2 (I), 1.8 (Cs)	1.06 (I), 1.04 (Cs)	Phébus
Primary circuit	1.1 (I), 1.2 (Cs)	1.09 (I), 1.2 (Cs)	Expert judgment
In-tube retention	Time variant	1.5	ARTIST
Break stage	Aerosol-size variant	1.5	ARTIST
Far-field stage I-VII	1.05	1.21	ARTIST
Top of shroud	1.20	1.09	Expert judgment
Separator	1.20	1.06	ARTIST
Recirculation	Model	Model	MELCOR, SR5
Downcomer	1.10	1.05	Expert judgment
Intra-volume	1.10	1.07	Expert judgment
Dryer	1.20	1.09	ARTIST
Dome	1.10	1.05	ARTIST

Multiple SA Code Analyses for Model Uncertainties for the same APET Branch

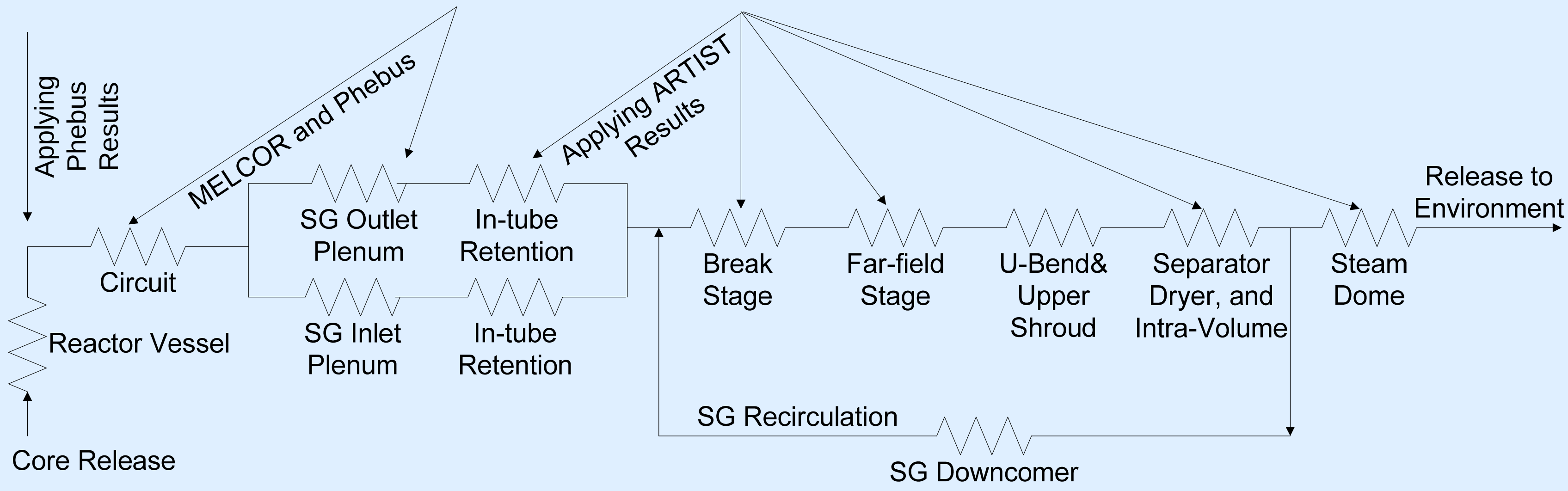
SGTR Frequency	**Category**	**Rupture Size**	**Rupture Location**		**Accident Mitigation**		
	No SA-Induced SGTR	No Multiple-Tubes	No U-Bend Rupture	No Cold-Leg Rupture	No Reclosed RV	No Refilled SG	FP Release Fraction
Node A	B	C	D	E	F	G	H



SGTR Accident Progression Event Tree

Retention Stages from Core to SG Steam Outlet

- *For each APET sequence, consider a series of retention stages in the fission product release path from the core to the environment*
- *For retention stages of the SG, the lumped parameter model is used*



Multiple SA Code Results: An example

Temperature predictions from MELCOR and SCDAP/RELAP5

Running multiple cases to estimate the temperature distribution

- SGTR sequence from NPP – Beznau PSA L2
  - SRV stuck-open at the affected SG
  - SRV opened manually at the intact SG at core exit temperature>923K
- Calculation stops at lower head failure

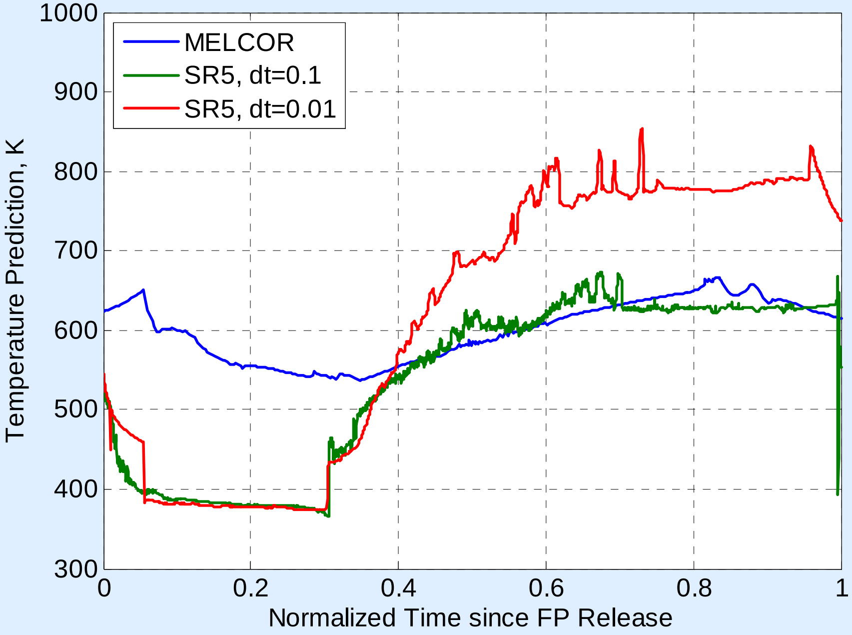
(a) MELCOR

(b) SR5, dt=0.1

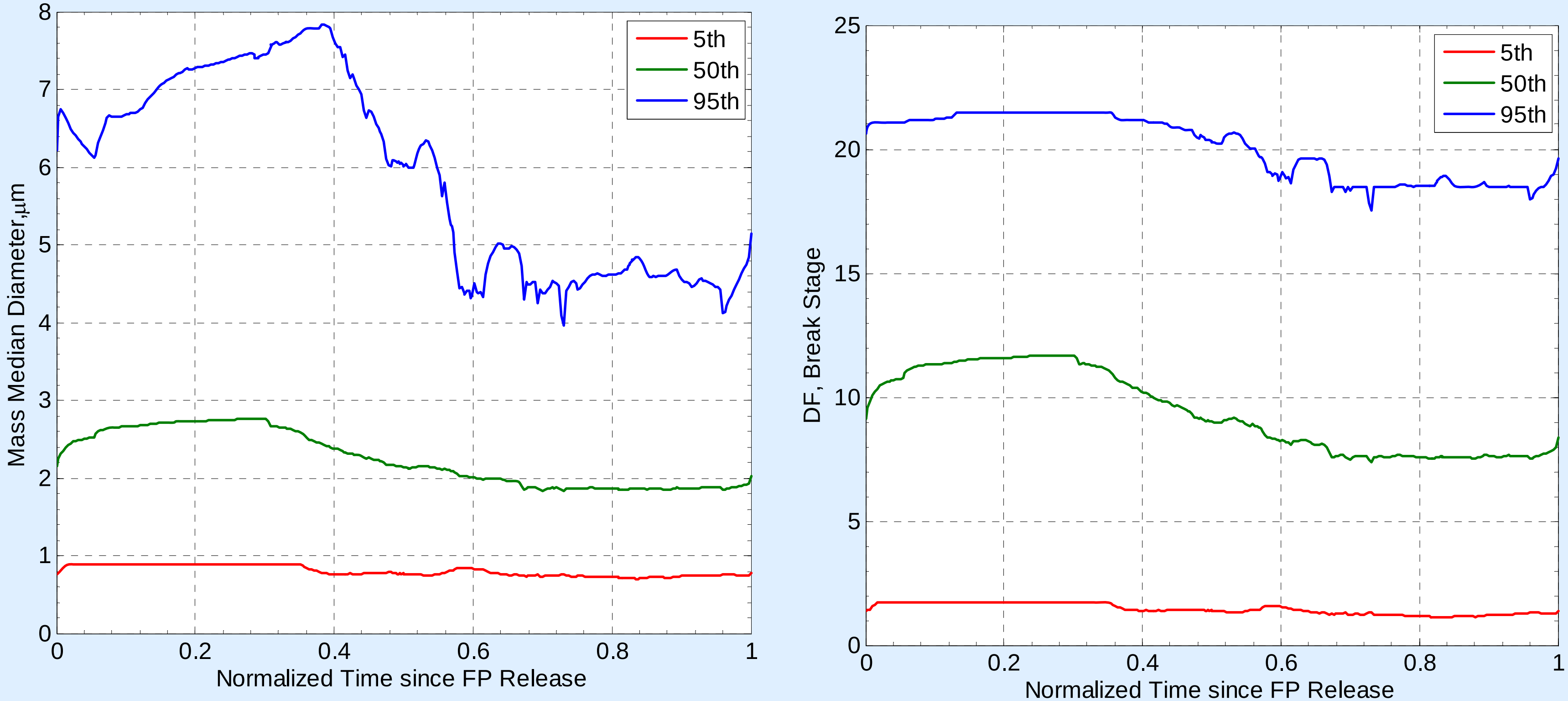
SCDAP/RELAP5, max. time step=0.1s

(c) SR5, dt=0.01

SCDAP/RELAP5, max. time step=0.01s

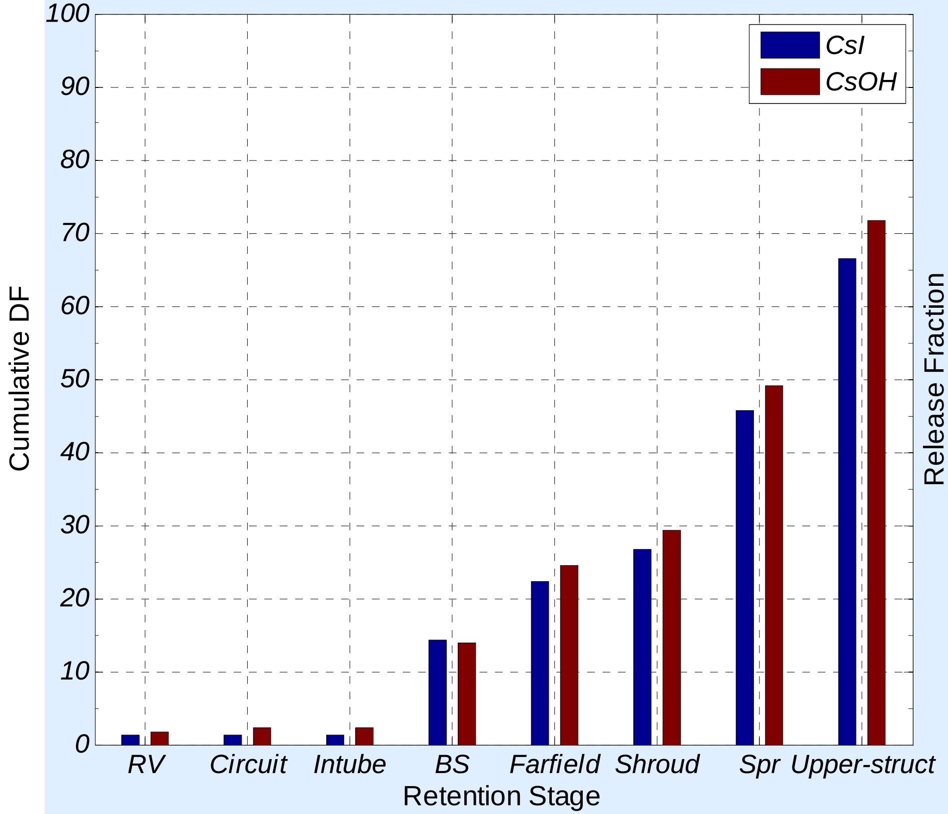


Monte-Carlo Simulation: Examples of 90% confidence interval of Particle Diameter and Decontamination Factor in Break Stage

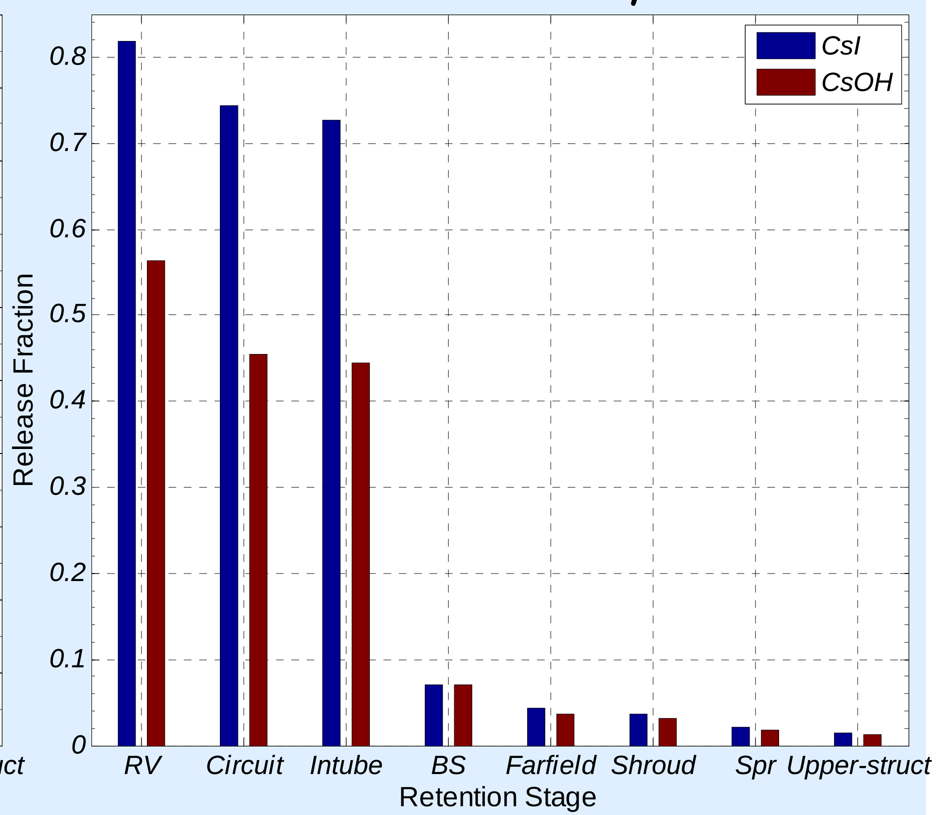


Cumulative Retention in/Release Fraction from Individual Retention Stages for Specific SGTR Sequence

Stage-wise mean decontamination factor

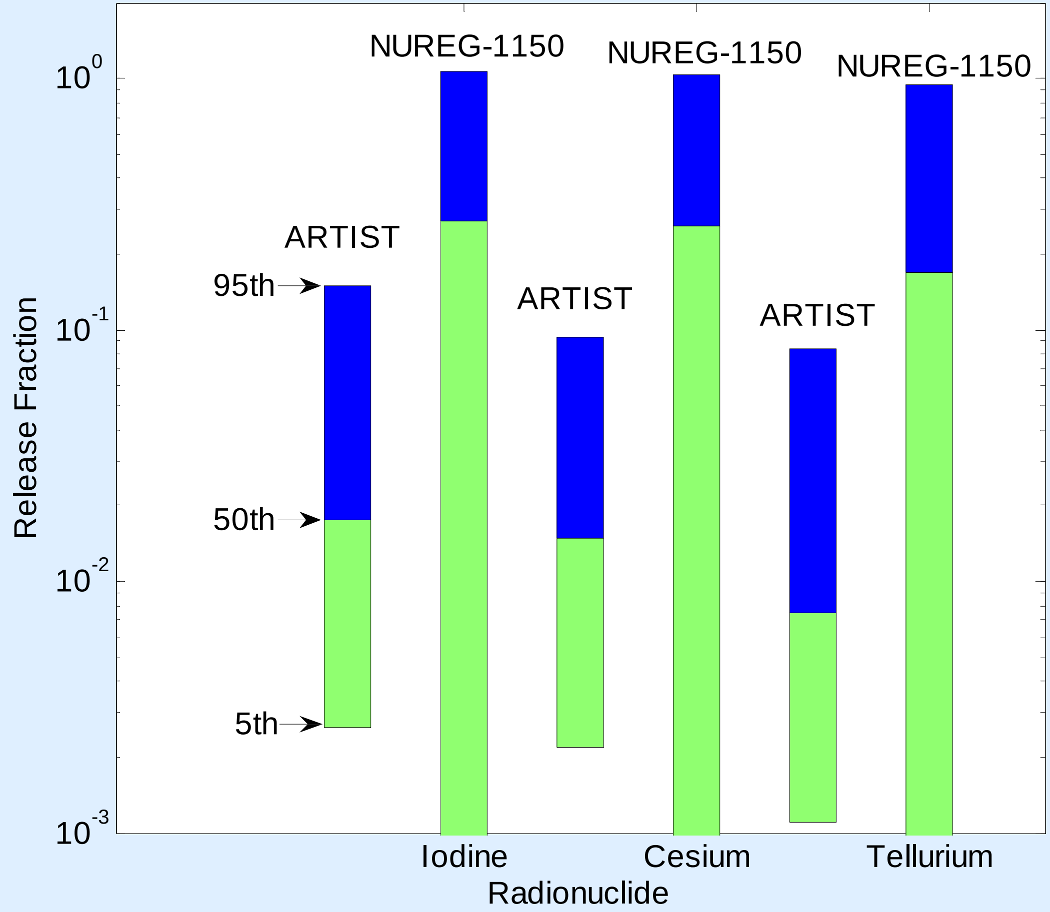


Mean release fraction of core inventory



Preliminary results

90% confidence interval of release fractions, comparing to NUREG-1150



Assessment of Methodology (1:2)

- MELCOR 1.8.6 runs for point estimates of source term
  - use of ARTIST data through „filter function"
  - Superimposing user input „aerosol size" to overwrite MAEROS

- Three MELCOR runs
  - Standard MELCOR 1.8.6 for the same SGTR sequence
  - MELCOR 1.8.6 with ARTIST DFs
  - MELCOR 1.8.6 with ARTIST DFs + PHÉBUS inferred temperature dependent particle size

With MELCOR default vapor and aerosol physics

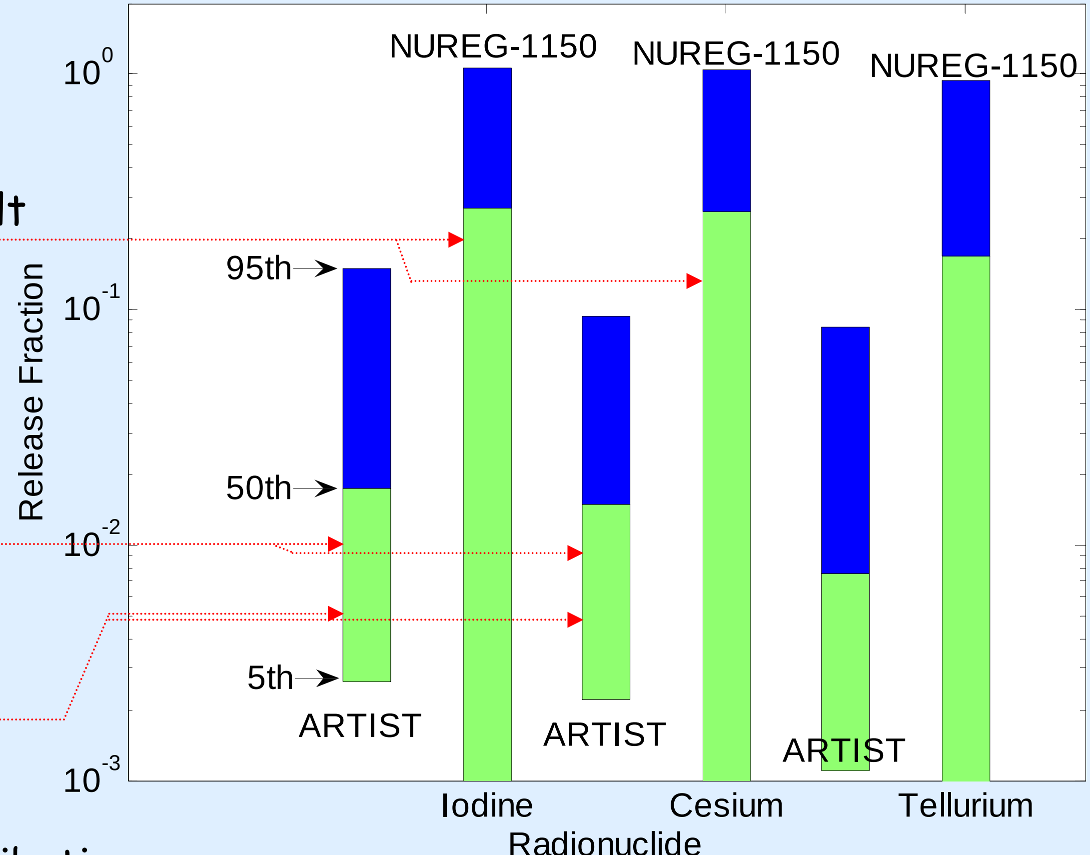
Assessment of Methodology (1:2)

Comparison of PSI-Risk Model Results to MELCOR Point Value Estimates

Point estimate of MELCOR default

Point estimate of MELCOR using MAEROS with incorporation of ARTIST DFs

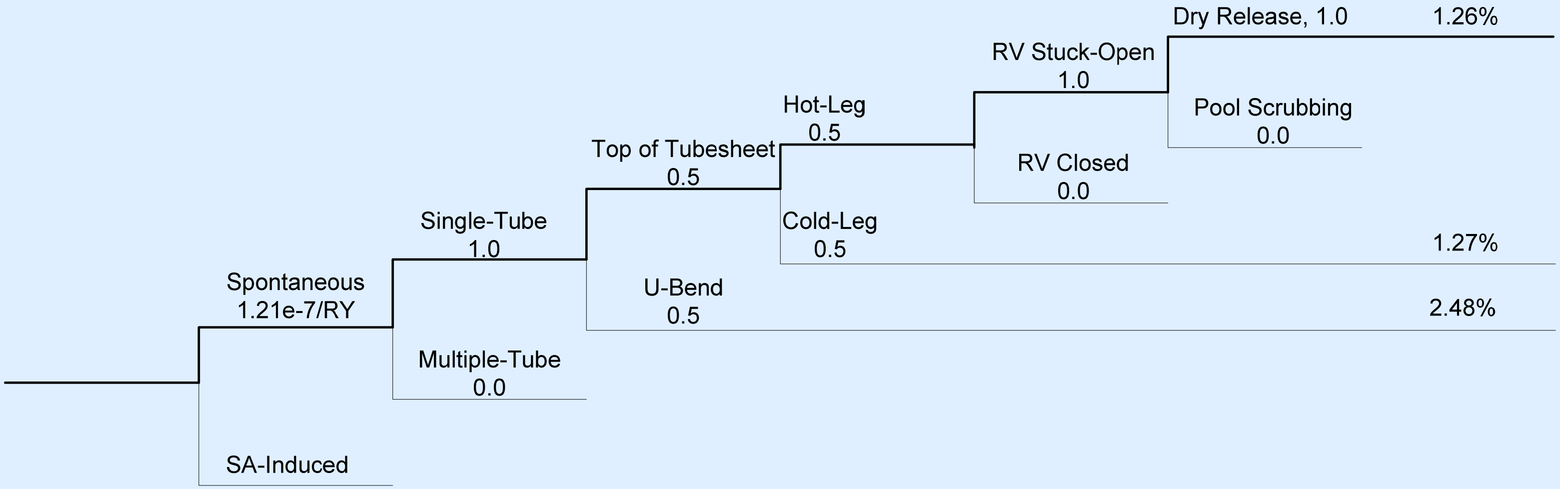
Point estimate of MELCOR using PHEBUS[1] with incorporation of ARTIST DFs



[1]superimposing particle size distribution

APET: branching fractions

SGTR Frequency	Category	Rupture Size	Rupture Location		Accident Mitigation		
	No SA-Induced SGTR	No Multiple-Tubes	No U-Bend Rupture	No Cold-Leg Rupture	No Reclosed RV	No Refilled SG	FP Release Fraction
Node A	B	C	D	E	F	G	H



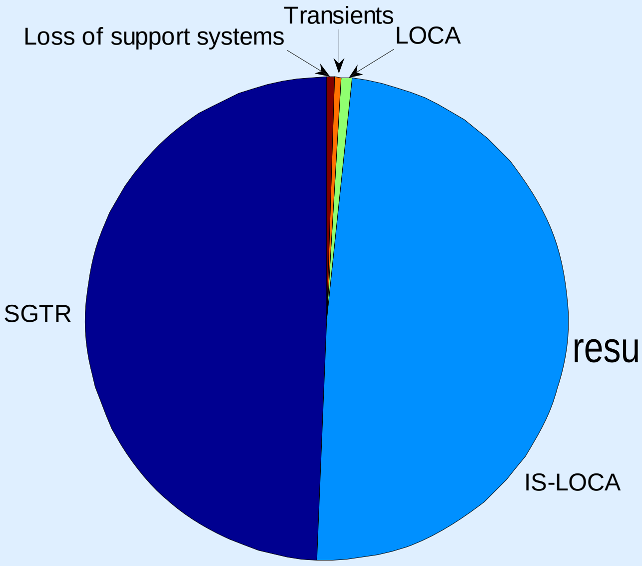
Case presented

Other cases not presented

SGTR Accident Progression Event Tree

Preliminary Risk Profile of NPP-Beznau Spontaneous SGTR

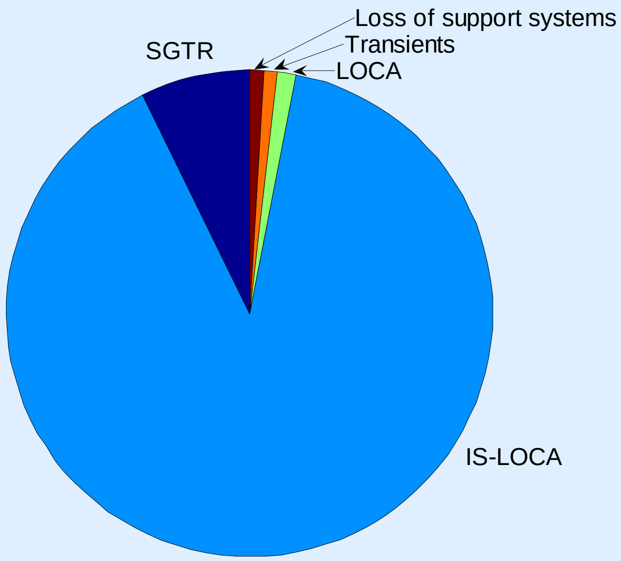
Comparison of the SGTR (without SG Reflooding) Risk significance to other internal initiating events for the Beznau NPP



NPP Beznau: PSA L2 BERA: 2002

SGTR risk reduction

resulting from using the ARTIST data



Conclusions

- Methodology consistent with Point values from MELCOR
- Further development for inclusion of other dependencies and their uncertainties (e.g., DF (dp, C))
- Generic model requires user to input from plant specific SA analysis
- APET to be revised with plant specific information (frequencies, split fractions)

Final Remarks

- PSI data supported by additional data from CIEMAT (Spain) for break stage retention and from VTT (Finland) for in-tube deposition/resuspension, both at low flows
- CFD Simulations of flow[1] and particles[2] by CFD (FLUENT) by Ringhals, AVN[1], CIEMAT[1], JNES [1,2] and NRC[1,2] (Sandia)
- Model development for aerosol removal in flooded bundle (IRSN) and in break stage (CIEMAT)
- 4 PhDs (de-agglomeration, aerosol motion through DNS+LES, bubble hydrodynamics in bundle) at PSI
- 3 PhDs (removal in far field, break stage hydrodynamics, aerosols) at UPM and CIEMAT
- 1 PhD (particle motion in SG pipe) at Sandia
- 1 masters (flow fields by CFD in Separator) at AVN
  - ➢ with involvement of 7 Universities

PSI thanks for all supporting and participating organizations in ARTIST

Phases V and VI: Flooded Bundle and Droplet Retention in Separator & Dryer

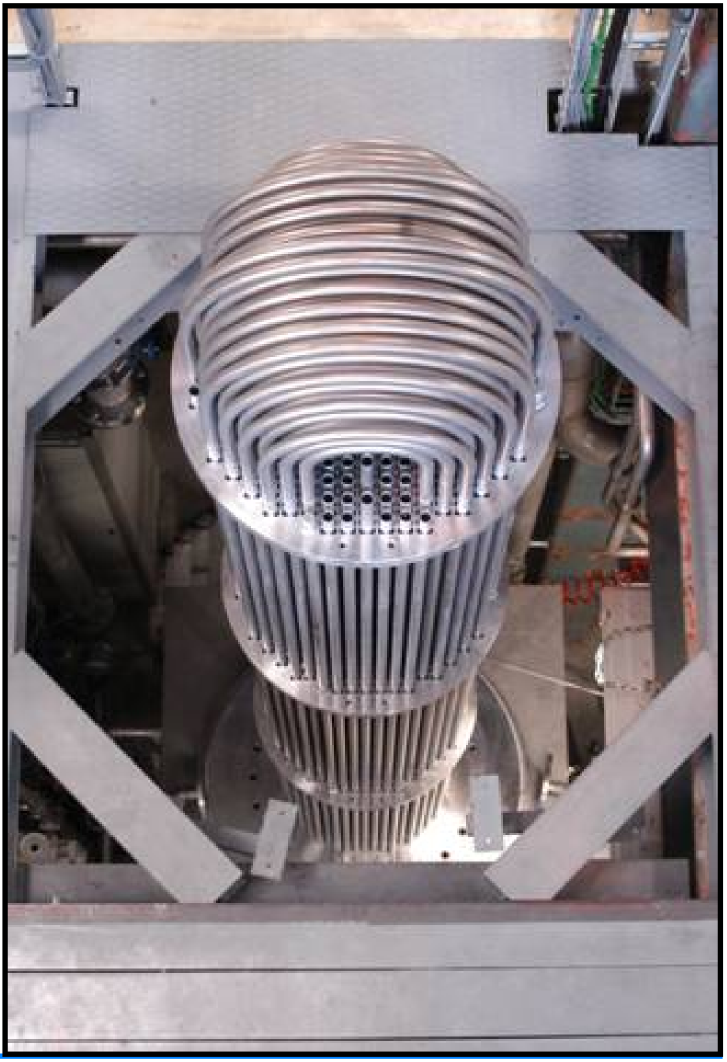
NRC does not participate in ARTIST Project Phases V and VI, however, the following information is introduced for those in ACRS who have interest in the Aerosol Scrubbing in Bundle Environment from High Jet Flows and Dissolved Activity (Iodine, mostly) Retention/Release by Droplets during the initiation of aSGTR event

Phase V: retention in the flooded bundle (1:2)

- **2 tests (+3 EU-SGTR)**
- **Decontamination Factor**

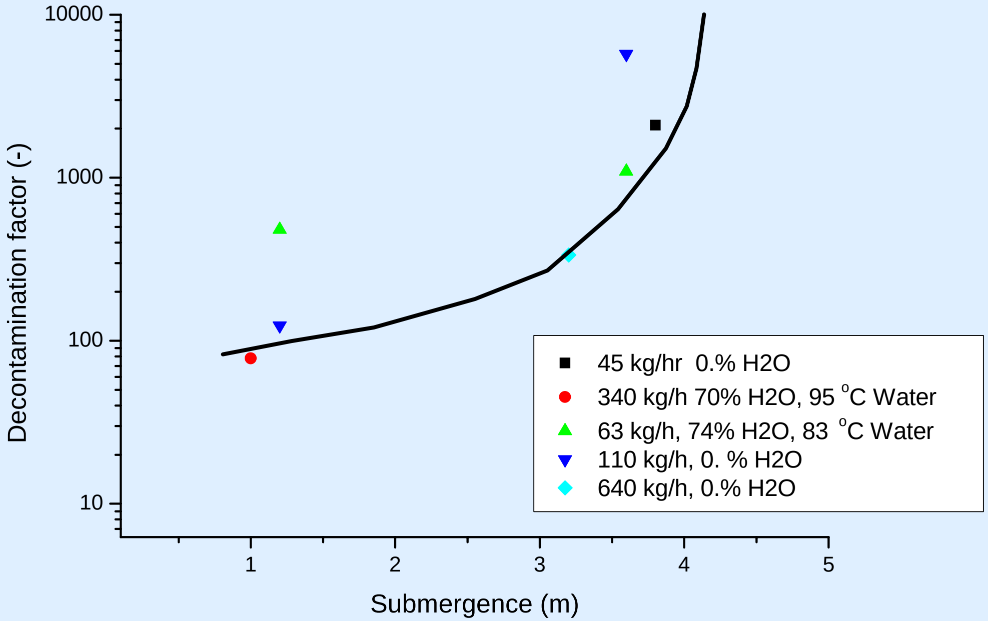
o Determined for relatively large submersion

DF	flow rate	submersion
2 100	45 kg/h	3.8 m
335	640 kg/h	3.2 m



Phase V: retention in the flooded bundle (2:2)

- Very high DF due to bundle-hydrodynamic interactions, especially at the break; models not able to reproduce DF
- Aerosol removal in hot pools without bundle: ~ **DF 20** (PSI – POSEIDON, 1991- 1996)



tests	Main features	Submergence m	Experimental DF	IRSN Model DF
A02	Steam, hot, medium flow rate	1.3	**50-100**	**352**
A03	NC, cold, low flow rate	1.2 2.3 3.6	**124** **1251** **5739**	**37** **54** **60**
E04	NC, cold, low flow rate	3.80	**2097**	**46**
E06	NC, cold, high flow rate	3.20	**271-465**	**67**

Iodine Source Term during Steam Generator Tube Rupture Initiated Design Basis Accidents: Introduction

- Spontaneous or initiated Steam Generator Tube Rupture

  => activity release until the operators can reduce the RCS pressure to the secondary side level

  => activity release at least 30-40 minutes (so-called "grace period")

- SGTR event is a design basis event

- The amount of activity release controlled by:

  a) amount of dissolved activity in the primary system (leaking rods, iodine spiking (reactor trip) and pressure change)

  b) the submergence of the leak; single or multiple tube ruptures; total break flow

  c) pH and iodine chemistry in the secondary side

  d) iodine mass transfer from the boiling pool

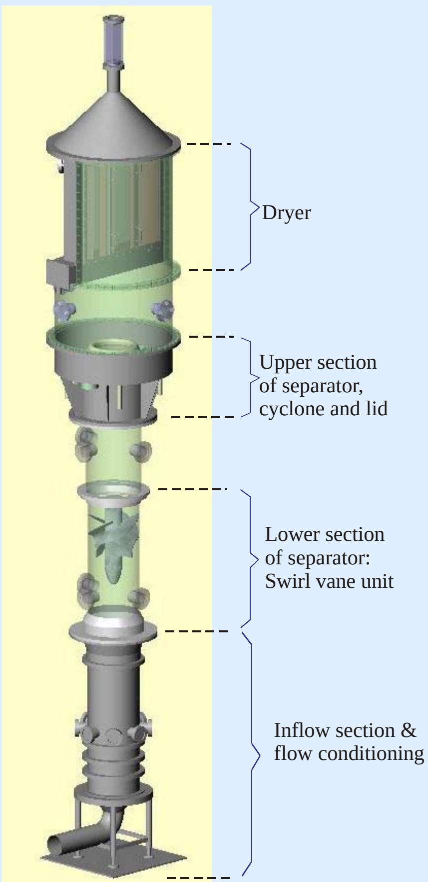
  e) The break at the tube bend

  <= 80-85 % of primary water in droplet form as a result of flashing

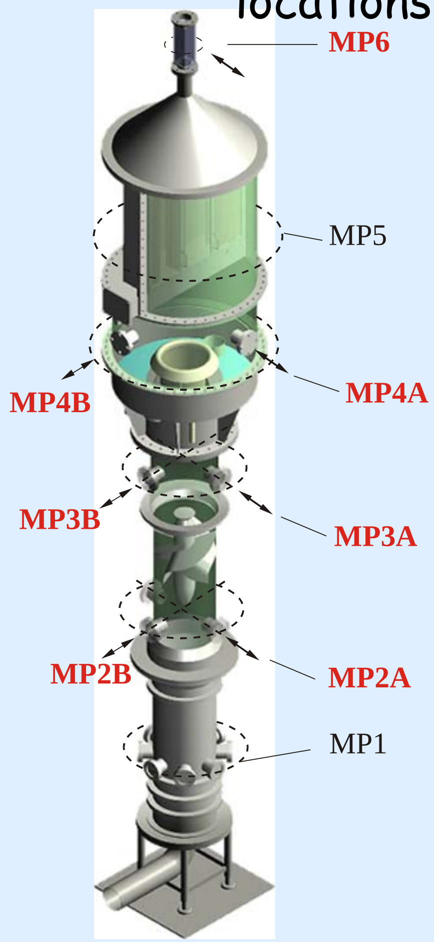
  => efficiency of separator and dryer to retain droplets

  ➔ ARTIST - Phase VI

Phase VI: Droplet retention in Separator and Dryer

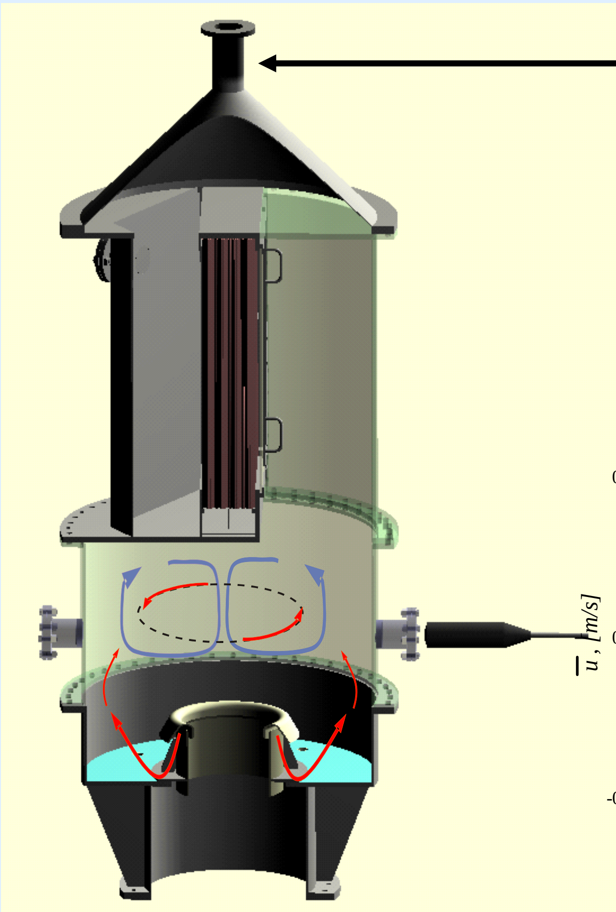


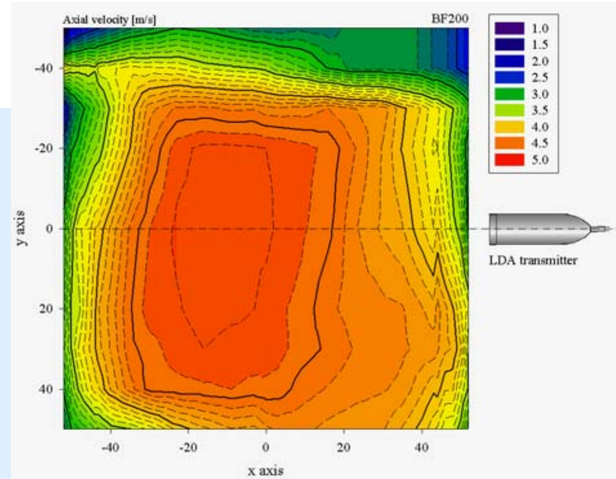
Measurement locations

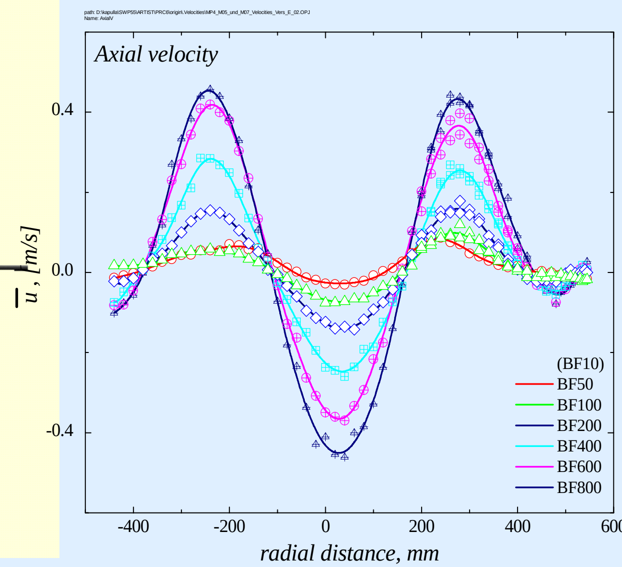


- Non-evaporating DEHS as droplet medium
- Spraying DEHS producing droplets
- Constant gas flow (10-800 kg/h)
- Known droplet inlet flux
- Known droplet size distribution at inlet (AMMD 10-50μm)
- **LDA, PDA, PIV**
- **Liquid Collection for DF**

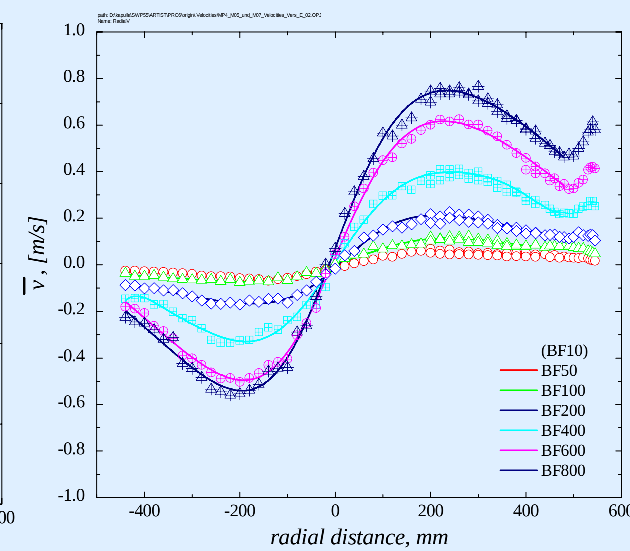
Flow velocity distribution





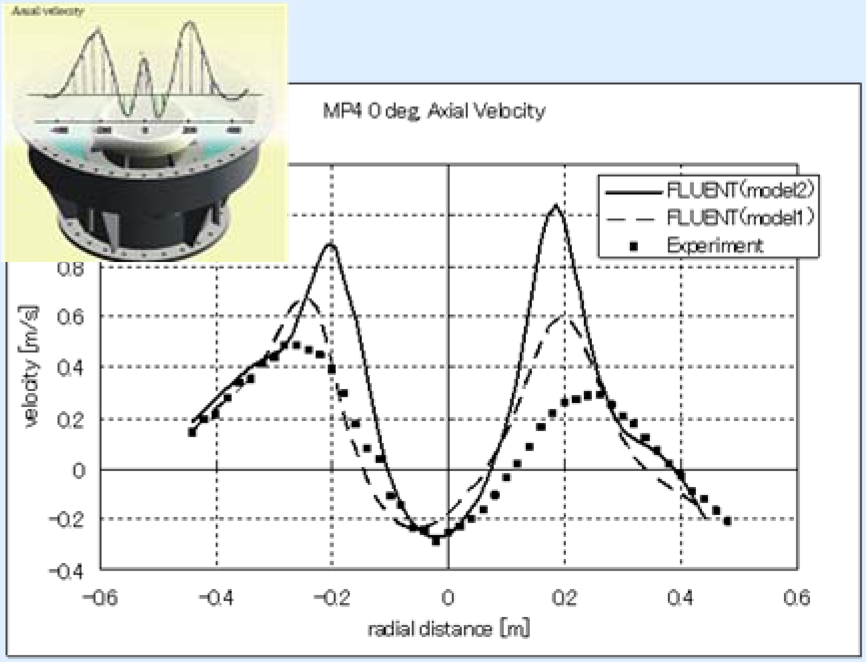


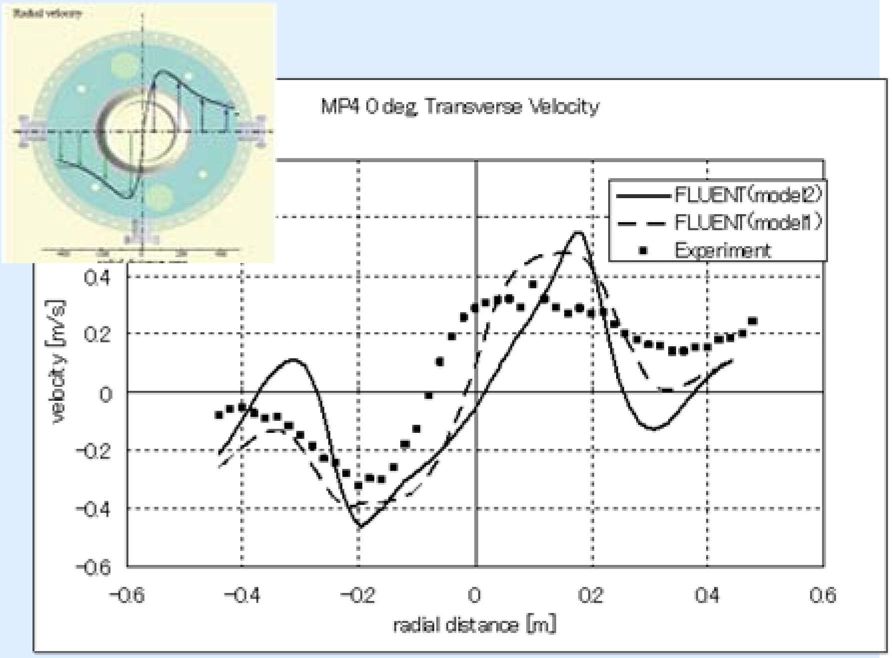
Mean axial velocity



Mean transverse velocity

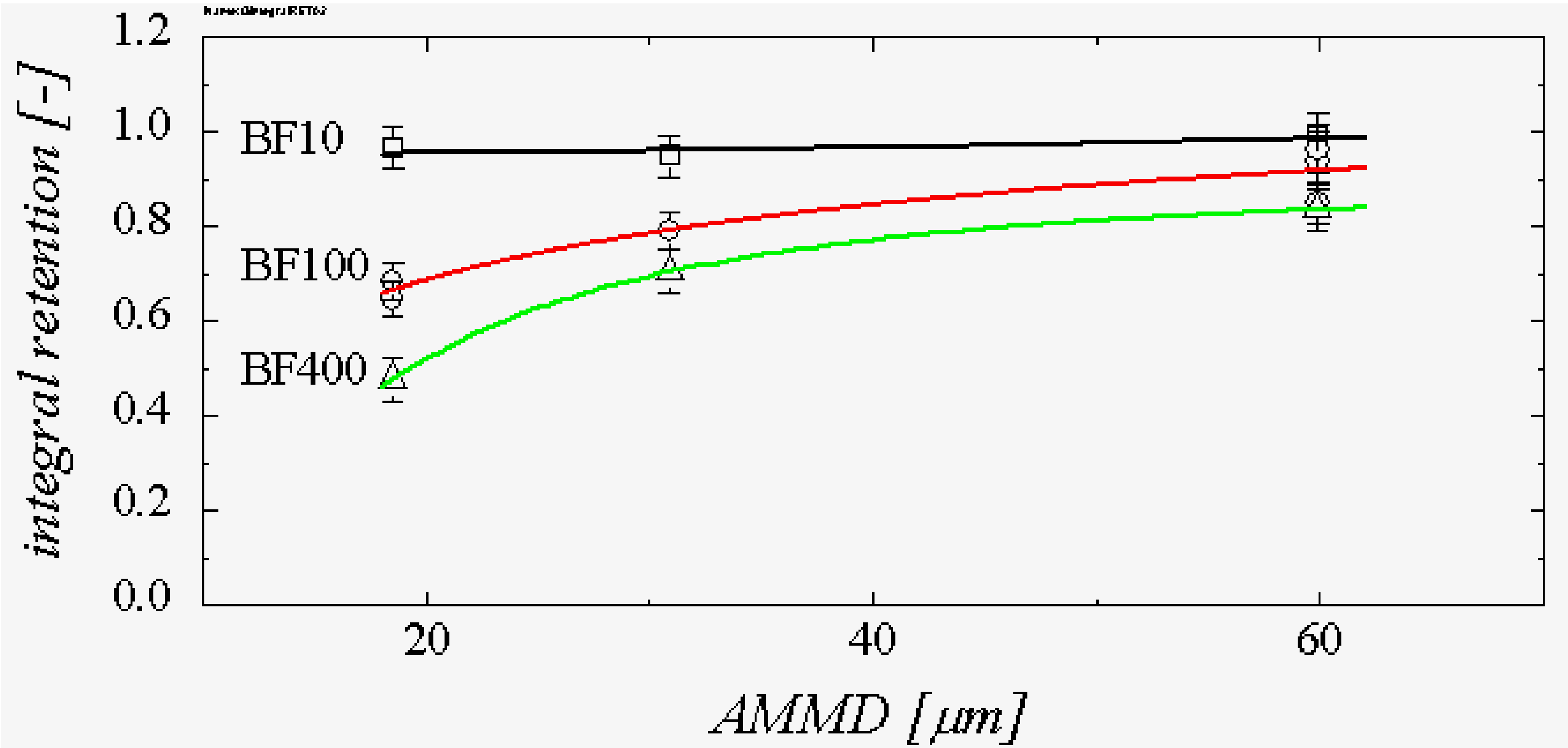
JNES FLUENT Simulations





- RSM turbulence model much better than $\kappa - \varepsilon$ model for rotating flow.
- Mesh resolution at lid controls quality of velocity profile above Lid plane
- Importance of adequate resolution of wall boundary layer

Integral retention across the separator & dryer



Particle Decontamination by FLUENT with PSI discrete-particle tracking model (JNES)

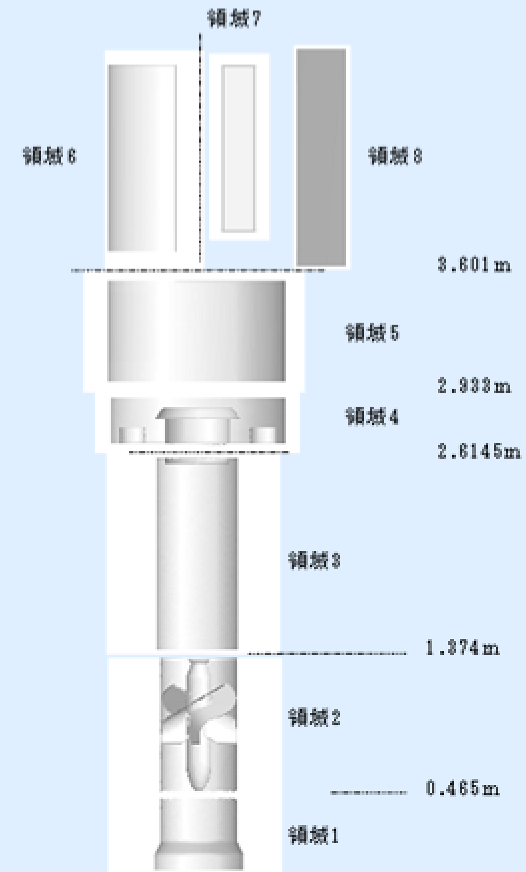
3D flow simulation at whole separator & dryer

Velocity field → **3D aerosol tracking at separator & dryer**

Inlet velocities → **2D flow simulation at chevron vane** → **2D aerosol tracking at chevron vane**

Integration with velocities

→ **Aerosol deposition at whole separator & dryer**



Particle Decontamination by FLUENT with PSI discrete-particle tracking model (JNES)

DF(300kg/h)

	1 μm	3 μm	10 μm
Separator	1.25	1.32	1.35
Dryer	1.09	1.14	1.25
Total	1.36	1.51	1.68

- Capturing hydrodynamic behavior is crucial prerequisite for aerosol behavior
- PSI discrete-particle tracing considers particle turbulence based on DNS simulations
- JNES predicted Overall retention is in agreement with Phase IV test results

Standardization of Operational Event Risk Assessments

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Division of Risk Analysis
Office of Nuclear Regulatory Research

June 4, 2008

United States Nuclear Regulatory Commission
Protecting People and the Environment

Presentation Outline

- Purpose
- Background
- Concepts of operational event risk assessment
- Implementation of standardization tasks
- Ongoing and future work
- Conclusions

Purpose

- To describe the activities undertaken by RES and NRR to standardize the risk assessment of operational events.
- To provide background to findings in draft NUREG-1635, Vol. 8, “Review and Evaluation of the Nuclear Regulatory Commission Safety Research Program,” Chapter 10, “Operational Experience.”
- To summarize the status of completed and ongoing RES activities in support of the standardization of operational event risk assessments.

Background

- In 2004, the staff initiated the Risk Assessment Standardization Project (RASP) as a collaborative effort between NRR, RES, and regional Senior Reactor Analysts (SRAs).
- The purpose of RASP is to provide consistent methods for risk analysis of conditions in the ASP and SDP Phase 3 programs and the risk analysis of events/conditions in the ASP and MD 8.3 programs, while recognizing differences in purpose among the programs.

Risk Assessment of Operational Events at NRC

- ***Significance Determination Process (SDP):*** Risk analysis of inspection findings (e.g., conditions with performance deficiencies) to determine the safety significance of inspection findings. (Regions, NRR)
- ***NRC Incident Investigation Program (MD 8.3):*** Risk analysis of initiating events and conditions to determine the appropriate level of reactive inspection in response to a significant event. (Regions, NRR)
- ***Accident Sequence Precursor (ASP) Program:*** Risk analysis of initiating events and conditions to identify significant precursors, adverse trends, and insights. (RES)

Event Risk Assessment – Introduction

- The aim of event risk assessment is to identify what else could have happened in an incident, which did not necessarily happen during the incident, and that would lead to core damage.
- The event risk assessment is *future-oriented*
  – What is probability that a similar event, occurring in the future, would lead to core damage?

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Event Risk Assessment – Basic Concepts

- The figures of merit are conditional core damage probability (CCDP) for initiating events and change in core damage probability (ΔCDP) for degraded conditions.
  - The CCDP given the event and the nominal or adjusted failure probabilities of the components and operator actions that did not fail, yields a measure of how close we came to core damage.
- The "failure memory concept"
  - All *failures* observed in the event are modeled as failures in the risk analysis:
    - Basic events representing failed components and operator actions are modeled as failed (e.g., with TRUE house events).
  - System and operator action *successes* receive a different treatment:
    - Basic events representing successes are ignored (i.e., successes are not set to FALSE house events).
    - These basic events remain at their nominal failure probability, or adjusted to represent complications observed during the event.

Standardization Approach

- Document methods and guides for event risk analysis
  - Internal event analysis
  - External event analysis, including internal fire and flood events
  - Low-power/shutdown (LP/SD) event analysis
  - Large early release frequency (LERF) calculation
- Improve SPAR model fidelity
  - Enhance Rev. 3 internal events SPAR models to better reflect the risk of the as-built, as-operated plant
  - SPAR models for external events, shutdown events, and LERF/Level 2
- Enhance analysis methods; provide technical support

User Need Tasks for RES

- Task 1: Develop guides for the analysis of internal events during power operations.
- Task 2: Develop new methods and guides for the analysis of the following events:
  - External events, including internal fire and flood
  - Internal events during low-power and shutdown (LP/SD) operations
  - Calculation of large early release frequency (LERF) for containment-related events
- Task 3: Make enhancements to SPAR models and SAPHIRE/GEM code
- Task 4: Provide ongoing technical support.

Tasks 1 & 2 – Guides for Event Risk Analysis

- RASP handbook (Rev. 1) issued January 2008 (publically available):
  - Volume 1, Internal Events (ML080070303)
  - Volume 2, External Events (ML080300179)
  - Volume 3, SPAR Model Reviews (ML080300182)
- Volumes 1 and 2 based on existing methods used in previous SDP and ASP analyses; Vol. 3 based in part on PRA Review Guide (NUREG/CR-3485) and PRA Standard (ASME RA-Sb-2005).
- Inspection Manual Chapter (IMC) 0609, "Significance Determination Process," references use of handbook.
- Internal reviews by NRC and contractor staffs; Rev. 0 of Vols. 1 and 2 been in trial use for 2 to 3 years.

Task 3 – SPAR Model Development

- Internal events models:
  - Detailed cut-set-level reviews against most licensee's PRAs
  - Updates to station blackout/loss of offsite power models
  - Updates to SPAR model parameters based on NUREG/CR-6928[1]
  - Updates to SPAR model QA plan for Rev. 3 SPAR models
  - Other enhancements based on staff and licensee feedback
- External events models: 15 integrated Rev. 3 SPAR models
- Shutdown events models: 5 integrated Rev. 3 SPAR models
- LERF/Level II models: 2 preliminary Level II SPAR models

1. NUREG/CR-6928, "Industry-Average Performance for Components and Initiating Events at U.S. Commercial Nuclear Power Plants," February 2007 (http://www.nrc.gov/reading-rm/doc-collections/nuregs/contract/cr6928/)

Task 3 – SAPHIRE and GEM

- A new version of SAPHIRE code being developed to meet requirements for:
  - New user interface for conducting SDP Phase 2 assessments
  - Improved user interface for conducting SDP Phase 3 and ASP analyses
  - Improved features and capabilities for SPAR model development and use (e.g., LERF modeling approach, support integrated models)
  - New modeling and calculation methods (e.g., common-cause failure analysis, phase mission time analysis)
- Beta testing and peer review to be performed during 2008 and 2009 to support release of SAPHIRE Version 8 by end of 2009.

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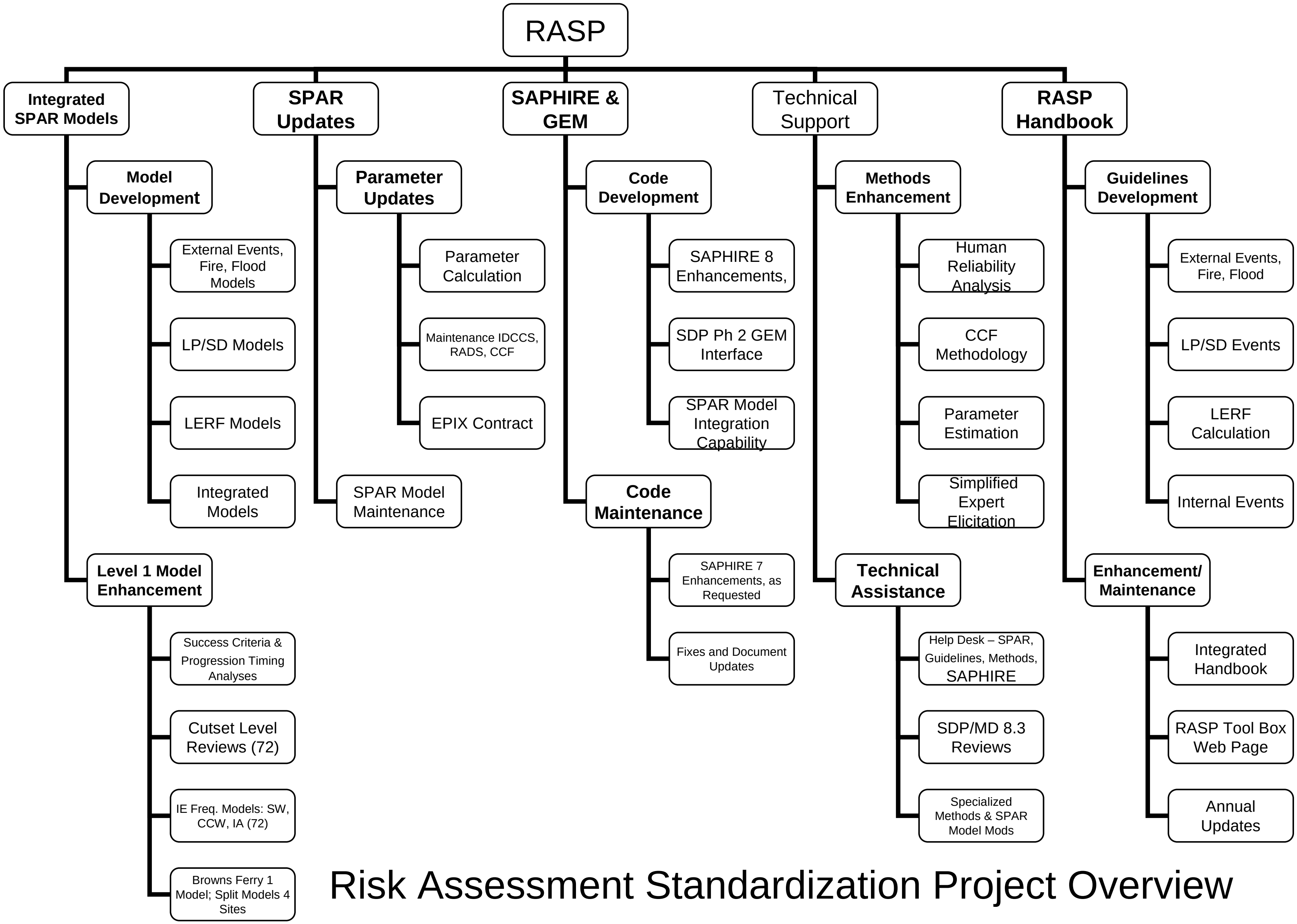
Task 4 – RES Technical Support

- Technical support provided to NRR analysts and Senior Reactor Analysts on methods, models, and analysis.
- Training provided at SRA counterpart meetings.
- Areas of support for event risk analysis include:
  - Common-cause failure modeling, parameter estimation
  - HRA and simplified expert elicitation applications
  - Uncertainty and sensitivity analyses
  - Internal event analysis guidance and SPAR model application
  - External event analysis guidance and SPAR model application
  - LP/SD event analysis guidance and SPAR model application
  - LERF calculation guidance
  - SAPHIRE/GEM code
- RASP Tool Box Web page developed for analysts.



Risk Assessment Standardization Project Overview

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Ongoing and Future Work – Methods and Guides

- RASP Handbook
  - Complete Volume 1: Guides for CCF modeling, parameter estimation and updates, uncertainty/sensitivity analysis, HRA, simplified expert elicitation, convolution analysis).
  - Revise Volumes 1, 2, and 3 based on user feedback.
  - Develop new volume for analysis of LP/SD events.
  - Develop new volume for LERF analysis of containment events.
- Technical support
  - Enhance methods
    - CCF methodology for event assessment (draft NUREG/CR)
    - HRA (based on results of international HRA benchmarking project)
    - Update pipe break LOCA frequencies (draft NUREG/CR)
  - Provide training support.
  - Provide on-call SDP analysis assistance.

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Ongoing and Future Work – SPAR Models

- Internal events SPAR model enhancements
  - Success criteria re-evaluation of key sequences based on thermal hydraulic analyses.
  - Work with industry to resolve key technical issues affecting SPAR and licensee PRA models (through NRC/EPRI Memorandum of Understanding).
  - Complete detailed cut-set-level reviews for 4 remaining models.
- Shutdown SPAR model development
  - Continue model development for shutdown events.
- SAPHIRE/GEM Version 8 development
  - Complete beta testing.

Conclusions

- RASP handbook widely in use by risk analysts and SRAs in the risk analysis of operational events in NRC programs:
  – Conditions in the ASP and SDP Phase 3 programs
  – Initiating events and conditions in the ASP and MD 8.3 programs
- ASP Program changed to eliminate duplicative analysis of SDP inspection findings.
- Communications and documented guidance improved consistency among analysts and enhanced knowledge transfer.
- Enhanced SPAR models better reflect the risk of the as-built, as-operated plant.

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Backup Slides

Past Briefings to the ACRS (Full and Subcommittees) on RES Risk Activities

- SPAR model development (10/10/2003)
  - Internal events (9/9/2005, 9/15/2005, 11/17/2005)
  - External events, including internal fire and flooding (11/18/2005)
  - shutdown event (11/11/2002, 10/10/2003, 11/18/2005)
  - Large early release frequency (LERF) (11/18/2005)
- SAPHIRE development (1/25/2002, 10/10/2003)
- Risk methods and databases
  - SPAR-H human reliability analysis method (10/09/2003, 12/15/2005, 3/22/2007)
  - Common-cause failure method, RADS/EPIX (12/15/1999, 04/6/2000)
  - Uncertainty (10/10/2003, 11/16/2004, 12/19/2007)
- Accident Sequence Precursor (ASP) Program (12/15/1999, 3/10/2006)

NRR User Need Requests

- "User Need Request for Support in the Development of Standard Procedures and Methods for Risk Assessments of Inspection Findings and Reactor Incidents," J. Dyer Memo to A. Thadani, February 17, 2004 (NRR-2004-005)
  - Task 1: Guides for risk analysis of internal events
  - Task 2: Guides for risk analysis of external events, LP/SD, and LERF
  - Task 3: SPAR model and SAPHIRE/GEM enhancements
  - Task 4: Technical support (methods, models, SDP analyses, handbook updates)
- "Supplement to User Need Request for Support in the Development of Standard Procedures and Methods for Risk Assessments of Inspection Findings and Reactor Incidents," Dyer Memo to B. Sheron, June 22, 2006 (NRR-2004-005)
  - Initiating event fault trees for cooling water systems (e.g. service water)
  - Revised models of success criteria for specific sequences using thermal hydraulic analyses

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NRC/EPRI MOU

- **SPAR model/industry PRA key technical issues:**
  - Support system initiating event analysis
  - Treatment of loss of offsite power
  - Standard guidance for event tree development
  - Treatment of injection following containment failure (BWRs)
  - Treatment of containment sump recirculation during small and very small loss of coolant accident
  - Human reliability analysis dependencies and recovery modeling issues
- **Other NRC/industry technical issues:**
  - Treatment of uncertainty in risk analyses
  - Aggregation of risk metrics
  - Human reliability analysis
  - Digital instrumentation & control risk methods
  - Advanced reactor PRA methods

RASP Tool Box Web Page

- http://www.internal.nrc.gov/RES/RASP/index.html *(Internal to NRC)*
- Provide web links to tools and access to references for Senior Reactor Analysts and risk analysts, e.g.,
  - RASP handbook volumes
  - Handbook references
  - SPAR models
  - SAPHIRE/GEM codes and manuals
  - Parameter estimation references (NUREG/CRs)
  - Databases and calculators (ASP, CCF, EPIX, LERs, RADS)
  - Plant information
  - PRA training manuals
  - PRA related references (NUREG/CRs)
- RASP Handbook kept current in the Tool Box.

Point-of-Contacts

- Accident Sequence Precursor Program: Chris Hunter (RES/DRA)
- RASP Handbooks
  - Vol. 1, Internal Event Analysis: See-Meng Wong (NRR/DRA), Don Marksberry (RES/DRA), Paul Bonnett (NRR/DIRS)
  - Vol. 2, External Event Analysis: Selim Sancaktar (RES/DRA)
  - Vol. 3, SPAR Model Reviews: Pete Appignani (RES/DRA)
- Risk Analysis Methods for Event Risk Analysis
  - CCF, parameter estimation, and RADS and CCF calculators: Jack Foster (RES/DRA)
  - SPAR-H HRA enhancements: Pete Appignani (RES/DRA)
  - Uncertainty/sensitivity analysis, simplified expert elicitation: Gary DeMoss (RES/DRA)
- Risk Databases (EPIX, LER, RADS, CCF): Bennett Brady (RES/DRA)
- SAPHIRE/SDP User Interface: Dan O'Neal (RES/DRA)
- Significant Determination Process: Paul Bonnett (NRR/DIRS)
- SPAR Models: Pete Appignani (RES/DRA)
- SPAR Model Success Criteria Re-Evaluation: Rick Sherry (RES/DRA)

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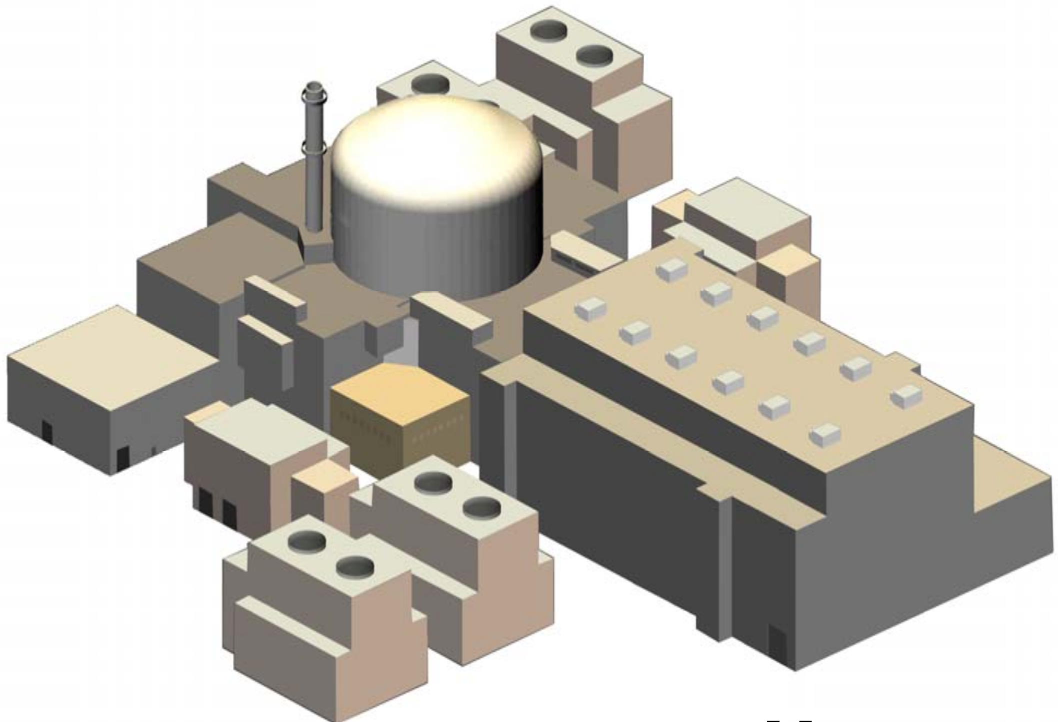
Abbreviations

• ASP	accident sequence precursor
• CCDP	conditional core damage probability
• CCF	common-cause failure
• EPIX	Equipment Performance and Information Exchange System
• EPRI	Electric Power Research Institute
• GEM	Graphical Evaluation Module
• HRA	human reliability analysis
• LER	Licensee Event Report
• LERF	large early release frequency
• LP/SD	Low-power/shutdown
• MD	Management Directive
• NRR	Office of Nuclear Reactor Regulation
• NRR/DIRS	Division of Inspection and Regional Support, NRR
• NRR/DRA	Division of Risk Assessment, NRR
• PRA	probabilistic risk assessment
• QA	quality assurance
• RADS	Reliability and Availability Data System
• RASP	Risk Assessment Standardization Project
• RES	Office of Nuclear Regulatory Research
• RES/DRA	Division of Risk Analysis, RES
• SAPHIRE	System Analysis Programs for Hands-on Integrated Reliability Evaluations
• SDP	Significance Determination Process
• SPAR	Standardized Plant Analysis Risk (model)
• SRA	Senior Reactor Analyst

ACRS Meeting:

U.S. EPR Design Overview

June 4, 2008



Introduction

Sandra M. Sloan
Manager, Regulatory Affairs
New Plants Deployment

Presentation Goal

To provide an overview of the U.S. EPR design, identifying the relationship to currently operating PWRs and different features, especially those of particular safety significance.

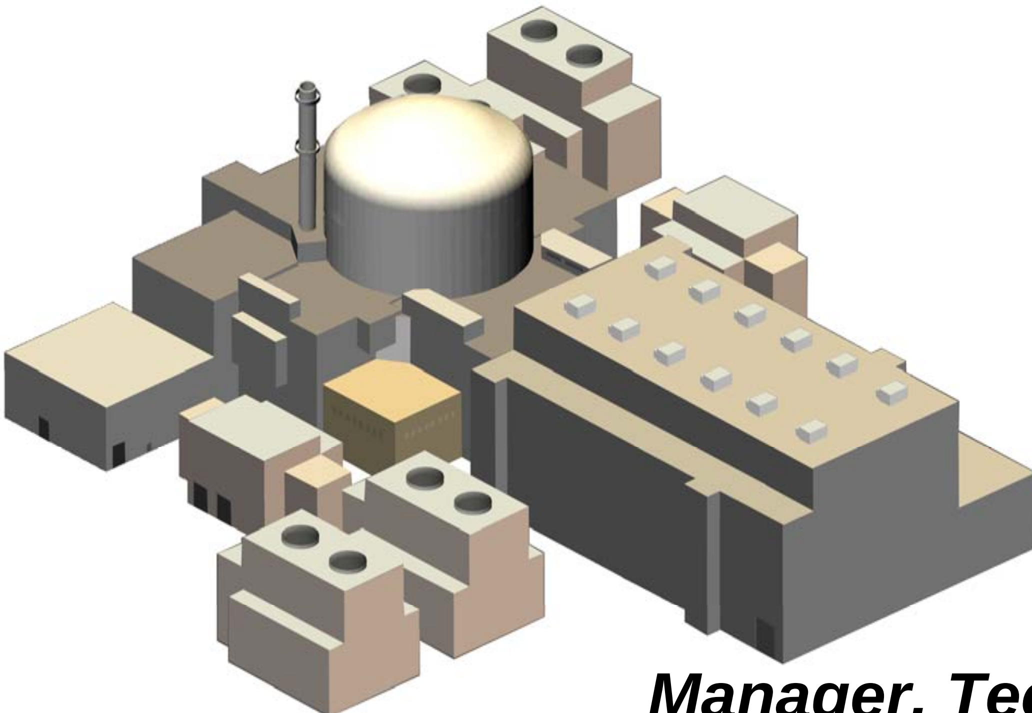
NRC-Identified Areas of Potential Schedule Uncertainty

- Post-accident containment mixing
- Seismic and dynamic qualification of mechanical and electrical equipment
- Unanticipated axial growth in M5™ guide tubes
- Four methodology-related topical reports
  - Realistic Large Break LOCA
  - Reactivity Insertion Accident
  - Incore Trip Setpoint and Transient Methodology
  - Fuel Assembly Mechanical Analysis
- Emergency Core Cooling System (ECCS) strainer downstream effects (GSI-191)

Presentation Topic Areas

- General design objectives
- Plant layout
- Safety systems
- Core design
- Instrumentation and controls
- Severe accident mitigation
- SGTR and SBLOCA mitigation
- Probabilistic risk assessment
- Operating experience feedback

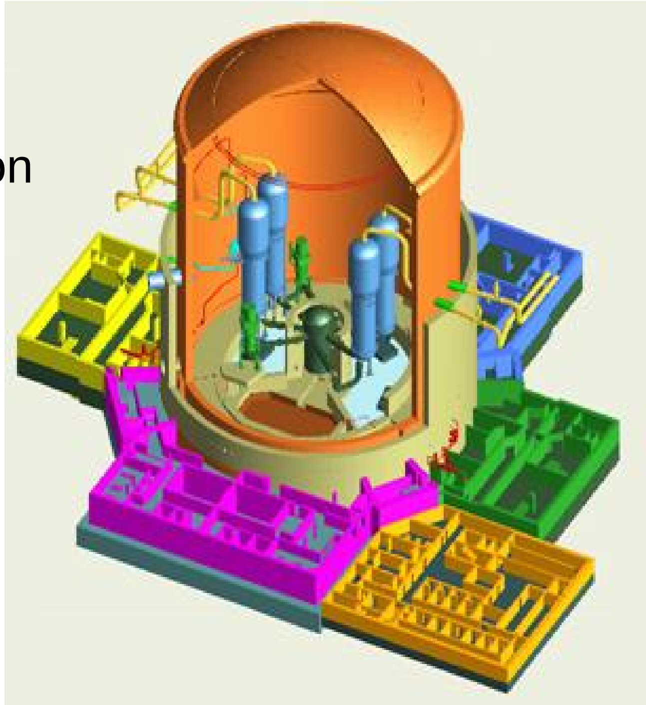
U.S. EPR Design Overview



Marty Parece
Chief Engineer
Manager, Technology Integration
New Plants Deployment

EPR Development Objectives

- **Evolutionary design** based on existing PWR construction experience, R&D, operating experience and "lessons learned"
- **Safer**
  - Reduce occupational exposure and LLW
  - Increase design margins
  - Increase redundancy & physical separation of safety trains
  - Reduce core damage frequency (CDF)
  - Accommodate severe accidents and external hazards with no long-term local population effect
- **Improved Operations**
  - Reduce generation cost by at least 10%
  - Simplify operations and maintenance
  - 60-year design life



Major Design Features

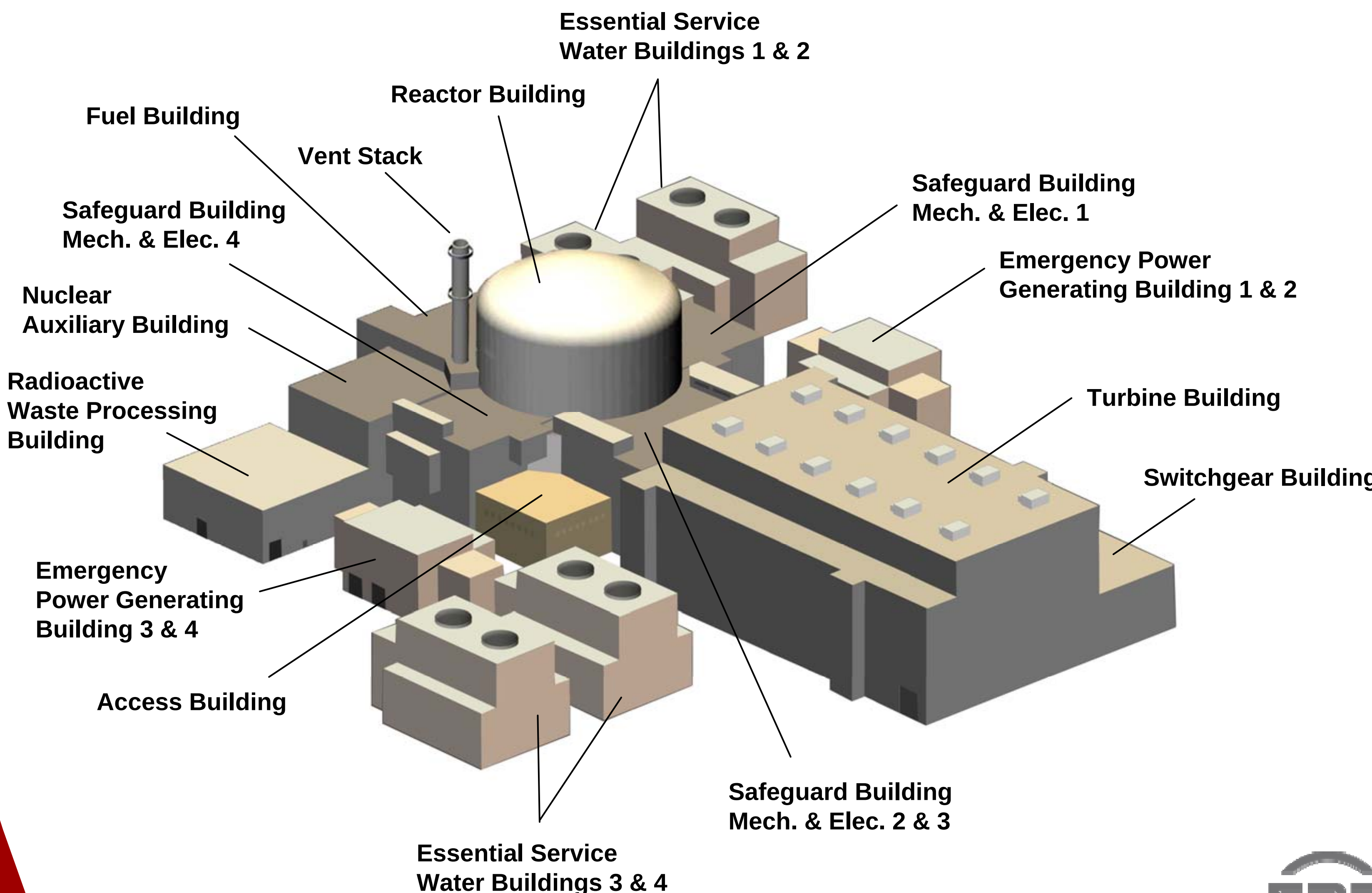
- Nuclear Island
  - *Proven Four-Loop RCS Design*
  - *Four-Train Safety Systems*
  - *Containment & Shield Bldg*
  - *In-Containment Borated Water Storage*
  - *Severe Accident Mitigation*
  - *Separate Safety Buildings*
  - *Advanced Control Room*

- Electrical
  - *Island Mode Operation*
  - *Four Emergency D/Gs*
  - *Two Smaller, Diverse SBO D/Gs*

- Site Characteristics
  - *Airplane Crash Protection (military and commercial)*
  - *Explosion Pressure Wave*

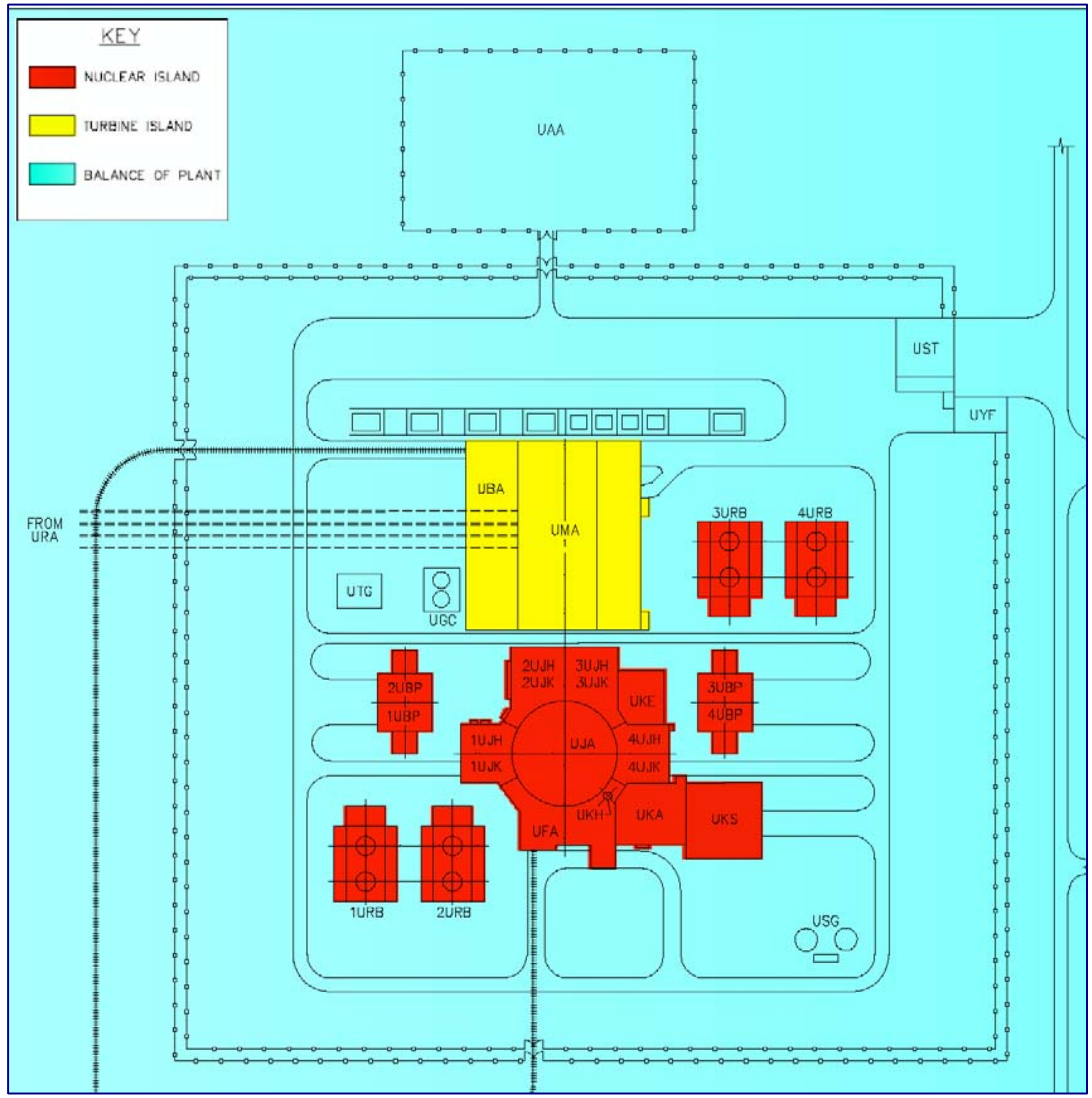
Reflects full benefit of operating experience and 21st century requirements.

The U.S. EPR


Essential Service
Water Buildings 1 & 2
Reactor Building
Fuel Building
Vent Stack
Safeguard Building
Mech. & Elec. 4
Nuclear
Auxiliary Building
Radioactive
Waste Processing
Building
Safeguard Building
Mech. & Elec. 1
Emergency Power
Generating Building 1 & 2
Turbine Building
Switchgear Building
Emergency
Power Generating
Building 3 & 4
Access Building
Safeguard Building
Mech. & Elec. 2 & 3
Essential Service
Water Buildings 3 & 4

U.S. EPR General Plant Layout

UAA	Switchyard
UBA	Switchgear Building
UBP	Emergency Power Generating Building
UFA	Fuel Building
UGC	Demineralized Water Storage Area
UJA	Reactor Building
UJH	Safeguard Building Mechanical
UJK	Safeguard Building Electrical
UKA	Nuclear Auxiliary Building
UKE	Access Building
UKH	Vent Stack
UKS	Radioactive Waste Processing Building
UMA	Turbine Building
URA	Cooling Tower Structure
URB	Essential Service Water Cooling Tower Structure
USG	Fire Protection Storage Tanks and Building
UST	Workshop & Warehouse Building
UTG	Central Gas Supply Building
UYF	Security Access Facility



Not To Scale

Radial Design
N+2 Approach

AREVA

The Four Train Concept

4 Independent Safety Trains

Arranged into 4 divisions

- Preventive maintenance during power operation
- Shorter outage time

Higher Availability

- Smaller components
- Fewer valves per train

Easier Maintainability

- Efficient hazard protection
- Reduced piping and components
- Optimized plant layout

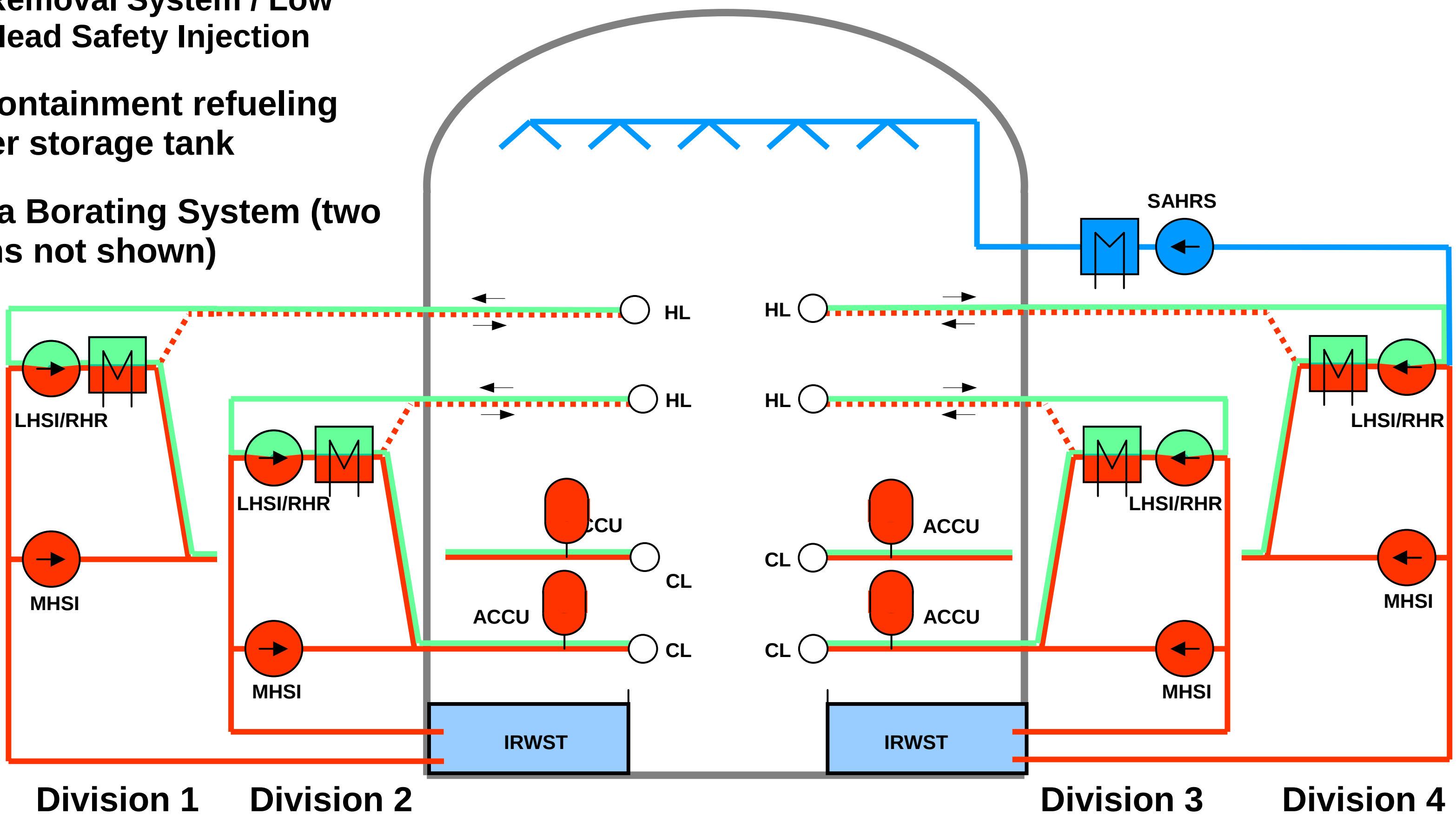
Lower Unit Cost

EPR

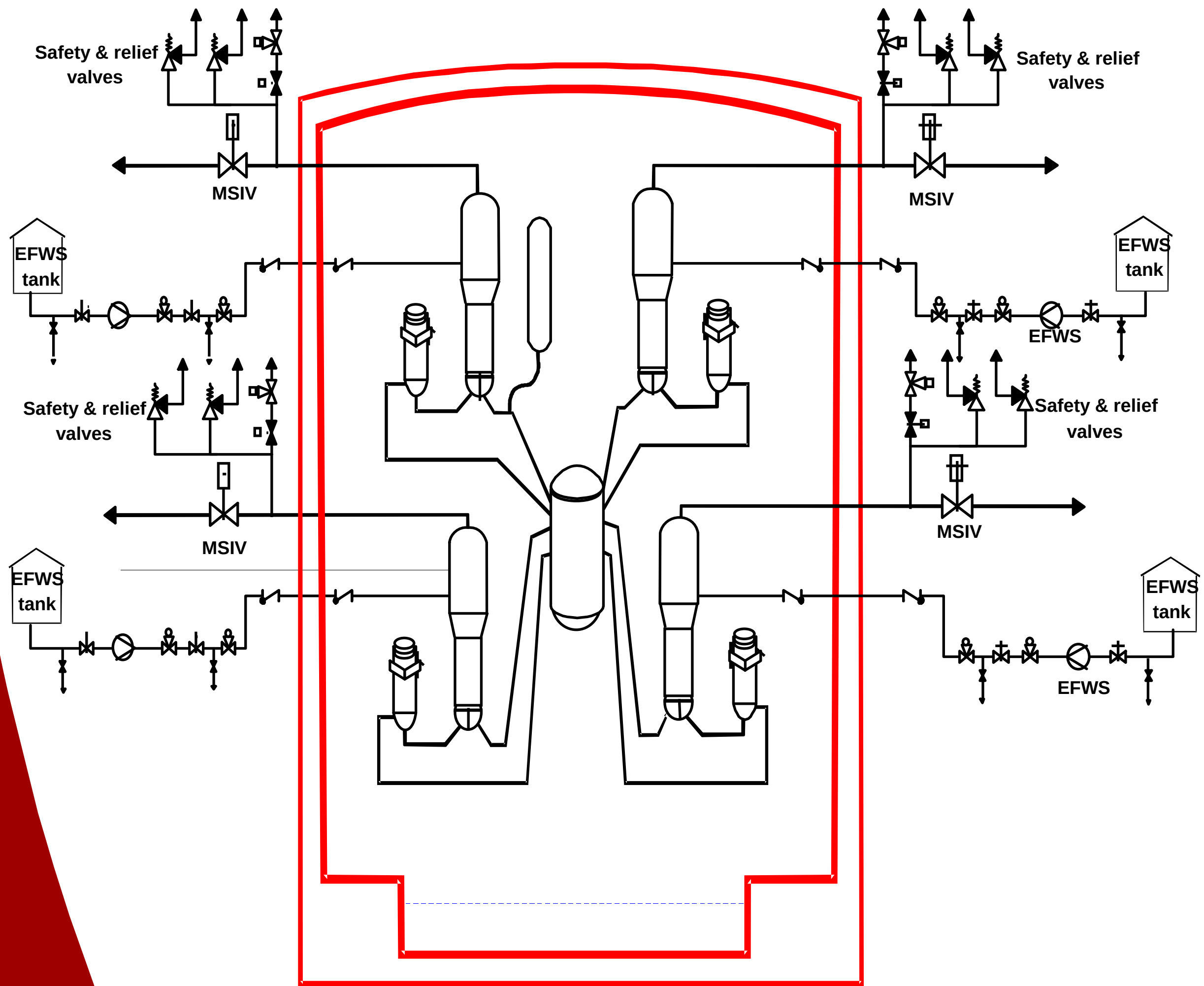
Main Safety Systems

- Four train Safety Injection System (SIS)
  - Medium head SI pumps
  - Combined Residual Heat Removal System / Low Head Safety Injection
- In-Containment refueling water storage tank
- Extra Borating System (two trains not shown)

- Non-safety containment spray for severe accident



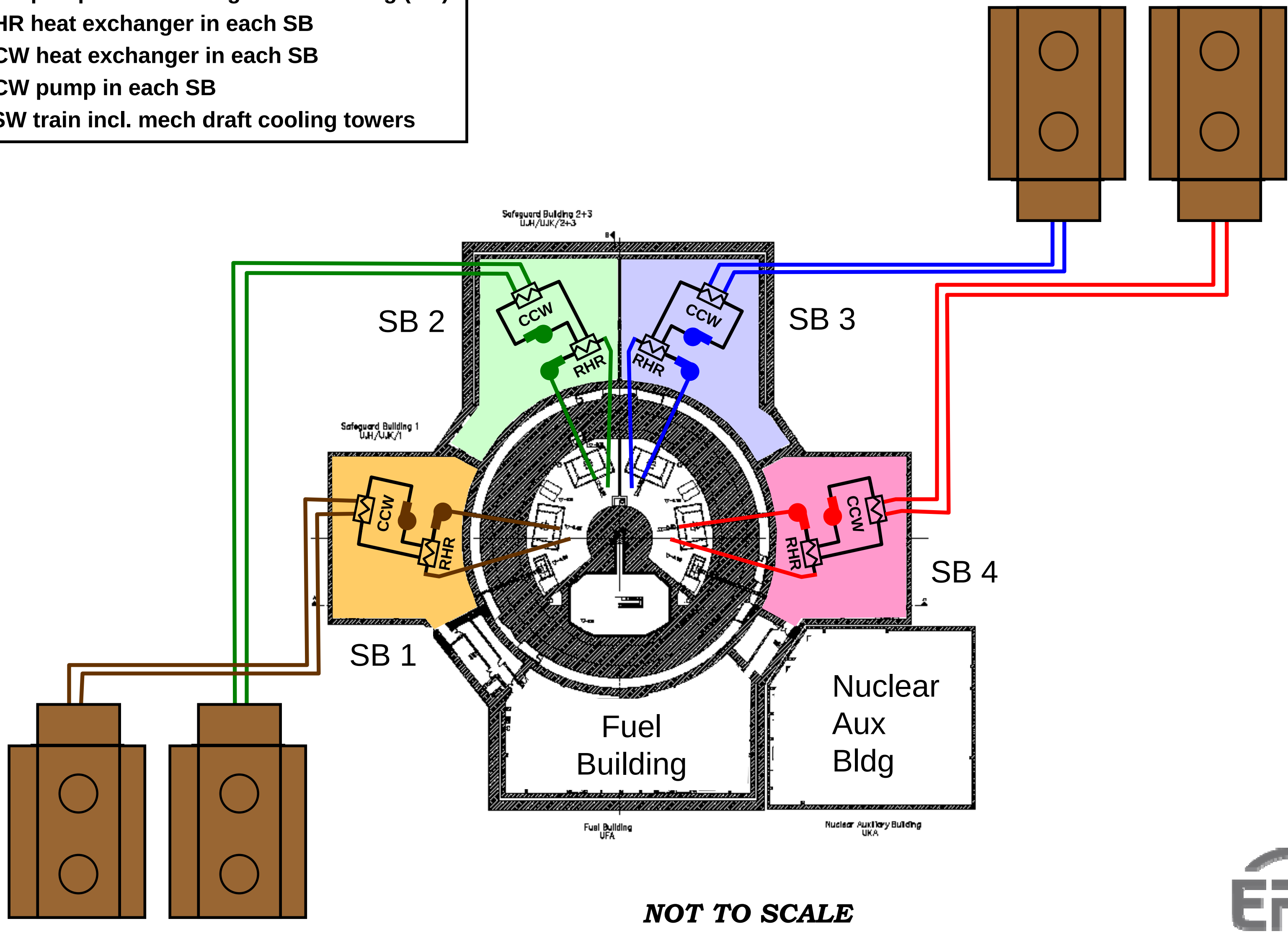
Main Safety Systems
Secondary Side



- Safety-related main steam relief train
- Four separate Emergency Feed Water Systems (EFWS)
- Separate power supply for each
- 2/4 EFWS also powered by Station Black Out (SBO) diesels
- Interconnecting headers at EFWS pump suction & discharge

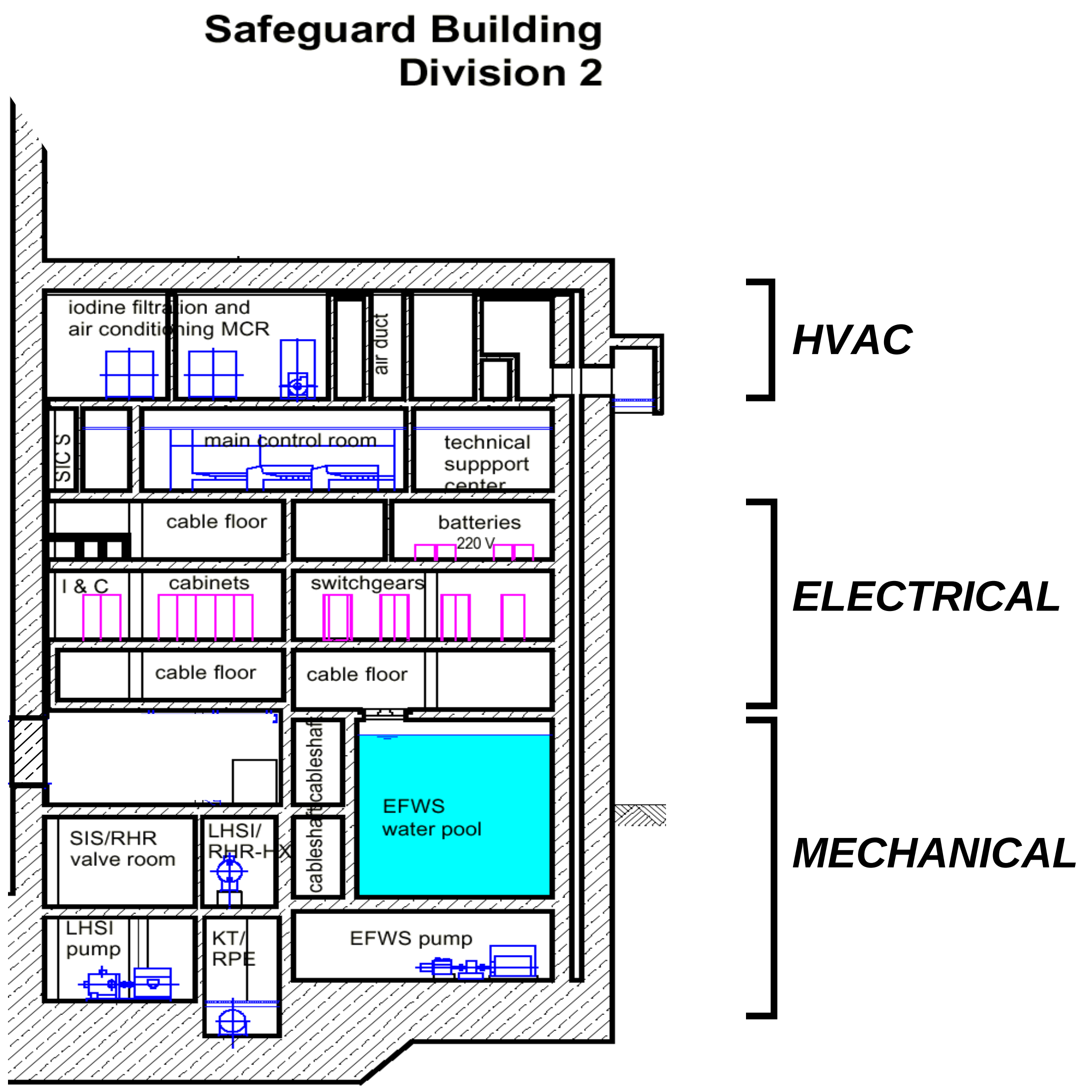
Example: RHR Systems

Each Train Connects to Different RCS Loop
- **1 RHR pump in each Safeguards Building (SB)**
- **1 RHR heat exchanger in each SB**
- **1 CCW heat exchanger in each SB**
- **1 CCW pump in each SB**
- **1 ESW train incl. mech draft cooling towers**



NOT TO SCALE

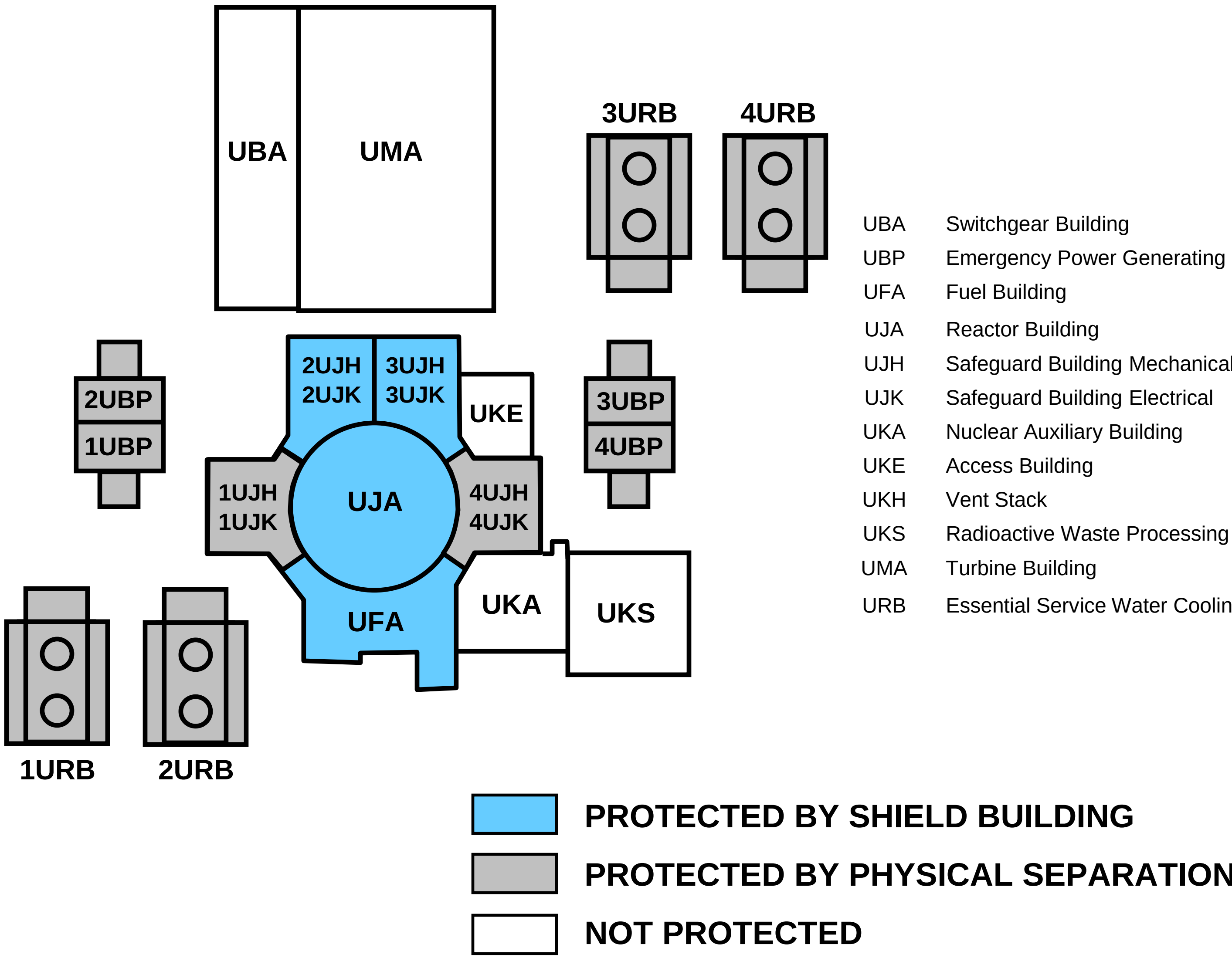
Safeguard Building Layout



Divisional Approach
Four Versus Two

- **Front-line safety systems 4 x 100%**
  - **Protection System**
  - **Emergency Power Supply System**
  - **ECCS**
  - **CCWS**
  - **ESWS**
  - **EFWS**
- **Many 2 x 100%**
  - **Annulus Ventilation**
  - **Safeguards & Fuel Building Iodine Filtration**
  - **Control Room Iodine Filtration**
  - **Containment Isolation**
  - **Extra Borating System**
  - **Spent Fuel Pool Cooling**

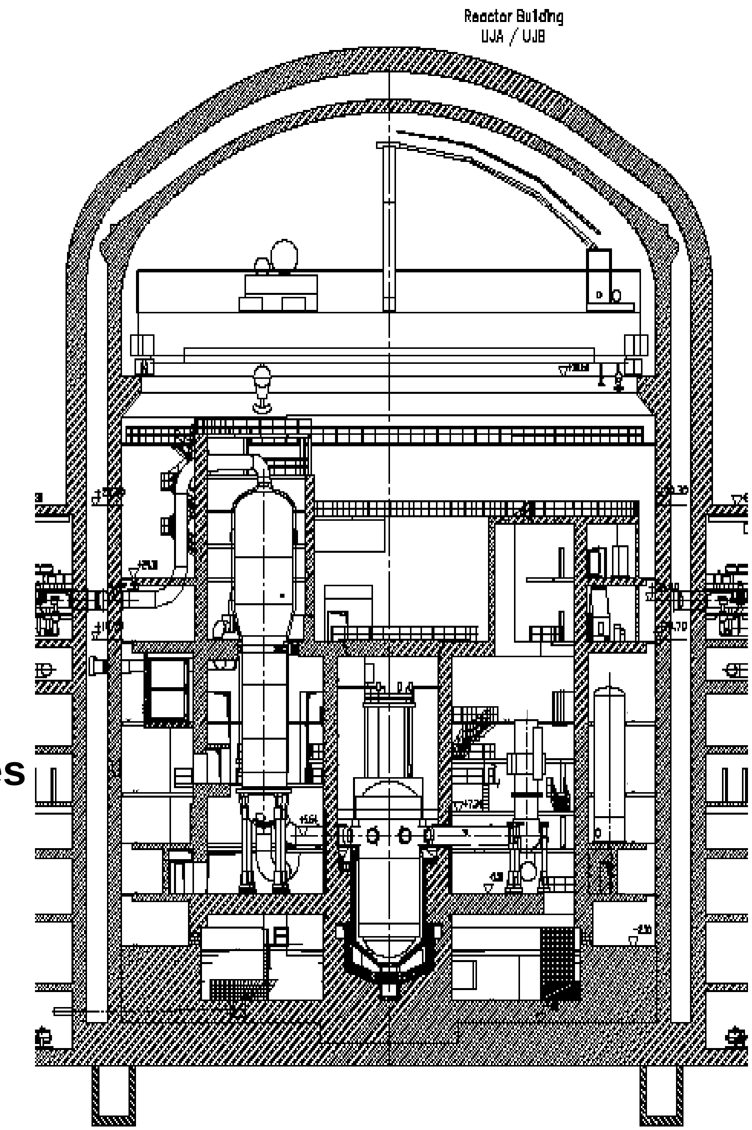
Protection From External Hazards



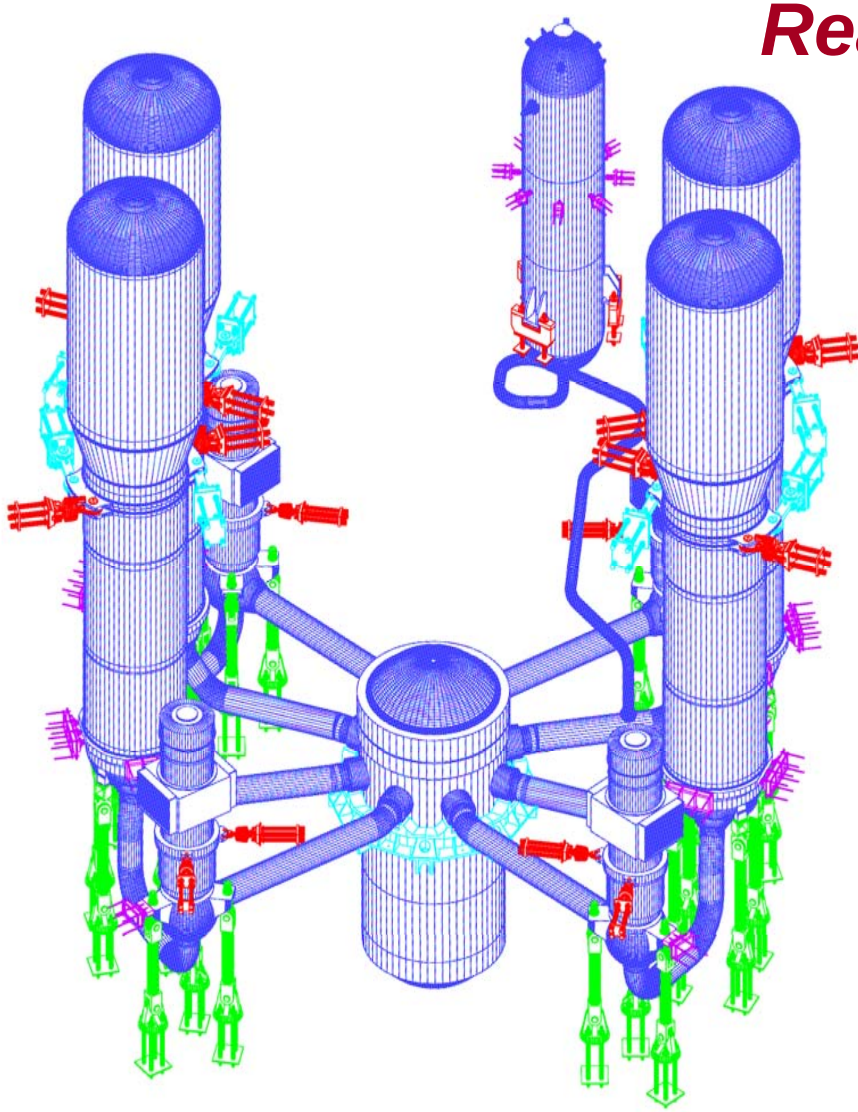
UBA	Switchgear Building
UBP	Emergency Power Generating Building
UFA	Fuel Building
UJA	Reactor Building
UJH	Safeguard Building Mechanical
UJK	Safeguard Building Electrical
UKA	Nuclear Auxiliary Building
UKE	Access Building
UKH	Vent Stack
UKS	Radioactive Waste Processing Building
UMA	Turbine Building
URB	Essential Service Water Cooling Tower Structure

EPR Reactor Building

- Containment wall post-tensioned concrete with steel liner
- Shield wall reinforced concrete
- Free volume = 2.8 Mft^3
- Design pressure = 62 psig
- Annulus filtered to reduce radioisotope release
- In-Containment Refueling Water Storage Tank (~500,000 gal)
- Severe accident mitigation features
- The design leak-rate at design pressure for a 24-hour period is less than 0.25 percent by volume



Reactor Coolant System



- Conventional 4-loop PWR design, proven by decades of design, licensing & operating experience.

- NSSS component volumes increased compared to existing PWRs, increasing operator grace period for many transients and accidents

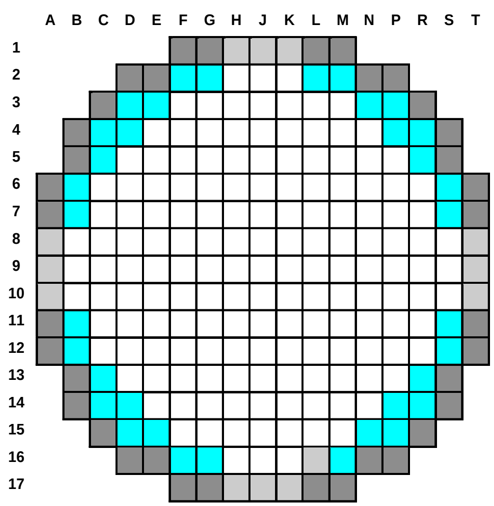
A solid foundation of operating experience.

U.S. EPR Plant Parameter Comparison

Parameter	4-Loop (Uprated)	U.S. EPR
Design Life	40	60
Thermal Power, MW	3565	4590
Electrical Power (Net), MW	1170	1595
Plant Efficiency, Percent	33	35
Hot Leg Temperature, F	618	624
Cold Leg Temperature, F	558	564
Reactor Coolant Flow Per Loop, gpm	100,500	124,700
Primary System Design Pressure, psia	2500	2550
Secondary System Design Pressure, psia	1200	1450
Primary System Operating Pressure, psia	2250	2250
Steam Pressure, psia	1000	1109
Steam Flow Per Loop, Mlb/hr	4.1	5.2
Total RCS Volume, cu.ft.	12,265	16,245
Pressurizer Volume, cu.ft.	1800	2650
SG Secondary Inventory at Full Power, lbm	101,000	182,000

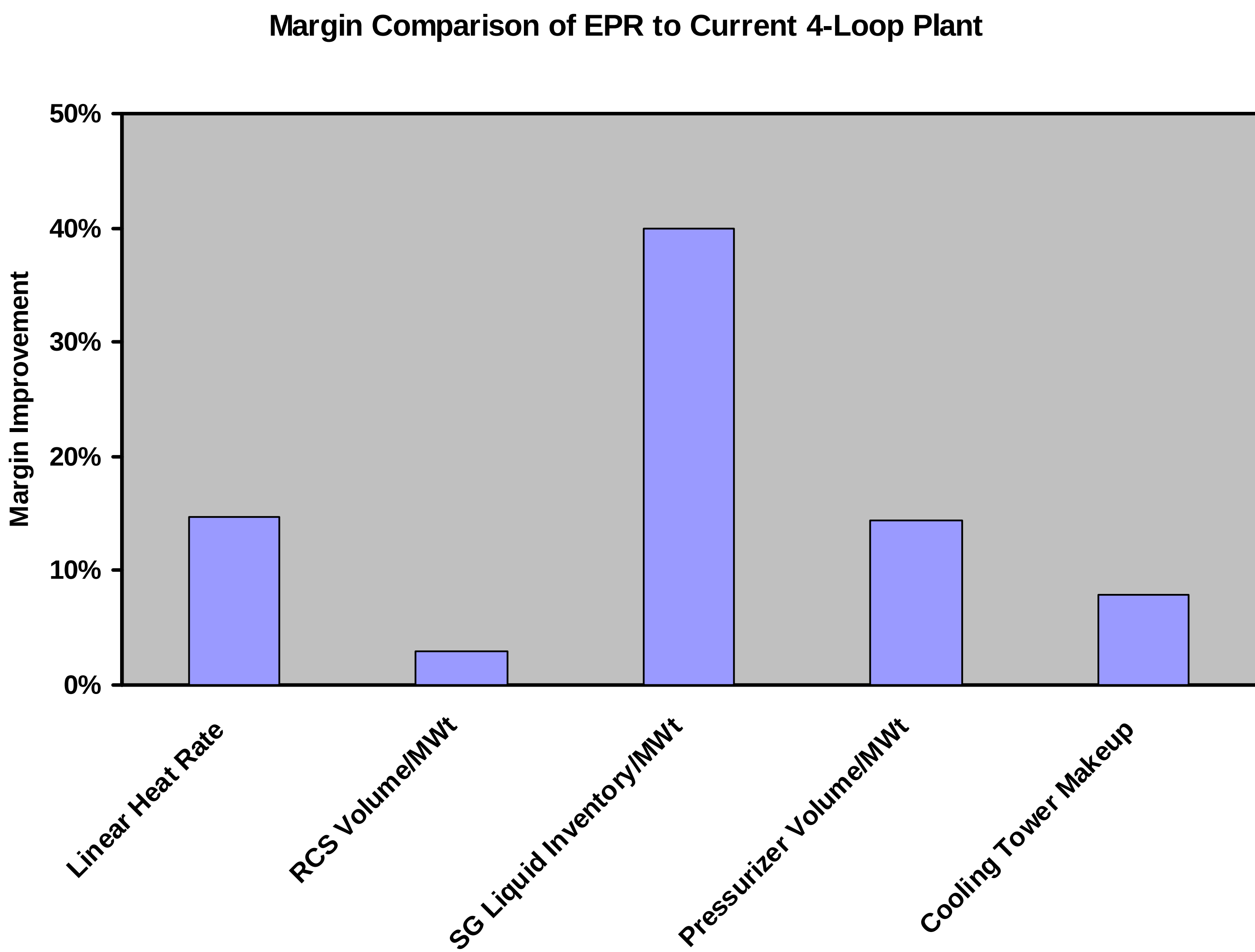
EPR Core Design Parameters

Parameter	Current 4-Loop (Uprated)	EPR
Core Thermal Power, MW	3565	4590
Number of Fuel Assemblies	193	241
Fuel Lattice	17x17	17x17
Active Fuel Length, ft	12	13.78
Rods Per Assembly	264	265
Average Linear Heat Rate, kw/ft	5.8	5.2
Peak Linear Heat Rate, kW/ft	14.6	13.8
Number of Control Rods	53	89



Type of Plant	No of Fuel Assy	
4-loop 1300 MWe	193	
4-loop N4	205	
U.S. EPR	241	

Comparison of EPR Design Margins to Typical 4-loop Unit



Digital Controls

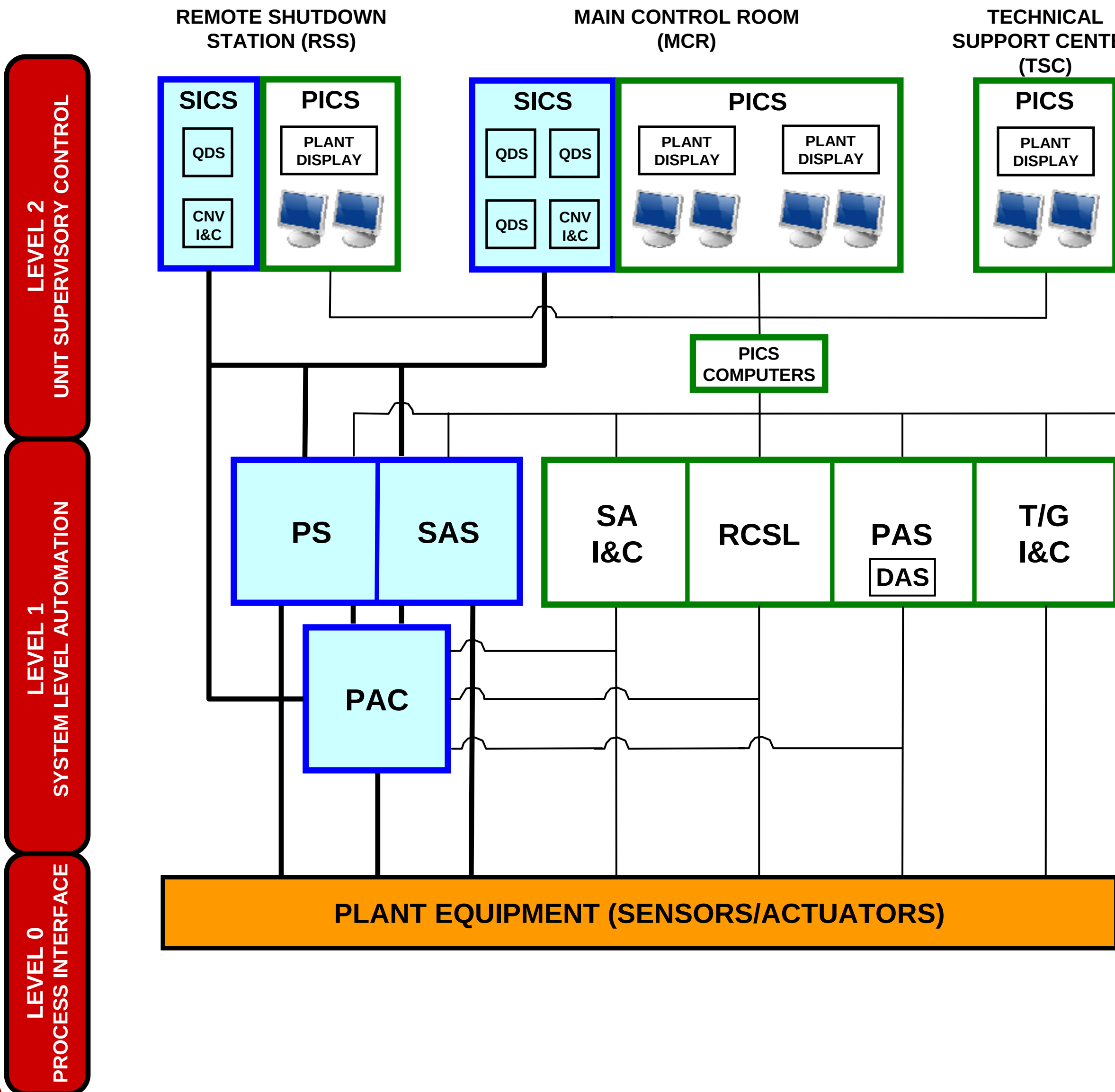
Operator-Friendly Man-Machine Interface

N4 Control Room

EPR Control Room

Capitalizing on nuclear digital I&C operating experience and feedback.

Digital I & C Architecture



CNV	Conventional
DAS	Diverse Actuation System
PAC	Priority and Actuator Control Module
PAS	Process Automation System
PICS	Process Information and Control System
PS	Protection System
QDS	Qualified Display System
RCSL	Reactor Control, Surveillance and Limitation System
SAS	Safety Automation System
SICS	Safety Information and Control System
SA I&C	Severe Accident I&C
T/G I&C	Turbine Generator I&C
	Safety I&C – Teleperm XS
	Operational I&C

Increased Protection & Automation

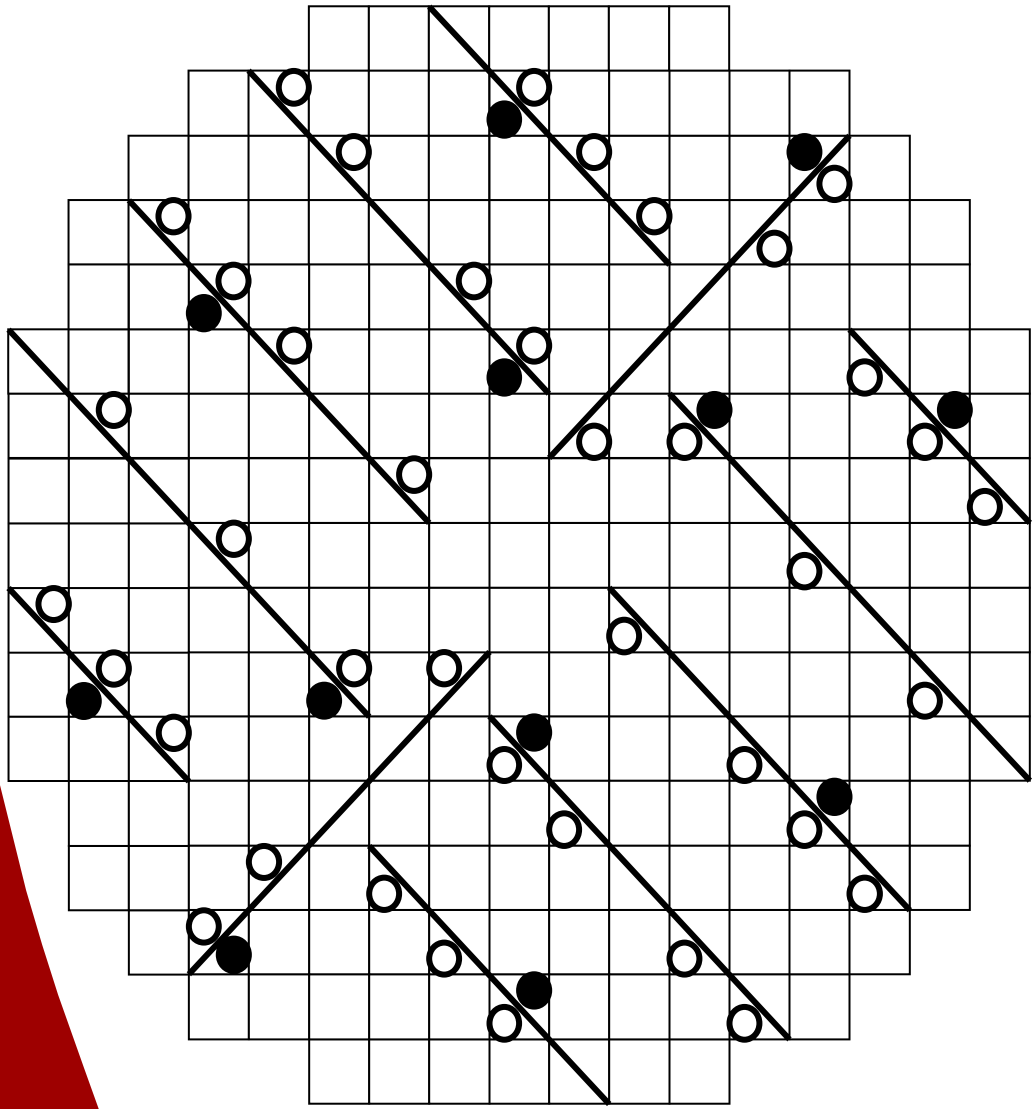
- Hot-channel DNBR trip
- High linear power density trip
- High SG pressure trip
- Protection System SG depressurization
- Automatic boron dilution detection
- Computer-controlled heat-up & cooldown
- On-line procedures
- Electronic tagging
- Self-checking

In-Core Monitoring

- **Fixed and moveable core monitoring systems**
- **Self-Powered Neutron Detectors continuously monitor core power**
  - **Provide input signals to POWERTRAX/E* software**
  - **Safety and non-safety functions are generated by SPNDs**
  - **SPND signal drift with burnup compensated by calibration**
- **The Aeroball Measurement System is used to calibrate SPNDs**
  - **About every 15 EFPD, the SPNDs are calibrated to the AMS reference signal**
  - **AMS is a moveable system that provides accurate 3-D core power maps**
  - **The AMS provides no signals to any protection or monitoring functions**

*** POWERTRAX/E provides a comprehensive system for on-line 3-D power distribution monitoring and for reactor operation support calculations**

Nuclear Instrumentation

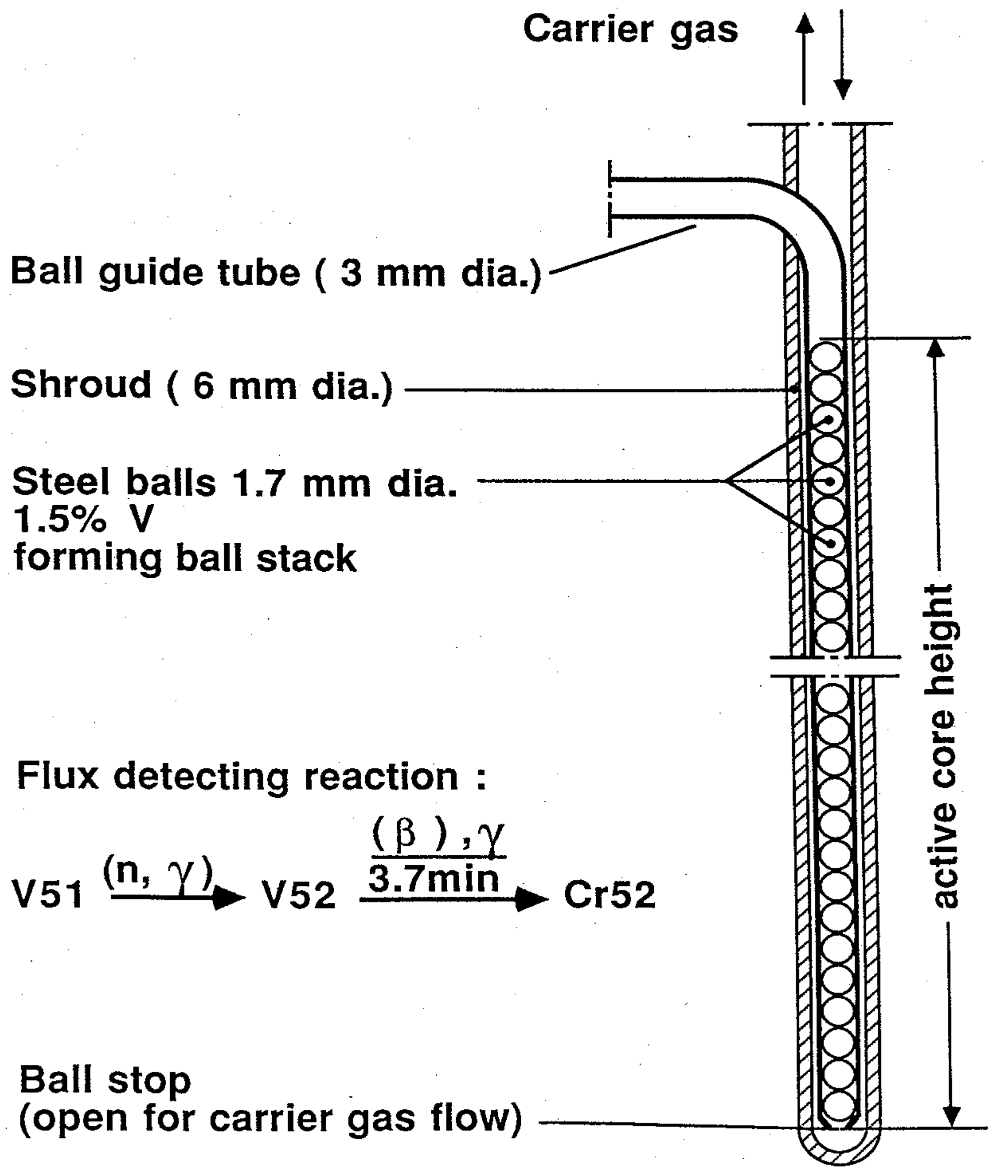


241 ASSEMBLIES

12 SPND FINGERS

40 AEROBALL PROBES

Aeroball Probe Schematic

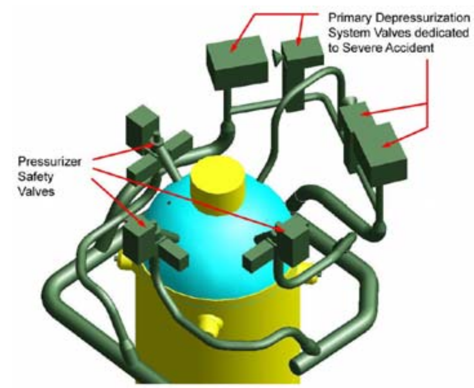


Flux detecting reaction :

$$V51 \xrightarrow{(n,\ \gamma)} V52 \xrightarrow[3.7min]{(\beta),\gamma} Cr52$$

Severe Accident Mitigation

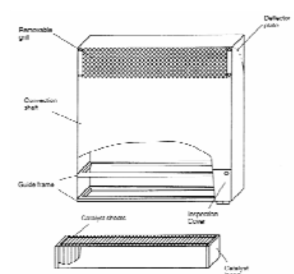
- Prevention of high-pressure melt-through using Primary Depressurization System



- Passive ex-vessel melt stabilization, conditioning and cooling

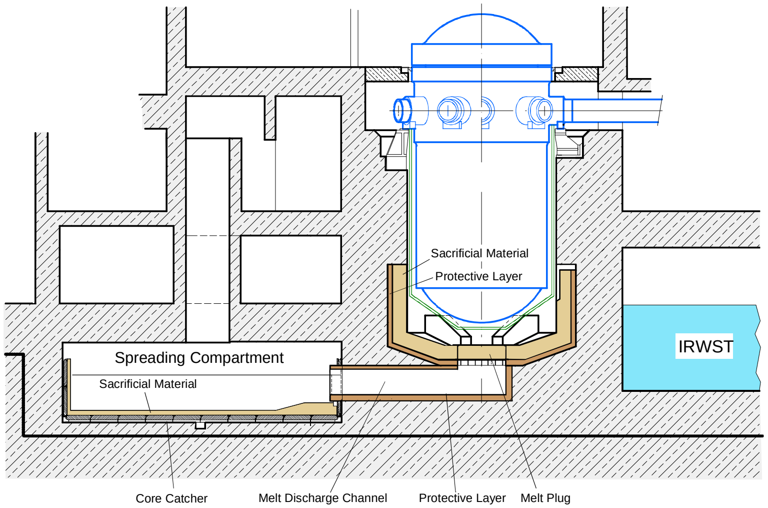
- Long-term melt cooling and containment protection using active cooling system

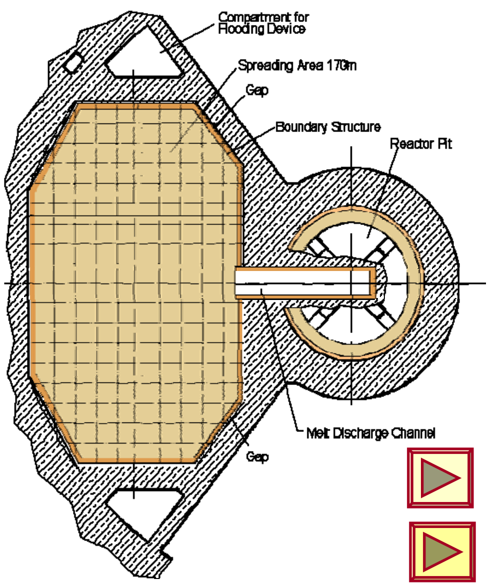
- Control of H_2 concentration using passive autocatalytic recombiners



Severe Accident Mitigation
Melt Conditioning and Stabilization

- **Reactor cavity temporarily retains molten core debris prior to spreading and stabilization processes**
  - Limits uncertainties associated with RPV release states
  - Corium/concrete interaction within reactor cavity lowers melting temperature of corium and promotes spreading
- **Melt spreading and relocation**
  - After melt plug failure, conditioned melt will relocate into spreading area (shallow crucible)
  - Large spreading area promotes cooling
  - Spreading area is dry at time of melt relocation to preclude ex-vessel steam explosion
- **Stablization**
  - Water from IRWST passively cools melt for up to 12 hours
  - Thereafter, severe accident heat removal system actively cools the melt and depressurizes containment

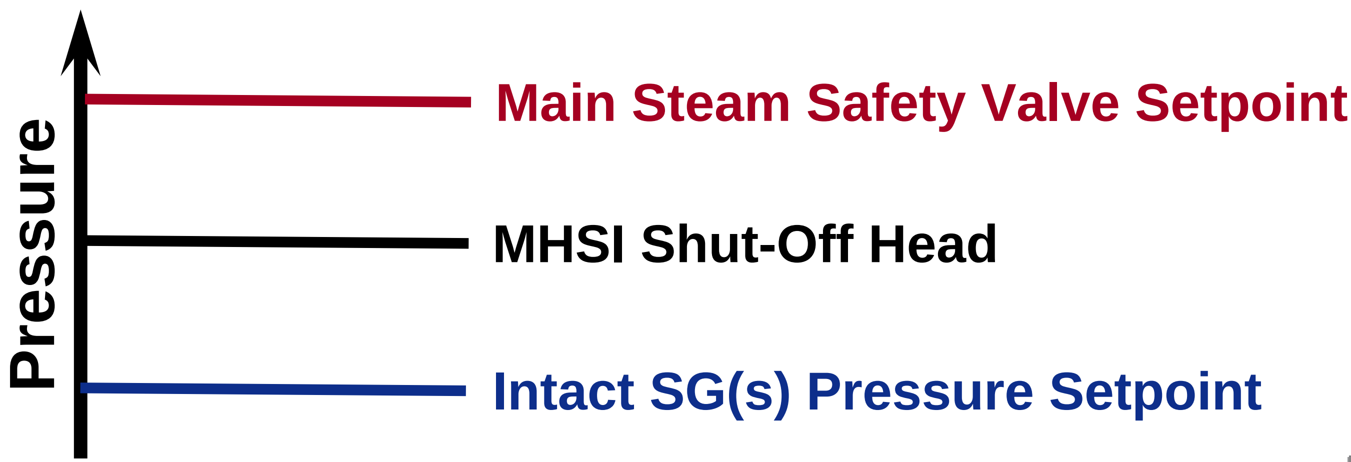




R&D

SGTR Mitigation
Safety Injection System

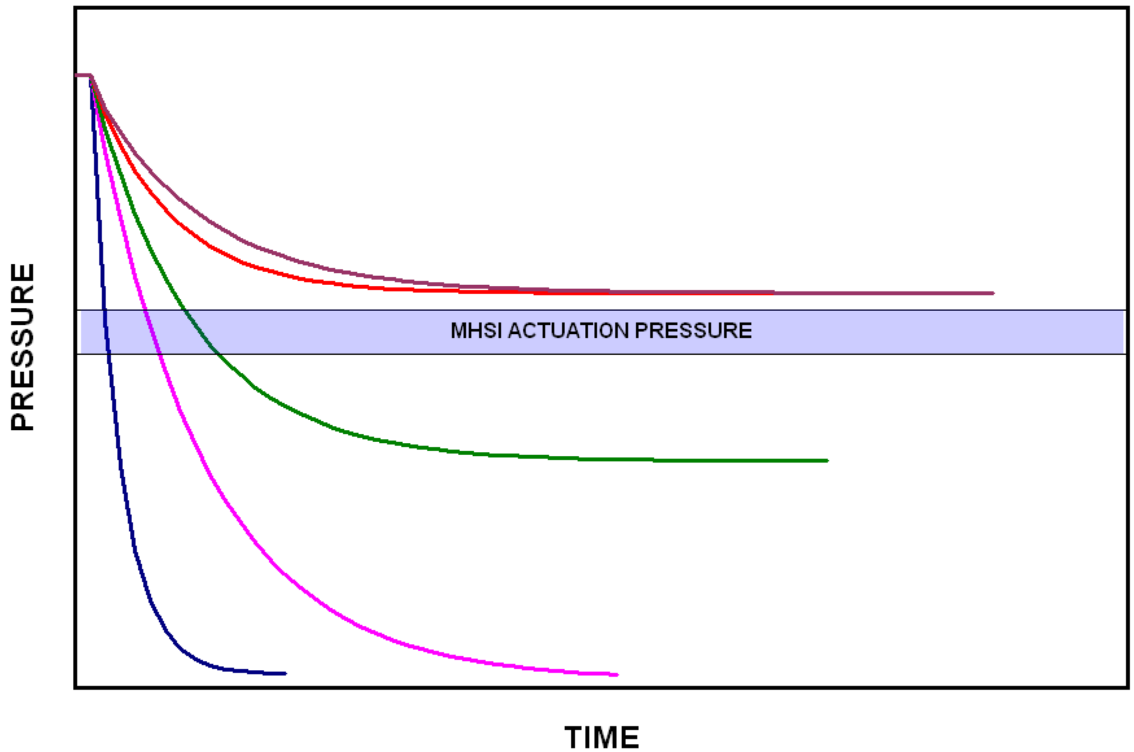
- **Medium Head Injection selected for SGTR mitigation:**
  - **Shutoff head below MSSV setpoint**
  - **Ensures no challenge to MSSVs during SGTR (no operator action required to throttle safety injection)**
  - **SGTR dose consequences meet safety goal by minimizing containment bypass (eliminate possibility of discharging reactor coolant)**



SGTR & SBLOCA Mitigation

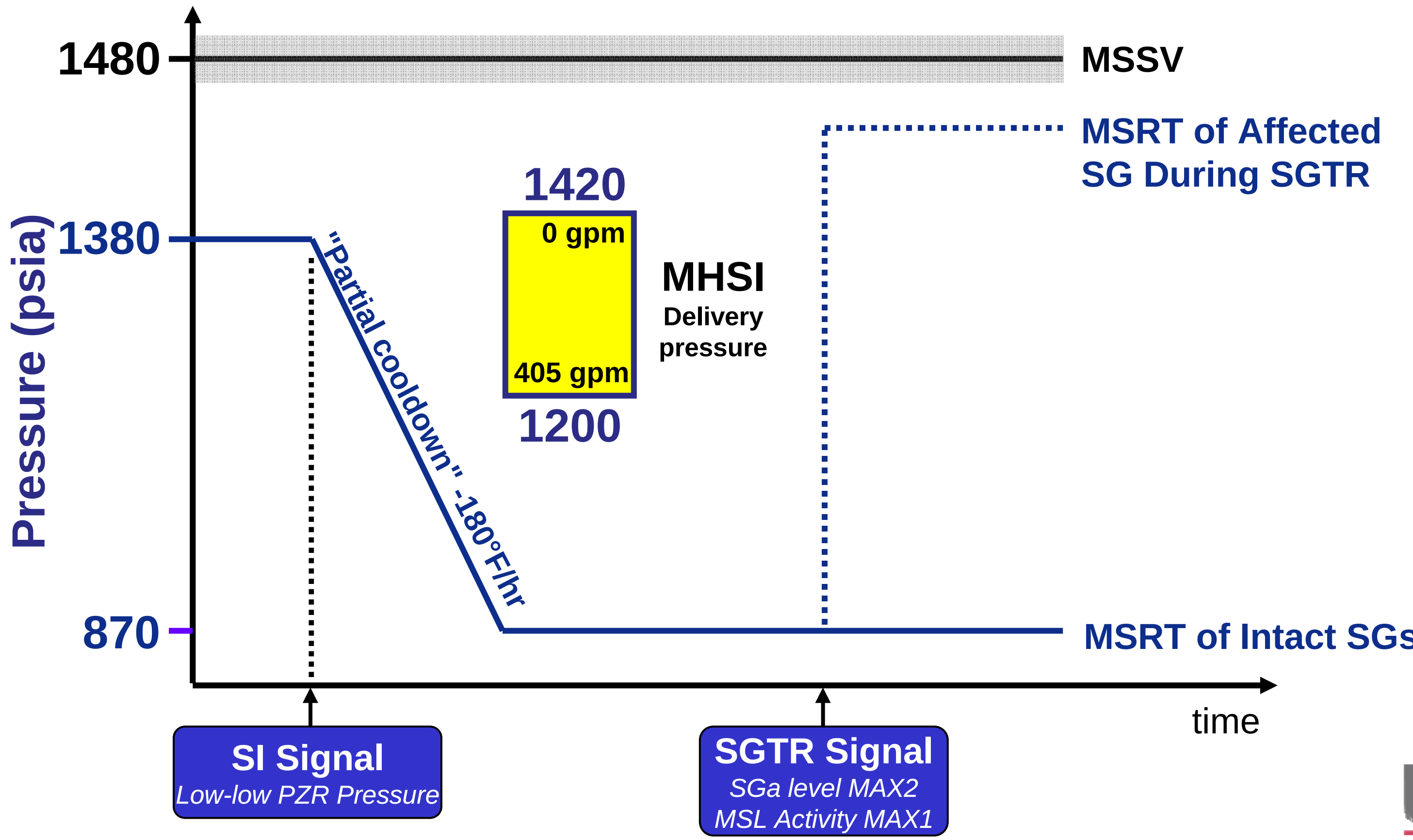
SBLOCA Spectrum Studies

- For very small LOCAs, RCS pressure "couples" to SG pressure because SG heat removal is maintained
- SI flow begins when RCS/SG pressure falls below the MHSI shut-off head



SBLOCA Mitigation
Partial Cooldown

- Safety-related function (Protection System)
- Depressurizes SGs to reduce T_{sat} at 180 F/hr
- Ensures adequate MHSI flow for SBLOCA



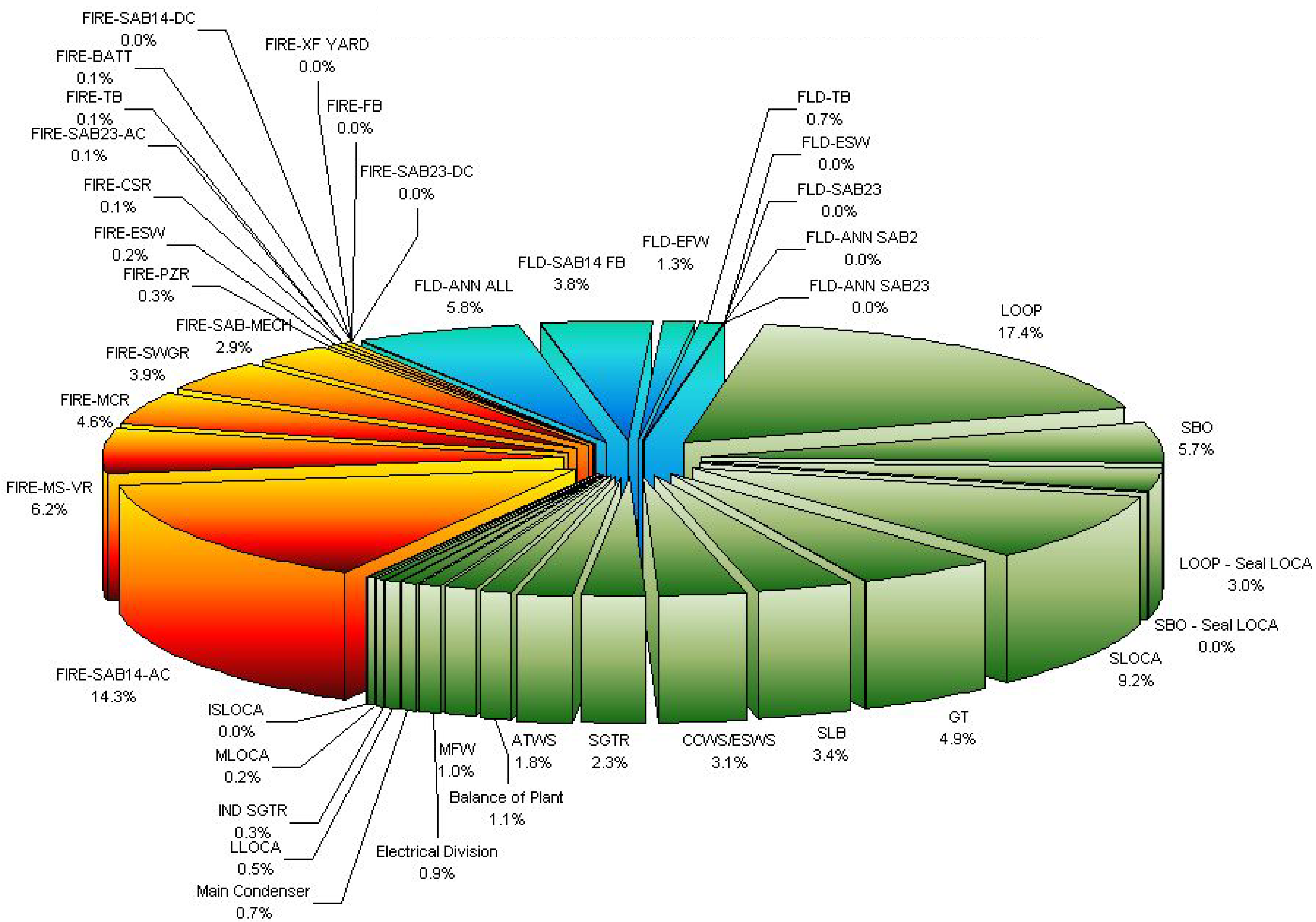
Probabilistic Objectives And Targets

- **Safety objective for integral core melt frequency (all plant states, all types of initiators): $< 10^{-5}$ per year**

- **Design target for core damage frequency for internal events**
  - **from power states: $< 10^{-6}$ per year**
  - **from shutdown states: less than power states**

- **Design target for core damage with large and early releases from containment: $< 10^{-7}$ /year**

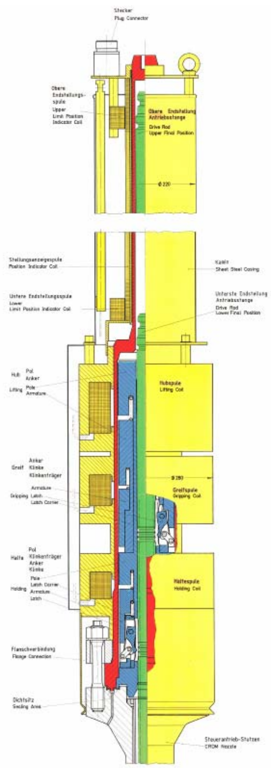
U.S. EPR CDF (At-Power Events)



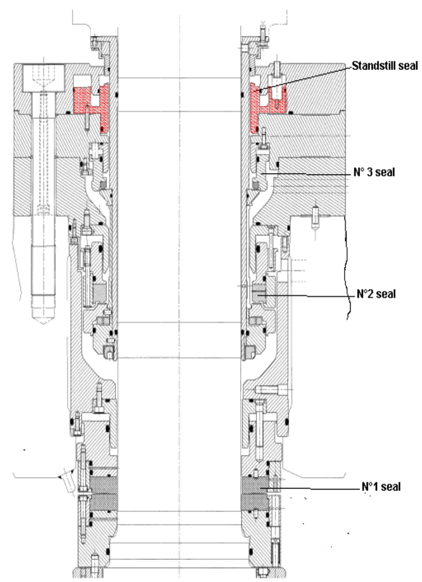
Level 1 At-Power, Internal Events CDF = 5.3 x 10^{-7}/yr
CDF For All Events < 5.8 x 10^{-7}/yr

Operating Experience Feedback

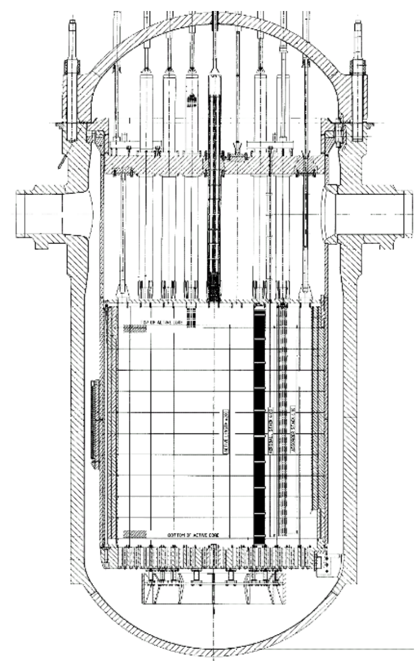
Martinsitic CRDM housing. Forced convection cooling of coils not req'd.



RCP stand-still seal eliminates leakage during SBO.



No penetrations in RV lower head.

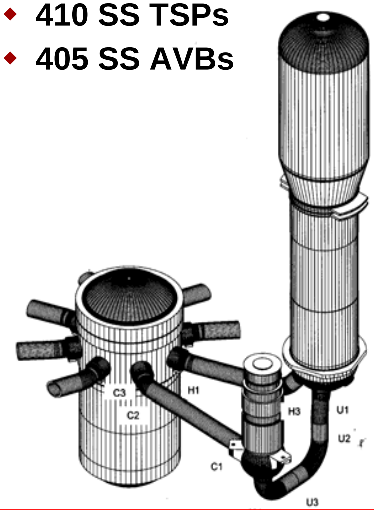


Operating Experience Feedback

Extensive use of forgings with integral nozzles.

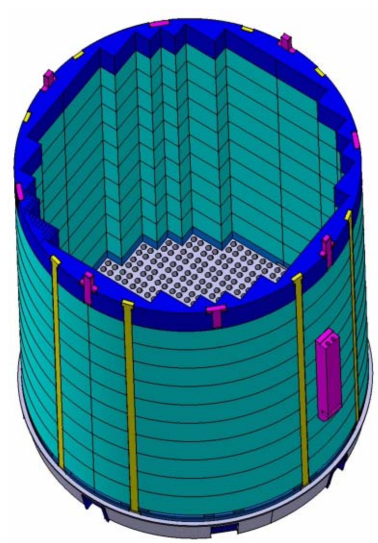
Materials resistant to corrosion and cracking

- **304L SS hot/cold legs**
- **316L SS surge line**
- **304L/316L RV internals**
- **308/309 SS cladding**
- **Alloy 690 SG tubes**
- **410 SS TSPs**
- **405 SS AVBs**

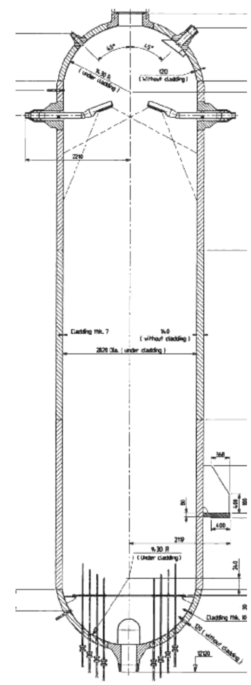


Conventional core baffle replaced by heavy reflector.

- **Eliminates bolting**
- **Improves neutron economy**
- **Reduces vessel fluence**



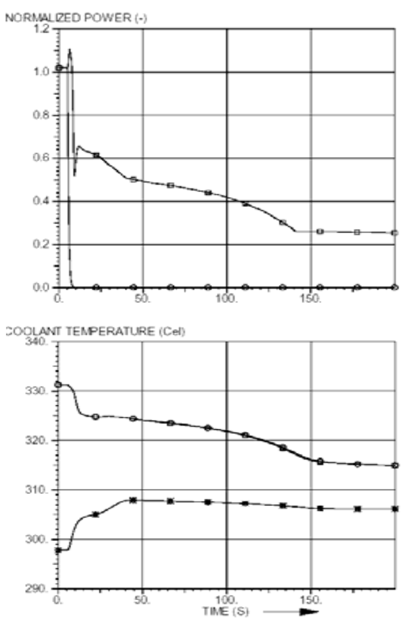
Two normal pzr spray (ea. from different CL) plus one aux spray



Operating Experience Feedback

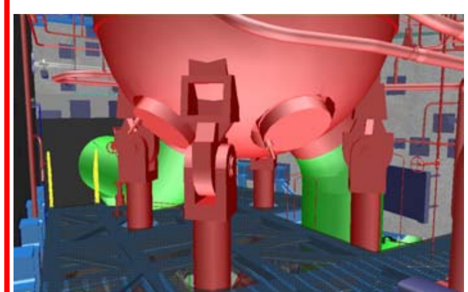
Reduction of single-point vulnerabilities

- Partial trip function
- Three 50% condensate pumps
- Bypass components for maintenance



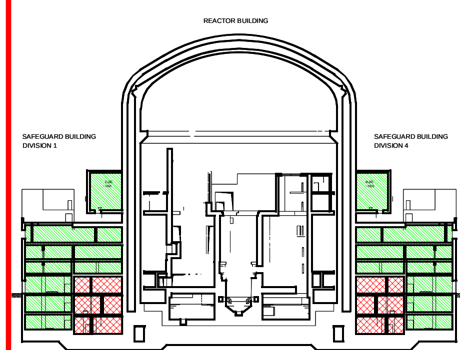
Facilitate maintenance

- Access room
- Permanent platforms
- Permanent maintenance power and air
- Pre-engineered haul routes & rigging points for component replacement



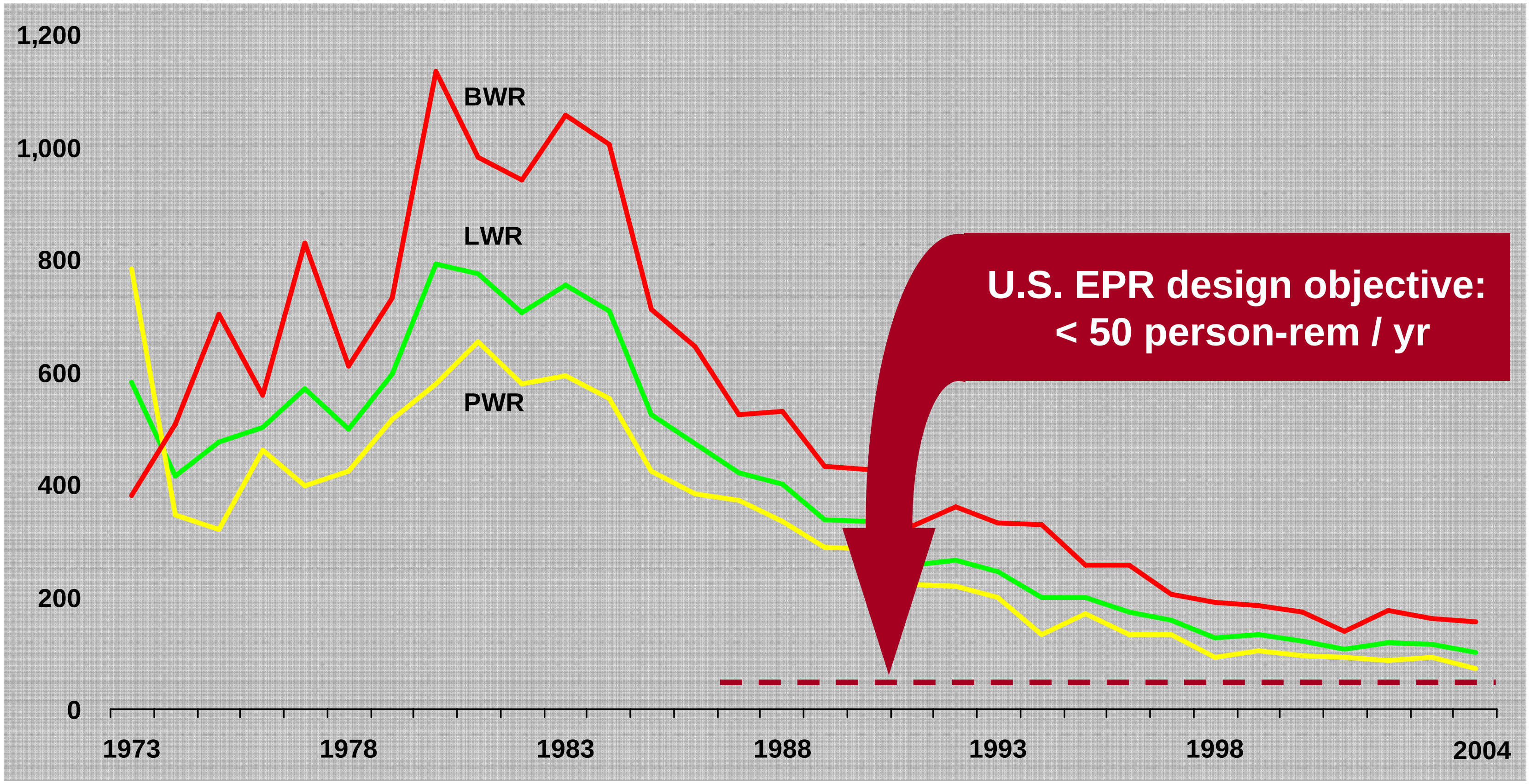
ALARA central in design

- Minimize cobalt
- Minimize deposits
- Use of “harsh” and “mild” zones



U.S. Industry-Average Dose Per Reactor

1973-2004, (Person-rem)



Source: Nuclear Regulatory Commission Occupational Radiation Exposure at Commercial Nuclear Power Reactors and Other Facilities 2004

Updated: 4/06

Design Summary

- U.S. EPR is evolutionary
- Most features are typical of operating PWRs
- Features included to
  - Improve safety
  - Protect critical systems from external events
  - Improve human factors
  - Enhance reliability

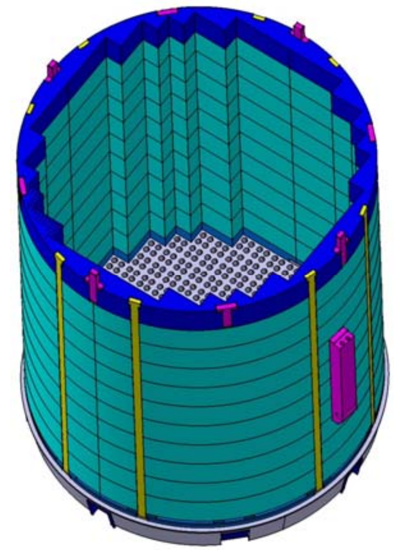
Backup Slides

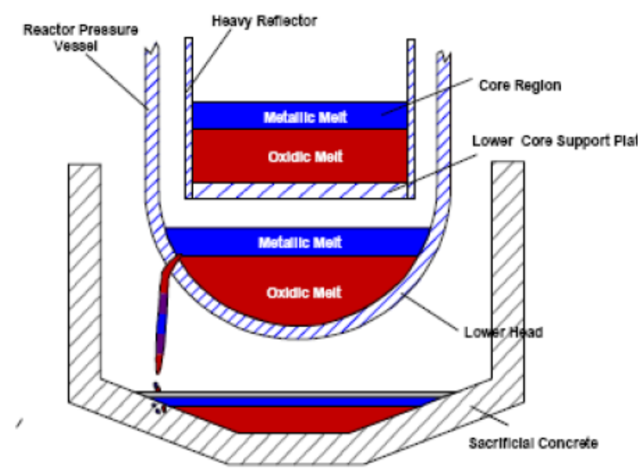
R&D Basis For Severe Accident Features

- **Melt Accumulation and Conditioning**
  - ACE (ANL) – 1D MCCI of prototypical oxidic melts
  - MACE (ANL) – 1D/2D MCCI of prototypical oxidic melts (flooded)
  - BALI (CEA) – Heat flux distribution in heated, agitated fluid
- **Behavior of the Melt Plug**
  - KAPOOL (FZK) – Transient thermit tests on gate attack
  - HTCM (SNU) – Effect of molten oxidic corium on metallic structures
  - CORESA (SNU) – Stability of zirconia-based materials
  - MVI (RIT) – Erosion and "hole-widening" effect of melt flows
- **Melt Spreading**
  - COMAS-EU (SNL) – 1D/2D melt spreading under dry conditions
  - VOLCANO (CEA) – 1D oxidic melt spreading of dry conditions
  - ECOKATS (FZK) – 2D demonstration test of EPR with flooding
- **Melt Stabilization and Cooling**
  - MACE (ANL) – 1D oxidic corium experiments with top flooding
  - MCCI-OECD (ANL) – 2D oxidic corium experiments with top flooding

In-Vessel Melt Progression

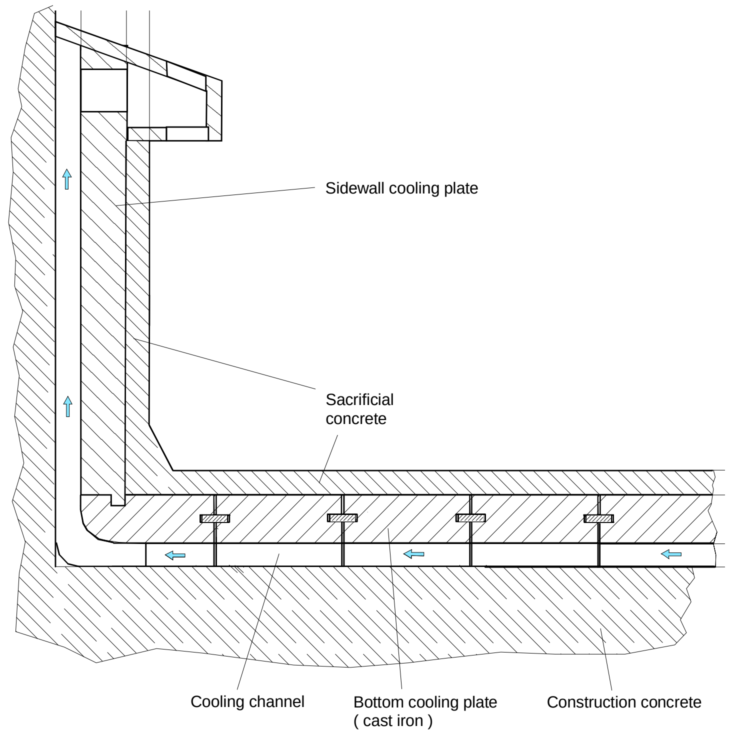
- **In-vessel melt progression is dependent on RPV internals**
- **Corium will accumulate in lower RPV head as melt progresses**
- **Accumulation in lower head can lead to RPV failure and relocation into reactor cavity**

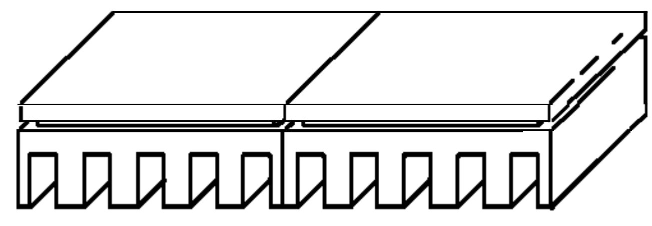




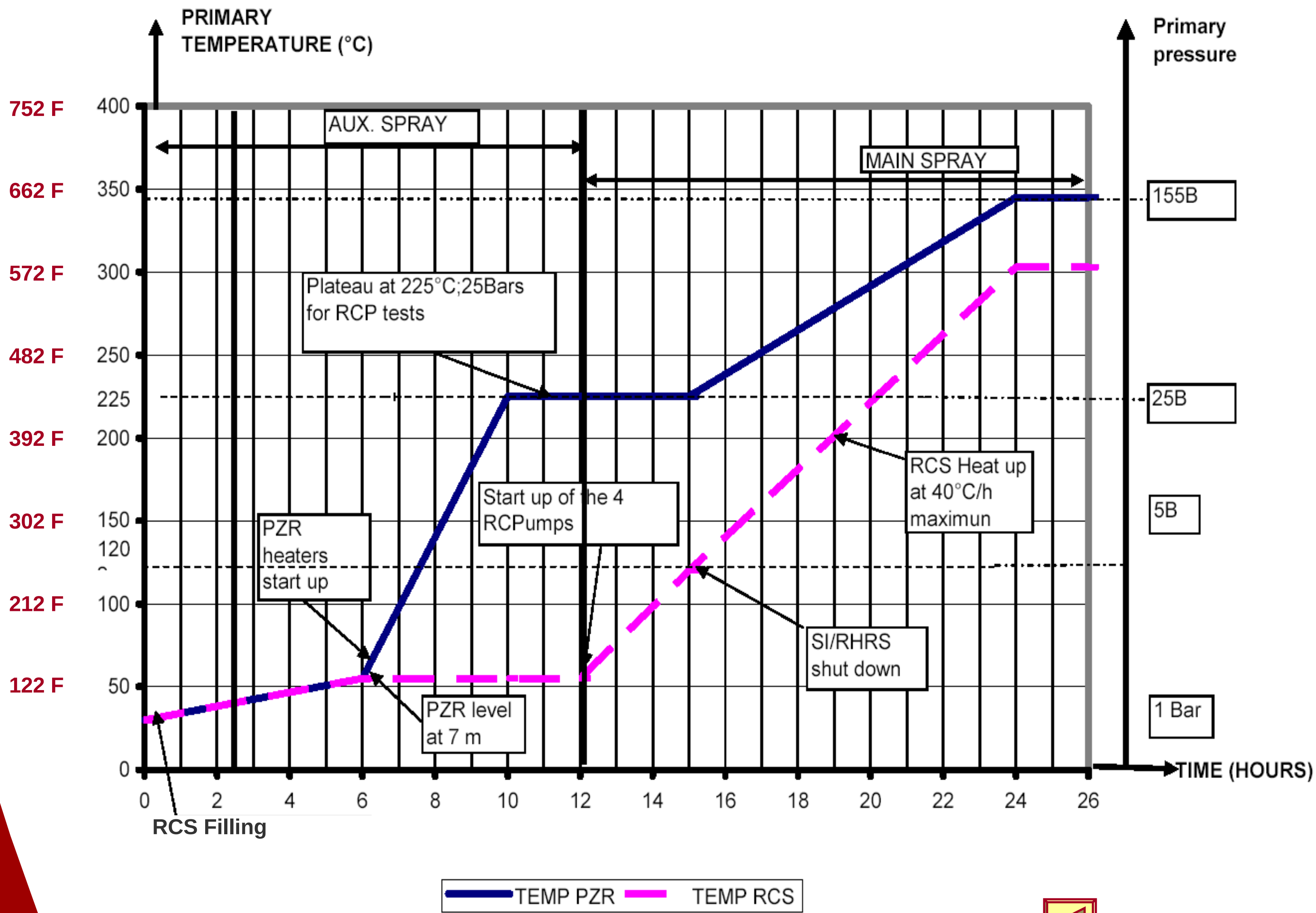
Spreading Area and Cooling Structure

- **Core melt is retained within spreading area and is passively cooled on all sides**
- **Cooling structure consists of finned iron elements that are protected from corium with sacrificial concrete**
- **Flooding of spreading area is initiated by thermally sensitive spring-loaded valves (passive)**
- **Water from IRWST gravity fills cooling channels and overflows onto melt surface**
- **Melt quenching is performed at low flow rates to minimize fuel coolant interactions**

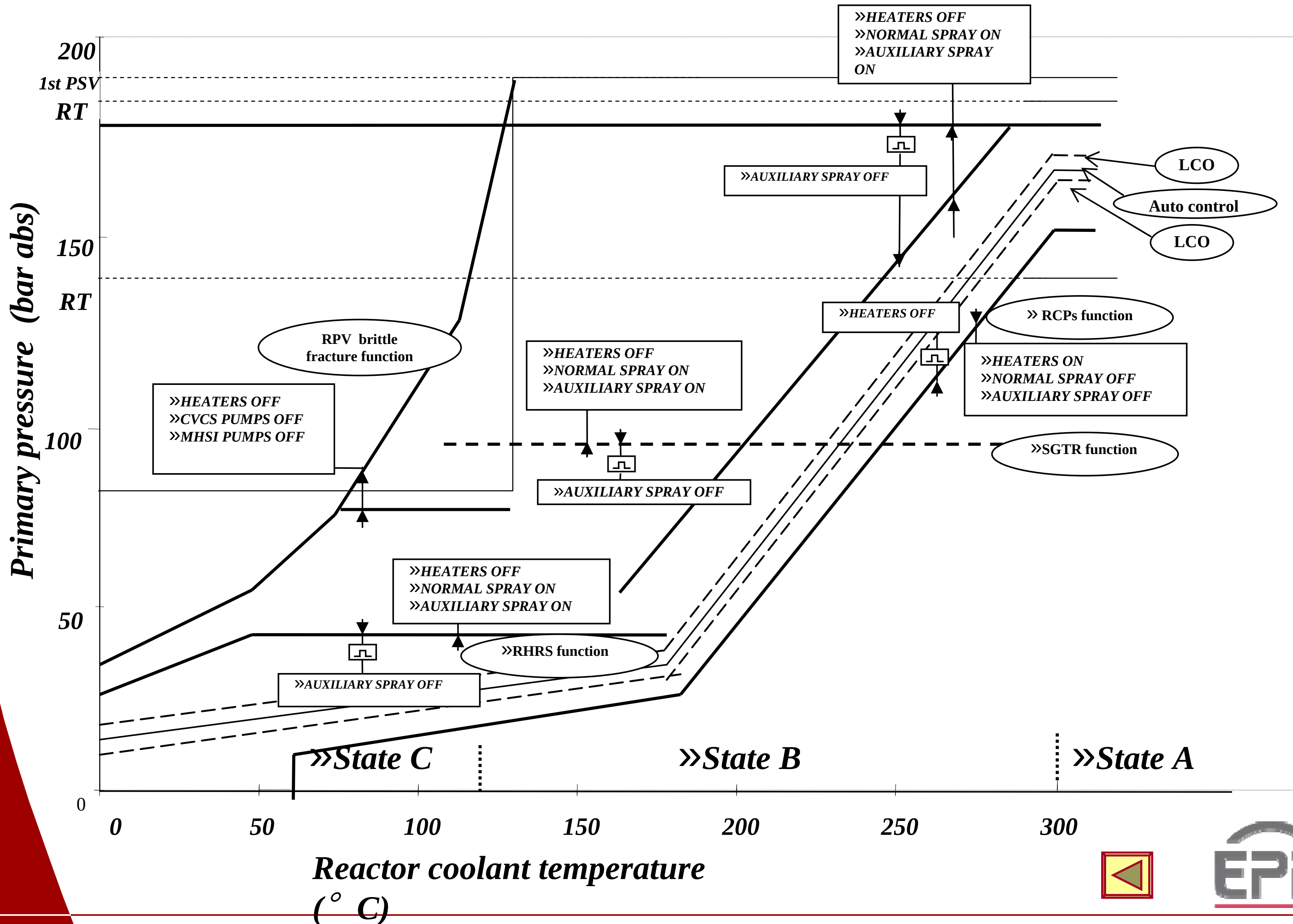




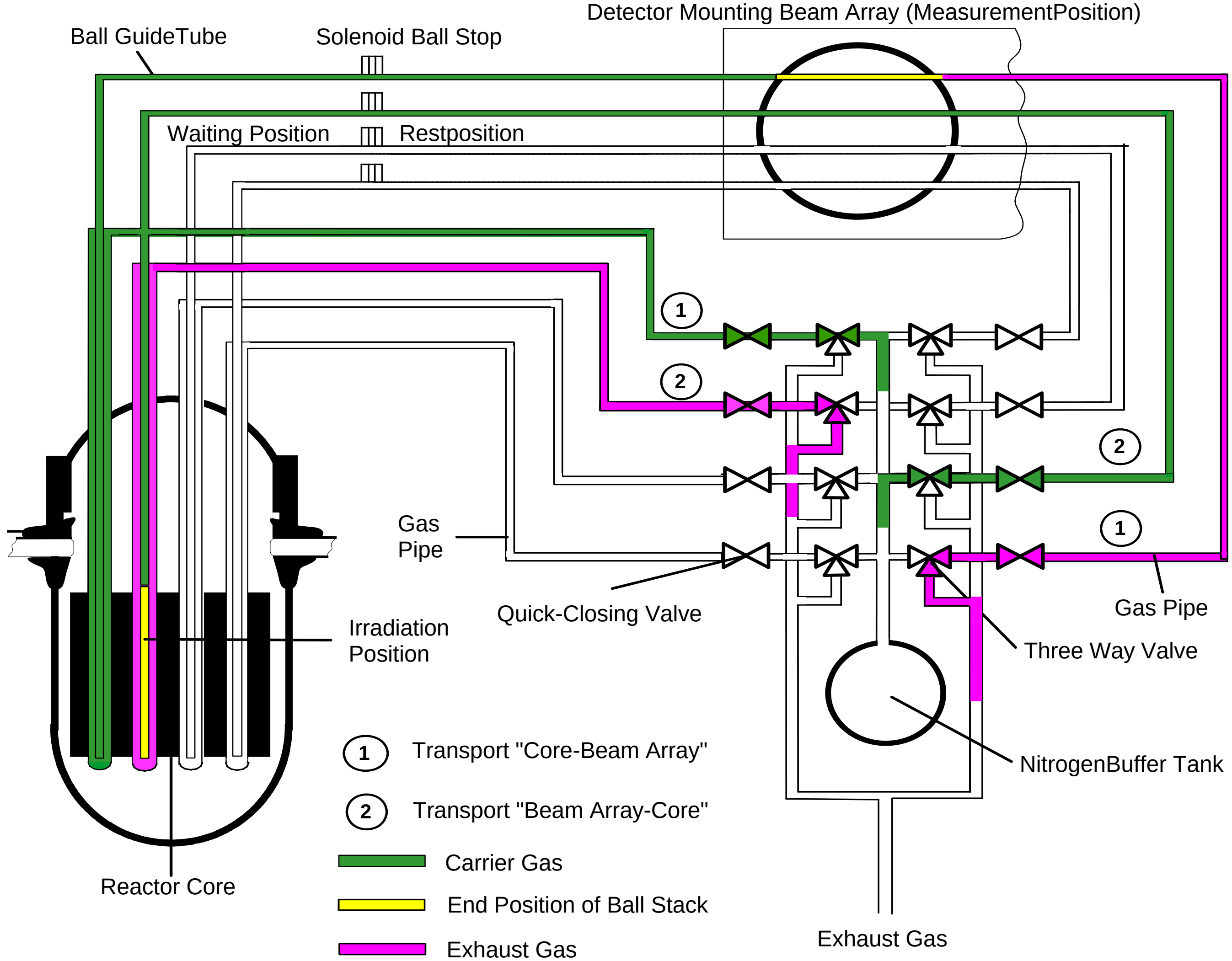
Start-up Diagram
Primary Side



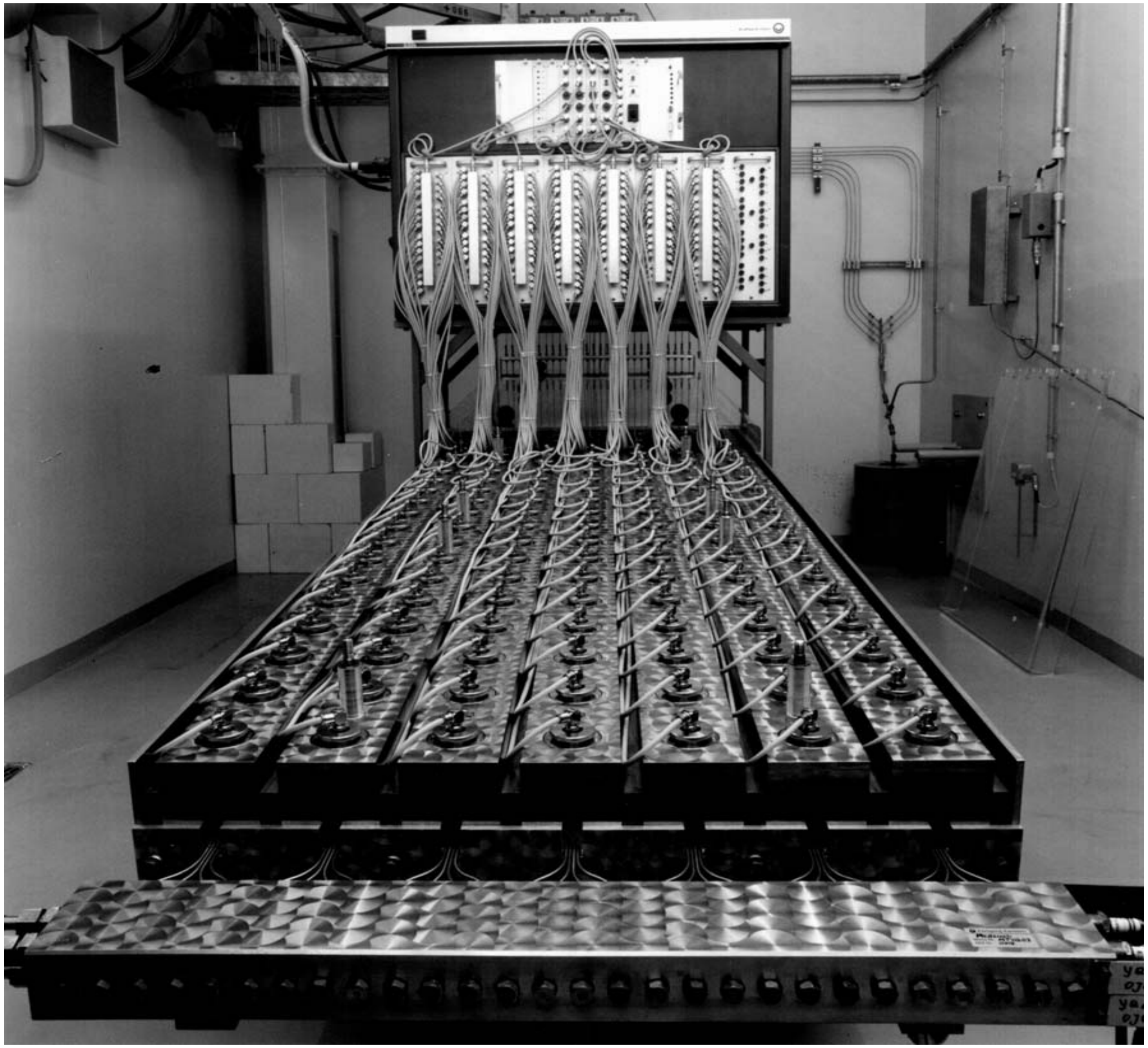
Primary Side Pressure And Inventory Control



Aeroball Pneumatic Transport System

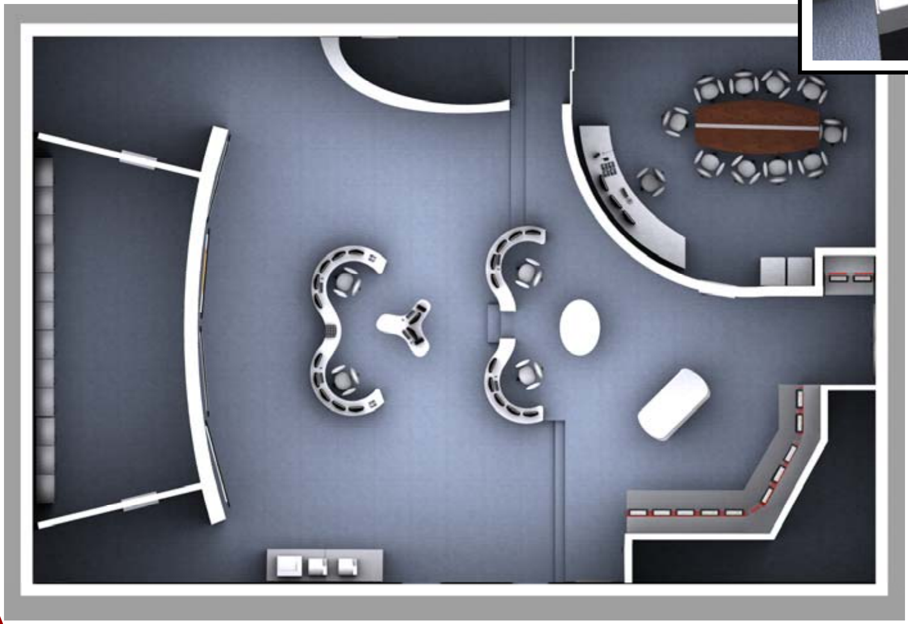


Aeroball System Measuring Table

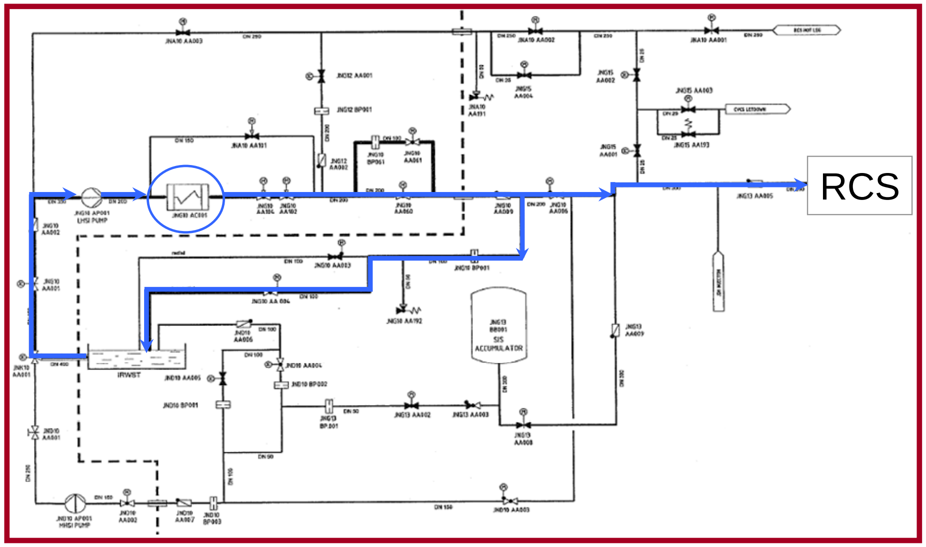


Note: Not identical to EPR equipment

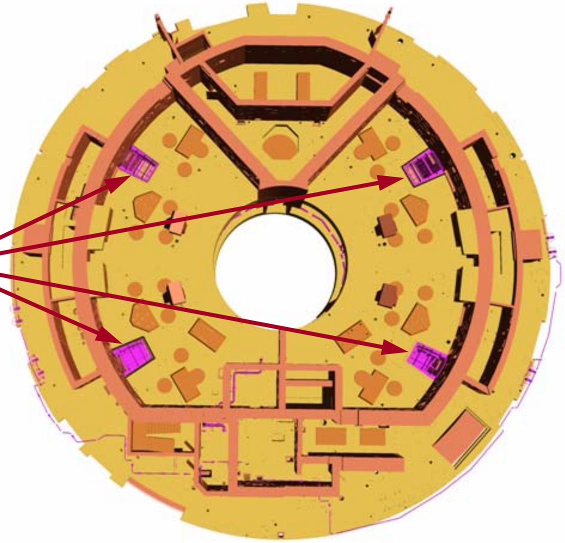
U.S. EPR
Main Control Room



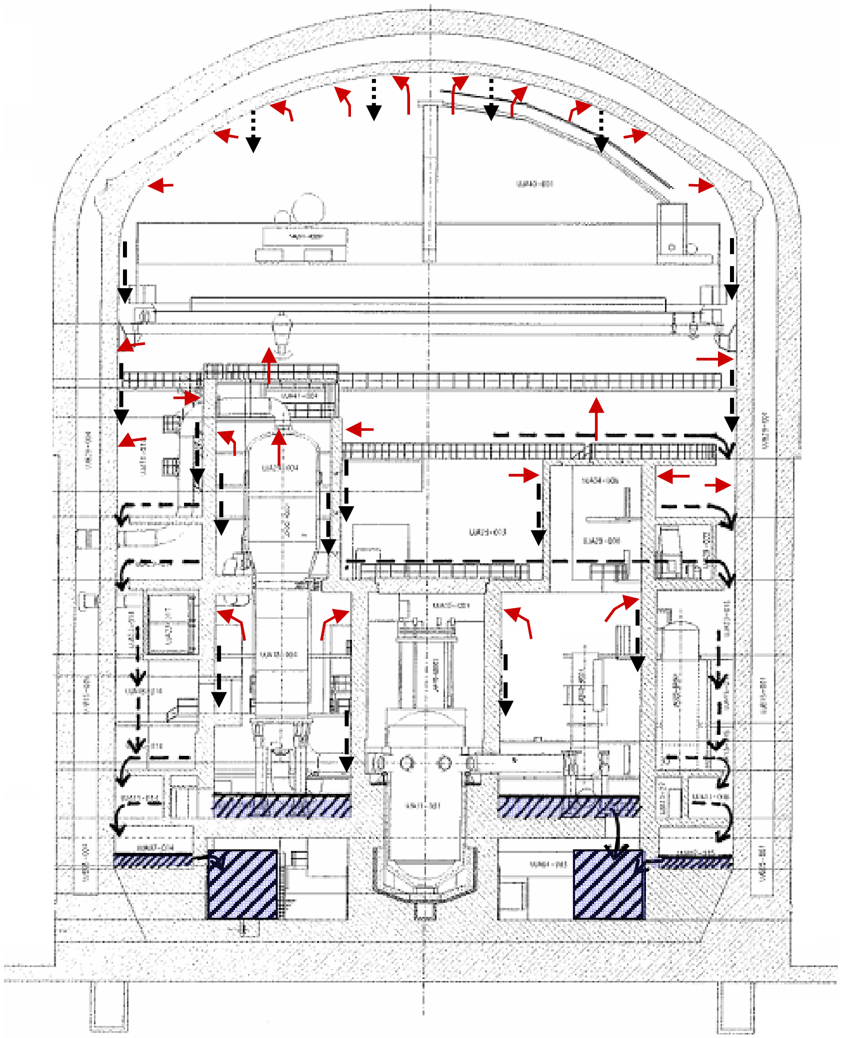
Recirculation During LOCA



DRAINS
TO
IRWST



HEAVY FLOOR



Security Rulemaking for Nuclear Power Plants

ACRS Presentation

June 4, 2008

Discussion Topics

- Status of Power Reactor Security Rulemaking
- Staff Draft Final Rule Text needing ACRS review
  - 50.54(hh) Imminent Attack/Mitigative Measures
  - 73.54 Cyber Security
  - 73.58 Safety/Security Interface
- Status of Regulatory Guidance

U.S.NRC
UNITED STATES NUCLEAR REGULATORY COMMISSION
Protecting People and the Environment

Security Rulemaking

- Part 73 Power Reactor Security Rulemaking (proposed rule published 10/06)
  - 50.54 (hh) Mitigative Strategies and Response Procedures for Potential or Actual Aircraft Attacks
  - 73.54 Protection of Digital Computer and Communication Systems and Networks
  - 73.55 Physical Security for Power Reactors
  - 73.56 Personnel Access Authorization Requirements for Nuclear Power Plants

Security Rulemaking (cont.)

- Part 73 Power Reactor Security Rulemaking (proposed rule published 10/06)
  - 73.58 Safety/Security Interface Requirements for Nuclear Power Plants
  - Appendix B to Part 73 – Section VI, Nuclear Power Reactor Training and Qualification for Personnel Performing Security Program Duties
  - Appendix C to Part 73 – Licensee Safeguards Contingency Plans

Status of Rulemaking

- FRN developed
- Begin formal concurrence on 6/16/2008
- Provide to EDO on 6/30/2008

ACRS Review for Rulemaking

- 50.54 (hh) Mitigative Strategies and Response Procedures for Potential or Actual aircraft Attacks
  - DG-50XX (July 2008)
- 73.54 Protection of Digital Computer and Communication Systems and Networks
  - DG 5022
- 73.58 Safety/Security Interface Requirements for Nuclear Power Plants
  - DG 5021 Safety/Security Interface

Draft Final Rule Text for 50.54 (hh) as of 6/4/2008

- Mitigative Strategies and Response Procedures for Potential or Actual aircraft Attacks
  - Contained in Appendix C of proposed rule
  - Moved to 50.54, Conditions of License
  - Supplemental rule published in Federal Register 4/10/2008
  - Comments received; incorporated into FRN
- Guidance to be developed from existing advisories, information (DG 50XX)

Draft Final Rule Text for 73.54 as of 6/4/2008

- Protection of Digital Computer and Communication Systems and Networks
  - Programmatic requirements for addressing cyber security
  - Included as part of DBT 73.1 issued March 2008
- DG 5022 Cyber Security Programs for Nuclear Facilities
  - Completed 6/1/08 (OUO)
  - In process of distribution to appropriate licensees (by 6/6/2008)

U.S.NRC
UNITED STATES NUCLEAR REGULATORY COMMISSION
Protecting People and the Environment

Draft Final Rule Text for 73.58 as of 6/4/2008

- Safety/Security Interface Requirements for Nuclear Power Plants
  - Requires coordination of potential adverse interactions between security activities and other plant activities
  - Addresses PRM 50-80, in part
- DG 5021 Safety/Security Interface
  - Published in Federal Register July 24, 2007
  - Public Meeting held; comments received & under consideration

Summary

- Security Rulemaking proceeding
- Supporting Regulatory Guidance for 50.54(hh) not developed
- Supporting Regulatory Guidance for 73.58 and 73.54 developed and drafts published or distributed

Security Rulemaking for Nuclear Power Plants

ACRS Presentation

June 4, 2008

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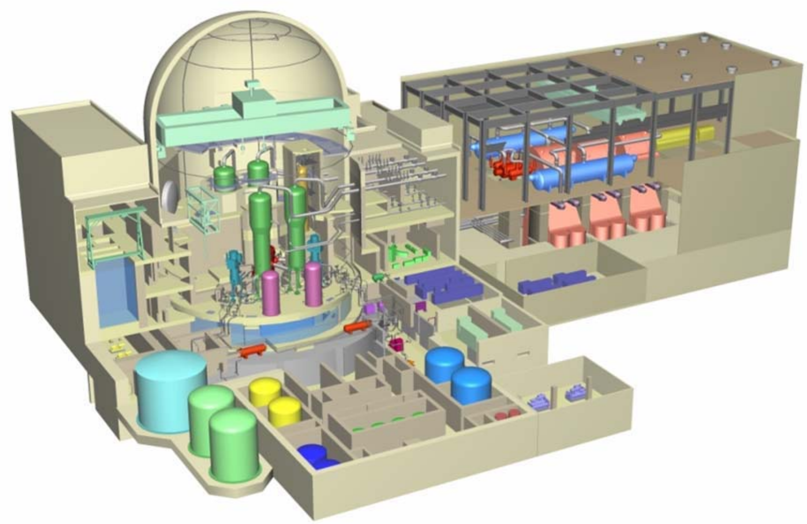
MITSUBISHI **US-APWR**

Opening Remarks
What is MHI and MHI Commitment



June 6, 2008

Kiyoshi Yamauchi
Executive Officer, Senior Vice President,
Nuclear Energy Systems Headquarters

Contents

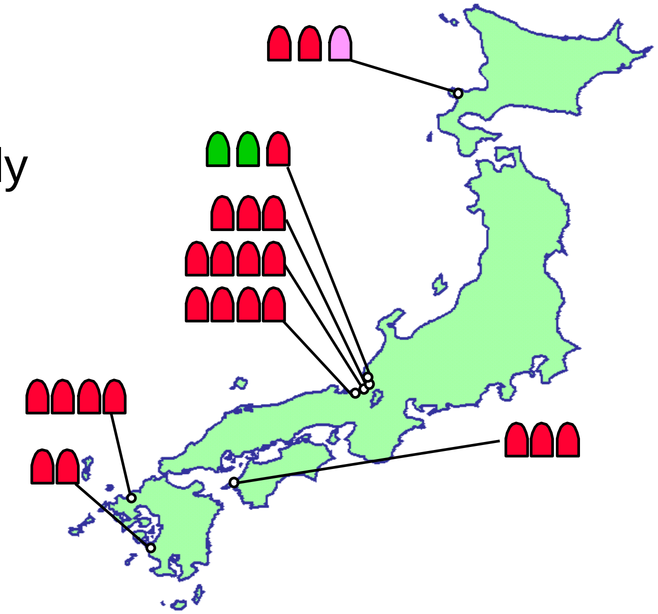
1. MHI Experience
2. MHI Technologies
3. MHI Commitment to Nuclear Safety
4. Conclusions

1. MHI Experience (1/2)
-Half Century of Mitsubishi Nuclear Engineering-

US-APWR

➢ Contribution to All of the 26 Japanese PWR Plants

- ✓ From 1960s first nuclear ship MUTSU, first PWR power plant MIHAMA 1 to the 21st century's latest APWRs
- ✓ New construction, major upgrade and replacement projects continued constantly even in the 80-90s "Nuclear Stagnation" era



- 23 units in operation
- 1 unit under construction
- 2 units (APWRs) in Licensing

➢ MHI Own Technology as MHI Core Competence

- ✓ Developed our own technology throughout a long history that have become MHI core competencies
- ✓ Established infrastructure for global deployment

1. MHI Experience (2/2) -Worldwide Component Supply-

✓Extensive experience of RV,VH, SG, RCP and Turbine exports

	Europe	Americas	Asia	Total
Reactor Vessels	1(1)	-	2	3(1)
Vessel Heads	3	16(5)	-	19(5)
Steam Generators	16 (8)	6(4)	-	22(12)
Reactor Coolant Pumps	-	-	8(4)	8(4)
Turbines	2	2	6(4)	10(4)

As of Jan. 2008
( ) : in progress

2. MHI Technologies (1/6) -Total Plant Capability-

➢ Total Plant Capability with “Single Point Responsibility”

- ✓ R&D, design and engineering, manufacturing, construction, maintenance services, and fuel supply

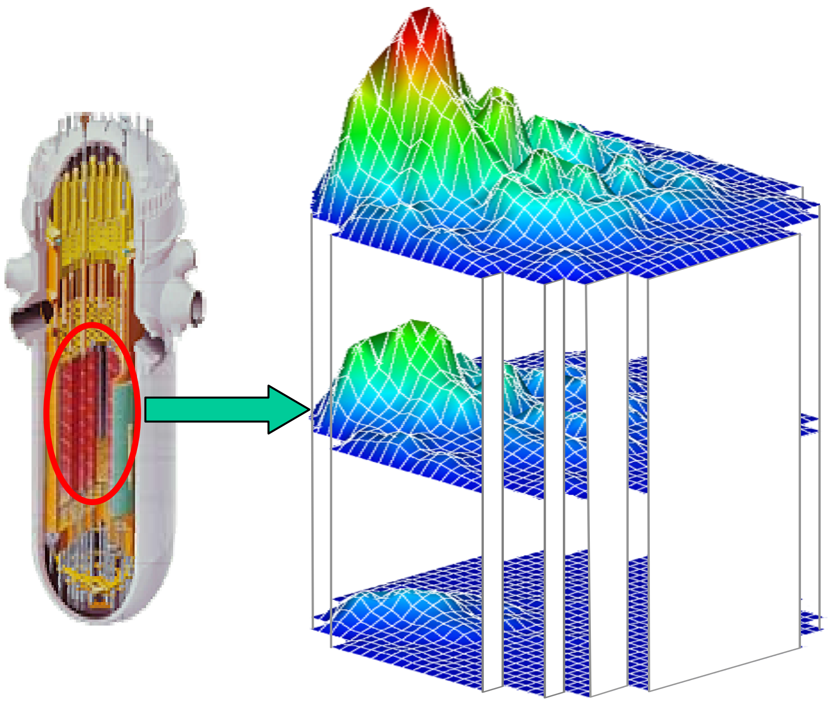
➢ Globalized Quality Assurance

- ✓ Supporting export of nuclear components, e.g., steam generators, reactor vessels, reactor vessel heads or turbines …

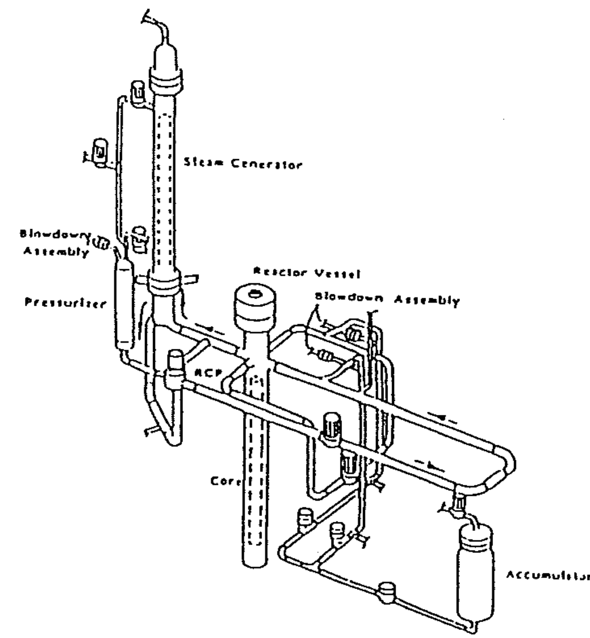
2. MHI Technologies (2/6) -Reactor Core Design & Safety Analysis-

➢ State-of-the-Art Reactor Core Design and Safety Analysis

- ✓ Advanced analytical program
- ✓ Verification using demonstration test facilities
- ✓ Licensing support



Power distribution after the rod ejection from 3-D calculation

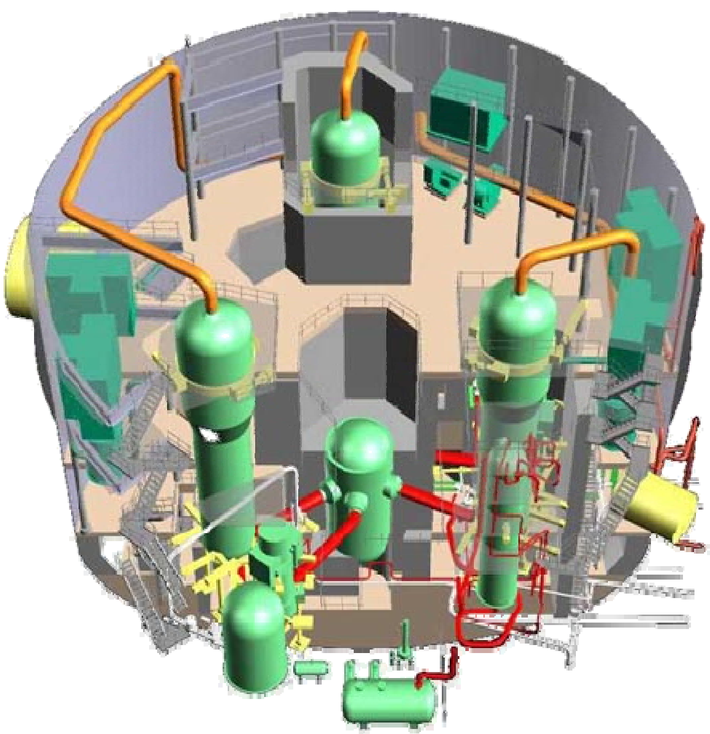


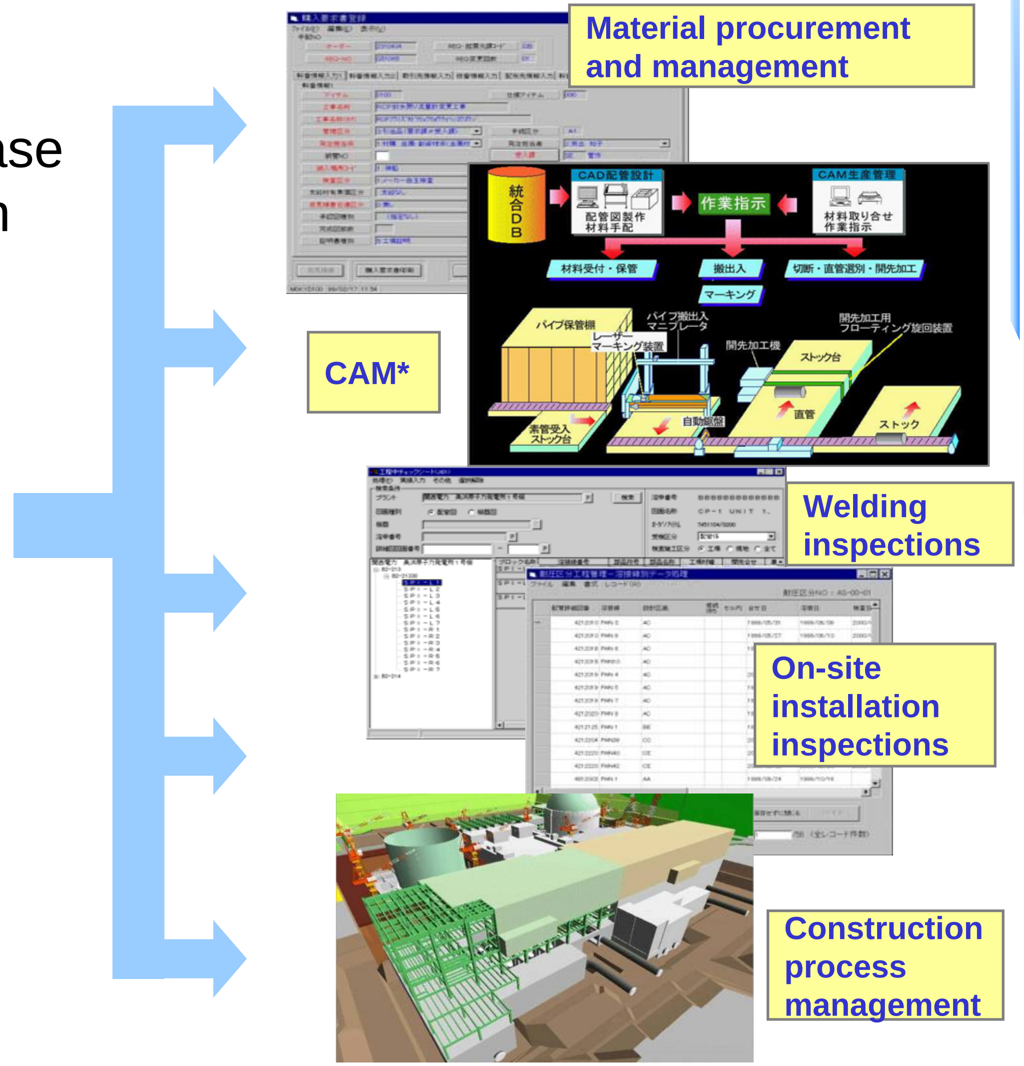
A demonstration test facility for the LOCA analysis methodology

2. MHI Technologies (3/6)
-Plant Engineering and Procurement-

➢ 3D-CAD :

- ✓ Integrated common database from design to construction



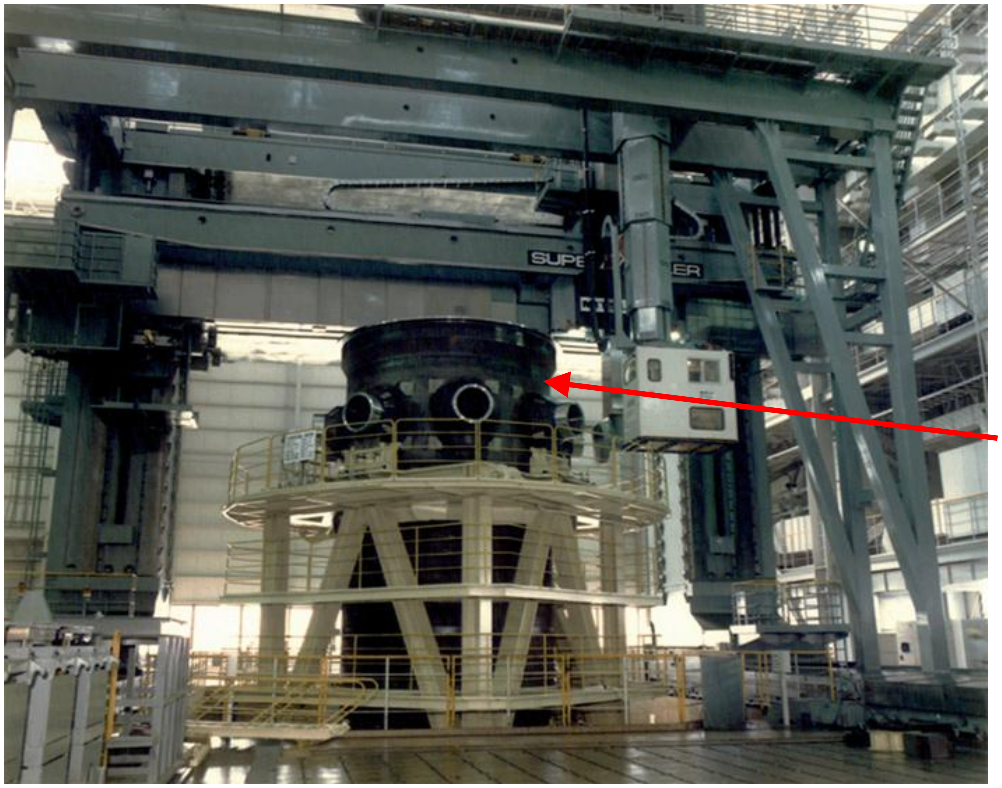


•CAM : Computer-Aided Manufacturing

2. MHI Technologies (4/6) -Manufacturing-

➢Super-large combined machine tool

"Super Miller"



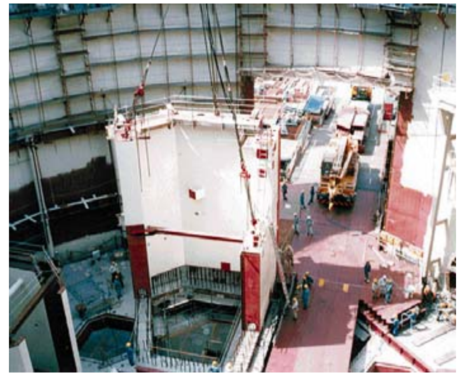
✓ High-accuracy, high-quality processing in upright installation position

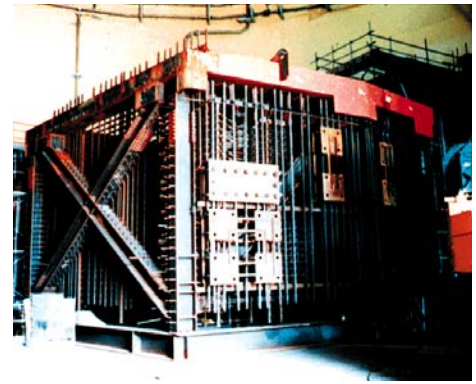
Reactor Vessel

2. MHI Technologies (5/6)
-Plant Construction-

➢Various On-Site Work Reduction Techniques

- ✓ Super large-capacity cranes
  (On-site containment
  Welding and formation)
- ✓ Comprehensive coordination of civil & construction work





➢Module Utilization

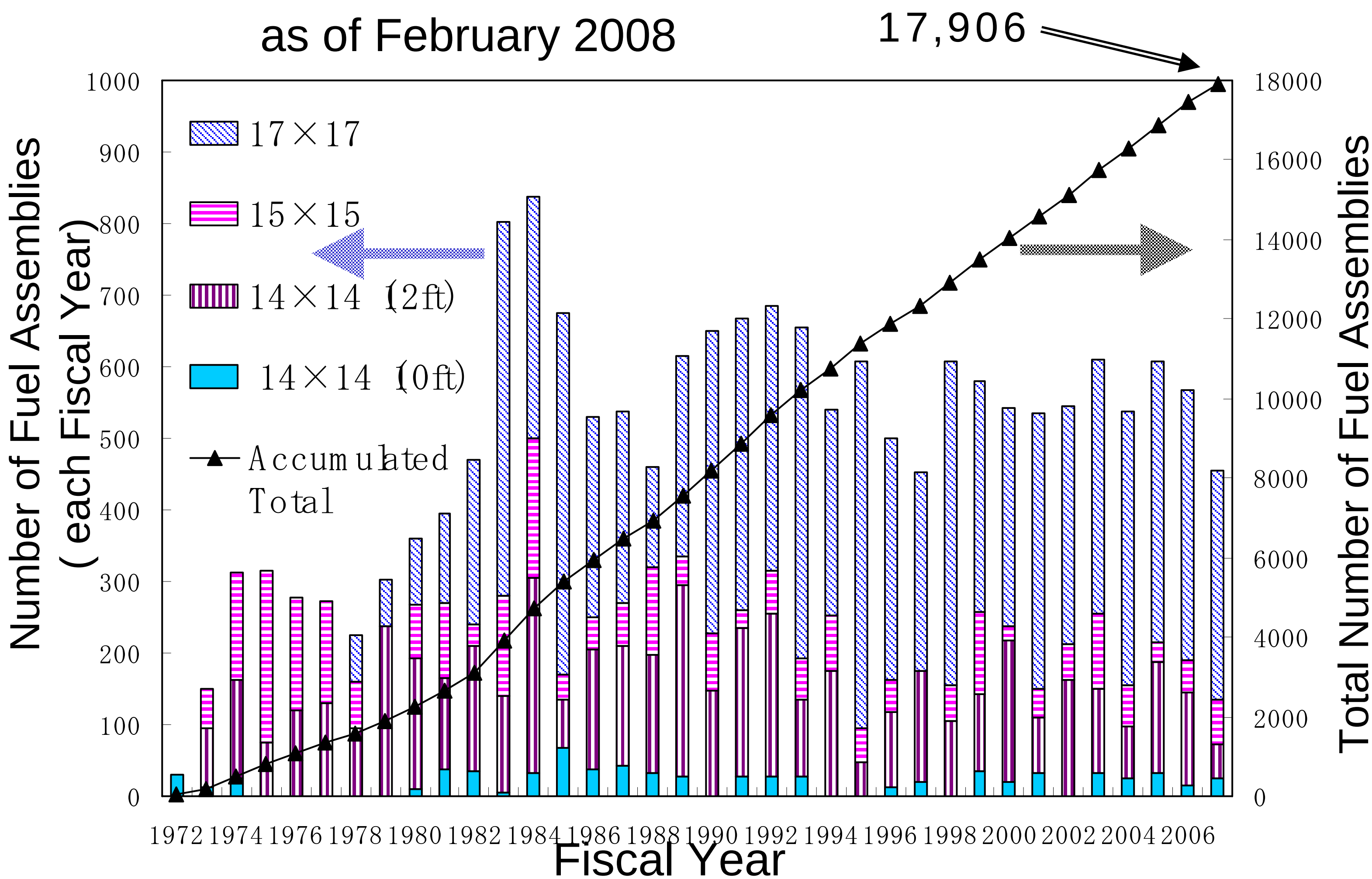
- ✓ Internal structures using SC (-Left)
  (Steel plate reinforced concrete)
- ✓ Large prefabricated blocks (-Right)

Typical achievements
(1st Concrete to fuel loading)

2 loop	: 34.5 months
3 loop	: 37.5 months
4 loop	: 40.0 months

2. MHI Technologies (6/6) -PWR Fuel Supply-

- **Leading edge technology based on abundant manufacturing / irradiation experience**



3. MHI Commitment to Nuclear Safety

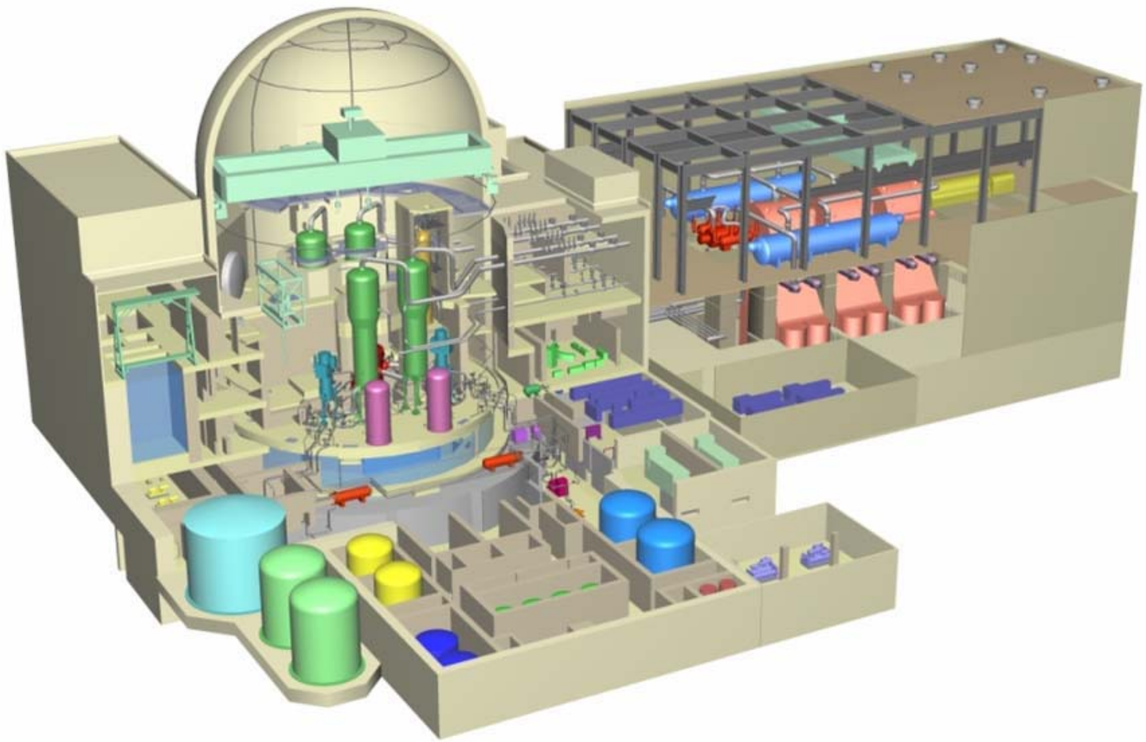
- **The US-APWR is in compliance with the U.S. regulatory requirements, guidance, and industry codes and standards**
- **The US-APWR design approach**
  - ✓ Use of proven, accepted technologies with improvements to enhance safety
  - ✓ Enhanced safety design
    - Highly reliable prevention function
    - Well-established mitigation systems with active safety functions and passive safety functions
    - Functions against beyond design basis accidents

4. Conclusions

- MHI is committed to providing the highest quality global nuclear products and services using its core competencies and supported by outside strategic alliances.
- MHI infrastructure, i.e., various technologies based on the expertise, know-how, human resources and quality assurance systems have been developed and maintained throughout MHI's long history.
- The US-APWR will demonstrate the commitment to quality and safety worldwide.

MITSUBISHI US-APWR

DESIGN FEATURES



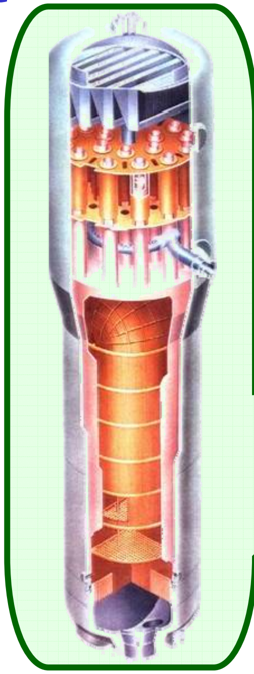
June 6, 2008

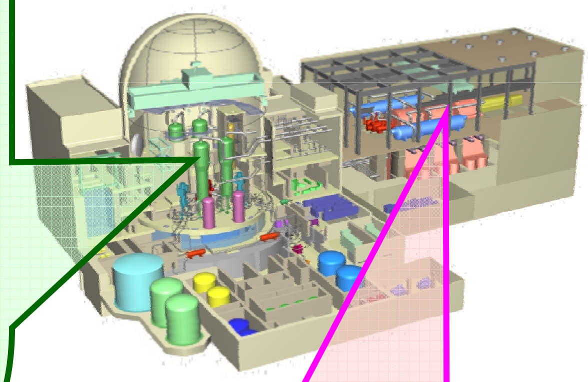
MITSUBISHI HEAVY INDUSTRIES, LTD.

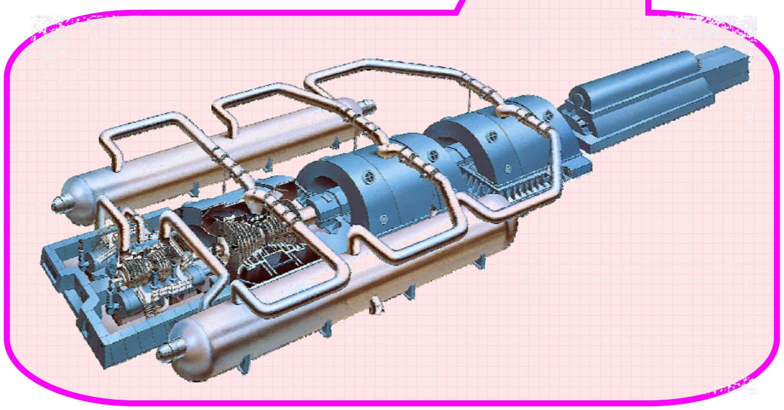
Contents

1. What is the US-APWR

- **The world largest class output 1,700 MWe based on proven, fully tested technologies**
  - ✓ **Thermal efficiency: 39%**
  - ✓ **Increased SG heat transfer area ($91,500\ ft^2$/unit) with triangular lattice of SG tubes**
  - ✓ **High performance steam-water separators generate high quality steam**
  - ✓ **High performance LP-turbine system with 70-inch class integral shroud blades**



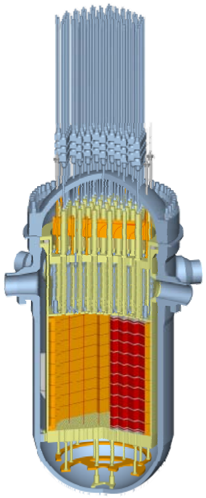


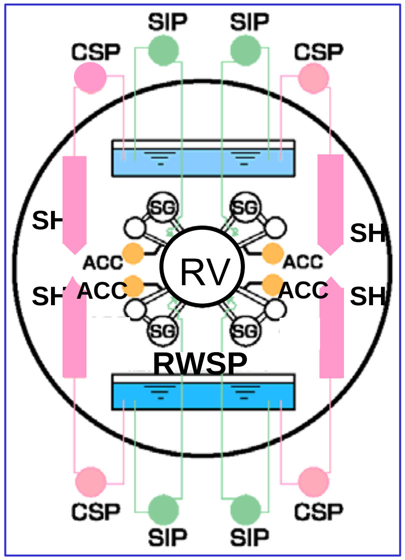


- **US-APWR meets U.S. utility's requirements and provides enhanced safety with features that address R.G. 1.206**

➢ Conformance with the URD* for safe, reliable and economical plant

- ✓ Top mounted ICIS eliminating penetrations at the RV bottom
- ✓ Full 4-train safety systems, with an optimized mix of passive and active systems allowing On-Line Maintenance (OLM)
- ✓ 14ft fuel creates additional thermal margin, achieving 24-month extended cycle operation to enhance fuel economy
- ✓ Full digital I&C technology with Japanese domestic operating experience
- ✓ Due consideration for protection against airplane crash and long-term containment integrity to mitigate postulated severe accidents



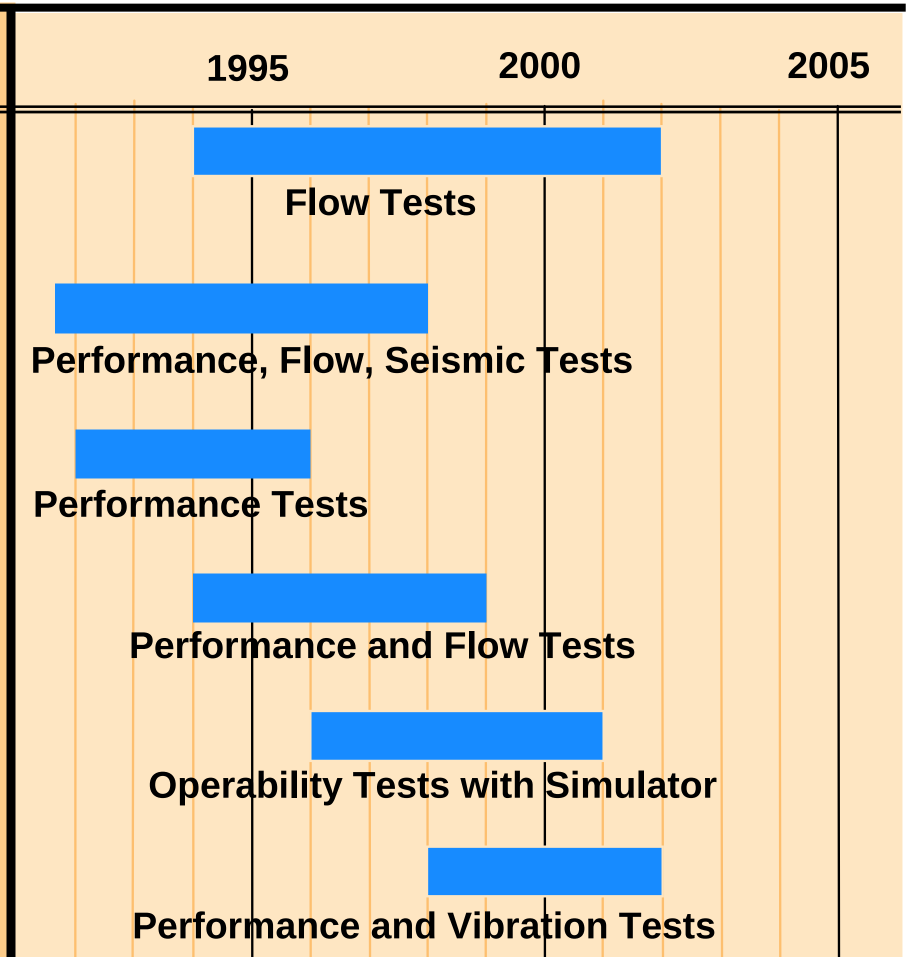


*URD : Utilities Requirements Document

Proven, Fully Tested Technologies

Verification of Advanced Features for APWR

- Reactor Internals and Neutron Reflector
- Compact SG and Improved Separator
- Advanced Accumulator
- High-performance RCP
- Advanced I&C System
- Turbine




Reactor Flow Test


SG Separator Test


LP Turbine Test

Comparison of Output & Main Components

		US Current 4 Loop	APWR	US-APWR
Electric Output		1,180 MWe	1,538 MWe	1,700 MWe Class
Core Thermal Output		3,411MWt	4,451 MWt	4,451 MWt
Steam Generator	Model	54F	70F-1	91TT-1
	Tube size	7/8 in.	3/4 in.	3/4 in.
Reactor Coolant Pump	Model	93A-1	100A	100A
Turbine	LP last-stage blade	44 in.	54 in.	70 in. class

- **APWR**
  - ✓ 1,538 MWe output is achieved by large capacity core and large capacity main components such as SG, RCP, turbine, etc.
- **US-APWR**
  - ✓ 1,700 MWe class output is achieved by a 10% higher efficiency than APWR.
    - Same core thermal output as APWR
    - High-performance, large capacity steam generator
    - High-performance turbine

Comparison of Fuel, Core & RIs

		US Current 4 Loop	APWR	US-APWR
Core Thermal Output		3,411 MWt	4,451 MWt	4,451 MWt
Core and Fuel	No. of FA	193	257	257
	Fuel Lattice	17 x 17	17 x 17	17 x 17
	Active Fuel Length	12 ft	12 ft	14 ft
Reactor Internals (RIs)		Baffle/former structure	Neutron Reflector	Neutron Reflector
In-core Instrumentation		Bottom mounted	Bottom mounted	Top mounted

- **APWR**
  - ✓ Large capacity core by increasing number of fuel assemblies
  - ✓ Installation of neutron reflector to enhance reliability and fuel economy
- **US-APWR**
  - ✓ Low power density core using 14 ft fuel assemblies with the same reactor vessel as APWR to enhance fuel economy for 24- month operation
  - ✓ Top mounted ICIS enhances reliability and maintainability of reactor vessel

Comparison of Systems, CV and I&C

			US Current 4 Loop	APWR	US-APWR
Safety Systems	Trains	Electrical	2 trains	2 trains	4 trains
		Mechanical	2 trains	4 trains	4 trains
	Systems	HHSI pump	100% x 2	50% x 4(DVI)	50% x 4(DVI)
		LHSI pump	100% x 2	-	-
		ACC	4	4 (Advanced)	4 (Advanced)
	RWSP		Outside CV	Inside CV	Inside CV
Containment Vessel			PCCV	PCCV	PCCV
I & C	Control Room		Conventional	Full Digital	Full Digital
	Safety I&C		Conventional		
	Non-Safety I&C		Full Digital		

➢ **APWR**
- ✓ **Enhanced safety by simplified and reliable safety systems**
  - **Mechanical 4 train systems with direct vessel injection design**
  - **Elimination of LHSI pump by utilizing advanced accumulators**
  - **Elimination of recirculation switching by in-containment RWSP**

➢ **US-APWR**
- ✓ **Enhanced safety by 4 train safety electrical systems**
- ✓ **Enhanced on-line maintenance capability**

2. Fuel and Core Design

US-APWR

Fuel Assembly

- Flexible Operation
- Enhanced Economy
- Improved Reliability

17x17 Fuel rod array

14 ft Fuel active length

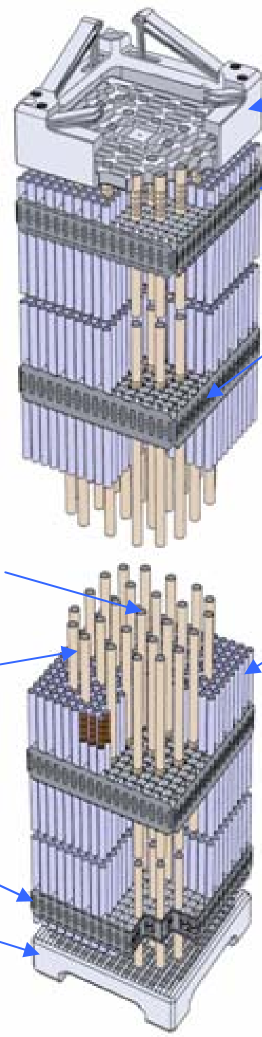
In-core instrumentation guide tube

Control rod guide thimble

Bottom grid spacer

Bottom nozzle

Anti-debris design with built-in filter



Top nozzle

Top grid spacer

Intermediate grid spacer

High DNB performance design

Shorter grid spacing with 11 grids

Fretting resistant spring

Zircaloy-4

Fuel rod

Higher density pellet (97%TD)

Corrosion resistant cladding Material (ZIRLO™)

Higher gadolinia content pellet (10wt%)

Large plenum volume

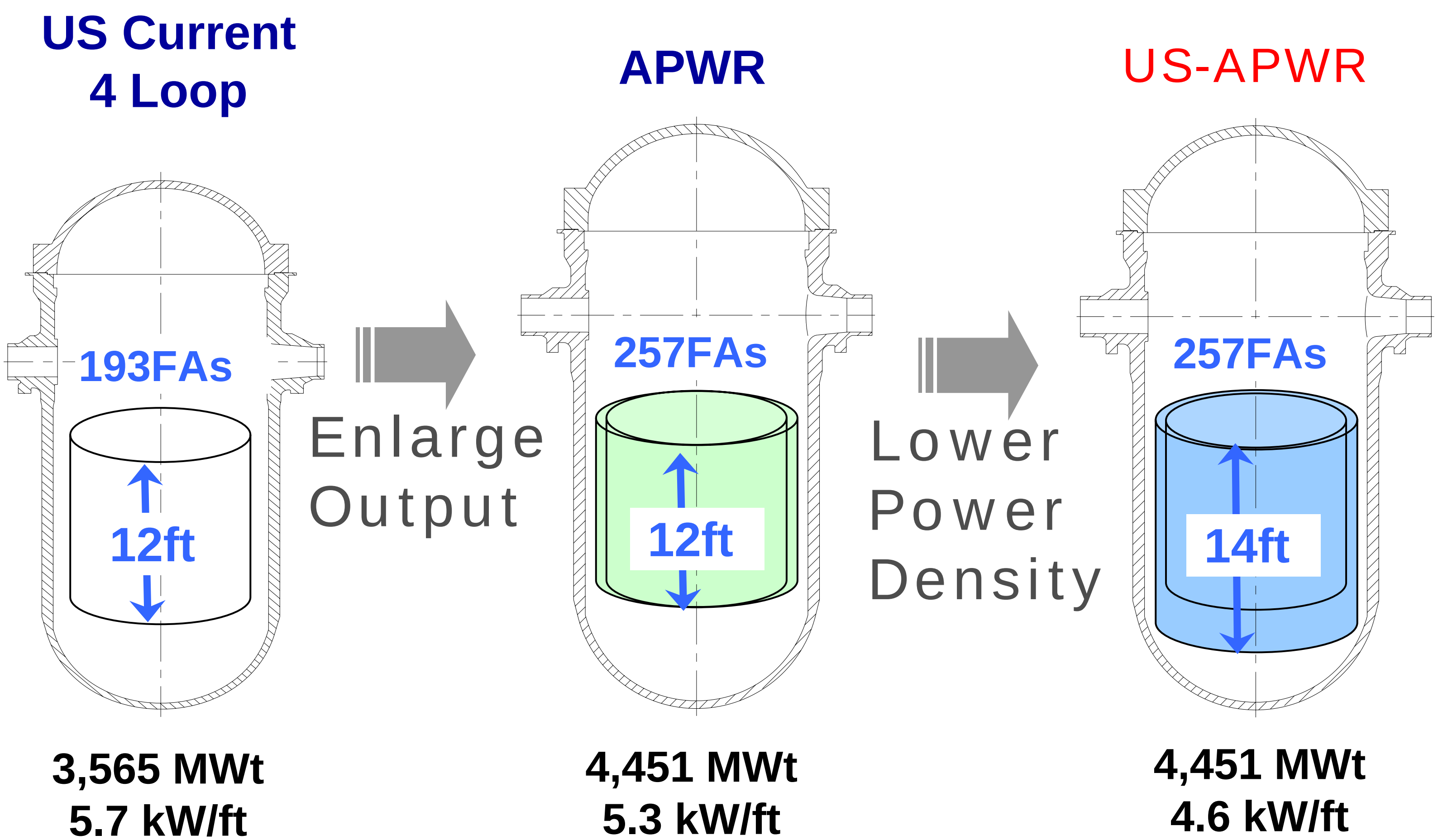
Lower power density

Fuel Design

	US Current	APWR	US-APWR
Fuel Assembly			
Fuel Rods Array in Fuel Assembly	17 x 17	17 x 17	17 x 17
Number of Fuel Rods per Fuel Assembly	264	264	264
Number of Control Rod Guide Thimbles	24	24	24
Number of in-core Instrumentation guide tube	1	1	1
Number of Spacer Grids	8 / 10	9	11
Fuel Rod			
Outside Diameter	0.374 in.	0.374 in.	0.374 in.
Cladding Thickness	0.022 in.	0.022 in.	0.022 in.
Active Fuel Length	12 ft / 14 ft	12 ft	14 ft
Enrichment	Max. 5 wt%	Max. 5 wt%	Max. 5 wt%
Gadolinia Content	Max. 8 wt%	Max. 10 wt%	Max. 10 wt%
Pellet Density	95 % TD	97 % TD	97 %TD
Material			
Cladding	ZIRLO™	MDA / ZIRLO™	ZIRLO™

Thermal Power & Density

Large thermal output + Low power density



Core Design

➢ Low power density for flexible operation

✓ **Longer cycle operation for a given cycle burnup**

- **24-month 2-batch cycles sustainable with**
  - **U235 enrichment < 5 wt%**
  - **Maximum rod burnup ≤ 62 GWd/t**

✓ **Large thermal margin ($F_{\Delta H}$ ~1.7 F_Q ~2.6)**

➢ Negative reactivity feedback

✓ **Doppler feedback against rapid reactivity insertion**

✓ **Moderator temperature coefficient with negative feedback effect during operation**

➢ Steel neutron reflector

✓ **Reduced neutron leakage to enhance neutron economy**

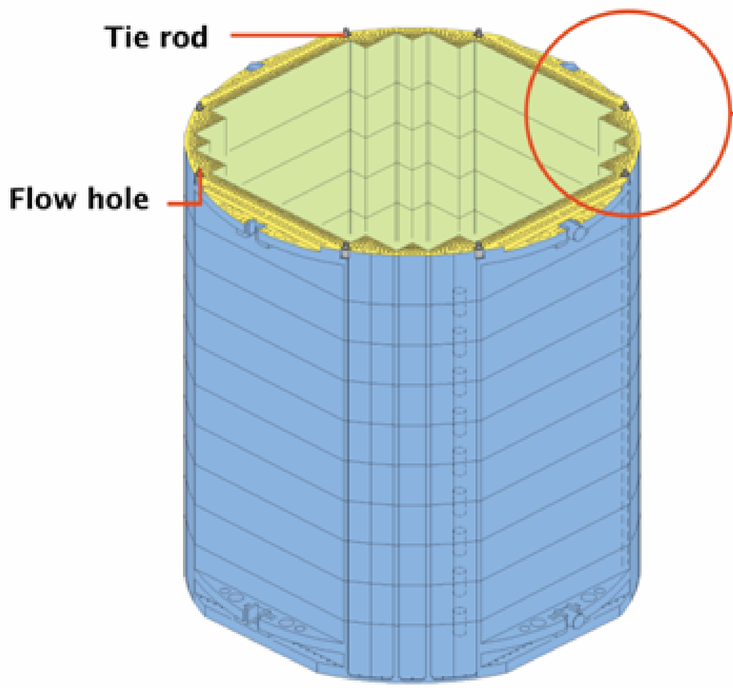
Neutron Reflector

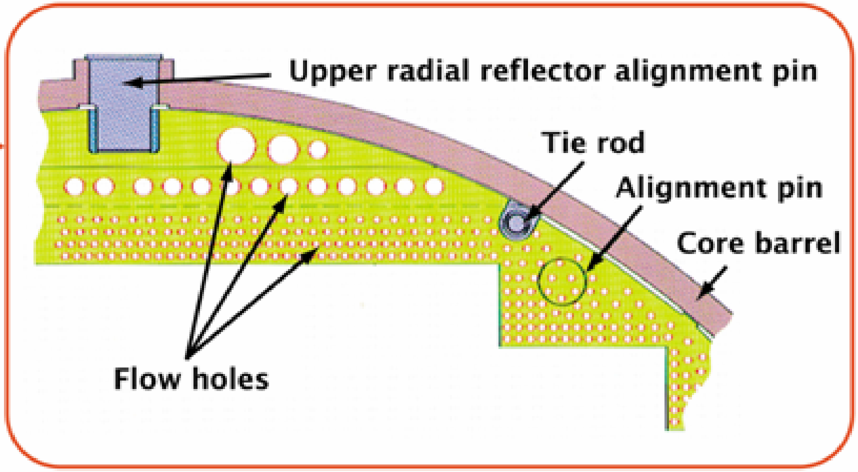
Improved reliability

- Significantly simplified and reliable structure
  - Number of bolts reduced significantly and located only at out of core region
  - No welds

Enhanced performance

- Reduced neutron exposure rate
  - 1/3 of current 4 loop design without neutron shield





Methodology and Codes

➢Fuel Design

✓ FINE

- Fuel rod design code developed by MHI
- Significant post irradiation examinations and out-of-pile test
- Topical report on verification and applicability to US-APWR fuel is under NRC review

✓ FINDS

- Fuel assembly seismic analysis code developed by MHI
- Topical report on verification and applicability to US-APWR fuel is under NRC review

Methodology and Codes (cont.)

➢ Nuclear Design

✓ **PARAGON/ANC**

- **PARAGON: Heterogeneous 2D lattice physics code**
- **ANC: 3D two-group diffusion core simulator code**
- **Applicability of PARAGON/ANC to PWR nuclear design was approved by NRC**

➢ Thermal and Hydraulic Design

✓ **VIPRE-01M / WRB-2**

- **Core subchannel analysis code and DNB correlation for DNBR evaluation**
- **Widely used in US PWRs**
- **Topical report on the applicability to US-APWR fuel is under NRC review**

✓ **RTDP : Revised thermal design procedure**

- **Statistical DNBR evaluation methodology was approved by NRC**

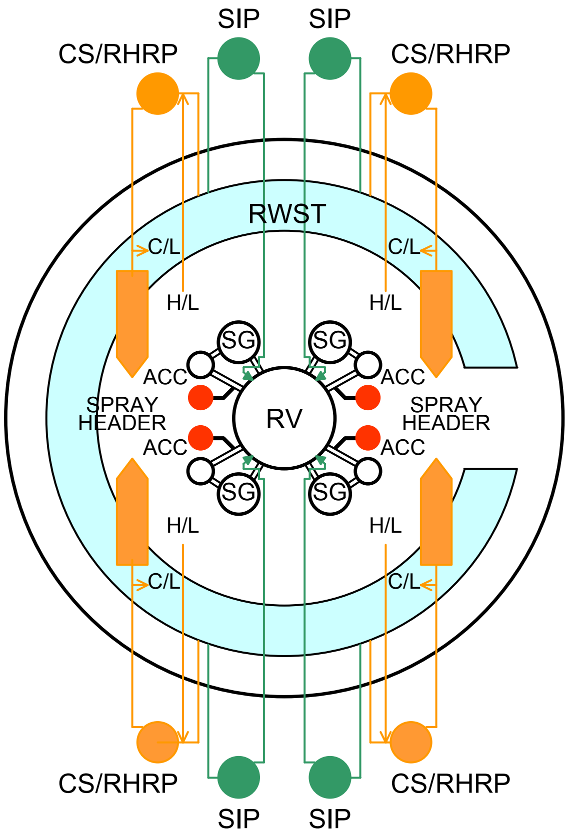
3. System Design & Safety Features

Reactor Coolant System

- **Larger main components**
  - ✓ Larger diameter and height of Reactor Vessel
  - ✓ Larger heat transfer area in SG contributes high efficiency due to high steam pressure
  - ✓ Larger reactor coolant flow rate of RCP with 8000 HP motor
- **Enhanced plant control**
  - ✓ Larger Pressurizer volume assures greater margin for transients

Specifications	US Current 4 Loop Plant	US-APWR
Core thermal output	3,565 MWt	4,451 MWt
SG Heat transfer area	55,000 ft^2	91,500 ft^2
Reactor Coolant Flow	93,600 gpm	112,000 gpm
Pressurizer Volume	1,800 ft^3	2,900 ft^3

ECCS and CSS/RHRS



➢ High Reliability

- ✓ **4 train configuration (50% x 4 for large break LOCA)**
- ✓ **In-containment RWSP (eliminate recirculation switchover)**

➢ Simplification

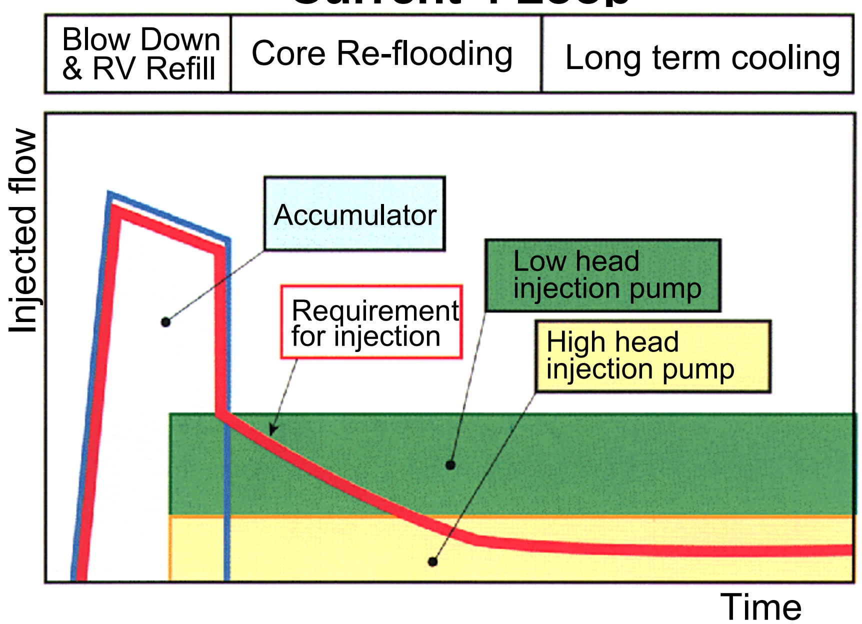
- ✓ **Advanced accumulators (ACC) (Integrated function of low head injection system)**
- ✓ **ECCS train includes an accumulator and high head injection system**
- ✓ **Direct vessel injection (no inter-connection between trains)**
- ✓ **Common use of CSS and RHRS**

ECCS and CSS/RHRS (cont.)

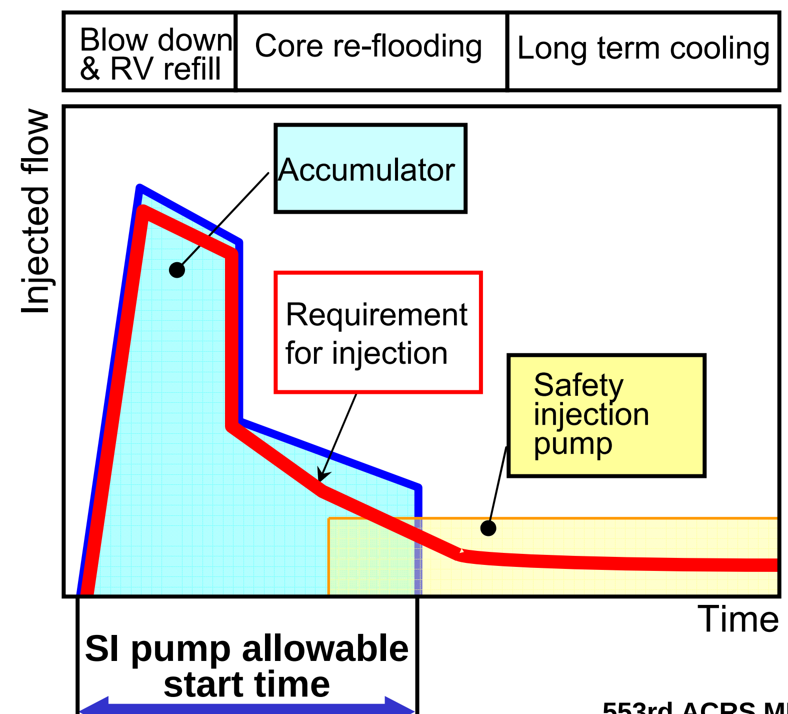
Advanced Accumulator

- ✓ **Passive switching of injection flow rate**
- ✓ **Integrated function of low head injection system**
- ✓ **Long accumulator injection time allows more time for safety injection pump to start (allows use of gas turbine generator for EPS)**
- ✓ **Topical report on Advanced Accumulator is under NRC review**

Current 4 Loop

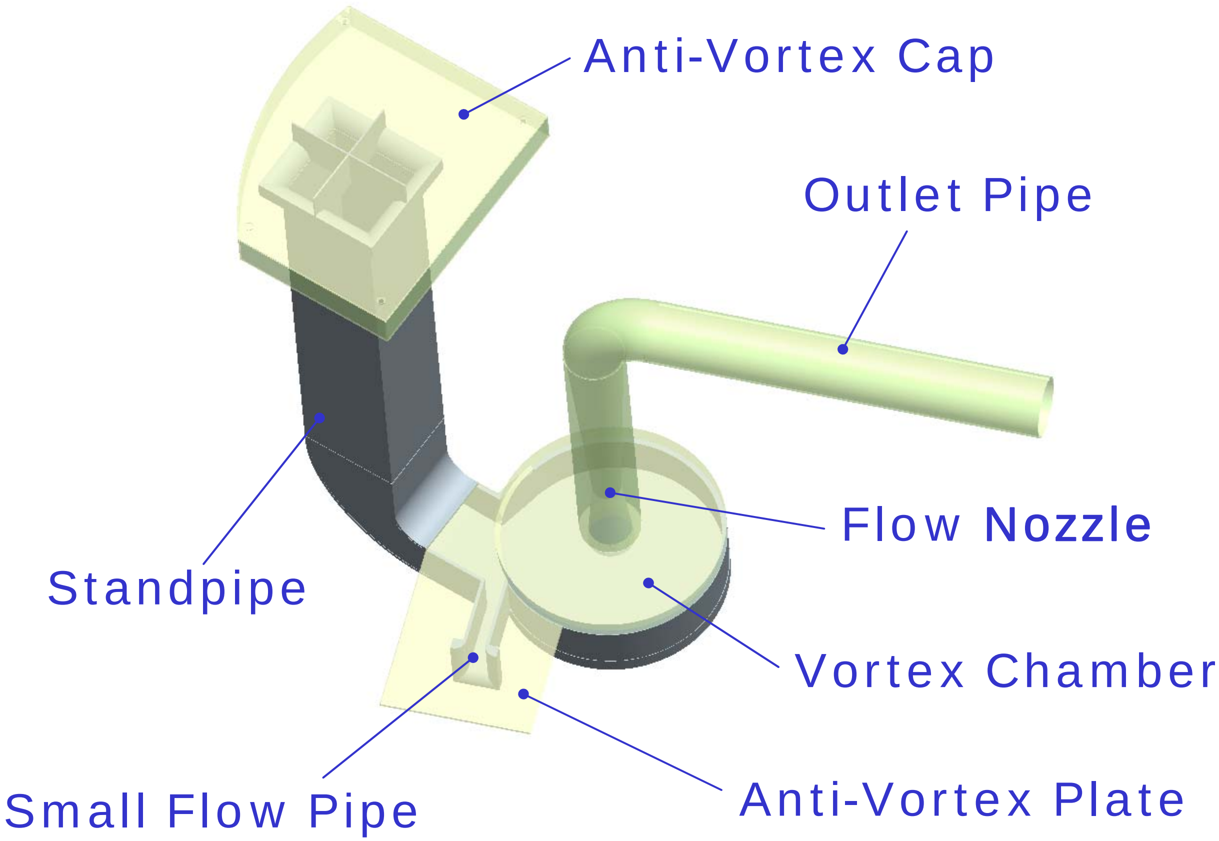


US-APWR

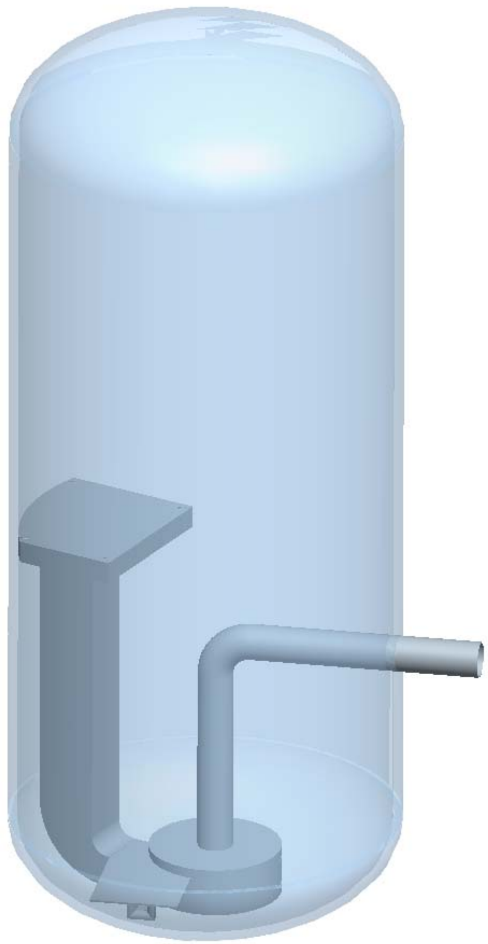


ECCS and CSS/RHRS (cont.)

Structure of Advanced Accumulator



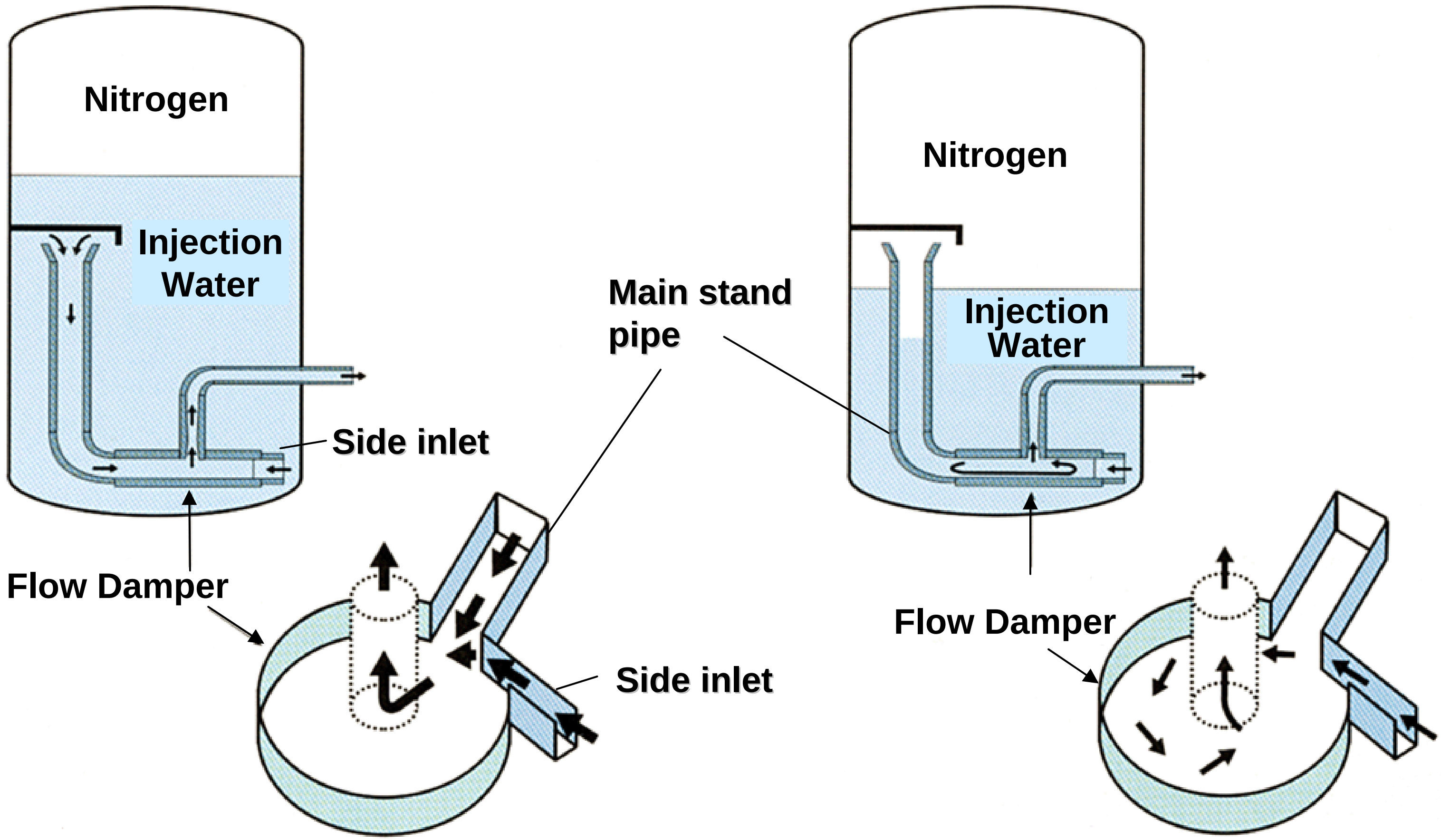
Flow Damper



ECCS and CSS/RHRS (cont.)

Mechanism of Advanced Accumulator

Flow damper passively switches the flow rate



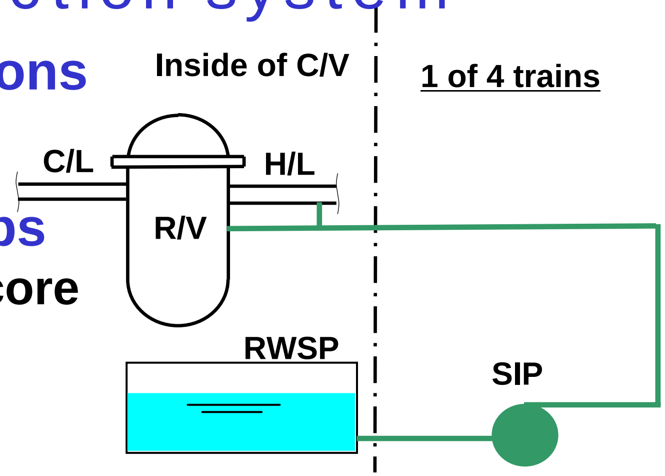
Large Flow Rate

Reduced Flow Rate

ECCS and CSS/RHRS (cont.)

Design feature of high head injection system

- 4 independent trains without interconnections between trains
- Sufficient capacity of safety injection pumps
  - ✓ Meets the safety injection requirement for core reflooding stage

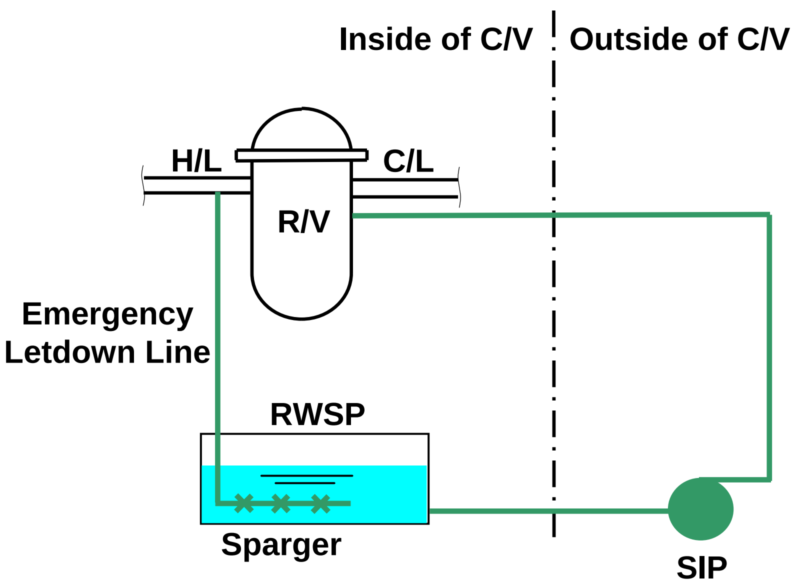


Item	US Current 4 Loop	US-APWR	Reason and/or Advantage
Trains	2 trains	4 trains	•Enhanced reliability •Achieve OLM under single failure
High head Injection	Loop injection 2 SIP + 2 CH/SIP	DVI 4 SIP	•No interconnection between trains
Refueling Water Storage Pit	Outside CV	Inside CV	•Eliminate recirculation switchover

ECCS and CSS/RHRS (cont.)

Feed & Bleed for Boration to Achieve Safe Shutdown

➢ Design Features

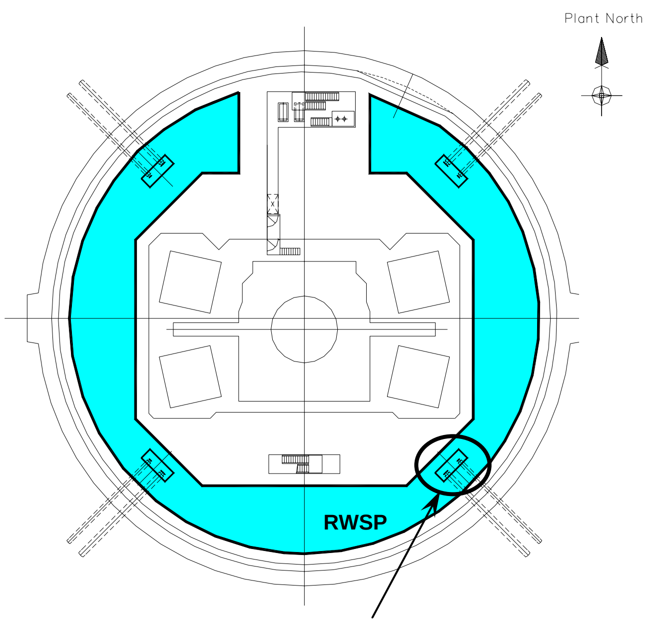


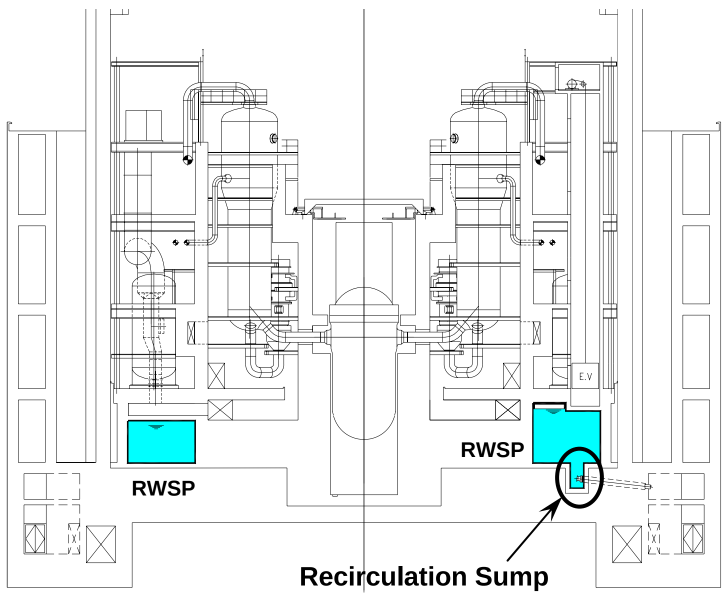
- ✓ Emergency Letdown Lines are installed from H/L to RWSP
- ✓ In Safe Shutdown operation, emergency boration source is RWSP
- ✓ The borated water is injected by Safety Injection pump
- ✓ The volume control of RCS is achieved by Feed & Bleed with SIP and Emergency Letdown Line

ECCS and CSS/RHRS (cont.)

In-Containment Refueling Water Storage Pit

- Located at the lowest part of containment
- Provides a continuous suction source for both safety injection and CS/RHR pumps
  (Eliminates switchover of suction source)
- 4 recirculation sumps are installed

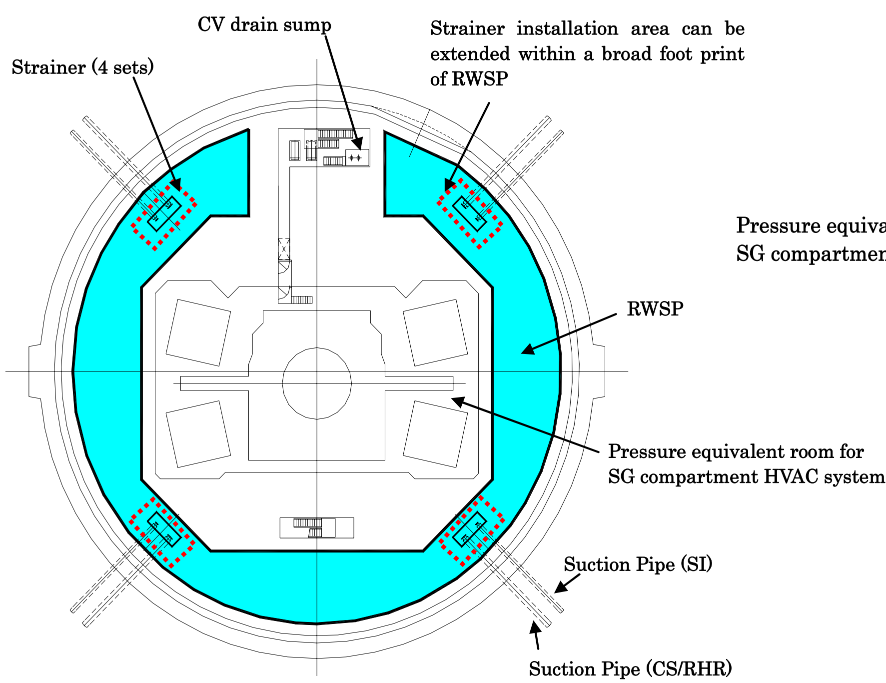




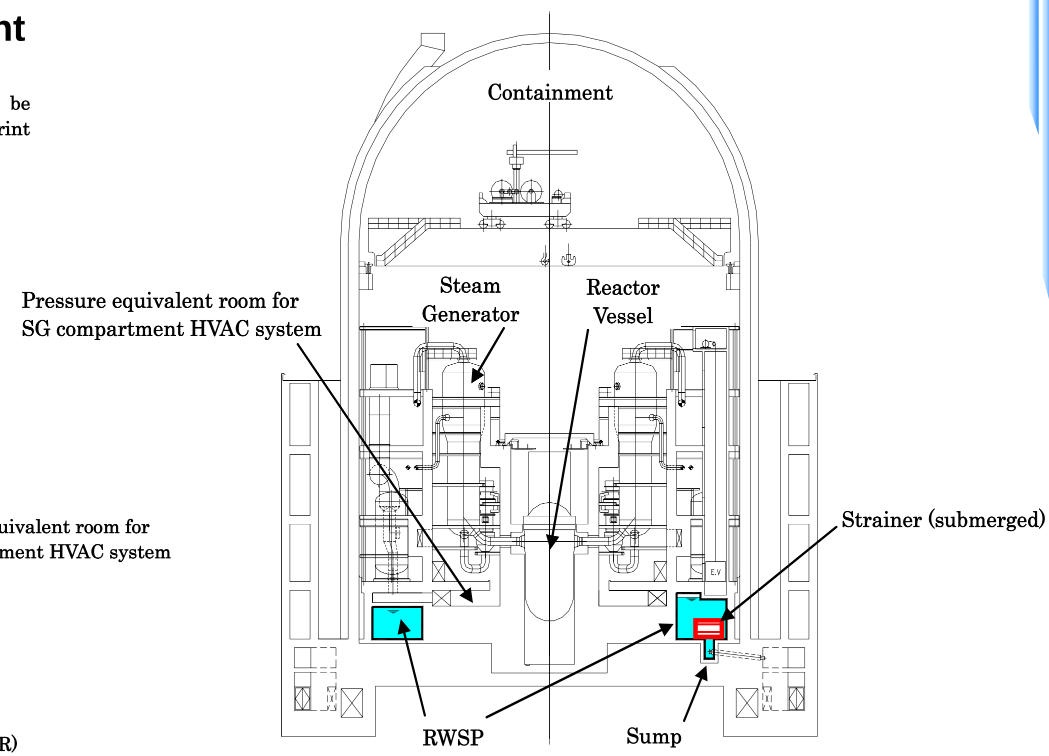
ECCS and CSS/RHRS (cont.)

Conservative Countermeasures for GSI-191

- ➢ **Robust arrangement of sump strainer systems**
  - ✓ **4 redundant passive strainer system**
  - ✓ **Sufficient surface area of strainer**
- ➢ **Extremely low debris sources**
  - ✓ **Use of reflective metal insulation is maximized, minimal fibrous insulation is used**
  - ✓ **Cal-Sil insulation is excluded in CV**
- ➢ **Avoid using problematic chemicals and substances**
  - ✓ **NaTB used as a buffer agent**



Plan View



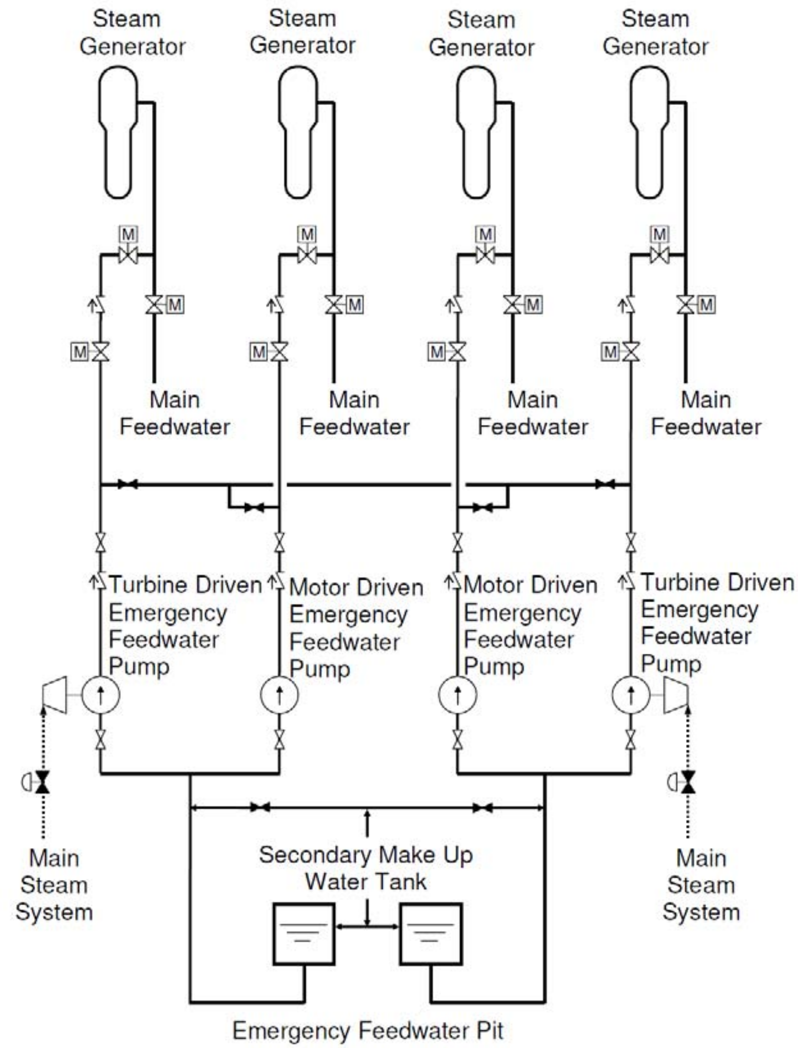
Emergency Feedwater System

➢ Design concept of the EFWS

- ✓ Achieve high reliability with simplified systems
- ✓ Allow On-Line Maintenance assuming single failure

➢ Feature of the EFWS

- ✓ Independent 4 train system
- ✓ 2 safety grade water sources
- ✓ Diverse power sources for the pumps
- ✓ Cross connections in the inlet and outlet of the pumps (normally isolated)



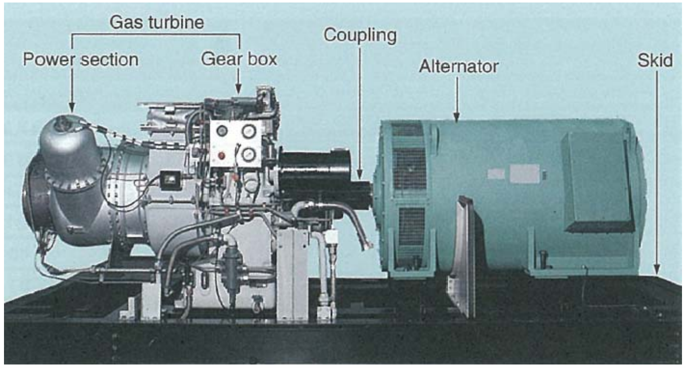
Emergency Feedwater System (cont.)

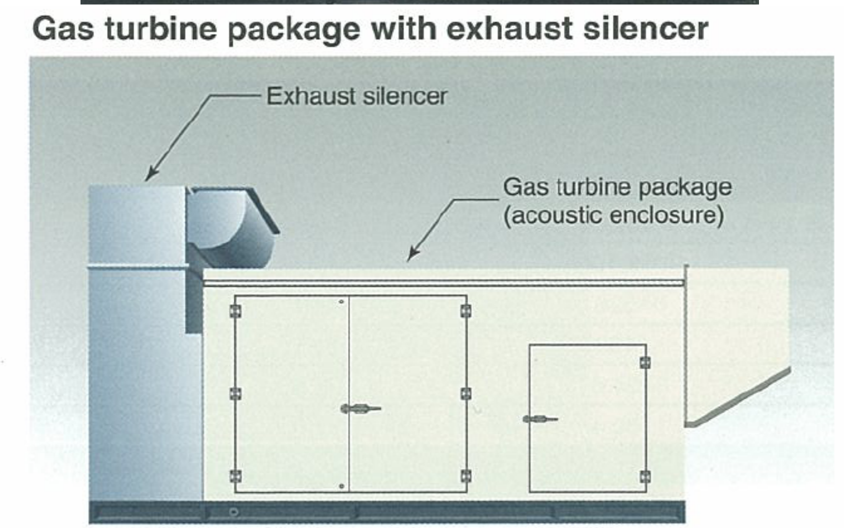
- **4 train configuration**
  - ✓ 4 pumps with diverse power sources
    - 2 motor-driven emergency feedwater pumps (50% x 2)
    - 2 turbine-driven emergency feedwater pumps (50% x 2)
  - ✓ Cross connected discharge of the pumps allows On-Line Maintenance (OLM)
- **2 safety grade independent feedwater sources**
  - ✓ Two emergency feedwater pits (50 % x 2)
  - ✓ Cross connected inlet of the pumps backs up each feedwater source

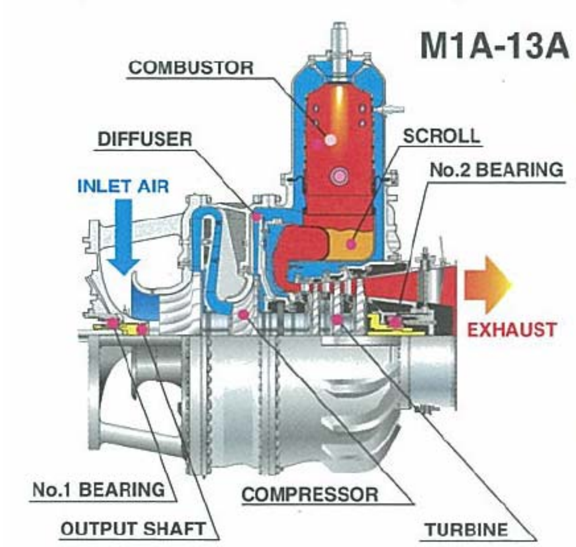
Item	US Current 4 Loop	US-APWR	Reason and/or Advantage
System Configuration	2 trains	4 trains	A pump is allowed OLM under the single failure
Emergency Feedwater Pump	M/D EFWP: 2 T/D EFWP: 1	M/D EFWP: 2 T/D EFWP: 2	Diverse power sources
Emergency Feedwater Source	1	2	2 independent pits (backup available)

Gas Turbine Generator for EPS (1/2)

- Gas-Turbine Generators are used as the Emergency Power Source
- Gas-Turbine Technical Report is under NRC review
  - ✓ Gas Turbine is a very simple rotating engine with few components
  - ✓ A water cooling system is not required
  - ✓ Rated output : 4,500 kW







Gas Turbine Generator for EPS (2/2)

US-APWR

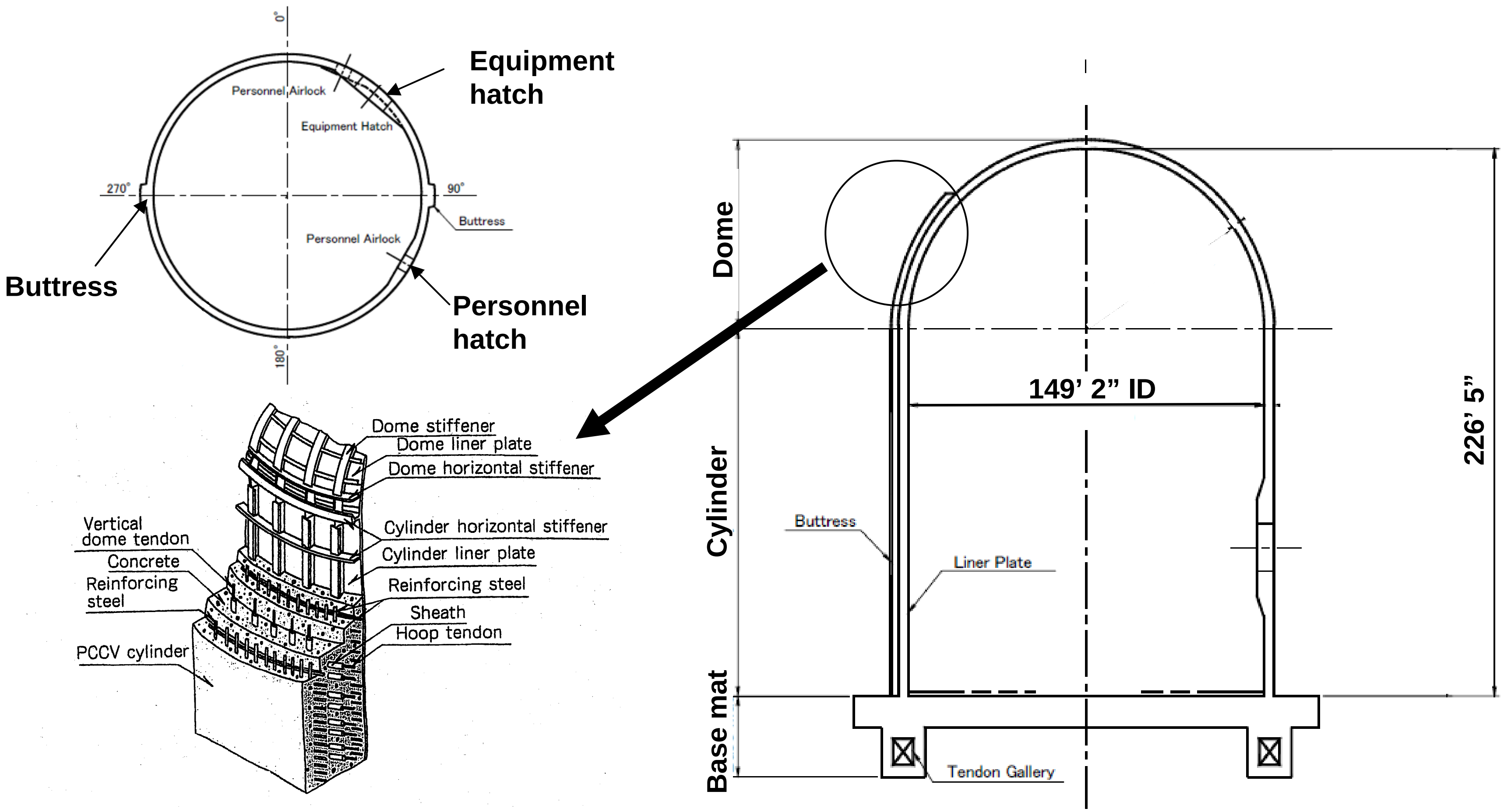
✓ **GT/G has been selected based on reliability and maintainability improvements when compared to DG**

	Gas Turbine Generator	*Diesel Generator*
Space	*Compact*	*Large*
Cooling Water	*Not Required*	*Required*
Routine Maintainability	*1/3 the parts of a DG*	*Complex*
Large Scale Overhaul	*Once or twice during plant life*	*Periodic Overhaul Required*
Reliability (failure/demand)	*10^{-3} based on Japanese experience*	*10^{-2}*
Starting Time	*40 sec*	*10 sec*

✓ Longer start time of GT/G is accommodated by the Advanced Accumulator design of US-APWR which allows 100 sec

PCCV

- Robust and reliable Pre-Stressed Concrete Containment Vessel with steel liner is used in US-APWR



Methodology and Codes for Safety Analysis

➢Large Break LOCA

- ✓ WCOBRA/TRAC code and ASTRUM methodology
- ✓ Approved by NRC
- ✓ US-APWR design features modeled:
  - Advanced Accumulator
  - Direct Vessel Injection
- ✓ Topical report on applicability to US-APWR is under NRC review

➢Small Break LOCA

- ✓ Appendix-K version of M-RELAP5 code
- ✓ Equivalent to RELAP5/MOD3.2 widely used in US
- ✓ US-APWR design features modeled:
  - Advanced Accumulator
  - Direct Vessel Injection
- ✓ Topical report on applicability to US-APWR is under NRC review

Methodology and Codes (Cont.)

➢LOCA Mass and Energy Release

- ✓ SATAN, WREFLOOD and GOTHIC codes
- ✓ Approved by NRC
- ✓ US-APWR design features modeled:
  - Advanced Accumulator
  - Direct Vessel Injection
- ✓ Topical report on applicability to US-APWR is under NRC review

➢Containment Pressure

- ✓ GOTHIC code
- ✓ Widely used in licensing analysis in US

Methodology and Codes (Cont.)

➢Non-LOCA

- ✓MARVEL-M, TWINKLE-M and VIPRE-01M codes
  - MARVEL-M : Plant system transient analysis code
  - TWINKLE-M : Multi-dimensional neutron kinetics code
  - VIPRE-01M : Core subchannel TH analysis code
- ✓Modified from MARVEL, TWINKLE and VIPRE-01 codes previously approved by NRC
- ✓Topical report on applicability of codes and methodology is under NRC review

➢Dose Evaluation

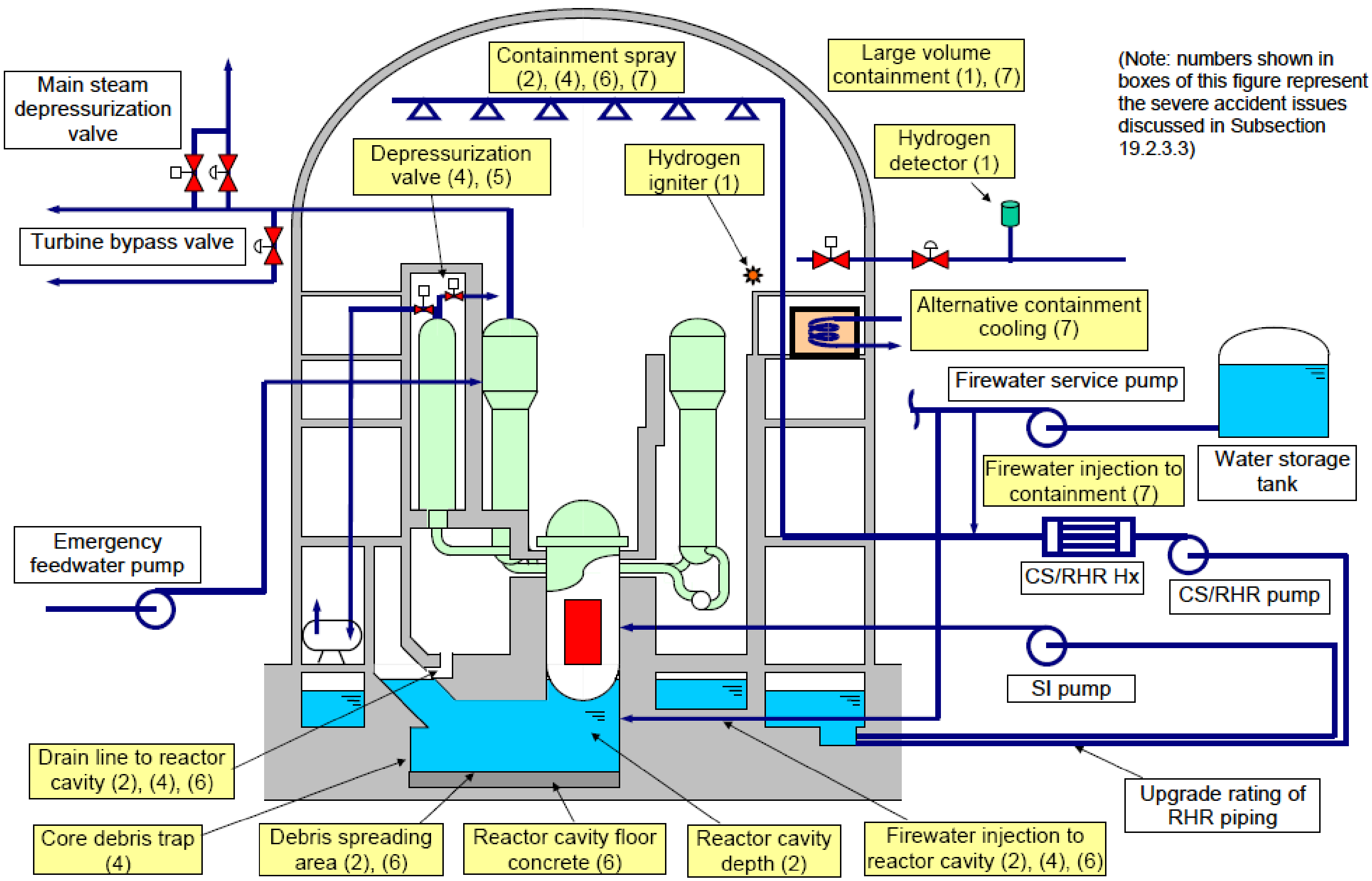
- ✓RADTRAD, PWR-GALE, etc.
- ✓Widely used in licensing analysis in US

Countermeasures for Severe Accident

- **US-APWR achieves higher safety to comprehensively address severe accident and mitigate consequences**
  - ✓ **Demonstrate compliance with current NRC regulations including TMI requirements for new plants**
  - ✓ **Demonstrate technical resolution of the applicable unresolved safety issues (USI), and the medium and high-priority generic safety issues (GSI) discussed in NUREG-0933**

Countermeasures for Severe Accident (cont.)

➢Severe Accident Mitigation Features



(Note: numbers shown in boxes of this figure represent the severe accident issues discussed in Subsection 19.2.3.3)

Addressed severe accident issues

(1) Hydrogen generation and control

(2) Core debris coolability

(3) Steam explosion

(4) HPME

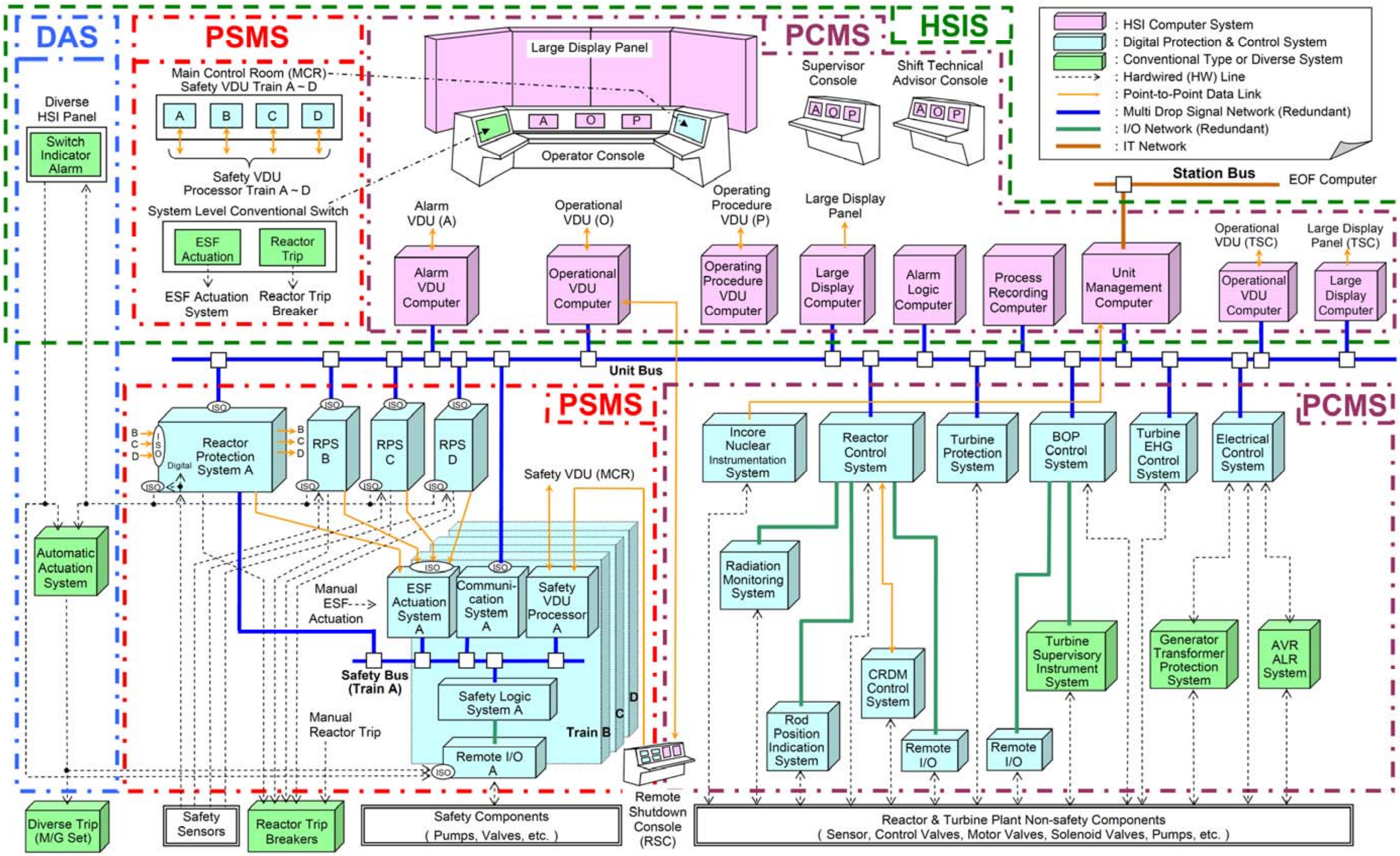
(5) TISGTR

(6) MCCI

(7) Long-term containment overpressure

(8) Equipment survivability

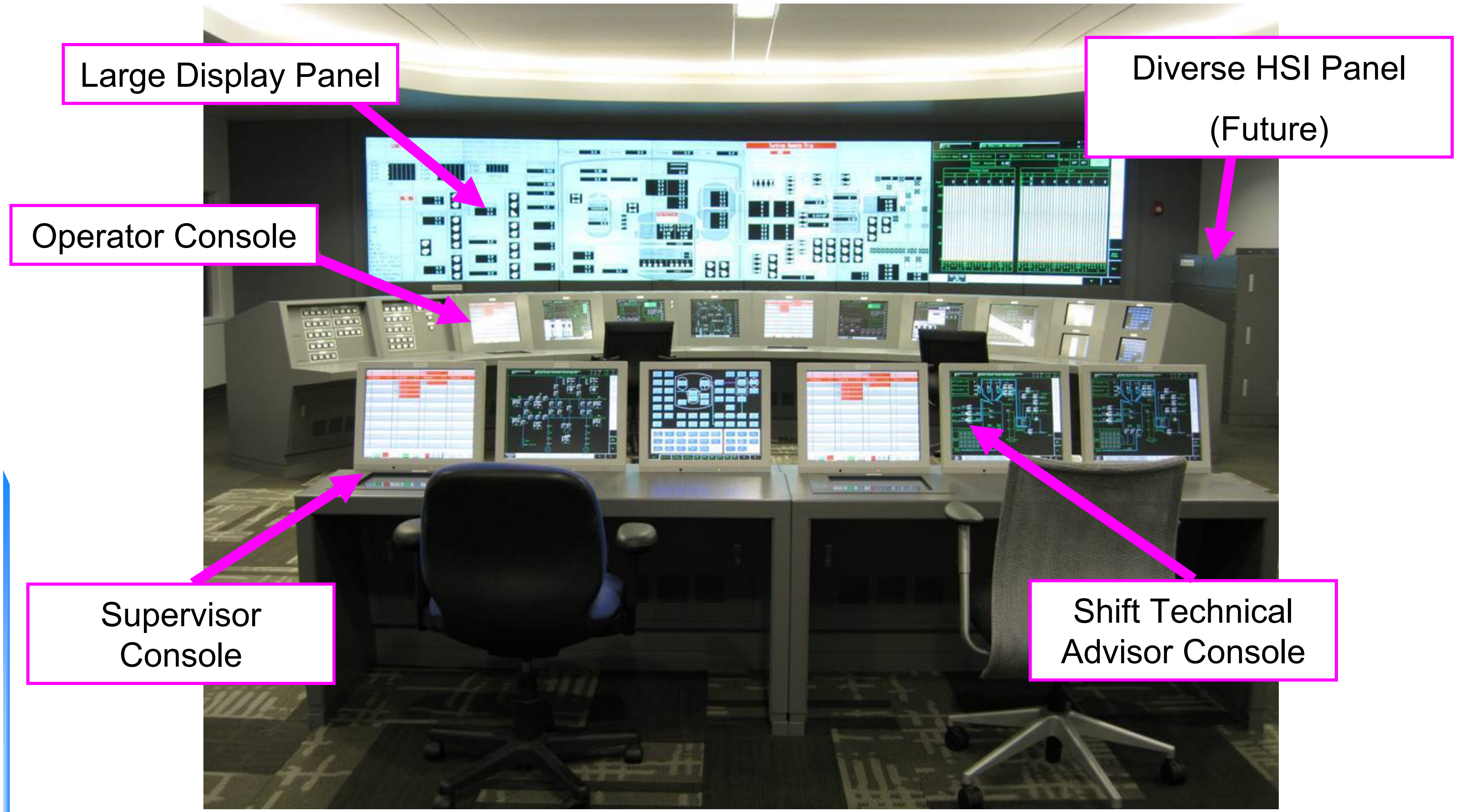
4. I&C System Architecture



DAS : Diverse Actuation System
PSMS : Protection and Safety Monitoring System
HSIS : Human System Interface System
PCMS : Plant Control and Monitoring System

HSI System Architecture

HSI Simulation Facility at MEPPI site – Pittsburgh



MEPPI: Mitsubishi Electric Power Products Inc.,

Overall I&C System Architecture

- Microprocessor based digital technology for most plant I&C (no electro-mechanical relays)
- Complete four train redundancy for safety I&C with each division in separate fire zone
- Distributed architecture for non-safety I&C with redundancy
- Fully multiplexed and duplicated signal transmission networks from local areas to I&C equipment rooms and Main Control Room, and between I&C systems
- Common digital platform for safety and non-safety I&C
- Diverse Actuation System based on analog technology
- Fully computerized Main Control Room

History of MHI Digital Application

➢ Non-safety Application History

- ✓ Development began in 1985 with initial goal of non-safety applications and long term goal of safety applications
- ✓ Platform originally developed in compliance with US standards, including communications independence (cyber security)
- ✓ First installation for non-safety systems / components
- ✓ Average 10 years operation for five operating plants
- ✓ Applied to all non-safety I&C, 50 applications per plant
- ✓ Over 20 million hours total operating experience
- ✓ No unexpected shutdown caused by I&C
- ✓ No system malfunction caused by S/W or H/W failure

History of MHI Digital Application (cont.)

- The same MELTAC digital platform is currently being applied to safety systems
- Current application for Reactor Protection and ESF Actuation System in Japan
  - ✓ Tomari #3 (Under construction, C/O 2009)
  - ✓ Tsuruga #3/4 (APWR) (Under licensing, C/O 2015)
  - ✓ Ikata #1/2 (Digital Upgrade 2009)
  - ✓ Takahama #1/2/3/4 (Digital Upgrade 2009 – 2012)
  - ✓ Ohi #1/2/3/4 (Digital Upgrade 2009 – 2013)

Note: Above RPS/ESFAS basic architecture is the same as US-APWR

Development of Computerized MCR

- **Computerized Main Control Room Developed with Japanese Operators**
  - ✓ **Development began in 1987**
  - ✓ **V&V tests (3 times) with Japanese PWR utilities shift operators (from12 sites) 1998-2001**
    - **Full-Scale Simulator**
    - **Performance Check**
    - **Review and Comment**
  - ✓ **Established Standard Design Specification for New PWR, APWR and Plant Modernization**
- **Current applications in Japan**
  - ✓ **Tomari #3 (Under construction, C/O 2009)**
  - ✓ **Tsuruga #3/4 (APWR) (Licensing, C/O 2016)**
  - ✓ **Ikata #1/2 (Modernization 2009)**

HSI Verification & Validation

- ➢ **HSI Verification & Validation is being conducted with U.S. operators**
- ➢ **Dynamic validation will be performed using Full-Scale Simulator with 8 U.S. operating crews**
  - **Performance Check**
  - **Review and Comment**
  - **Normal and accident scenarios**
  - **Normal and degraded HSI conditions**

  ✓ **Established Standard Design Specification**

  ✓ **Results will be issued as a technical report this year.**
- ➢ **NRC Staff visited MEPPI on June 4th.**

  Demonstrated plant operation using the simulator.

5. Conclusions

- US-APWR design is similar to the Japanese APWR currently in the stages of licensing review
- US-APWR is 1,700 MWe class large NPP based on MHI proven, advanced technology to improve reliability and enhance safety
- US-APWR meets U.S. utility's requirements and provides enhanced safety with features that address R.G. 1.206

United States Nuclear Regulatory Commission

Protecting People and the Environment

US-APWR
Standard Design Certification
Project Overview

To:
Advisory Committee on Reactor Safeguards

By:
Jeff Ciocco, Senior Project Manager
U.S. Nuclear Regulatory Commission

June 6, 2008

Purpose & Agenda

- Purpose
  - Provide an informational briefing to familiarize the Committee with the application, licensing review process, and current status of the US-APWR standard design certification application.
  - Address the Committee's questions.
- Agenda
  - Application Status
  - Review Schedule
  - Design Control Document Chapters and Topical Reports

US-APWR Application Status

- Pre-application review meetings began July 2006.
- Topical Report submittals began January 2007.
- Received Mitsubishi Heavy Industries (MHI), LTD, US-APWR standard design certification (DC) application on December 31, 2007.
- Acceptance review completed and docketed application on February 29, 2008. (Docket Number is 52-021).
- Phase 1 licensing review underway, preparing Preliminary Safety Evaluation Report and issuing RAIs.
- Luminant selected the US-APWR technology for Comanche Peak Nuclear Power Plant Units 3 & 4.
  - COL application expected September 2008.

Review Schedule Background

- The Design Control Document (DCD) identifies an approach to replacing proposed design criteria with detailed design information from Technical Reports and Audits for components and piping and digital I&C.
- The application references 13 MHI Topical Reports and 50 MHI Technical Reports.
- MHI's goal is to minimize the scope and number of Open Items in the staff's Safety Evaluation Report issued at the end of Phase 2.
- If necessary, the staff will review and re-baseline the review schedule after completion of Phase 2.
- Will coordinate with ACRS staff the review of the Safety Evaluation Report in Phases 3 & 5.

US-APWR Design Certification Review Schedule

Phase	Name	End date
Phase 1	Preliminary Safety Evaluation Report (SER) and Request for Additional Information (RAI)	June 2009
Phase 2	SER with Open Items	March 2010
Phase 3	ACRS Review of SER with Open Items	June 2010
Phase 4	Advanced SER with No Open Items	May 2011
Phase 5	ACRS Review of Advanced SER with No Open Items	August 2011
Phase 6	Final SER with No Open Items	September 2011

DCD Chapters and Topical Reports

Chapter Project Manager	DCD Chapter	Topical Reports (SER Dates)
Jeff Ciocco	Ch 1 – Introduction & General Description of the Plant	
Mike Takacs	Ch 2 – Site Characteristics	
Bill Ward	Ch 3 – Design of Structures, Systems, Components, & Equipment	
Jin Chung	Ch 4 – Reactor	- Mitsubishi Fuel Design Criteria & Methodology (07/2009) - Mitsubishi Thermal Design Methodology (04/2009) - FINDS: Mitsubishi Fuel Assemblies Seismic Analysis Code (05/2009)
Bill Ward	Ch 5 – Reactor Coolant & Connecting Systems	
Jin Chung	Ch 6 – Engineered Safety Features	- The Advanced Accumulator (03/2009) - LOCA Mass and Energy Release Analysis Code Applicability Report (10/2008)
Steve Monarque	Ch 7 – Instrumentation & Controls	-Safety I&C System Description and Design Process (06/2009) -Safety System Digital Platform -MELTAC- (06/2009_ -Defense-in-Depth and Diversity (10/2008)
Ngola Otto	Ch 8 – Electric Power	
Bill Ward	Ch 9 – Auxiliary Power	

DCD Chapters and Topical Reports

Chapter Project Manager	DCD Chapter	Topical Reports (SER Dates)
Mike Takacs	Ch 10 – Steam & Power	
Ngola Otto	Ch 11 – Radioactive Waste Management Systems	
Ngola Otto	Ch 12 – Radiation Protection	
Mike Takacs	Ch 13 – Conduct of Operations	
Ngola Otto	Ch 14 – Initial Test Programs	
Mike Takacs	Ch 15 – Transient & Accident Analyses	- Non-LOCA Methodology (05/2009) - Large Break LOCA Code Applicability (05/2009) - Small Break LOCA Methodology (04/2009)
Peter Hearn	Ch 16 – Instrumentation & Controls	
Jeff Ciocco	Ch 17 Quality Assurance & Reliability Assurance	- Quality Assurance Program (QAP) Description for Design Certification (01/2008)
Steve Monarque	Ch 18 – Human Factors Engineering	- HFE Process & HSI System Design (09/2008)
Jin Chung	Ch 19 – PRA & Severe Accidents	

Summary

- Provided an overview of the US-APWR application and review schedule.
- Phase 1 licensing review is underway.
- Will coordinate the ACRS review of the US-APWR SER in Phases 3 and 5.

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Protecting People and the Environment

Status and Path Forward for Generic Safety Issue 191, Pressurized Water Reactor Sump Performance

Presented by:
Michael Scott
Office of Nuclear Reactor Regulation
Presented to:
Advisory Committee on Reactor Safeguards
June 6, 2008

Background

- Generic Safety Issue 191 involves performance of PWR emergency core cooling and containment spray systems in recirculation mode in the presence of debris after a loss-of-coolant accident/high-energy line break
- Generic Letter 2004-02 requested licensees, by end of 2007, to:
  - Determine plant-specific debris generation and transport
  - Make needed modifications to show compliance with regulations in presence of plant-specific debris loading

Current Status of GSI-191

- Essentially all PWRs have installed much larger sump strainers
- Many have done other modifications (e.g., removed insulation or replaced sump buffer)
- Fort Calhoun implementing water management initiative
- Staff and industry believe risk of strainer clogging reduced significantly
  – Significant uncertainties remain
  – Plants can continue to operate safely for same reasons as stated in GL 2004-02
- Integrated head loss testing (including chemicals) ongoing
  – Staff reviewing and commenting on protocols
  – Staff observing and commenting on representative tests intended to show adequate strainer function

Current Status (Continued)

- Most licensees received additional time beyond 12/31/07 to complete certain corrective actions
  - Downstream effects analyses
  - Integrated head loss testing
  - Plant modifications
- Most extensions for a few months; a few into 2009
- All plants submitted supplemental responses to GL 2004-02 in February/March 2008 (incomplete responses for plants with extensions)

Chemical Effects

- Many plants did not complete integrated head loss testing with chemical effects by end of 2007
- Completion delayed by:
  - Late recognition by industry of difficulty of the issue
  - Limited number of testing vendors, requiring queuing
  - Challenges resolving staff issues with chemical effects topical report
  - Staff issues with testing methods used or planned by test vendors
- Staff issued safety evaluation (SE) on chemical effects topical report in December 2007

Chemical Effects Peer Review

- Staff screened peer review issues in 2007 to identify those warranting further evaluation
- Office of Nuclear Regulatory Research commissioned study of aspects that earlier staff review could not disposition
- Staff currently reviewing study results
- Likely result is need for additional consideration of some of these effects
- Will report to Committee on this later in 2008

Downstream Effects

- Ex-vessel (pumps, valves, etc.)
  - SE on ex-vessel downstream effects topical report issued December 2007
  - Some licensees have requested extensions to complete these analyses
- In-vessel (core flow blockage)
  - Received topical report WCAP-16793-NP June 2007
  - Draft SE issued in March 2008
  - Met with ACRS Thermal-Hydraulics Subcommittee March 19
  - Subcommittee had questions and concerns
  - Staff and PWR Owners Group working to address issues
  - Will return to Subcommittee as soon as issues resolved
  - Description of method in draft WCAP and some preliminary NRC staff conclusions discussed in backup slides

ACRS T/H Subcommittee Questions and Concerns

- Flow resistance at the core inlet or first spacer grid as a consequence of deposits (maximum loss permitted and whether that could occur)
- Temperature at the screen vs. that at the core inlet and its effect of solubility of chemical compounds
- More information on local subchannel blockage and its potential for temperature hot spots
- Bypass testing and assumptions
- Driving head for flow into the core
- Potential for and consequences of debris inhibiting boric acid mixing

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Path Forward on WCAP-16793

- Staff has provided additional information to the subcommittee that may address some aspects of these questions
- PWR Owners Group plans additional testing to reduce uncertainty regarding potential for blockage at core inlet
- Staff needs to evaluate responses being developed by PWR Owners Group
- Staff and PWR Owners Group plan to return to brief subcommittee
- Timeline dependent on completion of adequate Owners Group-sponsored testing and/or evaluation

Head Loss Testing

- Staff has questioned whether various aspects of the licensee-sponsored vendor-performed head loss testing are conservative or prototypical
  - Debris preparation and introduction
  - Near-field settling
  - Thin bed testing
- Staff's questions and concerns have had impacts on licensee test schedules
- Staff has found that most vendors now have conservative protocols – though some licensees completed testing under previous protocols with which staff has had concerns
- Licensees can use any approach that they can show to be conservative or prototypical

Head Loss Testing (Cont'd)

- One recent test of a uniform flow strainer conducted by adding full particulate load followed by sufficient fine fiber (only) to create a thin debris bed resulted in high head loss without chemicals
- Challenge for licensees is to develop conservative or prototypical, but not excessively conservative, test protocol
- Potentially challenging for high-fiber and maybe for medium-fiber plants

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GL Supplemental Response Reviews

- Staff has begun review of supplemental GL responses
- Because of extensions, many licensees will need to submit an additional response
- Likely to send requests for additional information (RAIs) to most plants
  - For low-fiber plants, few RAIs – maybe limited to in-vessel downstream effects
- Result is final closure in 2009

Closing GL 2004-02 and GSI-191

- Staff plans to close these issues for each plant based on:
  - Review of licensee supplemental responses
  - Results of Region inspections of licensee corrective actions
  - Review of licensee responses to audit open items (as applicable)
- If a plant has not completed all modifications but has a satisfactory strainer evaluation in place and a specific plan for completing remaining modifications, staff plans to close the GL and GSI for that plant
- Staff will track all corrective actions to completion at all plants

Subjects Proposed for Future ACRS Review

- In-vessel downstream effects
- Integrated head loss testing protocols and results
- Results of staff review of licensee supplemental responses
- Results of chemical effects peer review scoping analyses
- Results of additional confirmatory chemical effects testing at Argonne National Laboratory

Disparities in Treatment for PWRs and BWRs

- BWR strainer issues resolved in 1990s
- For various reasons, treatment of debris-induced clogging issues has varied for PWRs and BWRs
  - Different strainer, ECCS, and core designs
  - Issues addressed at different times and based on different states of knowledge
- Learned a lot from PWR work – applicable to BWRs?
- NRR has sent User Need to ask RES to evaluate differences and recommend additional actions if warranted – RES has begun work
- Encouraging BWR Owners Group to take initiative to address potential issues
- Will consider further regulatory actions based on BWROG and RES activities

Conclusions

- GSI-191 remains an extraordinary complex and difficult issue to resolve
- Licensees have made substantial progress in reducing vulnerability to strainer clogging and related issues
- Additional modifications may be needed (e.g., remove problem materials from containment) if licensees cannot show success in the near future with conservative testing and evaluation
- Staff expects issue resolution in 2009

Backup Slides

WCAP-16793 Approach to In-vessel Effects

- Limit on the maximum temperature of fuel clad is established based upon a conservative value that prevents fuel damage (in accordance with 10 CFR 50.46)
- Industry-recognized models for deposition of solids and calculation of temperature increases based on heat transfer coefficients are used
- Flow simulation code (WCOBRA/TRAC) is used to assess limit on flow reduction and still achieve adequate core cooling
- Entire chemical effects source term from topical report WCAP-16530 assumed to be available for deposit on core surfaces

Approach to In-vessel Effects (Cont’d)

- Size and quantity of fibrous material entering the lower core region is estimated from the containment sump screen dimensions and plant fiber bypass tests
- Deposition of this material on the lower core plate, leading to flow blockage, is assessed
- Particulate and fibrous matter that passes through the lower core plate is evaluated for local flow blockage and deposition effects
- Thickness of fuel deposits (oxide + crud + chemical deposit) formed is calculated using LOCADM based on fuel decay heat, the mass of materials present, and the core surface area

Licensee Use of WCAP-16793

- Licensees are likely to take credit for WCAP-16793-NP as bounding for their plants in showing that in-vessel downstream effects will not cause unacceptable impacts on the fuel
- Application of WCAP-16793-NP is to be in accordance with conditions and limitations contained in the NRC SE (when published)
- Licensees are expected to verify that the assumptions in the WCAP-16793-NP methods are conservative with respect to their individual plants
- Licensees may choose to develop and substitute plant-specific data, such as debris content, chemicals, strainer efficiency, etc.

Staff Review of WCAP-16793

- Staff noted a number of conservatisms in WCAP-16793
  - Most of core entrance assumed blocked with debris – flow still adequate
  - Assumed buildup of debris on core surfaces conservative
  - Thermal conductivity value conservative
  - Worst-case local heating well below limit
  - Chemical source term assumptions conservative
  - Large margin between the chemical deposit predicted for a high-fiber plant with large amounts of calcium silicate insulation and the amount of deposit that would cause the maximum peak clad temperature to exceed the acceptance criteria