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

July 8, 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 552$^{nd}$ FULL COMMITTEE MEETING HELD ON MAY 8-10, 2010 IN ROCKVILLE, MARYLAND

The minutes of the subject meeting were certified on June 21, 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

June 21, 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 552nd MEETING OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS (ACRS), MAY 8-10, 2008

I certify that based on my review of the minutes from the 552nd 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/Sunsi |
| **NAME** | SMeador | CSantos/sam |
| **DATE** | 06/ 21 /08 | 06/ 21 /08 |

**OFFICIAL RECORD COPY**

CERTIFIED Date Certified: 06/21/2008

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MINUTES OF THE 552nd ACRS MEETING

May 8-10, 2008



**Appendices**

During its 552nd meeting, May 8-9, 2008, the Advisory Committee on Reactor Safeguards (ACRS) discussed several matters and completed the following reports, letters, and memoranda.

REPORTS

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

- Draft NUREG/CR-6962, "Approaches for Using Traditional Probabilistic Risk Assessment Methods for Digital Systems," and Related Matters, dated May 19, 2008

- PHEBUS Fission Product (PHEBUS-FP) Program, dated May 30, 2008

LETTERS

Letters to R. W. Borchardt, Executive Director for Operations, NRC, from William J. Shack, Chairman, ACRS:

- Interim Letter 3: Chapters 4, 6, 15, 18, and 21 of the NRC Staff's Safety Evaluation Report with Open Items Related to the Certification of the ESBWR Design, dated May 23, 2008

- Response to the January 17, 2008 EDO letter regarding Susquehanna Steam Electric Station Units 1 and 2 Extended Power Uprate (EPU), dated June 3, 2008

MEMORANDA

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

- Multiple Draft Regulatory Guides to be Issued as Final, dated May 15, 2008

- Withdrawal of Regulatory Guide 3.42, dated May 15, 2008

- Draft Regulatory Guides 1178, 1190, 1198 and 1200, dated May 15, 2008

- Draft Regulatory Guides 3024 and 3033, dated May 15, 2008

# MINUTES OF THE 552nd MEETING OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

## ROCKVILLE, MARYLAND

The 552nd meeting of the Advisory Committee on Reactor Safeguards (ACRS) was held in Conference Room 2B3, Two White Flint North Building, Rockville, Maryland, on May 8-10, 2008. Notice of this meeting was published in the *Federal Register* on April 23, 2008 (72 FR 21994-21995) (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. 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. D. Joseph Armijo 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 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. Bonaca 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.

HIGHLIGHTS OF KEY ISSUES

II. Selected Chapters of the NRC Staff's Safety Evaluation Report with Open Items Related to the Certification of the ESBWR Design

The Committee met with representatives of the NRC staff and General Electric-Hitachi Nuclear Americas, LLC (GEH) to discuss Chapters 4, "Reactor," 6, "Engineered Safety Features," 15, "Accident Analysis," 18, "Human Factors Engineering," and 21, Computer Code Validation," of the draft, Safety Evaluation Report (SER) with open items related to the ESBWR design certification application. The staff presented a brief summary of the status of various issues in the above Chapters related to power flow stability, reliability of the vacuum breaker system, and applicability of the TRACG code for the ESBWR ATWS analysis. In addition, GEH and the staff discussed the ESBWR thermal-hydraulic issues raised by the EWBWR Subcommittee during its meetings on January 16-17 and April 9, 2008.

The Committee issued a letter to the EDO on this matter, dated May 23, 2008, identifying several issues that merit additional attention, including: confirmation of coupled neutronic and thermal-hydraulic stability including interactions between the core and chimney; assurance of the proper operation of the vacuum breaker system by appropriate surveillance testing, leakage monitoring, and isolation capability; demonstration of the performance of passive safety systems addressing issues such as gap binding; and assurance that the proposed principals of Human Factors Engineering are appropriately integrated into the ESBWR design.

III. Insights from the PHEBUS-Fission Product (PHEBUS-FP) Program

[Note: Dr. Hossein Nourbakhsh was the Designated Federal Office for this portion of the meeting.]

The Committee met with representatives of the NRC staff and the French Institut de Radioprotection et de Sûreté Nucléaire (IRSN) to discuss the results of the PHEBUS-FP tests. PHEBUS-FP is an international program carried out in Caderache, France. The PHEBUS facility is a small scale representation of the core, reactor coolant system, and containment. Five experiments were run varying the steam flow, irradiation of the fuel, control rod material (silver-indium-cadmium and boron carbide), and pH of the sump water in the containment module. The PHEBUS-FP experiments simulate the major aspects of a severe accident, including degradation of irradiated fuel, release of fission products, transport of fission products through a simulated reactor coolant system, and injection of the fission products into a simulated reactor containment.

The staff stated that the data from these integral tests were valuable for validating and refining the models and computer codes used for reactor accident analysis, in particular the MELCOR code, and in assessing the adequacy of accident source terms for use in reactor-related regulatory analyses. The results supported many of the modeling assumptions in MELCOR and the source term described in NUREG-1465. The first two tests have been used to improve the

modeling of fuel slumping in MELCOR. The tests also indicated that cesium was not transported predominantly as cesium hydroxide as previously believed. Silver vaporized from control rod alloys transported to the containment sumps, where it reacted with iodine to form insoluble compounds that limit free iodine in the containment atmosphere. In the tests with a boron carbide control rod, there were other reactions of iodine that limited its release to the containment atmosphere. The nature of these reactions will be studied in follow-on programs. The release of iodine from the sump was observed to be independent of pH over a range from 5 to 9.

The Committee issued a report to the NRC Chairman on this matter, dated May 30, 2008, stating that PHEBUS-FP is an example of a successful international cooperative research program. It is yielding data for validating and refining of severe accident analysis computer codes as well as assessing the appropriateness of the accident source terms. The strategy for developing a mechanistic understanding of gaseous iodine behavior is appropriate and the planned work should be supported.

## IV. Draft NUREG/CR Report on PRA Methods for Digital Systems

[Note: Mrs. Christina Antonescu was the Designated Federal Official for this portion of the meeting.]

The Committee met with representatives of the NRC staff and its contractor Brookhaven National Laboratory (BNL) to discuss draft NUREG/CR- 6962, "Approaches for Using Traditional Probabilistic Risk Assessment Methods for Digital Systems."

The principal objective of the current BNL project is to determine the capabilities and limitations of using traditional reliability modeling methods to develop and quantify digital system reliability models, with the desired goal of supporting the development of regulatory guidance for assessing risk evaluations involving digital systems.

This draft NUREG/CR specifically addresses the development of a draft criteria and lays out the process by which the first reliability study of an example digital system will be performed. Preliminary work on these tasks indicates that the traditional methods of event tree/fault tree and Markov modeling appear to be useful for the PRA of Digital Instrumentation & Control (DI&C) systems, but it also reveals limitations in the state-of-the-art using traditional PRA methods and areas where additional research and development are needed.

As part of this BNL project, the traditional event tree/fault tree and Markov methods were applied to two example systems (referred to as "benchmark" test cases). The first benchmark test case involves a digital feed-water control system (DFWCS) of a two-loop pressurized water reactor and the second involves a reactor protection system. Detailed information was only available for the DFWCS system. Therefore, the DFWCS was used in the NUREG/CR report to illustrate how the traditional reliability modeling methods will be applied in the later tasks of the project (i.e., in the actual benchmark studies).

The Committee issued a report to the NRC Chairman on this matter, dated May 19, 2008. The Committee recommended that the draft NUREG/CR-6962 be revised before publication to state clearly that its methods do not address software failures and that it employs simulation in addition to traditional PRA methods. The revised NUREG/CR report should focus on failure mode identification only. The Committee also stated that the staff should establish an integrated program that focuses on failure mode identification of DI&C systems and takes advantage of the insights gained from the investigations on traditional PRA methods and on advanced simulation methods. In addition, the quantification of the reliability of DI&C systems should be deferred until a good understanding of the failure modes is developed.

## V. Plant License Renewal Subcommittee Report

The Chairman of ACRS Subcommittee on Plant License Renewal provided a report to the Committee regarding the interim review of the Carolina Power & Light Company's (CP&L) license renewal application for the Shearon Harris Nuclear Power Plant and the associated NRC staff's Safety Evaluation Report (SER) with Open Items.

The current operating license for the Shearon Harris Nuclear Power Plant expires on October 24, 2026. CP&L submitted the license renewal application on November 14, 2006, and the staff's draft SER was issued on March 18, 2008, and contains one open item. The open item is related to the classification applicable to the feedwater regulating and bypass valves. The staff's position is that these valves are safety-related equipment, and should be scoped under 10 CFR 54.4(a)(1). The applicant believes, however, that they are not safety-related, and therefore, should be scoped under 10 CFR 54.4(a)(2). The staff and CP&L are in the process of resolving this open item. The Committee plans to discuss the final SER related to the license renewal application for the Shearon Harris Nuclear Power Plant in a future meeting.

## 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 April 11, 2008, to comments and recommendations included in the March 6, 2008, ACRS report on the review and evaluation of the NRC Safety Research Program. The staff has agreed with the ACRS recommendations. The Committee will be afforded opportunities to discuss staff's responses during its future meetings on specific programs. The Committee decided that it was satisfied with the EDO's response.

- The Committee considered the EDO's response of April 17, 2008, to comments and recommendations included in the March 20, 2008, ACRS report on the review of Entergy Nuclear Operations' license renewal application for the Vermont Yankee Nuclear Power Station and the associated NRC staff's final SER. The Committee decided that it was satisfied with the EDO's response.

- The Committee considered the EDO's response of April 24, 2008, to comments and recommendations included in the March 20, 2008, ACRS report on the final review of Entergy Nuclear Operations' license renewal application for the James A. FitzPatrick Nuclear Power Plant and the associated NRC staff's final SER. The Committee decided that it was satisfied with the EDO's response.

- The Committee considered the EDO's response of April 24, 2008, to comments and recommendations included in the March 20, 2008, ACRS letter on Chapters 9, 10, 13, and 16 of the staff's SER related to the certification of the ESBWR design. The Committee decided that it was satisfied with the EDO's response.

B. Report of the Planning and Procedures Subcommittee Meeting

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

Member assignments and priorities for ACRS reports and letters for the May 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 workload for ACRS members through July 2008 was discussed and the objectives were to:

- 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

Proposed Topics for Meeting With the NRC Commissioners

The ACRS is scheduled to meet with the Commission on Thursday, June 5, 2008, between 1:30 and 3:30 p.m. The following topics were approved by the Commission:

1. Overview (WJS/SD)

   - Accomplishments
   - License Renewal
   - New Plant Activities
   - Ongoing / Future Activities

2. Safety Research Program Report (DAP/HPN)
3. Digital I&C Matters (GEA/CEA)
4. State-of-the-Art Reactor Consequence Analyses (WJS/HPN)
5. ESBWR Design certification (MLC/DEB)

6. Extended Power Uprate and Related Technical Issues (MVB/ZA)

The proposed presentation slides prepared by the cognizant staff engineers were sent to the lead members for review and comment. The current package reflects incorporation of comments received from the lead members. Please note that the presentation slides for the Digital I&C item will be revised to include comments and recommendations from the Committee letter scheduled to be completed during the May meeting.

The Committee needs to approve the presentation slides at the May meeting subject to including additional recommendations on Digital I&C and minor editorial changes. Following the May meeting, complete presentation slides will be sent to the members prior to transmitting them to the Commission on May 28, 2008.

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

During the April 2008 meeting, the Committee decided to visit the Byron Nuclear Plant, and meet with the Region III Administrator to discuss items of mutual interest. Since the licensee needs to prepare for the NRC staff team inspection, they suggested that the members visit the Braidwood Nuclear Plant. A proposed schedule agreed to by the Committee 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

A detailed schedule along with the arrangements will be provided to the members during the June meeting.

OIG Report on the Audit of NRC's Power Uprate Program

The Office of the Inspector General (OIG) performed an audit of the NRC's Power Uprate Program and documented its findings and recommendations in a report dated March 11, 2008. A summary list of OIG recommendations is as follows:

- Revise Inspection Procedure (IP) 71004 to provide more specificity with regard to the use of the inspection procedure.
- Provide cross references from baseline and other inspection procedures that are called for in IP 71004.
- Document or index cumulative IP 71004 and other uprate-related inspection activities in a centralized location or in an easily retrievable way so that internal and external stakeholders can easily find the results.
- Develop training for technical reviewers and project managers that is specifically focused on writing or contributing to a safety evaluation.
- Implement internal controls to ensure communication of the safety evaluation, highlighting the recommended areas of inspection and regulatory commitment sections to the regions and resident inspectors.
- Strengthen and communicate the coordinating authority of Power Uprate and Generic Communications Branch or assign a coordinating authority to be responsible for all aspects of power uprate activities.
- Identify and communicate roles and responsibilities for headquarters and regional points of contact for power uprates.
- Develop a tool for project managers to share and record information, monitor trends, and capture best practices and lessons learned.

In a letter dated March 24, 2008, the EDO responded to the OIG recommendations. The EDO states that the staff agrees with six of the eight recommendations. The staff does not agree with recommendations 2 and 4.

Cancelling Scheduled Subcommittee Meetings and Full Committee Meeting Items

Subcommittee meetings and items for the full Committee meetings are scheduled in mutual agreement between the staff and the Committee members. In accordance with the Federal Advisory Committee Act (FACA) requirements, notice for individual ACRS meetings is published in the Federal Register 15 days in advance of the meeting. Cancelling a meeting a few days before the scheduled date will not provide adequate time for the ACRS staff to issue an amendment to the Federal Register Notice to inform the public and industry of the cancellation of the meeting; it takes about 5-6 days to publish an amendment to the Federal Register Notice.

Recently, the Thermal-Hydraulic Phenomena Subcommittee meeting scheduled for Tuesday, April 8, 2008 to discuss PWR sump performance issues was cancelled on Thursday, April 3, 2008, at the request of the staff. Even though an amendment was issued, it did not get published until after the meeting. As a result, several industry people showed up for the meeting. Another Subcommittee meeting on Reliability and PRA scheduled for April 18, 2008 to discuss a NUREG document on uncertainty and sensitivity analyses was cancelled late Friday, April 11, 2008. Once again, there was no time to amend the Federal Register Notice. The staff and EPRI were not happy about such a late cancellation of the meeting.

The long-standing policy of the Committee is as follows:

- If a Subcommittee meeting is noticed in the Federal Register, it should not be cancelled. However, under inevitable circumstances (e.g. inclement weather), it could be cancelled in coordination with the meeting participants. The ACRS staff should amend the Federal Register notice to notify the external stakeholders about the cancellation of the meeting.
- If an item is scheduled for the full Committee meeting, it should not be cancelled no mater what. The staff should be asked to provide at least a status briefing to the Committee.

Scheduling Subcommittee Meetings

It has become more difficult to find hotels with available rooms at the government rate that are in close proximity to the NRC. During certain times of the year, hotel reservations need to be made up to 4 months in advance. This problem is extremely difficult when subcommittee meetings are scheduled with relatively short notice.

Trip to Salem Nuclear Plant

At the invitation of the licensee, Drs. Banerjee, Bley, and Mr. Maynard along with several ACRS staff engineers, visited the Salem Nuclear Plant on April 24, 2008 to look at the replaced steam generators and containment sump screen. A brief report on the observations made by the members during this visit would be helpful to the Committee.

Withdrawal of Regulatory Guides

The staff proposes to withdraw the following Regulatory Guides and seeks Committee's endorsement:

- Regulatory Guide 1.139, "Guidance for Residual Heat Removal."

  This Guide describes an overly conservative and prescriptive method for complying with the regulations. 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 its 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.

- Regulatory Guide 3.42, “Emergency Planning for Fuel Cycle Facilities and Plants Licensed Under 10 CFR Parts 50 and 70.”

  This Guide provides guidance for applicants developing emergency plans for fuel cycle facilities licensed under 10 CFR Part 50 as well as for applicants of special nuclear materials licensed under 10 CFR Part 70. The staff is withdrawing this Guide because no fuel cycle facilities are currently licensed under both 10 CFR Part 50 and 10 CFR Part 70. For fuel cycle and materials facilities licensed under 10 CFR Part 70, Reg. Guide 3.67, “Standard Format and Content for Emergency Plans for Fuel Cycle and Materials Facilities,” provides adequate guidance for developing emergency plans.

Draft Final Regulatory Guides

The staff plans to issue several Regulatory Guides as final (pp. 19-22), which are in Division 3, “Fuels and Materials Facilities,” Division 6, “Products,” and Division 10, “General.” Proposed version of these Guides were provided previously to ACNW&M for possible review prior to issuing them for public comment. Since most of these Guides deal with process issues, the Committee does not need to review them.

Proposal Division 1 Regulatory Guides

The staff plans to issue the following Division I Regulatory Guides for public comment. These Guides are also listed in the attachment:

- Proposed Revision 2 to RG 1.200, An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results fro Risk-Informed Activities.

- Proposed Revision 2 to RG 1.125, Physical Models for Design and Operation of Hydraulic Structures and Systems for Nuclear Power Plants

- Proposed Revision 1 to RG 1.62, Manual Initiation of Protective Actions

- Proposed Revision 1 to RG 1.151, Instrument Sensing Lines.

Since these are Division I Regulatory Guides, the Committee may want to review the draft final version of these Guides after reconciliation of public comments.

In addition, the staff plans to issue the following Division 3 Reg. Guides for public comment:

- Proposed Revision 1 to Regulatory Guide 3.25 (DG-3033), Standard Format and Content of Safety Analysis Reports for Uranium Enrichment Facilities.

This Guide is being revised to endorse NUREG-1520, “Standard Review of License Application for a Fuel Cycle Facility.” Since this is a minor revision, the Committee does not need to review this Guide.

- Proposed Revision 1 to RG 3.5, (DG-3024), “Standard Format and Content of License Applications for Conventional Uranium Mills”

This Guide is being revised to update the references, such as NUREGs, and the format. These are not technical changes. Therefore, the Committee does not need to review this Guide.

Reorganization of the ACRS Office

The reorganization of the ACRS Office will become effective on June 1, 2008. Attached are the updated organizational chart, staff engineer assignments, staff biographies, and pictures.

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 are included in the attached calendars (pp. 39-42) and summarized below.

| Meeting Number | Dates | Days |
|---|---|---|
| --- | January 2009 | (No Meeting) |
| 559 | February 5-7, 2009 | Thursday-Saturday |
| 560 | March 5-7, 2009 | Thursday-Saturday |
| 561 | April 9-11, 2009 | Thursday-Saturday |
| 562 | May 7-9, 2009 | Thursday-Saturday |
| 563 | June 3-5, 2009 | Wednesday-Friday |
| 564 | July 15-17, 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 |

Dr. Abdel-Khalik suggests that the April 2009 meeting dates currently scheduled for April 9-11 be changed to April 2-4.

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.

C. Future Meeting Agenda

Appendix IV summarizes the proposed items endorsed by the Committee for the 553rd ACRS Meeting, June 4-6, 2008.

A list of documents that were provided to the Committee during the 552nd ACRS Meeting is listed in Appendix V.

The meeting was adjourned at 3:00 p.m. on May 10, 2008.

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

**Appendix II**

**April 17, 2008**

**SCHEDULE AND OUTLINE FOR DISCUSSION**
**552nd ACRS MEETING**
**MAY 8-10, 2008**

**THURSDAY, MAY 8, 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 - 10:30 A.M. Selected Chapters of the SER Associated with the ESBWR Design Certification Application (Open/Closed) (MLC/DEB)
   2.1) Remarks by the Subcommittee Chairman
   2.2) Briefing by and discussions with representatives of the NRC staff and General Electric – Hitachi Nuclear Energy (GEH) regarding selected Chapters of the NRC staff's Safety Evaluation Report (SER) With Open Items associated with the Economic Simplified Boiling Water Reactor (ESBWR) design certification application.

   **[Note: A portion of this session may be closed to protect information that is proprietary to GEH and its contractors pursuant to 5 U.S.C. 552b ( c) (4).]**

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

**10:30 - 10:45 A.M. ***BREAK*****

3) 10:45 - 12:30 P.M. Insights from PHEBUS – FP Tests (Open) (JSA/DEB/HPN)
   3.1) Briefing by and discussions with representatives of the NRC staff regarding the findings of the large-scale integral tests conducted in connection with the PHEBUS – FP Program and their implications on containment iodine behavior.

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

**12:30 - 1:30 P.M. ***LUNCH*****

4) 1:30 - 3:30 P.M. Draft NUREG/CR Report on PRA Methods for Digital Systems (Open) (GEA/CEA)

4.1) Remarks by the Subcommittee Chairman

4.2) Briefing by and discussions with representatives of the NRC staff and Brookhaven National Laboratory (BNL) regarding draft NUREG/CR – XXX Report on Approaches for Using Traditional PRA Methods for Digital Systems and other related matters.

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

**3:30 - 3:45 P.M. ***BREAK*****

5) 3:45 - 7:00 P.M. Preparation of ACRS Reports (Open)

Discussion of proposed ACRS reports on:

5.1) Selected Chapters of the SER Associated with the ESBWR Design Certification Application (MLC/DEB)

5.2) Insights from PHEBUS – FP Tests (JSA/DEB/HPN)

5.3) Draft NUREG/CR Report on PRA Methods for Digital Systems (GEA/CEA)

5.4) Response to the EDO Response dated January 17, 2008, to the December 20, 2007 ACRS Report on the Susquehanna Power Uprate Application) (SB/ZA)

**FRIDAY, MAY 9, 2008, CONFERENCE ROOM T-2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND**

6) 8:30 - 8:35 A.M. Opening Remarks by the ACRS Chairman (Open) (WJS/CS/SD)

7) 8:35 - 9: 15 A.M. Future ACRS Activities/Report of the Planning and Procedures Subcommittee (Open) (WJS/FPG/SD)

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

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

8) 9: 15 -9: 30 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) 9:30 – 10:00 A.M. Subcommittee Report (Open) (JWS/PW)
Report by and discussions with the Chairman of the ACRS Subcommittee on Plant License Renewal regarding the license renewal application for the Shearon Harris Nuclear Power Plant that was discussed during the Subcommittee meeting on May 7, 2008.

**10:00 – 10:15 A.M. ***BREAK*****

10) 10: 15 – 12:00 P.M. Preparation for Meeting with the Commission on June 5, 2008 (Open) (WJS, et al. /FPG, et al.)
Discussion for meeting with the Commission on June 5, 2008.

**12:00 - 1:30 P.M. ***LUNCH*****

11) 1:30 – 6:30 P.M. Preparation of ACRS Reports (Open)
Discussion of proposed ACRS reports on:

11.1) Selected Chapters of the SER Associated with the ESBWR Design Certification Application (MLC/DEB)
11.2) Insights from PHEBUS – FP Tests (JSA/DEB/HPN)
11.3) Draft NUREG/CR Report on PRA Methods for Digital Systems (GEA/CEA)
11.4) Response to the EDO Response dated January 17, 2008, to the December 20, 2007 ACRS Report on the Susquehanna Power Uprate Application) (SB/ZA)

**SATURDAY, MAY 10, 2008, CONFERENCE ROOM T-2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND**

12) 8:30 - 1:00 P.M. **(10:30-10:45 A.M. BREAK)** Preparation of ACRS Reports (Open)
Continue discussion of proposed ACRS reports listed under Item 11.

13) 1:00 - 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

**ADVISORY COMMITTEE ON REACTOR SAFEGUARDS**

NRC ATTENDENCE

April 8, 2008

| | NAME | NRC ORGANIZATION |
|---|---|---|
| 1 | J. Gilmer | NRO |
| 2 | J. Ashcraft | NRO |
| 3 | H. Wagage | NRO |
| 4 | A. Notafrancesco | RES |
| 5 | H. Esmaili | RES |
| 6 | A. Droed | NRO |
| 7 | G. thoma | NRO |
| 8 | W. Wang | NRO |
| 9 | T. Attard | NRR |
| 10 | P. Yausky | NRR |
| 11 | R. Gaoel | NRO |
| 12 | J. Bongarra | NRO |
| 13 | A. Cubbage | NRO |
| 14 | M. Scott | NRR |
| 15 | G. Hammer | NRO |
| 16 | S. LaVie | NSIR |
| 17 | A. Hiser | NRR |
| 18 | R. Taylor | NRR |
| 19 | D. Galvin | NRO |
| 20 | M. Yoder | NRR |
| 21 | J. Donoghue | NRO |
| 22 | J. Mitchell | RES |
| 23 | M. Snodderly | NRO |
| 24 | A. Kuracky | RES |
| 25 | S. Arndt | NRR |
| 26 | D. Santos | RES |
| 27 | A. Boatright | NRR |
| 28 | M. Salaj | RES |
| 29 | G. Kelly | NRO |
| 30 | M. Gutierrez | RES |
| 31 | R. Sydnor | RES |
| 32 | P. Rebstock | RES |
| 33 | C. Douitt | NRR |
| 34 | D. Helton | RES |
| 35 | C. Lui | RES |

OUTSIDE ORGANIZATIONS

| | | |
|---|---|---|
| 1 | D. Piedmeyer | GE Hitachi |
| 2 | J. Kinsey | GE Hitachi |
| 3 | G. Wadkins | GE Hitachi |
| 4 | A. Levin | AREVA |
| 5 | W. Marquino | GE Hitachi |
| 6 | J. Diaz-Quiroz | GE Hitachi |
| 7 | R. Stattel | GE Hitachi |
| 8 | M.D. Alamgir | GE Hitachi |
| 9 | T. Chu | BNL |

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

**Appendix IV**

**May 13, 2008**

**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
   2.3) 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. ***BREAK*****

3) 10:15 - 11:45 A.M. Risk Assessment Standardization Project (Open) (GEA/HJV)
   3.2) 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. ***LUNCH*****

4) 1:45 - 3:45 P.M. 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. ***BREAK*****

5) 4:00 - 5:00 P.M. Status of the Development of Rules and Regulatory Guidance in 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."

**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 P.M. Preparation of ACRS Report (Open)
Discussion of proposed ACRS report on:
13.1) ARTIST Test Program (JSA/DEB)

**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.

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Appendix IV

LIST OF DOCUMENTS FROM THE
552ND ACRS MEETING MAY 8-10, 2008

Agenda Item 2:
Selected Chapters of the SER Associated with the ESBWR Design Certification Application

1. Proposed Schedule
2. Status Report
3. References
4. ACRS ESBWR Letters Issued
5. ESBWR Subcommittee Consultant Reports

Agenda Item 3:
Insights from PHEBUS – FP Tests

6. Proposed Schedule
7. Status Report
8. References

Agenda Item 4:
Draft NUREG/CR Report on PRA Methods for Digital Systems

9. Table of Contents
10. Proposed Agenda
11. Status Report
12. Draft NUREG/CR “Approaches for Using Traditional PRA Methods for Digital Systems”, including the following:
    – Appendix A, “Summary Report of the External Review Panel Meeting on Reliability Modeling of Digital Systems (May 23–24, 2007)”
    – Appendix B, “Detailed FMEA of the DFWCS at Different Levels”
    – Appendix C, “Modeling of Software Failures”
    – Appendix D, “Other Methods for Modeling Digital Systems”

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

# Presentation to the ACRS Full Committee

ESBWR Design Certification Status Update
Chapters 4 and 15
Bruce Bavol - Project Manager

Chapters 6 and 21
Tom Tai – Project Manager

May 8, 2008

# ACRS Full Committee Presentation
# ESBWR Design Certification Status
# Chapter 4

## RAI Status Summary - Since January 2008

- Additional Resolved RAIs – 14
- New RAIs issued – 23
  - (Associated with Topical Report reviews)
- Current Number of Open Items - 39

# ACRS Full Committee Presentation
# ESBWR Design Certification Status
# Chapter 15

RAI Status Summary - Since January 2008

- Additional Resolved RAIs – 7
- New RAIs issued – 27
  - (Associated with Topical Report reviews)
- Current Number of Open Items - 45

# ACRS Full Committee Presentation
# ESBWR Design Certification Status
# Chapters 4 and 15

## New Topical Reports/Revisions Currently Under Review - Since January 2008

- NEDE-33243P Rev 1 “Marathon Control Rod Nuclear Design Report”
- NEDE-33244P Rev 1 “Marathon Control Rod Mechanical Design Report”
- NEDC-33326P “GE14E Initial Core Design Report”
- NEDC-33413P “Full Scale Critical Power Testing of GE14E and Validation of GEXL14”
- NEDE-33338P “ESBWR Feedwater Temp Operating Domain Accident Analyses”
- NEDC-33337P “ESBWR Initial Core Transient Analyses”

# ACRS Full Committee Presentation
# ESBWR Design Certification Status
# Chapters 4 and 15

## Staff Proposes Future Subcommittee Meetings

- NEDE-33338P "ESBWR Feedwater Temp Operating Domain Accident Analyses"
- Other topical reports as needed

# ACRS Full Committee Presentation
# ESBWR Design Certification Status
# Chapter 6

<u>RAI Status Summary - Since January 2008</u>

- Additional Resolved RAIs - 54
- New RAIs Issued – 0
- Current Number of Open Items - 37

# ACRS Full Committee Presentation
# ESBWR Design Certification Status
# Chapter 21

RAI Status Summary - Since January 2008

- Additional Resolved RAIs - 7
- New RAIs Issued - 4
- Current Number of Open Items - 31

# ACRS Full Committee Presentation ESBWR Design Certification Status Chapter 21 New RAIs

- List summary of new RAIs
  - RAI 21.6-111, Requested a description of the process for accounting for the void history bias in the TRACG04 nodal void reactivity coefficient
  - RAI 21.6-112, Requested GEH to address non-condensable gases and steam moisture flow in the GDCS lines
  - RAI 21.6-113, Requested information on how TRACG nodalization and flow regime maps in the chimney capture the phenomenon of non-fully developed flow at the inlet
  - RAI 21.6-114, Requested information on how TRACG nodalization and flow regime maps in the chimney capture the phenomenon of flow oscillation caused by the turbulence of the slug/chum regime or transition to angular flow regime
- Staff waiting for responses from GEH

# ACRS Full Committee Presentation
# ESBWR Design Certification Status
# Chapters 6 and 21

Staff Proposes Future Subcommittee Meetings:

- Discussion of Containment Analysis open items
- Discussion of TRACG Open Items

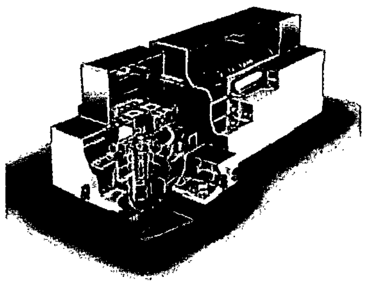
# Discussion/Committee Questions

# Gravity-Driven Cooling System (GDCS) Interaction with Steam and Non-Condensables

**ACRS Meeting**

MD Alamgir, PhD
May 8, 2008

GE Hitachi Nuclear Energy



1

## GDCS Interaction with Steam & Non-Condensables (ACRS Questions & RAI 21.6-112)

Q1: Will RPV Side Counter Current Flow Limiting (CCFL) Degrade GDCS Performance?
A1: Effect of CCFL on GDCS Flow Not Significant

Q2: Will Non-Condensables Degrade GDCS Flow?
A2: Insignificant Effect of Non-Condensables on GDCS Flow

- Summary
- Background/Supporting Information (PMK-2 Test Data)
- TRACG Sensitivity Results - Shows Insignificant Effects
- *RAI 21.6-112 Response to Contain Details (In Preparation)*

2

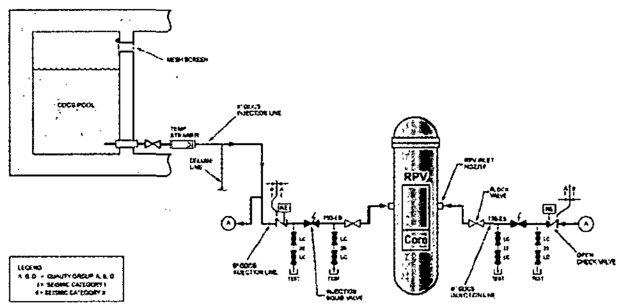
## SUMMARY

**Not Significant...**

- Effect of CCFL on GDCS Flow
  - Large Steam Condensing Capacity – No CCFL
  - Large GDCS Static Head Dominant Over CCFL
  - TRACG Models CCFL and Horizontal Stratified Flow
  - TRACG Sensitivity Confirms Insignificant CCFL Effect
- Effect of Non-Condensables on GDCS Flow Degradation
  - Insignificant Impact - No Realistic Mechanism
  - Good GDCS Pipe Routing/Desirable Slopes for Venting
  - Water Seal
  - TRACG Models Non-Condensables
  - TRACG Sensitivity with Non-Condensables in GDCS Line/Pool Confirms Insignificant Effect on GDCS Flow

3

## GDCS Schematic

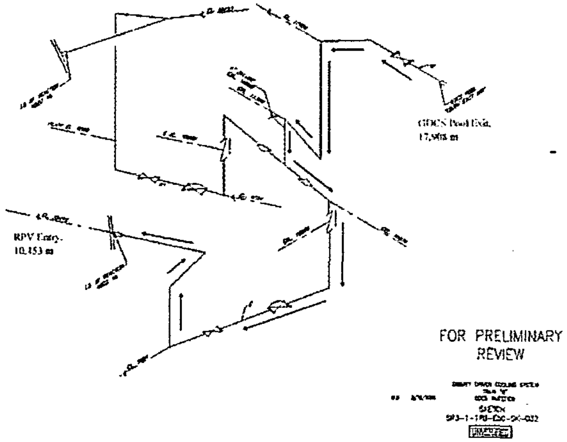



ESBWR Gravity Driven Cooling System Injection Piping Schematic Diagram
Division A Shown

UNVERIFIED

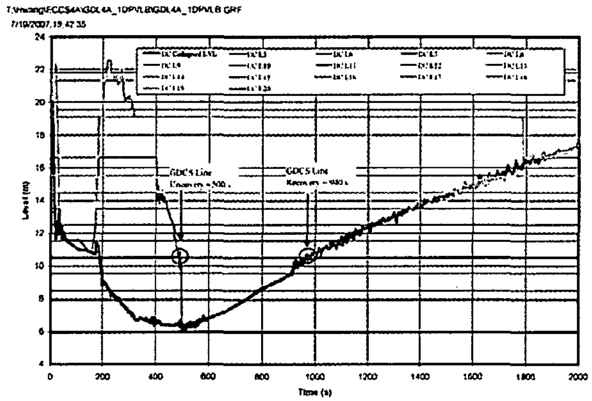
4

## GDCS Routing – Schematic (Unverified)




5

## GDCS Line Uncovery/Recovery (Downcomer Level)
## GDCS Drain Line Break LOCA

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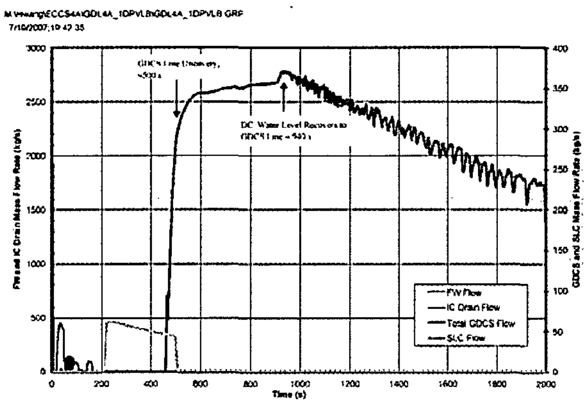
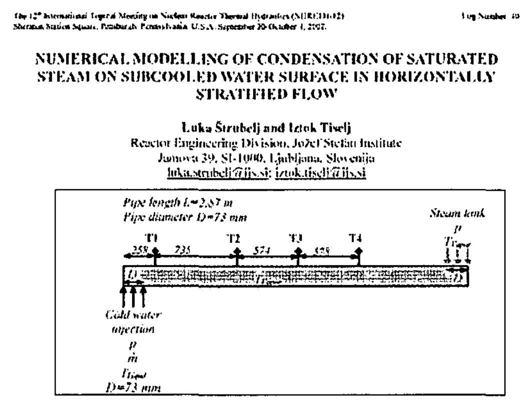
## GDCS Flow
## GDCS Drain Line Break LOCA




7

## Experiments & CFD Analysis Showing Quick Filling in Steam-Water Pipes


NUMERICAL MODELLING OF CONDENSATION OF SATURATED STEAM ON SUBCOOLED WATER SURFACE IN HORIZONTALLY STRATIFIED FLOW

Luka Štrubelj and Iztok Tiselj
Reactor Engineering Division, Jožef Stefan Institute
Jamova 39, SI-1000, Ljubljana, Slovenija



8

## PMK-2 Tests: Pipe Full in <10 secs

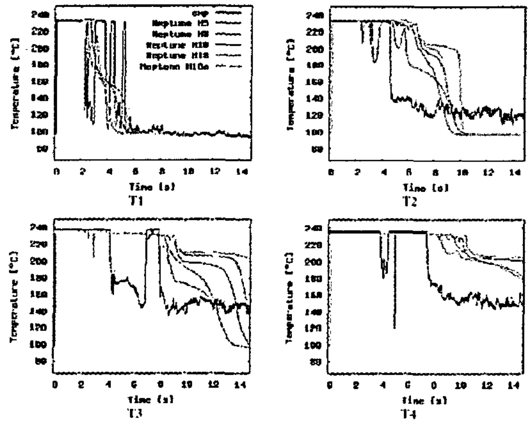



Figure 8 Local temperatures in experiment and simulation with Neptune_CFD on different grids.

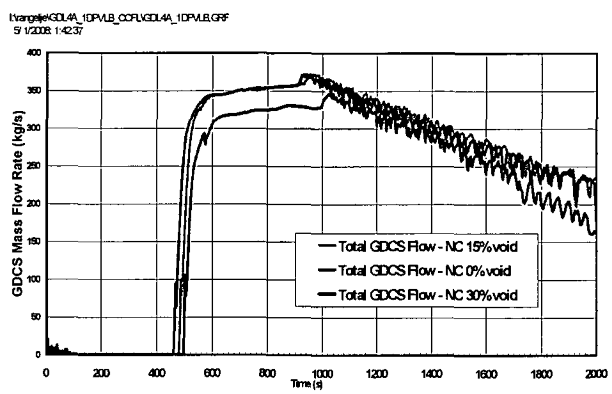
9

## Non-Condensables - Minor Impact on GDCS Flow

- TRACG Sensitivity Results with Unrealistically High 15% to 30% Initial Non-Condensable (NC) Void in GDCS Line/GDCS Pool
  - Uninterrupted GDCS Flow to RPV after LOCA
  - NC Vents Effectively to GDCS Pool Gas Space
  - Magnitude of Total GDCS Flow to RPV – Minor Impact
  - Minimum Chimney Level – Very Little Impact

10

## Initial NC Sensitivity (GDCS Drain Line Break LOCA) (Unverified/Preliminary)




11

# Conclusion

- Effect of CCFL on GDCS Flow
  - Not Significant
- Effect of Non-Condensables on GDCS Flow
  - Not Significant
- RAI 21.6-112 Response – In Preparation

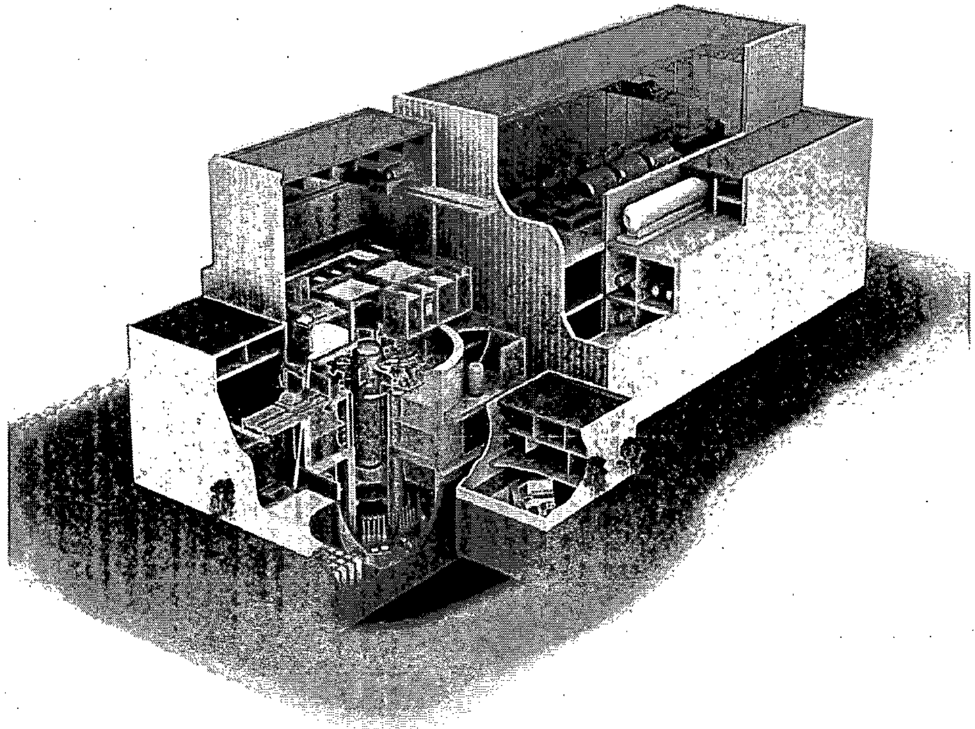
12

# ESBWR Loss-of-Coolant Accident (LOCA) Containment Response

**ACRS Meeting**

Wayne Marquino
Chester Cheung
May 8, 2008

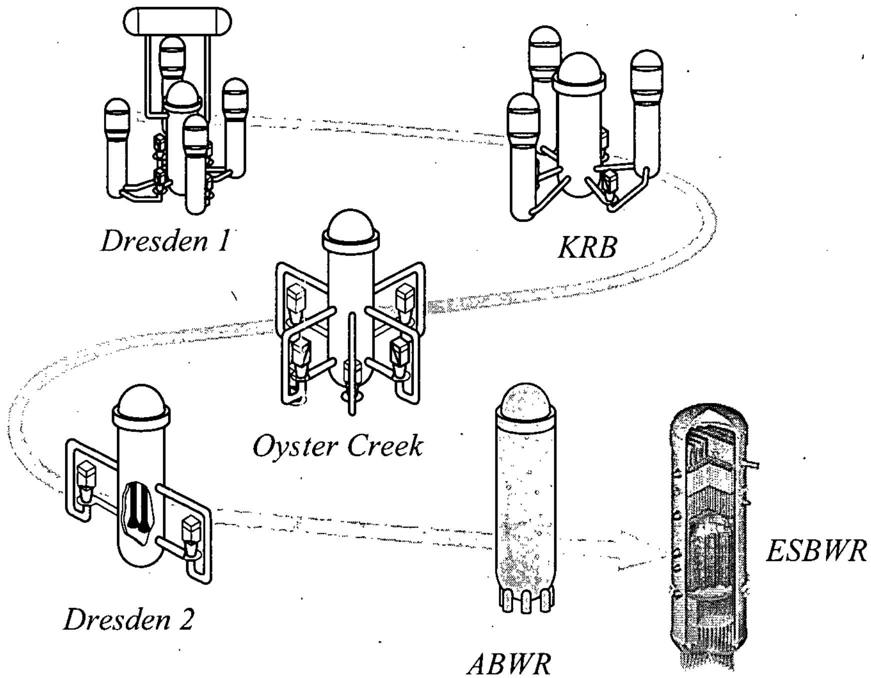
GE Hitachi Nuclear Energy



# ESBWR Chapter 6, Engineered Safety Features Selected Topics for ACRS Full Committee

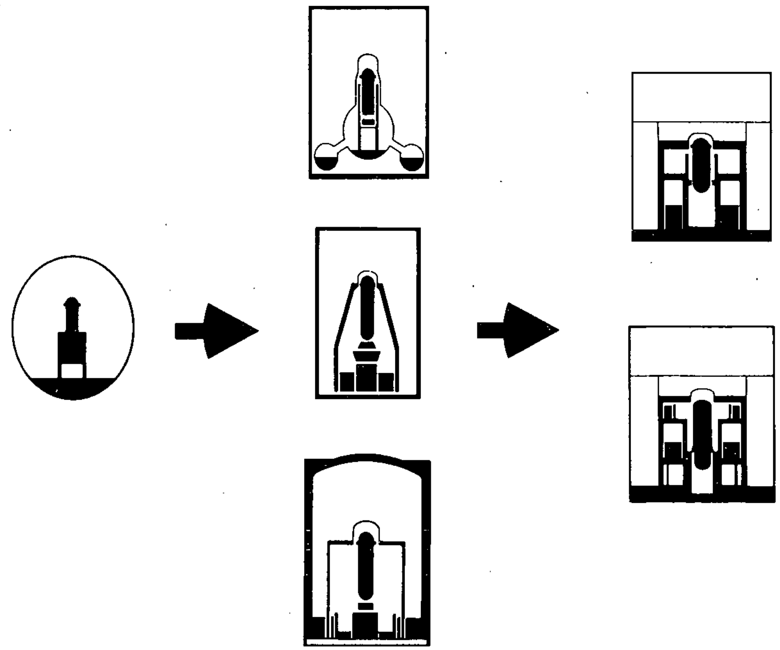
- Gravity-Driven Cooling System (GDCS) Drain with Reactor Pressure Vessel (RPV) Nozzle Uncovered
- LOCA Non-Condensable Gas Distribution
- ESBWR Vacuum Breakers and Vacuum Breaker Isolation Valves

# BWR Evolution

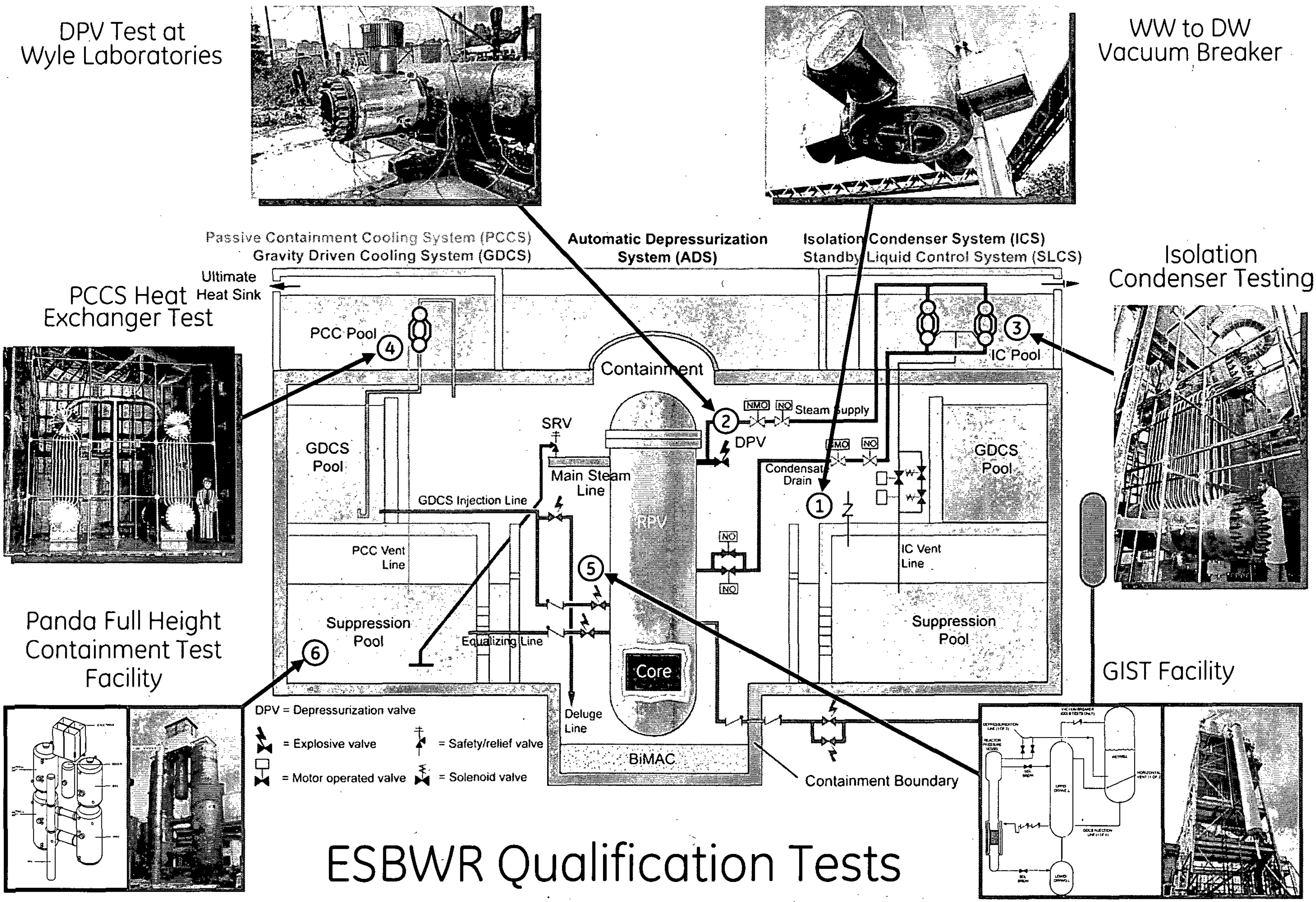



Gradual evolution of BWRs to ABWR to ESBWR

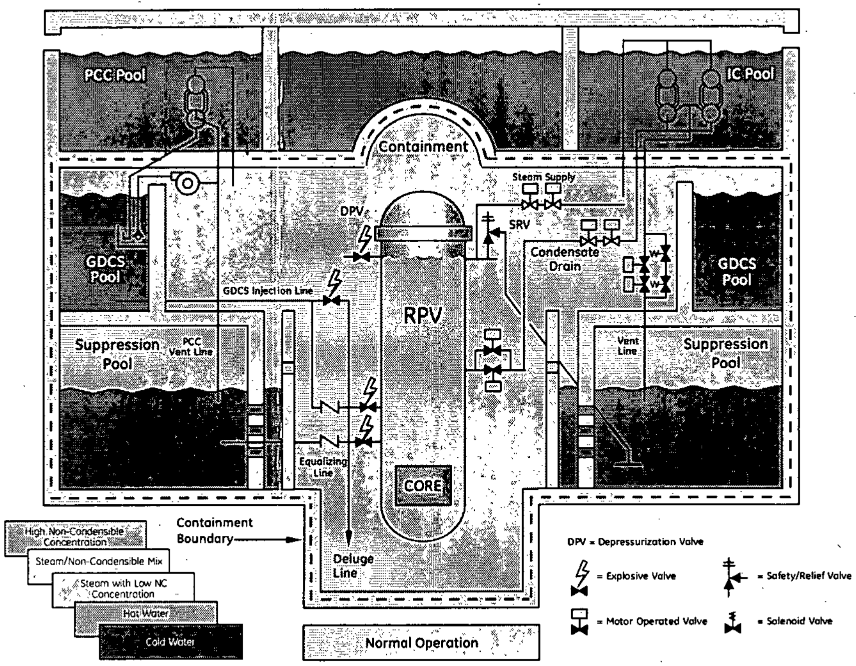
# BWR Containment Evolution

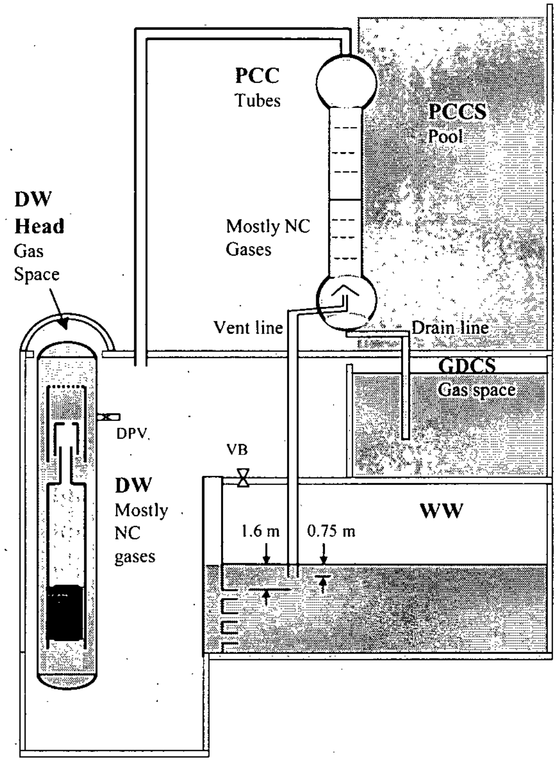


# ESBWR Qualification Tests



# Normal Operation

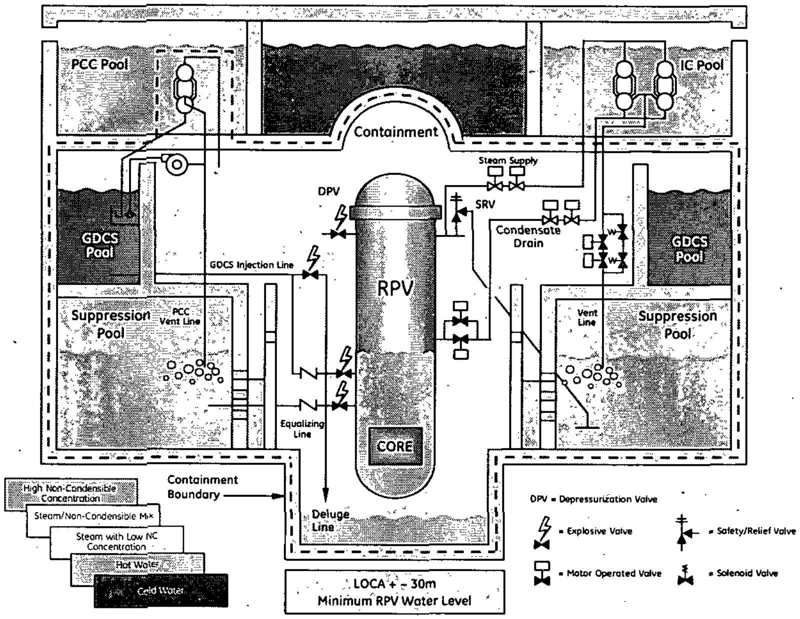


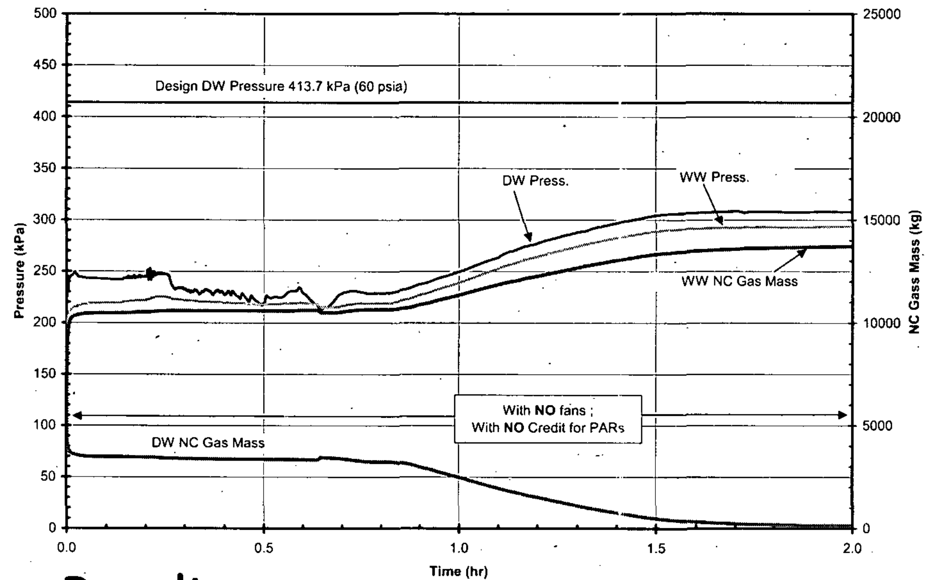




| Plant Parameter | Bounding Value |
|---|---|
| – RPV Power | 102% |
| – PCC pool temperature | 43.3°C (110°F) |
| – DW Pressure | 110.3 kPa (16.0 psia) |
| – DW Temperature | 46.1°C (115°F) |
| – WW Pressure | 110.3 kPa (16.0 psia) |
| – WW Temperature | 43.3°C (110°F) |
| – Suppression pool Temp. | 43.3°C (110°F) |
| – GDCS pool temperature | 46.1°C (115°F) |
| – Suppression pool level | 5.50 m |
| – DW relative humidity | 20% |
| – RPV pressure | 7.274 MPa (1055 psia) |
| – RPV Water Level | NWL*+0.3m |

| Model Parameter | Bounding Case | Bounding Value Used |
|---|---|---|
| Critical Flow (PIRT84) | - 2 sigma | 0.81 |
| Decay Heat | + 2 sigma | + 2 sigma |
| Surf. Heat Transfer (PIRT07) | Lower bound | 1 |
| PCC inlet Loss (k/A2) | + 2 sigma | 1440.2m-4 |
| PCC Heat Transfer (PIRT78) | - 2 sigma | 0.902 |
| VB Loss (k/A2) | + 2 sigma | 211.4m-4 |

# Main Steam Line Guillotine Break LOCA + ~30min Min. RPV Water Level



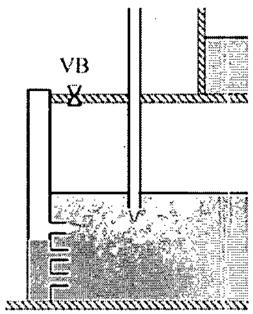



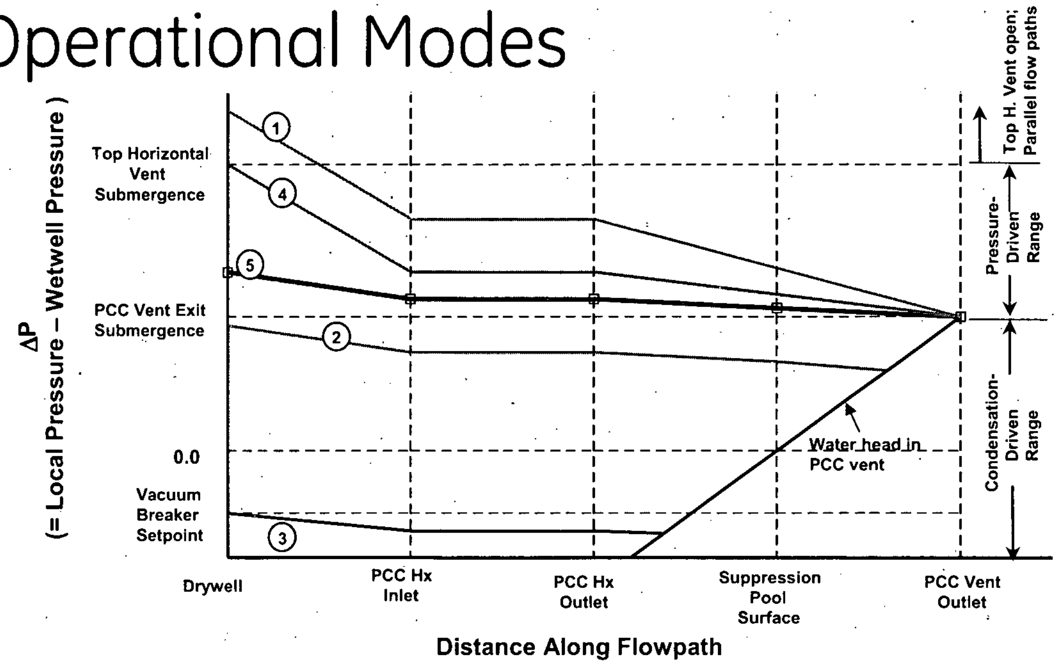

## Results

- Immediately after LOCA, DW pressure increases rapidly, leading to clearing of the PCC vent line and main vents
- During this blowdown period, ~ 50% of the DW NC gases are purged into the WW; WW pressure increases
- GDCS injection starts at ~ 0.2 hr; from 0.2 to 0.7 hr. the RPV steaming reduced due to subcooled GDCS water
- RPV steaming resumes after 0.7 hr.,
- The remaining NC gases in DW mix with the RPV steam, through the PCC and purge into the WW
- DW and WW pressures continue to increase as the NC gases are purged into the WW

# PCC Operational Modes

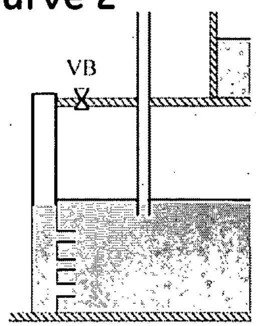
## Curve 1
- Blowdown period (0~10 min.)
- PCCS Power < Decay Heat
- Top Horizontal vent open
- Flows from H. vent and PCC vent



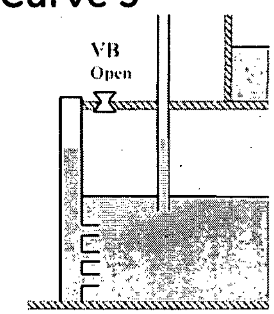




## GDCS period (~10 min. to ~ 1 hr.)
- PCCS Power > Net steaming rate in DW
- Top Horizontal vent close, and no PCC vent flow
- VB may open
- PCC flow driven by condensation

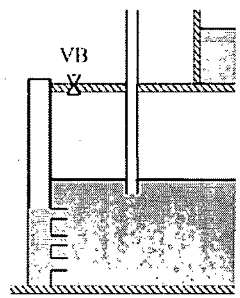
## Curve 2



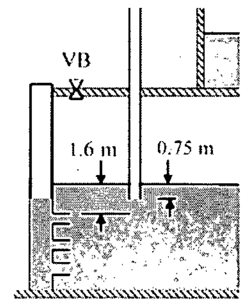

## Curve 3




## Curve 4
- Early part of long-term PCCS period (~1 to ~6 hrs.)
- PCCS Power < Decay Heat
- Top Horizontal vent close
- PCC vent flow only

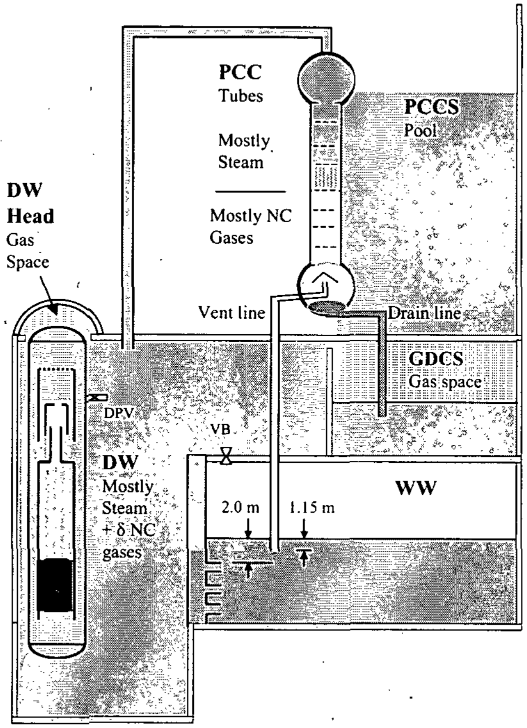



## Curve 5
- Long-term PCCS period (~6 to 72 hrs.)
- PCCS Power = Decay Heat
- Top Horizontal vent close
- PCC vent flow only




Initial Suppression Pool level

# Main Steam Line Break +15 hrs

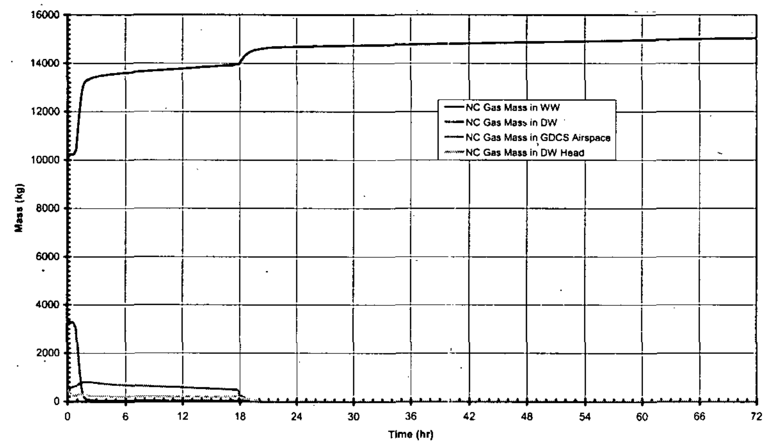



### 15 hrs. after MSL Break

- Sup. Pool Level increased 0.4 m
- Small amount of NC gases trapped in DW head and GDCS gas spaces
- Water accumulated in the lower DW ~ 3m deep

## Results

- PCCS Power = Decay Heat
- Top Horizontal vent close
- PCC vent flow only

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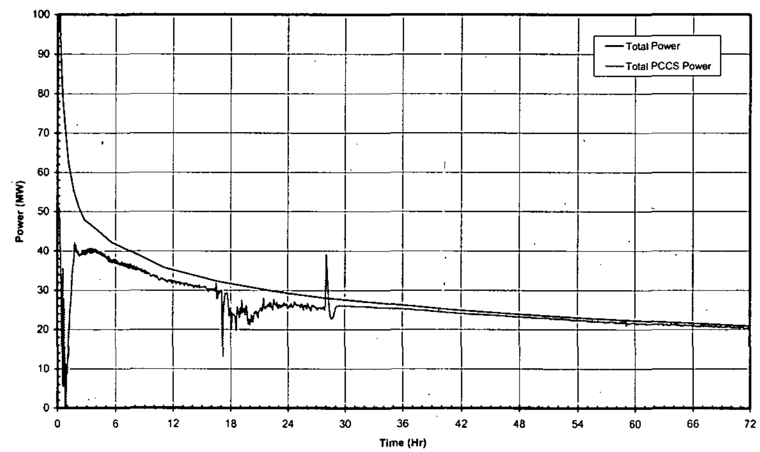
8/22/2007: 6:10:21




## Inputs

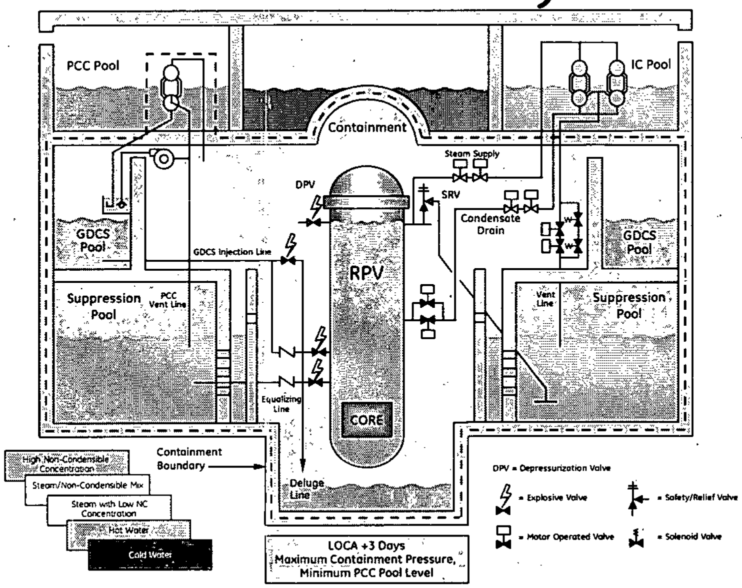
Radiolytic gas production in core

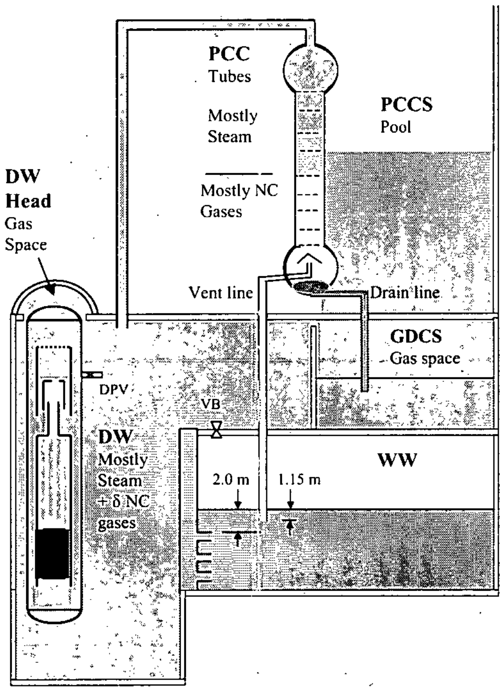
## Results

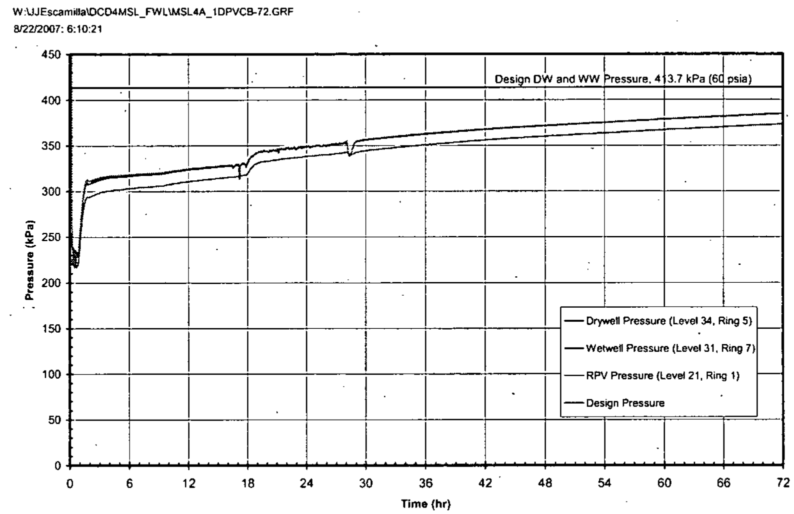
PCC continually venting NC gas to WW produce DW-WW pres. Dif.
Pres. Difference drives steam to WW airspace through leak path
Worst leak area which puts containment at design pressure 2 cm2
The steady increase in DW and WW pressures is due to:

(a) Continued generation of radiolytic gases and purging them into the WW
(b) Hot gases trapped in between the I-beams (at top of WW)

# LOCA +3 Days




## INPUTS

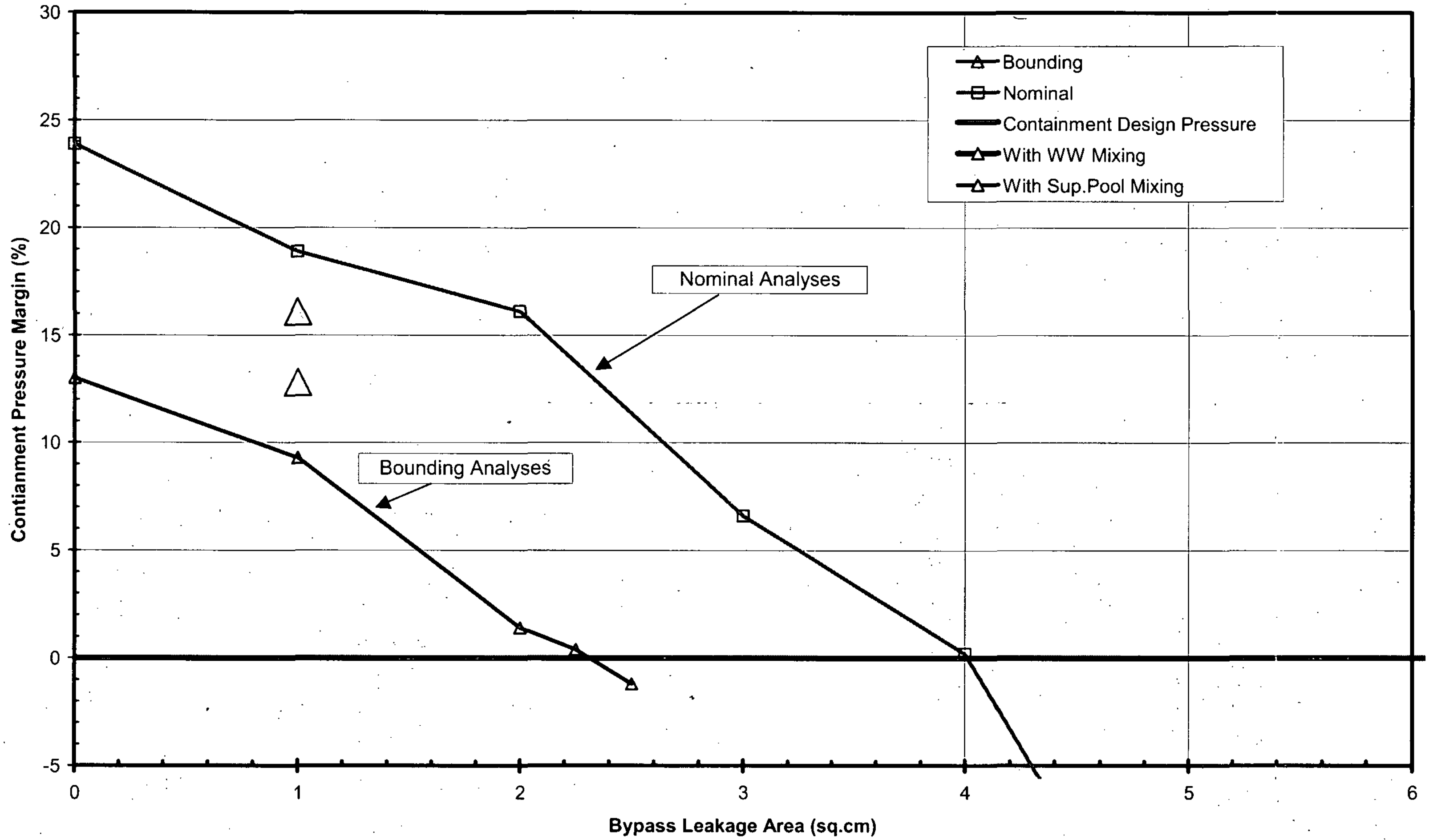
TRACG is a best estimate model, but application is conservative in selected areas:

- Radiolytic gas production per NRC Reg Guide
- Drywell nodalization to mix DW non-condensables
- Wetwell Airspace forced stratification
- Suppression pool forced stratification
- Heat conduction
- Lower DW to WW (incr. Pcont)
- WW to Reactor Building Heat Transfer (decr. Pcont, but no DW>RB credit)
- DW>WW leak set to put containment at design pres. For Rev. 5

## RESULTS

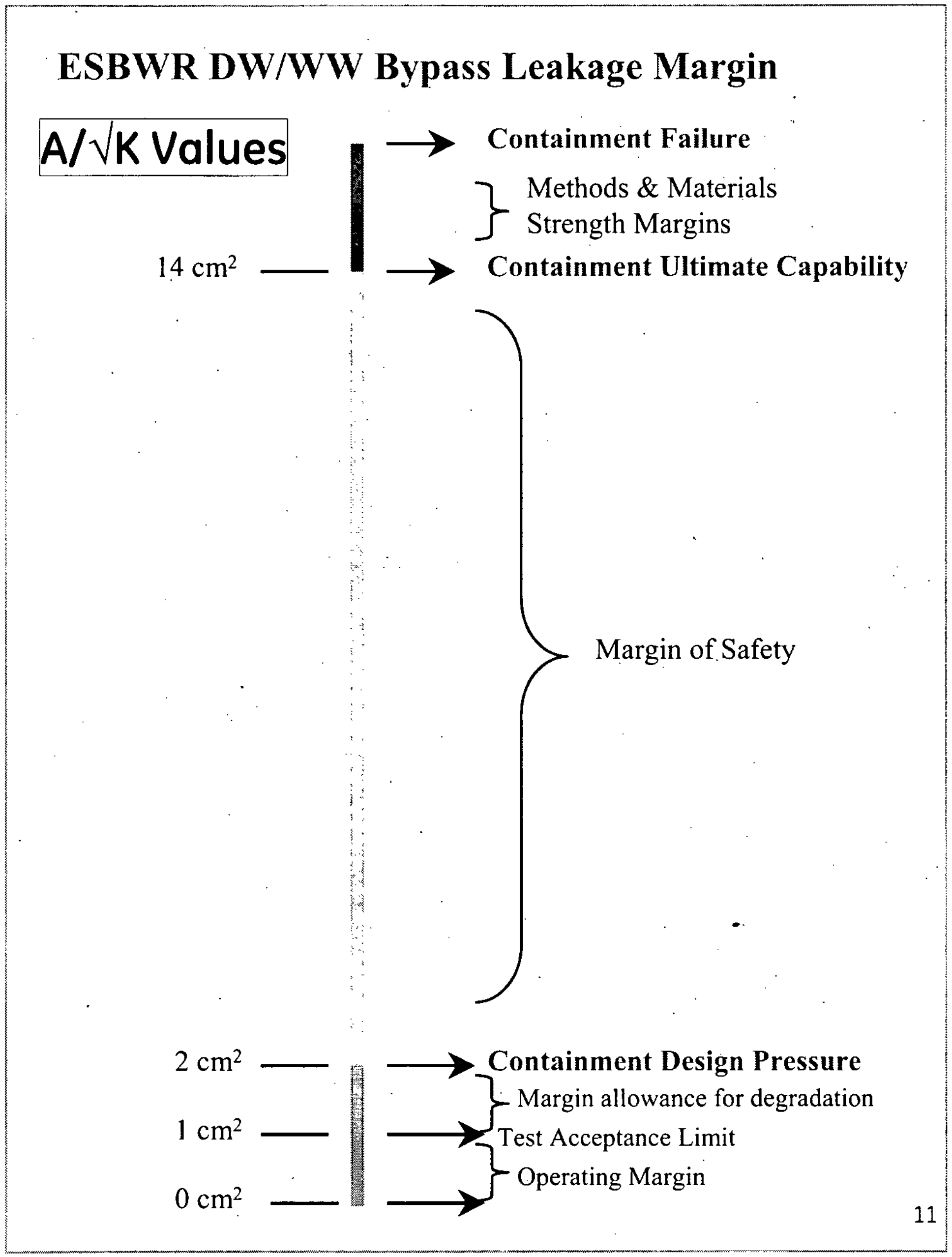
- Top 3/8 PCC tube uncovered
- Bottom 4/8 PCC tube filled with NC gas
- PCC operation is self-regulated to match the decay heat
- Pressure margin is ~ 9% at 3 days

# ESBWR DW Pressure Margins versus Leakage Area (from 72 hrs analyses)

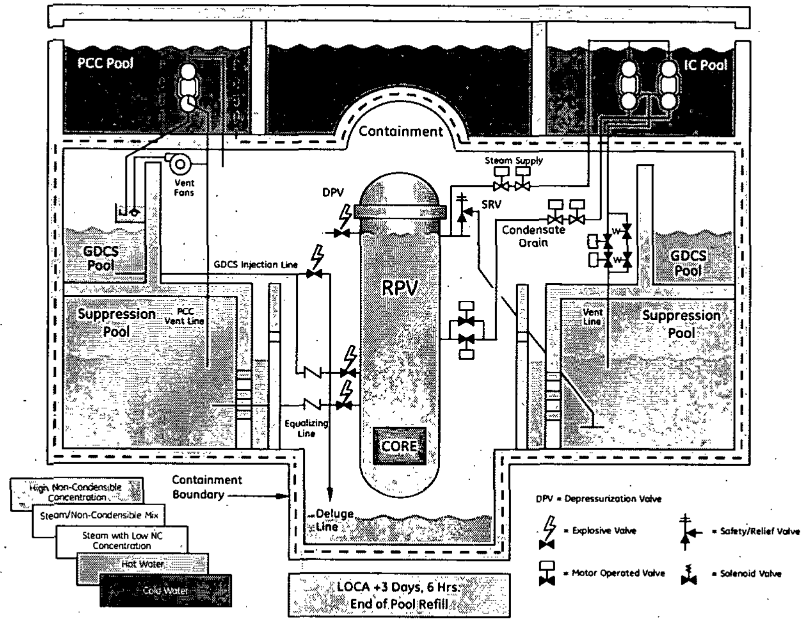


# Bypass Leakage Margin Illustration

- **Test acceptance limit**
  - > 1 $cm^2$
  - > Realistic leakage area through diaphragm floor << 1 $cm^2$
  - > 50% of the licensing basis
- **Licensing basis bypass leakage area**
  - > 2 $cm^2$
  - > Supported by bounding calculations with conservative modeling and plant conditions
    - – No mixing in WW gas space and sup. Pool for 72 hrs
    - – No credit for PARs
    - – Purging all NC gas from the hideout volumes in DW into WW
- **Large margin of safety to the Containment Ultimate Capability**

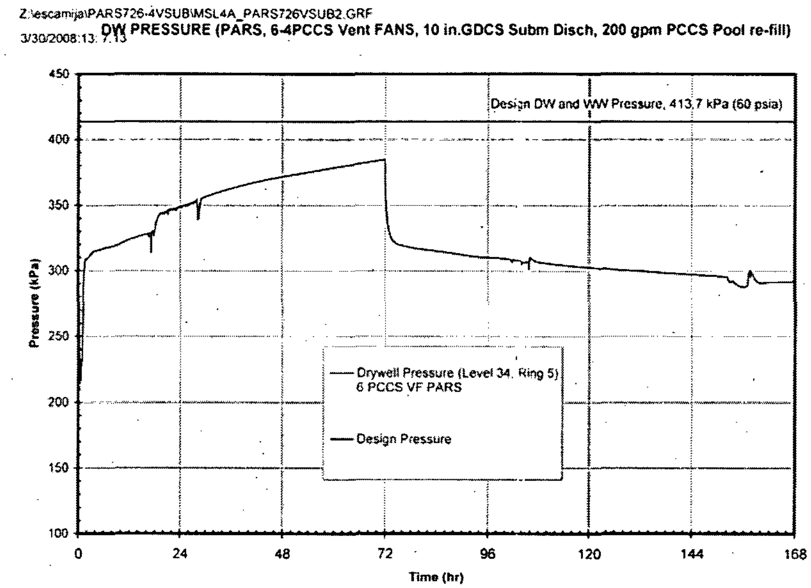


# LOCA +3 Days, 6 Hrs.




## Inputs

| | |
|---|---|
| Fan Flowrate at a calculated rated head of $(\Delta P/\rho) = 602\ m^2/s^2$ (6480 $ft^2/s^2$) and fluid density ($\rho$) of 2.27 $Kg/m^3$ (0.137 $lb/ft^3$). | 0.343 $m^3/s$ (727 CFM) |
| Pool refill flowrate at 310.9 K (100F). | 0.0127 $m^3/s$ (201 gpm) |
| Discharge submergence | 0.254 m (10 in) |
| PARS credited at | 72 hrs |



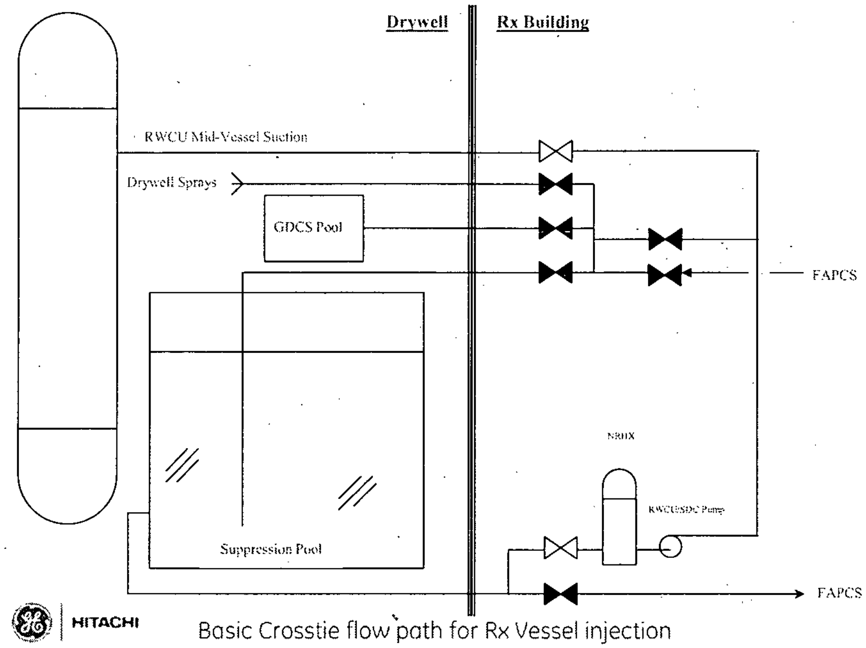

MSL4A_PARS726VSUB2_DCD_6-3_Fig1n.xls Pressure (72hr) 6-4V 0-7-16 (2)

## Results

- Pool refilling and PCC vent fans increase PCC heat removal rate
- DW pressure drops rapidly, resulting in vacuum breaker openings and reversed leakage flow
- NC gases flow from WW back into DW
- Leakage flow reverses from 3 to 6 days and back into the DW; instead of purging into the WW
- NC gases continue to relocate from WW to DW
- Continued relocation of NC gases from WW to DW results in continued reduction of pressures
- The pressure reduction is proportional to the amount of NC gas mass reduced in the WW
- Pressure margin increases from ~9% at 3 days to ~33% at 7days

# Post LOCA Recovery

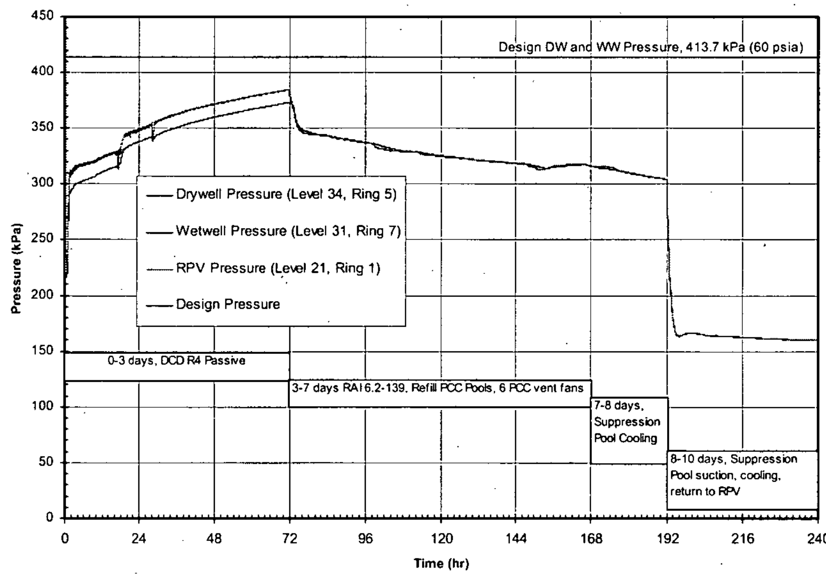
When conditions permit, active systems can be used to cool the suppression pool and the reactor vessel




Basic Crosstie flow path for Rx Vessel injection

## Inputs (RWCU/SDC)

| NRHX Characteristic | Value |
|---|---|
| Shell Side Flow Rate | 1590 $m^3/hr$ (7000 gpm) |
| Tube Side Flow rate (Suppression Pool Cooling mode) | 605 $m^3/hr$ (2665 gpm) |
| Tube Side Flow rate (Reactor Vessel injection mode) | 605 $m^3/hr$ (2665 gpm) |
| Shell Side Inlet temperature | 38.3°C (101°F) |
| Heat Exchanger K value | 4.6E+05 J/sec °C (8.7E+05 Btu/hr °F) |

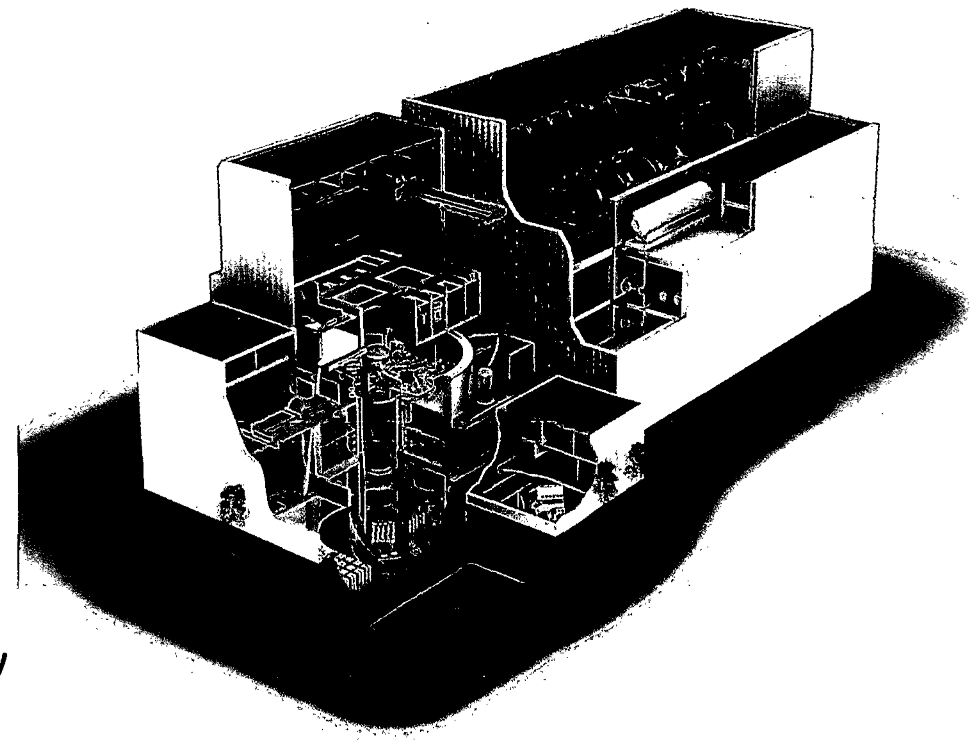


# Summary

- ESBWR/SBWR Testing qualified TRACG for treatment of non-condensables, tests included Air & Helium
- TRACG inputs for Non-condensable Gas and Heat Structure conservative for Containment Pressure
- Peak containment pressure limited by:
  - Pressure Suppression in the Suppression Pool
  - Primary Containment Cooling System (PCCS)
- After 3 days containment pressure is reduced and controlled by:
  - Refill of the PCCS pools
  - PCC Vent Fans evacuating non-condensable gases
  - Passive Autocatalytic Recombiners (recombine radiolytic H2-O2)
- When conditions permit, containment pressure can be further reduced using:
  - Cooling Suppression Pool water and returning to pool
  - Cooling Suppression Pool water and injecting in the RPV

# ESBWR Vacuum Breaker Isolation Valve (VBIV) Control Logic and Seat Configuration

**ACRS Meeting**



Jesus Diaz-Quiroz
May 8, 2008

GE Hitachi Nuclear Energy

# VBIV Control Primary Demand Logic

## VBIV Control Primary Demand Logic Features:

- Independent of Q-DCIS and other Instrumentation and Control Systems
  - Processed by ATWS/SLC NUMAC Components
- Logic for Each VBIV is Independent of the Other VBIVs
- Logic for Each VBIV uses 4 Divisions of Instruments to Prevent Inadvertent Closure and to Meet N-2 Requirements
- Failure of the Logic for One VBIV does Not Affect the Logic of the Other VBIVs
- Logic Does Not Attempt to Quantify the Vacuum Breaker (VB) Leakage Rate, Only Determine if a VB is Leaking Based on Sensed Differential Temperature

## The VBIV Primary Demand Logic Signal will be the Result of:

- Temperature Differential Between the VB Cavity and Wetwell Exceeds a Predetermined Setpoint, <u>AND</u>
- Division is NOT in Bypass, <u>AND</u>
- 2-out-of-4 Divisions Meet the Above Conditions, <u>AND</u>
- 2-out-of-4 Divisions Provide a Loss-of-Coolant Accident (LOCA) Permissive

# VBIV Control Secondary Demand Logic

## VBIV Control Secondary Demand Logic Features:

- Required if VB is OPEN when Not Desired Post Accident
- Each VB Contains 4 Proximity Probes to Indicate VB Disc Not Full Closed and 1 Proximity Probe to Indicate VB Full Open
- Same Logic Features as Primary Demand Logic Apply

## The VBIV will Close Automatically if:

- Temperature Differential Between VB Cavity and Wetwell Exceeds a Predetermined Setpoint, <u>AND</u>
- LOCA Permissive is Present, <u>AND</u>
- Proximity Probes Indicate that the VB is Full Open or Disc Not Full Closed

## VBIV Manual Control Features:

- Manual VBIV Control Available to the Operator in the Main Control Room to:
  - Open Each VBIV Individually
  - Close Each VBIV Individually
- Manual Controls are Independent for Each VBIV and are Independent of the VBIV Control Primary and Secondary Demand Logics

# Vacuum Breaker and Vacuum Breaker Isolation Valve

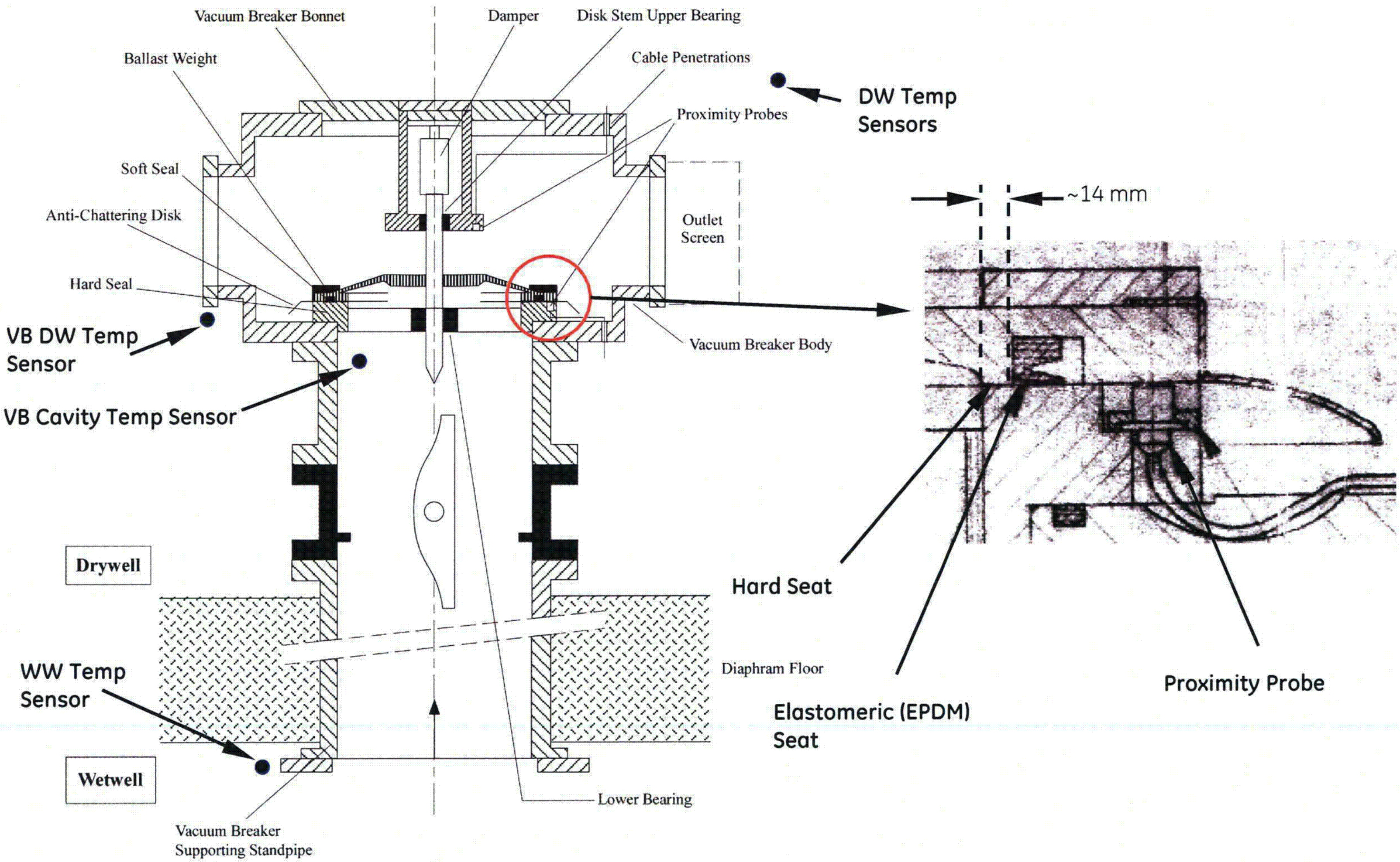



Figure 1 – Vacuum Breaker, Vacuum Breaker Isolation Valve, and Vacuum Breaker Seat Details

# LOCA Temperature Evaluation

- **VB Leakage Temperature Setpoint Evaluation Conducted to Confirm VBIV Isolation Logic for Large, Medium, Small and Very Small Break LOCAs**

- **LOCA Permissive Signal Established at: $T_{drywell} \geq 90^{o}C$:**
  - ➢ Enough Margin to Cover Normal Plant Operation
  - ➢ Reasonable Response Time to Reach Permissive Signal After LOCA
    - For Large, Medium and Small Break: ~ 0.5 to 50 seconds
    - For Very Small Break (Standby Liquid Control Line Break): ~ 600 seconds

- **VB Leakage Detection:**
  - ➢ Reference is Based on the Temperature Difference

    $\Delta T = (T_{drywell} - T_{wetwell})$
  - ➢ $\Delta T$ Varies From $25^{o}C$ to $90^{o}C$ During LOCA

# LOCA Temperature Evaluation

## VB Leakage Detection

| $\Delta T = (T_{drywell} - T_{wetwell})$ Transient after LOCA | | | |
|---|---|---|---|
| **Break** | **Time around blowdown** | **to** | **72 hours** |
| MSL | 60°C | to | 25°C |
| FWL | 68°C | to | 25°C |
| GDL | 65°C | to | 30°C |
| BDL | 90°C | to | 25°C |
| SLC | 40°C | to | 60°C |

- **When Leakage Occurs, $T_{cavity}$ will Increase and Approach $T_{drywell}$**
- **VB Leakage Signal:**

> - **$(T_{cavity} - T_{wetwell}) \geq$ X% of $(T_{drywell} - T_{wetwell})$**
> - **Where X = 80 (to be Finalized)**

# Vacuum Breaker (VB) Sealing Surfaces

- VB Primary Seat is Elastomeric EPDM Seat with a 2 mm Crush and a Secondary Hard Seat (See Figure 1)
- VB Qualification Test Program Confirmed VB Leak Tightness Under All Degradation and Aging Mechanisms (See Figure 2)
- Leak Tightness with Simulated LOCA Debris on Soft Seat Fully Tested in VB Qualification Test Program and Leak Tightness Confirmed
- Foreign Material Seat Sensitivity Tests:
  - Done at the End of the VB Qualification Program
  - Chips of Metallic Wire 12, 32, 50 mils Placed under Elastomeric Seal Where it Contacts Metal Surface
  - No Noticeable Leak Variation Occurred Compared with Leak Curve Obtained without Foreign Particles
  - Soft Seal Proved Unaffected by Scraps of Foreign Material Trapped in the Seat
- VB Disc Position Monitored by 4 Proximity Probes 90° Apart Located Below the Disc to Indicate VB Disc Full Closed or Off Seat and 1 Proximity Probe in the VB Stem Upper Bearing to Indicate VB Full Open (See Figure 1 for Details)

# Vacuum Breaker (VB) Test Program

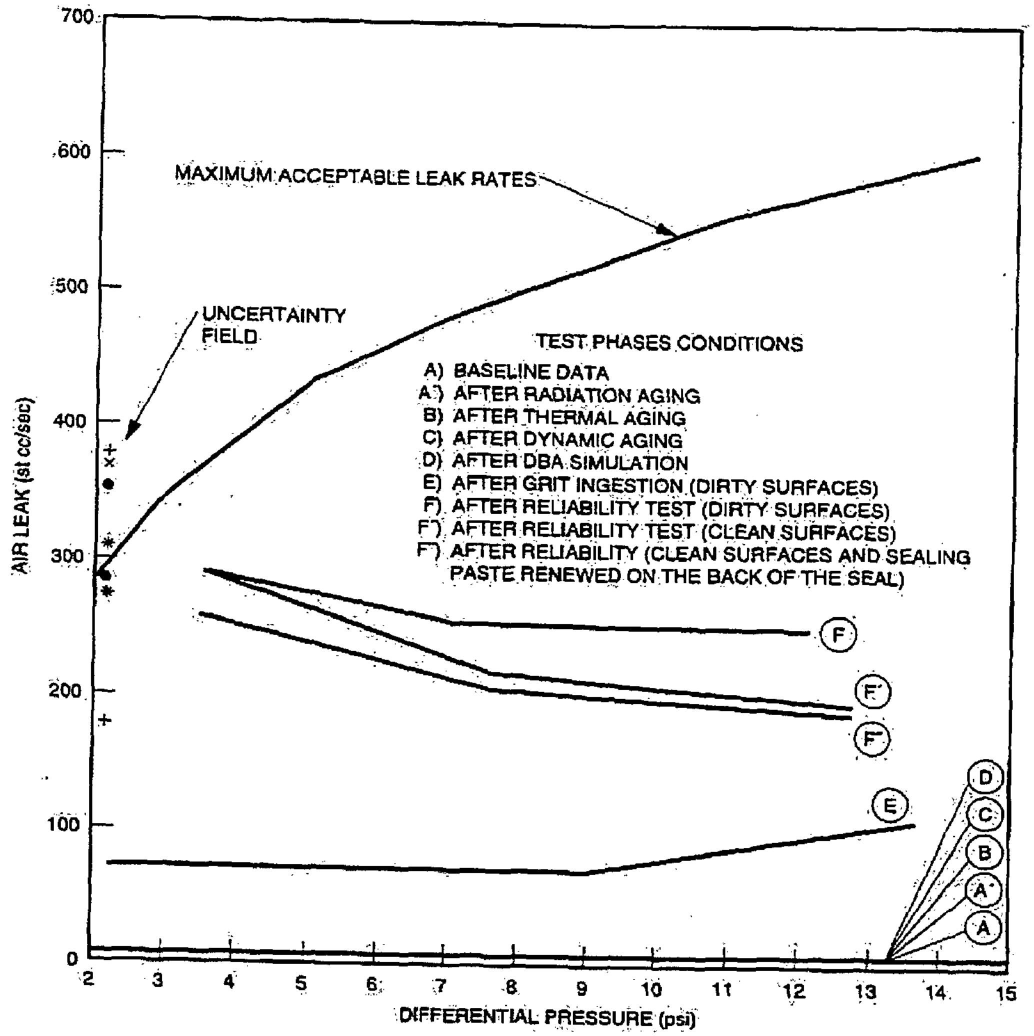
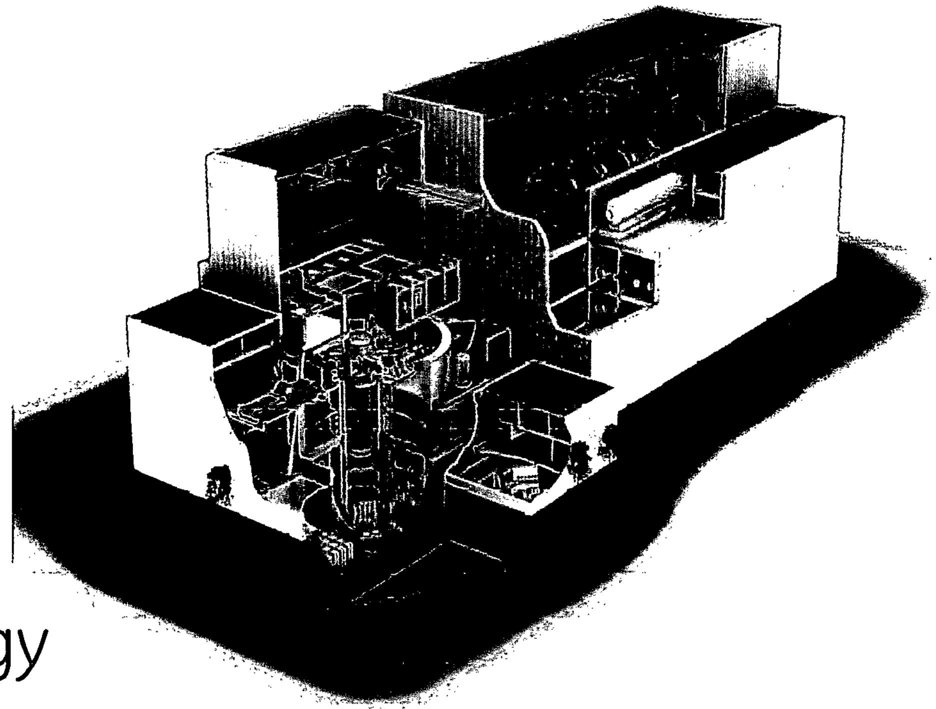



Figure 2 - VB Systematic Checks Primary Soft Seal Leak Test Results

# ESBWR Chapter 18 Human Factors

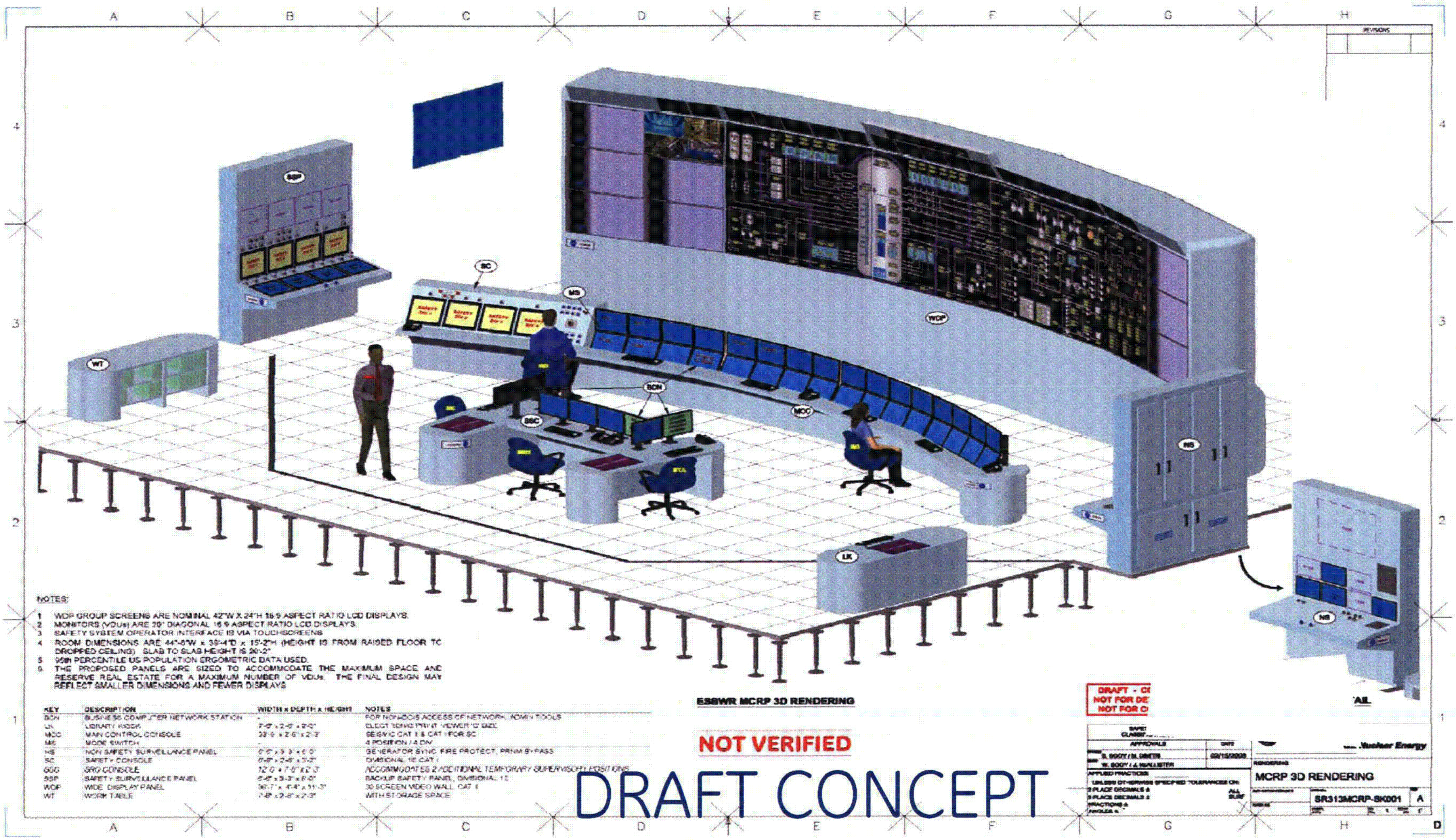
ACRS Meeting



Richard Stattel
May 8, 2008

GE Hitachi Nuclear Energy

# ACRS Meeting ESBWR Chapter 18 Human Factors



ESBWR Chapter 18 ACRS Review

# ESBWR HFE Program Highlights

- ESBWR is a "Human-Centered" Design
- Regulatory Guidance Bases is NUREG-0711 R2 and NUREG-0700 R2
- Program Experience Bases include:
  - Use of Nuclear and Non-nuclear Industry Operating Experience
  - Use of Predecessor Plant Data
  - Extensive Licensed Senior Reactor Operators (SRO) Input
- Other Program Features include:
  - Extensive Investment in Simulation and V&V
  - Integration of HFE into all Engineering Disciplines and Project Phases
  - Customer Participation and Acceptance

# Current Status – Program Planning

- DCD Chapter 18 and 12 Implementation Plans (LTRs) have been Submitted for Staff Review
- An Experienced Staff of 44 Dedicated Personnel is Established
- HFE Issue Tracking System (HFEITS) is Established

ESBWR Chapter 18 ACRS Review

# Current Status – Program Execution

- Substantial OE has been Analyzed
- Baseline Record of Predecessor Plant Data has been Collected
- Functional Requirements Analyses / Allocation of Function, Task Analyses Activities are being Performed
- Program Elements have been Prototyped through Procedures and Training

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UNITED STATES NUCLEAR REGULATORY COMMISSION
*Protecting People and the Environment*

# Presentation to the ACRS

ESBWR Design Certification Review
Chapter 18,"Human Factors Engineering"



## ACRS Presentation
## ESBWR Design Certification Review
## Chapter 18, Summary

Chapter 18 Sections
- 18.2 "HFE Program Management"
- 18.3 "Operating Experience Review"
- 18.4 "Functional Requirements Analysis and Function Allocation"
- 18.5 "Task Analysis"
- 18.6 "Staffing and Qualifications"
- 18.7 "Human Reliability Analysis"
- 18.8 "Human-system Interface Design"
- 18.9 "Procedure Development"
- 18.10 "Training Program Development"
- 18.11 "Human Factors Verification and Validation"
- 18.12 "Design Implementation"
- 18.13 "Human Performance Monitoring"

## ACRS Presentation
## ESBWR Design Certification Review
## Chapter 18, Summary

Review Team for Chapter 18

- Lead Project Manager
  - Dennis Galvin

- Lead Technical Reviewers
  - James Bongarra
  - James Higgins, BNL
  - John O'Hara, BNL



## ACRS Presentation
## ESBWR Design Certification Review
## Chapter 18, Summary

RAI Status Summary: SRP Chapter 18

- Original number of RAIs = 266
- Number of RAIs resolved = 225
- Number of Remaining Open Items = 41
- 25 RAIs resolved since the subcommittee meeting 4/9/2008
- 7 supplemental RAIs recently issued

## ACRS Presentation
## ESBWR Design Certification Review
## Chapter 18, Summary

- GEH has made considerable progress to address open items through RAI responses and document revisions
- Level of detail issues remain for some of the implementation plans
  - RAI responses submitted
  - Staff will perform an onsite review of detailed GEH procedures
- Based on progress to date, no major obstacles are expected to resolving the remaining issues
- GEH has used state of the art techniques in developing their HFE program and where staff has completed review, the program elements appear to be comprehensive



## ACRS Presentation
## ESBWR Design Certification Review
## Committee Questions

Discussion/Committee Questions

IRSN

# Main Phebus Lessons and the International Source Term Programme

**Presented to the**
**Advisory Committee on Reactor Safeguards**
**U.S. Nuclear Regulatory Commission**
**May 8, 2008**

*B. Clément*

*Institut de Radioprotection et de Sûreté Nucléaire*

*France*

IRSN

## OUTLINE

- Main lessons learnt from Phebus FP
  - Fuel degradation
  - FP and structure material release
  - FP and structure material transport in RCS
  - Thermal-hydraulics and aerosol behaviour in containment
  - Iodine chemistry
- Status of knowledge and implications
- International Source Term Programme
  - General objectives
  - Iodine studies
  - Boron carbide studies
  - Air ingress studies
  - Fission Product release studies
- Concluding remarks

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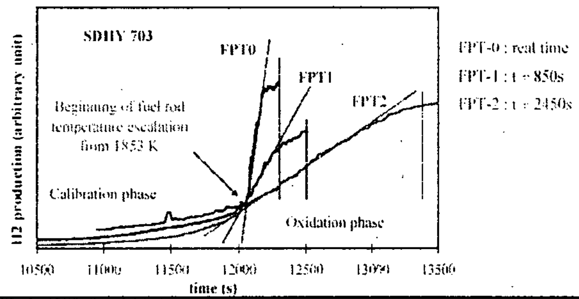
# MAIN OUTCOMES: FUEL DEGRADATION

## Cladding oxidation

More violent than expected cladding oxidation runaway in first test FPT-0)

Need to revise correlations for "cladding dislocation criteria" based on temperature and oxide scale thickness

New correct predictions of cladding oxidation and hydrogen produc



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# MAIN OUTCOMES: FUEL DEGRADATION

## Fuel relocation

Fuel liquefaction and transition from rod-like geometry to molten pool at temperatures (2600 ± 200 K) far below actual melting point of pure UO2 (3100 K)

Recent detailed analysis of FPT-0 and FPT-1 PIE show that:

- According to oxidation measurements and thermodynamic calculations, a deviation from stoichiometry of x = 0.15 must be taken into account for ceramic phase $(U_{1-y}\ Zr_y)\ O_{2+x}$
- According to recent measurements (Manara) this composition due to interactions between oxidised cladding and fuel can reduce the fuel collapse temperature to 2500-2600 K

In the mean time calculation codes can reproduce final state of degradation given suitable reduction of bulk fuel relocation temperature

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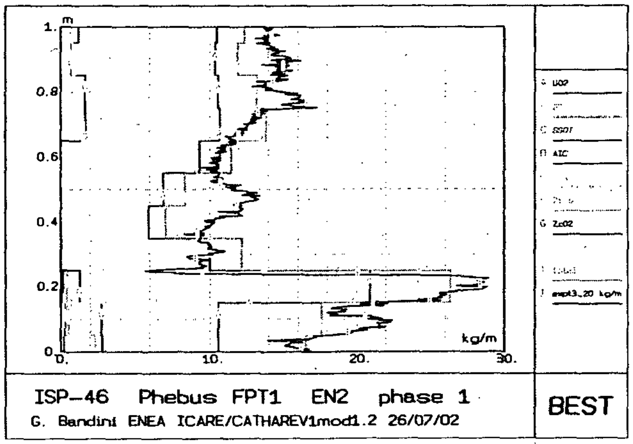
## MAIN OUTCOMES: FUEL DEGRADATION

### Fuel relocation

Example of calculated material axial distribution in FPT-1

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## FP AND STRUCTURE MATERIAL RELEASE

### FP Releases

Volatile generally well calculated even if CORSOR approach tends to overestimate kinetics at the beginning of the transient

Semi empirical models, though not describing all processes, able to do well for volatiles using consistent set of parameters for Phebus and separate-effect experiments

Situation more contrasted for less volatiles for which chemistry plays an important role

Insights gained from mechanistic codes describing repartition of fission products in different phases of the fuel and their changes with temperature and stoichiometry

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# FP AND STRUCTURE MATERIAL RELEASE

## Coupling with fuel degradation

In FPT-0 using 9-days irradiated fuel, early release of volatiles can only be explained by fuel dissolution during cladding oxidation phase

Barium release much smaller in Phébus than in separate-effect experiments - difference attributed to interactions of fuel with cladding material and maybe iron reducing barium volatility

Low release from molten pool

## Release from silver-indium-cadmium control rod

Governing phenomena well understood but modelling effort still needed especially for coupling with degradation processes

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# FP AND AEROSOL TRANSPORT IN RCS

## FP and structural material speciation

In the hot leg, iodine and cadmium were the only non condensed elements for FPT-0 and FPT-1 - CsOH was not the dominant for caesium transport

Codes calculating chemical speciation can reproduce caesium volatility and indicate the formation of caesium molybdate

Iodine is transported partly as a gas and partly as metal-iodides

Caesium iodide is not the only species for iodine transport as vapour and/or aerosol

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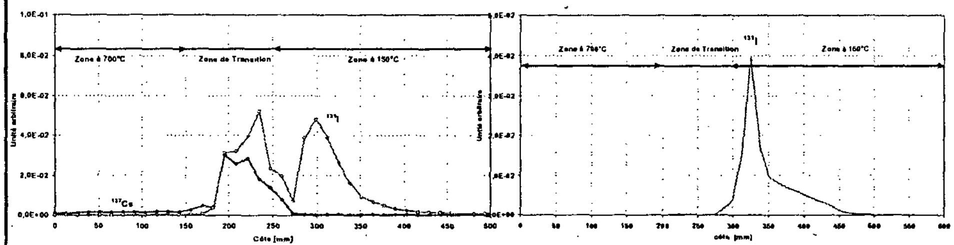
# FP AND AEROSOL TRANSPORT IN RCS

## FP and structural material speciation

Caesium and iodine deposition profiles for FPT-2



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# FP AND AEROSOL TRANSPORT IN RCS

## Deposition

- High deposition on the hot leg vertical section above the bundle where temperature drops down to 700°C underestimated by codes
- Can be accounted for by the effect of developing flow characterised by much higher mass transfer coefficients of vapours to the walls than for developed flow
- Deposition of aerosols by thermophoresis in the steam generator overestimated by codes
- Recent detailed studies taking into account the interaction between turbulence and aerosol particles give promising results

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## F/H AND AEROSOL IN CONTAINMENT

### Thermal-hydraulics

Governed by the balance between incoming steam and condensation - generally well calculated

### Aerosol depletion

Mainly by gravitational settling and diffusiophoresis - generally well calculated

Significant deposition on outer walls cannot be explained by Brownian diffusion - a model accounting for turbulence damping in the boundary layer can reproduce the results satisfactorily

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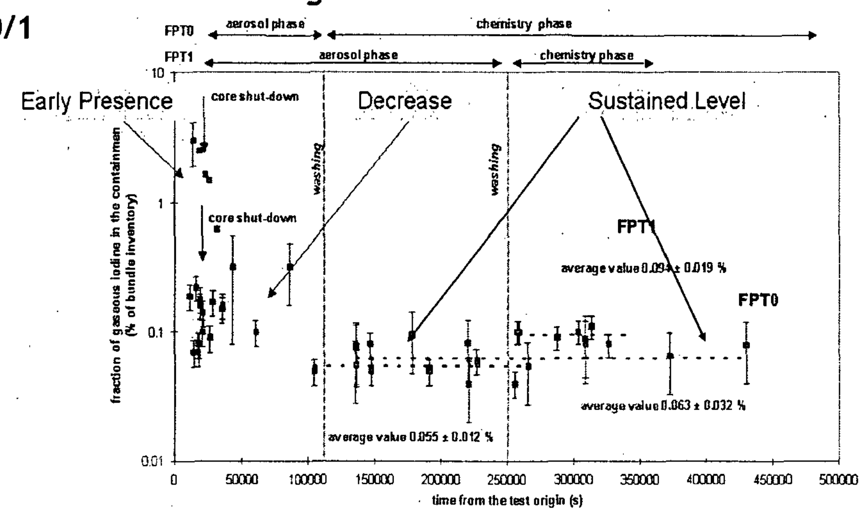
## IODINE CHEMISTRY

### General evolution of gaseous iodine in containment FPT-0/1

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# IODINE CHEMISTRY

## Early presence of gaseous iodine in containment

Likely to have been form ed in the primary circuit

Probably linked to non equilibrium chemical effects

Assumption supported by existence of sharp and large temperature gradients in the circuit especialy at the bundle exit and the steam generator inlet

Fully compatible with higher fraction in FPT-0 with lower concentrations as compared with FPT-1

Much higher fraction in FPT-3 - due to the absence of Ag-In-Cd? - due to the presence of boron carbide degradation products?

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# IODINE CHEMISTRY

## Liquid phase chemistry

Sump water does not contribute much to production of gaseous iodine in Phebus tests

- For FPT-0 and FPT-1, due to reaction with silver to form non soluble species and inhibiting gaseous iodine production by radiolysis processes despite acidic pH
- For FPT-2, due to alkaline pH
- For FPT-3, due to an excess of iodates as compared with iodides - iodates come from the radiolytic oxidation of I2 in gas phase (interpretation)

Note that efficient trapping of iodine by silver requires an excess of silver as compared with iodine (AgI decomposes under radiation)

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# IODINE CHEMISTRY

## Gas phase chemistry

- Volatile iodine concentration mostly determined by gas phase chemistry
- Importance of gaseous iodine injection from RCS
- Equilibrium between iodine formation/destruction processes and/or reversibility of iodine adsorption/desorption processes yield a steady-state concentration in the long term
- In FPT-0 and FPT-1 most of gaseous iodine organic in the long term
- Previous conclusion does not apply to FPT-2 and FPT-3
- Role of homogeneous gas phase radiolysis reactions determining speciation and evolution of iodine to be taken into account

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# STATUS OF KNOWLEDGE AND IMPLICATIONS

- A number of unexpected and/or badly quantified phenomena have been identified by previous and ongoing research programmes, especially Phebus FP, e.g.
  - Fraction of iodine entering the containment as a gas and not as an aerosol
  - Fp release and transport, cladding oxidation in air ingress conditions...
- The associated uncertainties have an impact on the results of Source Term assessment studies, e.g. for IRSN
  - S3 studies used for checking the adequacy of Emergency Planning
  - PSA level 2 studies
- The International Source Term Programme aims at reducing the uncertainties on Source Term assessment

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## GENERAL OBJECTIVES OF ISTP

Provide a set of data allowing the improvement or elaboration of models (to be) implemented in Severe Accident codes

Priorities were given in accordance with the outcomes of EURSAFE European Project as revised within the SARNET Network of Excellence (items with high safety significance end important lack of knowledge)

Set of separate-effect experiments dealing with

- Iodine behaviour in RCS and containment building
- Impact of boron carbide on the progression of a severe accident
- Air ingress situations
- Fission products release from fuel

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IODINE STUDIES

Main uncertainties for Source Term Studies

- **Iodine is risk dominant in the short term in case of a severe accident (from IRSN Source Term assessment studies)**
- **The partition of airborne iodine in the containment between aerosol particles, gaseous organic and gaseous inorganic is a key point (difference in retention by filters or other means)**
- **Part of iodine at the break in the RCS is in gaseous form (Phebus-FP results) – this fraction is badly quantified**

Fraction of iodine entering the Phébus containment in a gaseous form

| FPT-1 | FPT-2 | FPT-3 |
|---|---|---|
| 4 % | 0.6% | 85% |

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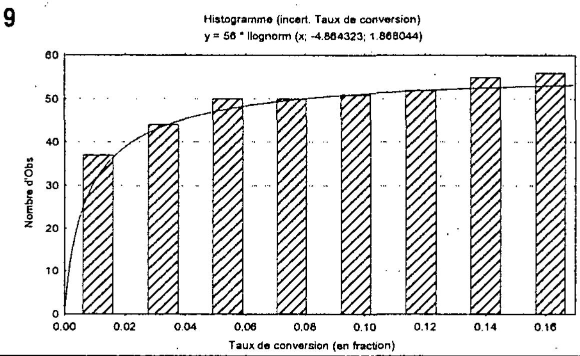
IODINE STUDIES

Main uncertainties for Source Term Studies

- Part of the gaseous inorganic iodine injected into the containment or produced by radiolytic reactions in liquid phase will be converted into gaseous organic iodine by reactions with atmospheric paints – the conversion factor is badly quantified: in IRSN PSA level 2 factor of 10 between median value and 9

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IODINE STUDIES

Main uncertainties for Source Term Studies

- Gaseous iodine reacts with air radiolysis products (ozone, nitrogen oxides) to form less volatile species - the fate of these species is badly known

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**IODINE STUDIES**

**Iodine chemistry in RCS: CHIP**

- Confirm and quantify the amount of gaseous iodine in the RCS
- Provide thermodynamic (when missing) and kinetic data for modelling

**Iodine chemistry in containment: EPICUR**

- Kinetics of formation of organic iodides through reactions with paints
- Kinetics of reactions in gas phase
- Kinetics of formation of volatile iodine in liquid phase

**Iodine chemistry in containment: PARIS**

- Interactions between iodine, surfaces and air radiolysis products
- Programme realised by Framatome-ANP under funding by IRSN – programme completed

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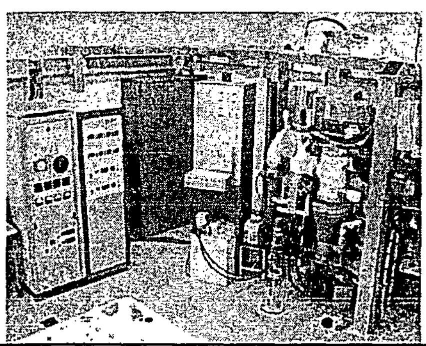
## IODINE STUDIES

### Iodine chemistry in RCS: CHIP

- Confirm and quantify the amount of gaseous iodine in RCS (phenomenological line)
- Provide thermodynamic (when missing) and kinetic data for modelling (analytical line)



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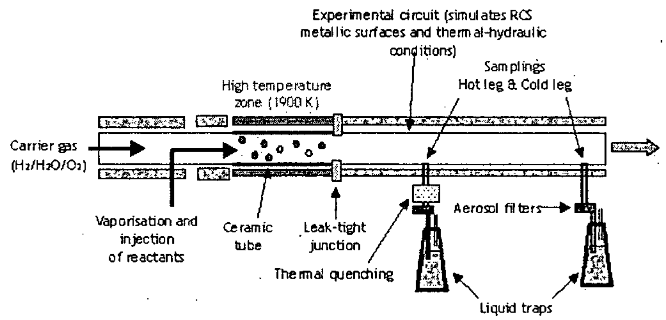
# IODINE STUDIES

## Iodine chemistry in RCS: CHIP

Sketch of the phenomenological line



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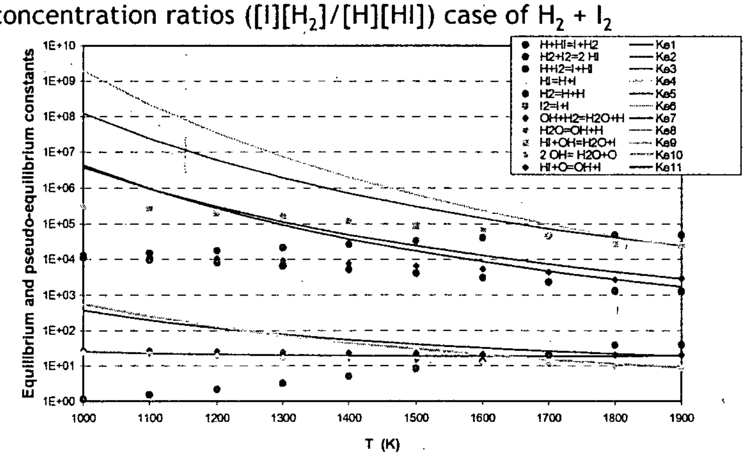
# IODINE STUDIES

## Iodine chemistry in RCS: CHIP analytical line

Kinetic pre test calcualtions of equilibrium constants and steady-state concentration ratios ([I][$H_2$]/[H][HI]) case of $H_2$ + $I_2$



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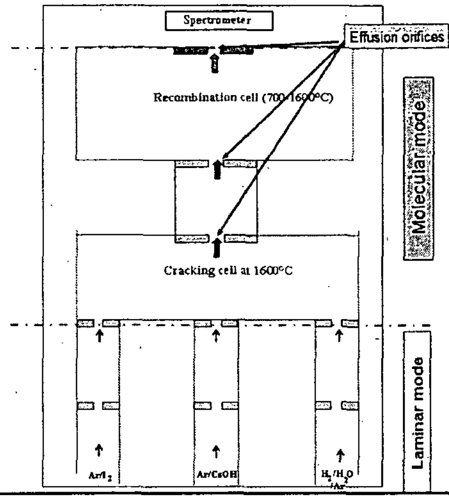
# IODINE STUDIES

## Iodine chemistry in RCS: CHIP analytical line

Scheme of experimental set-up



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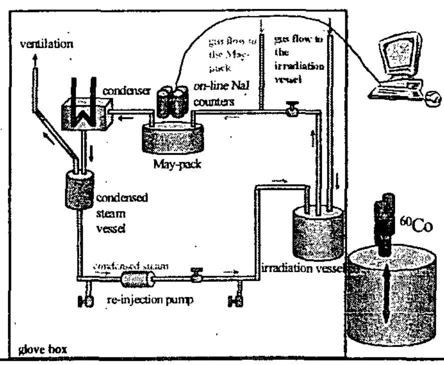
# IODINE STUDIES

## Iodine chemistry in containment: EPICUR

Kinetics of organic iodides formation through reactions with paints

Kinetics of reactions in gas phase

Kinetics of formation of volatile iodine in liquid phase



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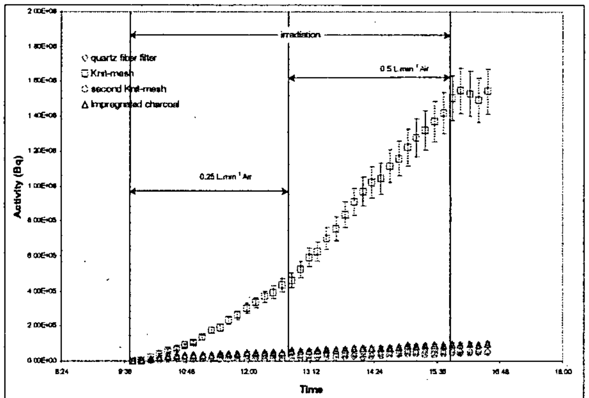
## IODINE STUDIES

### Iodine chemistry in containment: EPICUR

An example of on-line measurement for liquid phase chemistry




S1-8, 22/02/06, $[I^-] = 10^{-5}$ mol.$L^{-1}$, pH = 5, coupon peint dans la phase liquide, 120°C, 6h

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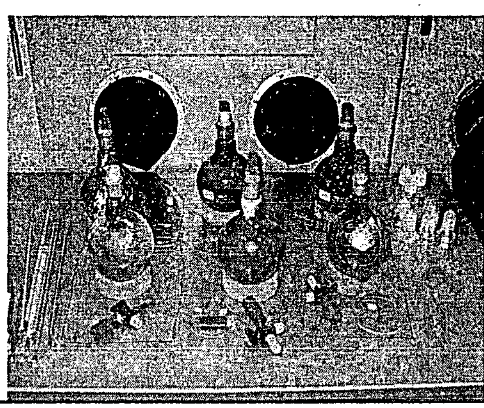
## IODINE STUDIES

### Iodine chemistry in containment: PARIS

Interactions between iodine, surfaces and air radiolysis products

Programme realised by AREVA NP under funding by IRSN



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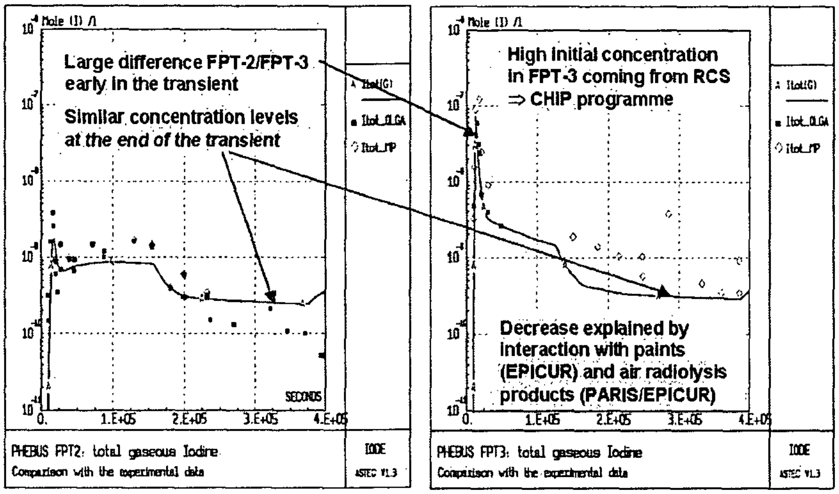
## IODINE STUDIES

### Links between ISTP and Phebus tests interpretation

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## BORON CARBIDE STUDIES

### Main Uncertainties for Source Term Studies

- Boron carbide used as neutron absorber in many reactors( BWRs, VVERs, most recent French PWRs, EPR)
- Possible impact of boron carbide degradation products (B-C-Fe-Zr) mixtures on fuel degradation
  - Might be an explanation for fuel degradation in FPT-3 - codes (ICARE/CATHARE, MELCOR, ATHLET-CD) not able to reproduce FPT-3 with present modelling
- Possible impact of boron carbide oxidation products (boric acids, CO, CO2, CH4) on fission product chemistry
  - Could this be an explanation for the 85% fraction of gaseous iodine at the break in FPT-3 - to be tested in CHIP

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## BORON CARBIDE STUDIES

### BECARRE programme

- Oxidation of pellets for determining oxidation kinetics-_ programme completed (VERDI)
- Oxidation of liquid B4C-stainless steel mixtures
- Degradation and oxidation of 30-cm long B4C rods for validation/improvement of models
  - 1 test without steam starvation
  - 1 test in same conditions with surrounding zircaloy structure to simulate claddings of neighbouring fuel rods
  - 1 test with steam starvation and a downstream "hot" and "long" circuit to favour methane production (if any)

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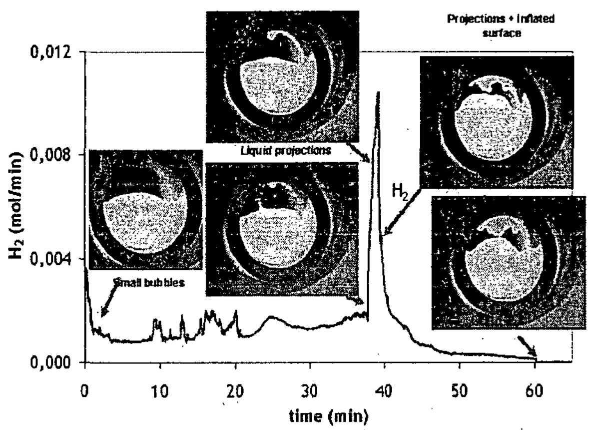
## BORON CARBIDE STUDIES

### BECARRE programme

An experiment on oxidation of liquid B4C/SS mixtures

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## AIR INGRESS STUDIES

### Main Uncertainties for Source Term Studies

- Air may be in contact with degrading fuel for several reactor accident scenarios: loss of coolant in shutdown situations, after melt-through of RPV lower head during a severe accident
- Under very oxidising conditions, ruthenium behaves as a volatile fission product and is largely released (AECL experiments)
- Ruthenium may be partly present as gaseous RuO4 in typical containment conditions (AEKI and VTT experiments)
- Radio-toxicity of ruthenium is comparable with that of iodine in short term and caesium in mid term
- Air ingress may result in fast cladding oxidation that may induce fuel degradation and FP release in spent fuel storage pool accidents

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## AIR INGRESS STUDIES

### Ruthenium behaviour in containment

- Tests on RuO4 adsorption/desorption on painted and steel surfaces
- Test on oxidation by ozone and re-volatilisation of RuO2 deposits from surfaces and from Ru trapped in liquid phase
- Same tests using EPICUR irradiator (representative air radiolysis products composition)
  - 1st series of tests show that a significant fraction of Ru remains gaseous
  - 2nd series show that Ru may be revaporised from deposits and from liquid phase (still underway)

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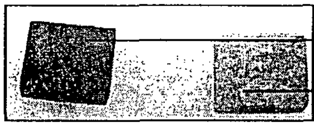
## AIR INGRESS STUDIES

### Ruthenium behaviour in containment

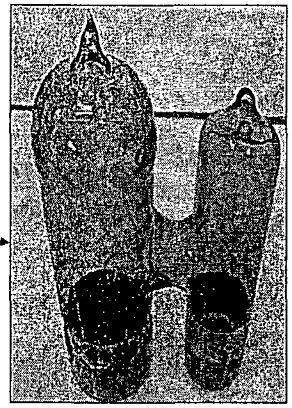
Example of revaporisation test with ozone



Initial sample with Ru deposit.

Same sample after being 24h in contact with $O_3$.

Flask used for irradiation in EPICUR



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## AIR INGRESS STUDIES

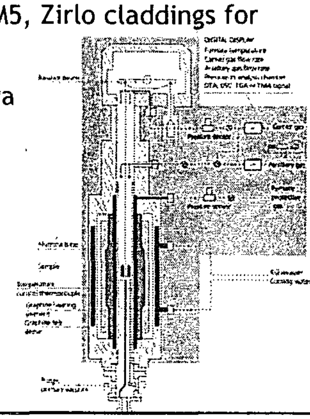
### Cladding oxidation by air: MOZART

Related to air ingress in reactor core, handling and spent fuel storage pool accidents

Determination of oxidation kinetics of Zy4, M5, Zirlo claddings for different regimes and conditions

Determination of kinetic transition (breakawa

Role of nitrogen



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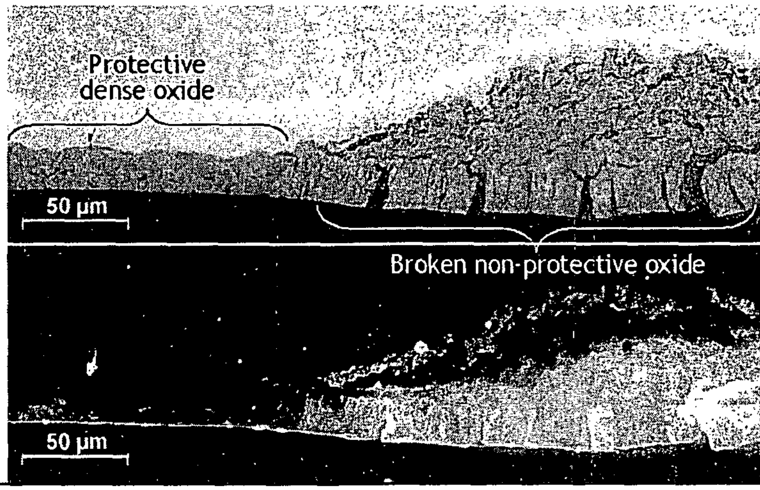
## AIR INGRESS STUDIES

### Cladding oxidation by air: MOZART

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## FISSION PRODUCTS RELEASE STUDIES

### Main Uncertainties for Source Term Studies

- Knowledge required for prediction of potential releases and residual power in corium
- Existing data from small scale and integral (Phebus FP) experiments: measured releases strongly depend on fuel burn-up and oxygen potential, not only on temperature
- Ned to extend the experimental data base to high burn-up and MOX fuels
- Need for predictive models (not only correlations)

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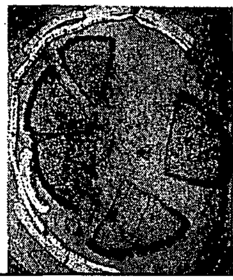
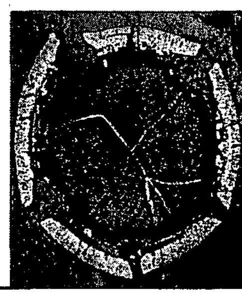
## FISSION PRODUCTS RELEASE STUDIES

### Micro analyses

Interpretation of FP release tests depends on hypotheses on formation and destruction of various compounds (molybdates, uranates... ) in the fuel during reactor operation and accidental transient

Validation of hypotheses by characterising samples of fuel annealed in FP release tests VERCORS then VERDON





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## CONCLUDING REMARKS

Among others, the Phébus FP programme provided novel information and evidenced some unexpected phenomena for severe core meltdown accidents

Some of these phenomena are still misunderstood or badly quantified and the corresponding uncertainties have a significant impact on the results of Source Term assessment studies

It is expected that the International Source Term Programme will largely contribute to the reduction of these uncertainties

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# Phébus-FP Findings on Iodine Behavior in Design Basis and Severe Accidents

Presented to the
Advisory Committee on Reactor Safeguards
May 8, 2008

R.Y. Lee, M. Salay
U.S. Nuclear Regulatory Commission
Washington, DC

# Evolving View of Radioactive Iodine

- Important fission product
  - ~750 million Curies in typical reactor core
- Nearly all reactors licensed originally to the TID-14844 Source Term
  - 100 % noble gas release
  - 50 % of iodine in gaseous form (1/2 deposits)
  - 1 % of all other radionuclides as particulate
- Following TMI a belief all iodine released to containment as particulate
  - Focus was on CsI
- Nagging evidence of some gaseous iodine

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# Alternative Source Term

- Iodine released to containment over time:
  - Gap Release (~5% of inventory)
  - In-vessel Release (25 to 35% of inventory)
- Most iodine released as unspecified particulate
  - usually interpreted as CsI
- 5% released as gaseous iodine
  - $I_2$
  - HOI
  - HI
  - $CH_3I$

3

# What Happens to Iodine in Containment ?

- Particulate Iodine
  - Agglomerates with other aerosol particulate
  - Gravitational settling
  - Diffusiophoresis to cool surfaces
  - Removal by engineered safety systems
- Gaseous Iodine
  - Voluminous research on iodine chemistry in chemically simple systems (UK, France, Canada, Poland, Germany, Switzerland)

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# Iodine Chemistry Complicated

- Most iodides very water soluble
  - Exceptions are AgI, TlI, CuI
- Aqueous iodine in water can assume any of 8 different oxidation states
  - Iodide, $I^-$, can be oxidized to molecular iodine, $I_2(aq)$, which can partition from water back into the gas phase
  - Sensitive to pH
  - Other reactions possible

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# Expectations

- Particulate and gaseous iodine released to containment would end up in sump waters
- Iodine would remain in sump if alkaline conditions maintained
  - Acidification from nitric acid formation and other radiolytic processes (cable insulation degradation, radiolysis of solvents from paint)
- If acidified, molecular iodine would partition back into the atmosphere
  - *Might form volatile organic iodides, $CH_3I$*
  - Rate of release from sump to atmosphere would increase as the sump approached boiling

6

# Phébus-FP Tests

- Opportunity to test expectations
  - Realistic configurations and chemical complexities

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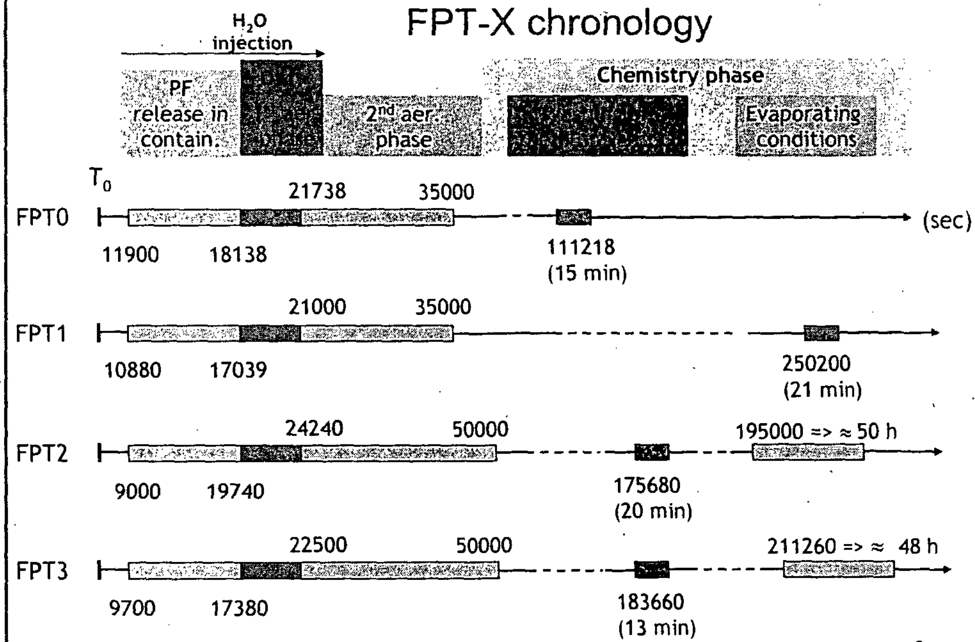
# Pertinent Tests

| | FPT-0 | FPT-1 | FPT-2 | FPT-3 |
|---|---|---|---|---|
| Fuel | Irradiated 9 days | 23 GWd/t | 32 GWd/t | 24 GWd/t |
| Flow | Steam rich | Steam poor | Steam poor + boric acid | Steam poor |
| Control Rod | Ag-In-Cd | Ag-In-Cd | Ag-In-Cd | $B_4C$ |
| Sump pH | Acid (pH = 5) | Acid (pH = 5) | Alkaline (pH = 9) | Acid (pH=5) |
| Sump | Condensing | Condensing | Condensing initially; Evaporating in Chemistry Phase | Condensing initially; Evaporating in Chemistry Phase |

## FPT-X chronology

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## Test Phases (Summary)

- Release Phase ~2.5 hours
- First aerosol phase with continued steam injection ~ 1 hour
- Second aerosol phase with no continued steam injection ~10s of hours
- Washing phase ~20 minutes
- Chemistry phase ~10s of hours

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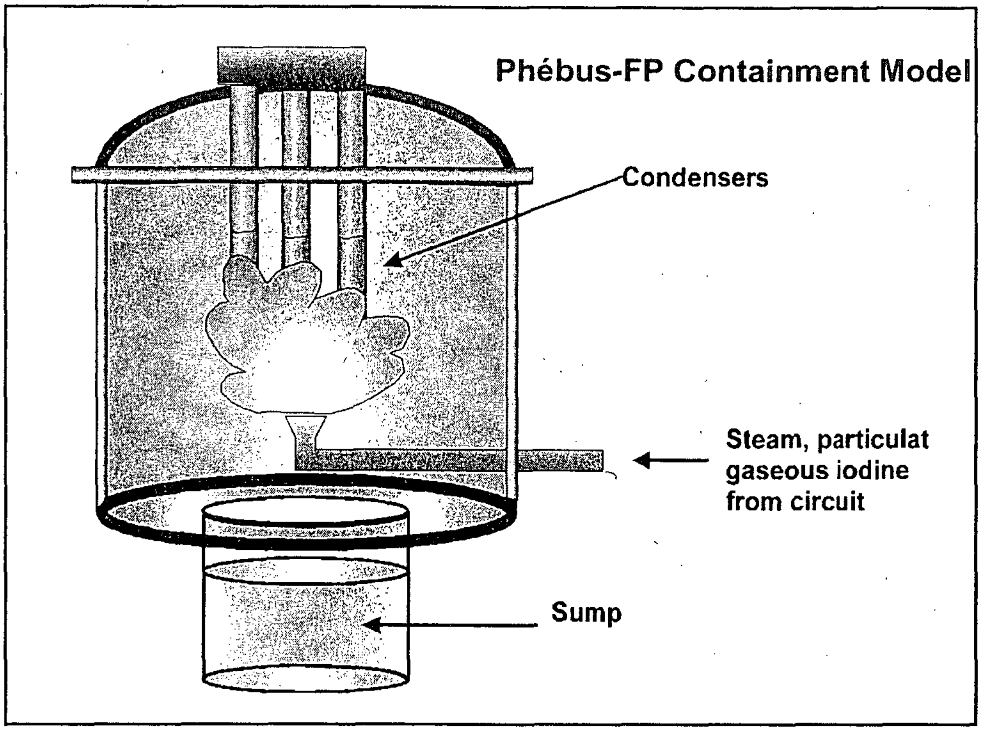
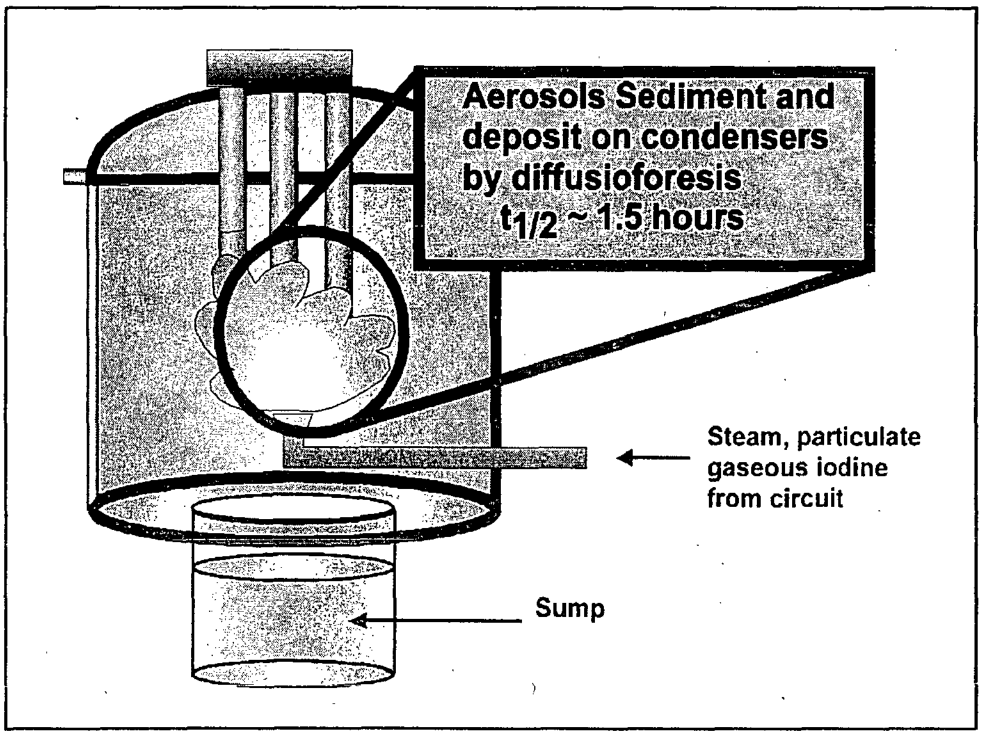
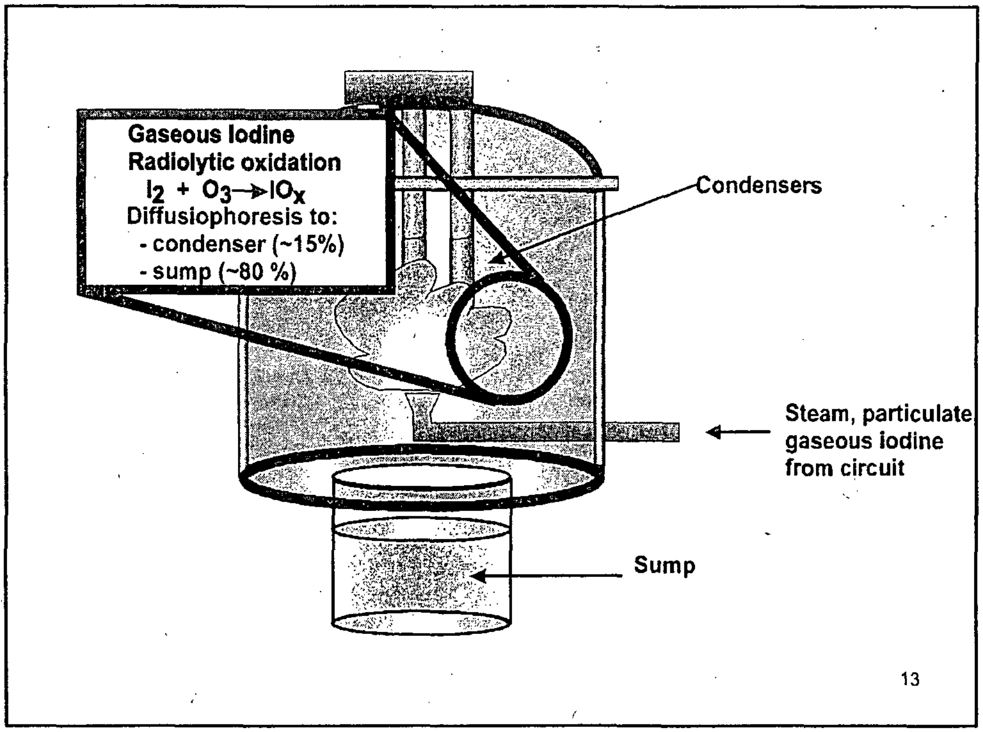
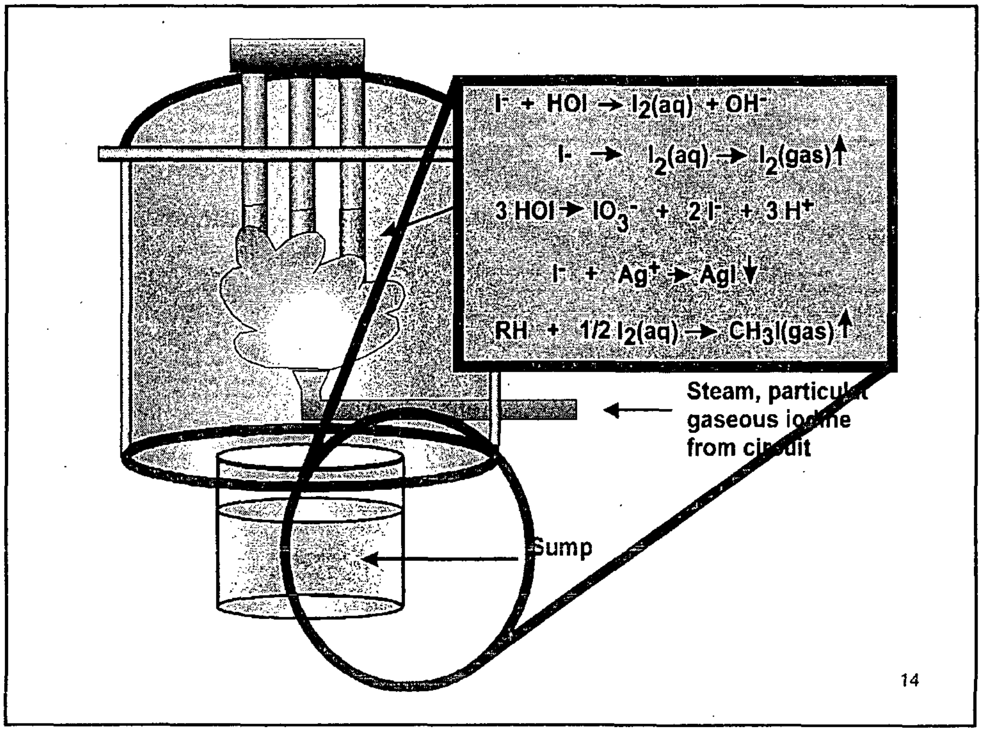
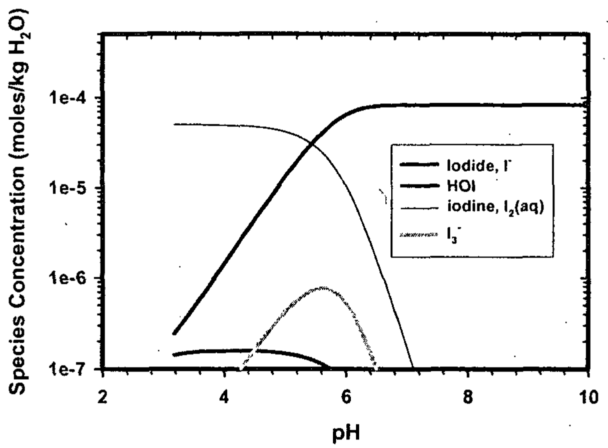

Phébus-FP Containment Model
Condensers
Steam, particulat
gaseous iodine
from circuit
Sump


Aerosols Sediment and
deposit on condensers
by diffusioforesis
$t_{1/2} \sim 1.5$ hours
Steam, particulate
gaseous iodine
from circuit
Sump


Gaseous Iodine
Radiolytic oxidation
$I_2 + O_3 \rightarrow IO_x$
Diffusiophoresis to:
- condenser (~15%)
- sump (~80 %)
Condensers
Steam, particulate
gaseous iodine
from circuit
Sump
13


$I^- + HOI \rightarrow I_2(aq) + OH^-$
$I^- \rightarrow I_2(aq) \rightarrow I_2(gas)\uparrow$
$3\,HOI \rightarrow IO_3^- + 2\,I^- + 3\,H^+$
$I^- + Ag^+ \rightarrow AgI\downarrow$
$RH + 1/2\,I_2(aq) \rightarrow CH_3I(gas)\uparrow$
Sump
14

## pH Sensitivity of Aqueous Iodine Speciation

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# Pertinent Results

| | FPT-0 | FPT-1 | FPT-2 | FPT-3 |
|---|---|---|---|---|
| Fraction of iodine released to containment as a gas | < 0.25 | ~0.02 | ~0.01 | ~0.85 |
| Aerosol Sedimentation half life<br>Phase 1 (hr)<br>Phase 2 (hr) | 1<br>1.5 | 1<br>1.5 | 1.5<br>1.8 | 4.5 |
| Sump pH | Acid | Acid | Alkaline | Acid |
| Ag : I in sump | 2000 | 45 | 10 | < 0.5 |
| Steady state iodine concentration in containment | Yes | Yes | Yes | Yes |

# Conclusions

- Indeed some fraction of iodine released to containment model as a gaseous species
  - Consistent with AST
  - CsI not the only particulate form of iodine ($CdI_2$, $NiI_2$, $InI_2$, AgI, etc.)
- Overall Iodine release rate consistent with expectations based on current severe accident modeling
- Aerosol sedimentation consistent with expectations

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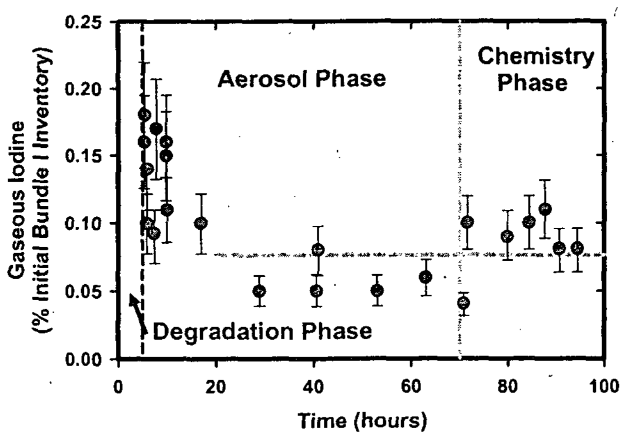
FPT-1 Gaseous Iodine in Containment

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# Observations

- A steady-state concentration of gaseous iodine developed in the containment model for all tests
  - Acid or alkaline sump did not affect qualitative observation
  - Ag precipitated iodine when abundant in sump
    (will iodine absorb on other materials in sumps following accidents?)
- Behavior contrary to expectations
  - Increases when sump condensing
  - Decreases when sump evaporating
- Complicated variations between molecular iodine ($I_2$) and volatile organic iodide ($CH_3I$)

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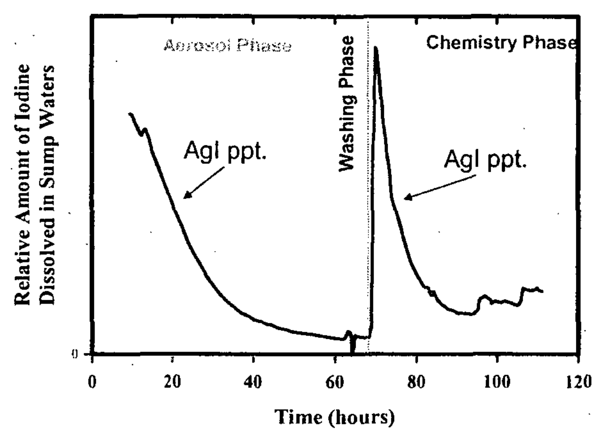
Iodine in Sump FPT-1

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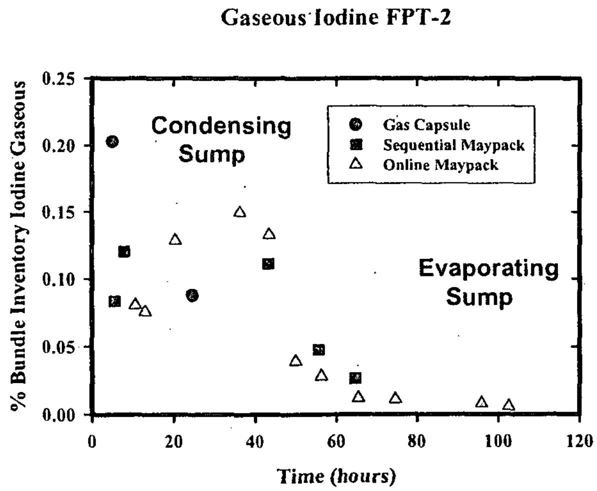


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# Conclusion (continued)

- **Observed behavior of gaseous iodine in the Phébus-FP tests is not driven by iodine partitioning from the sump!**
  - **Behavior likely involves interactions with painted condenser surfaces above the sump**
  - **The condensers simulate in some sense cooler structures with reactor containments during accidents**

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# Hypothesized Mechanism

- Gaseous iodine and particulate iodine swept to painted condensers by steam condensation
- Iodide ion (or some other soluble chemical form of iodine) rapidly absorbs from water film onto paint before water film can drain and be discharged to sump
  - Dissolves in pore water, reacts with polymer or with residual solvent
- Iodine on paint desorbs as a volatile species whether a water film is present or not
  - Irradiation releases $I_2$(gas) or $CH_3I$(gas)
- Gaseous iodine species radiolytically destroyed to form fine particulate iodine oxides or iodine nitrogen oxides
  - $IO$, $IO_2$, $IO_3$ ($IO_x$) or $INO_y$
- Iodine particulate sediments from containment atmosphere
  - $IO_x$ and $INO_y$ nucleate to form very fine particles

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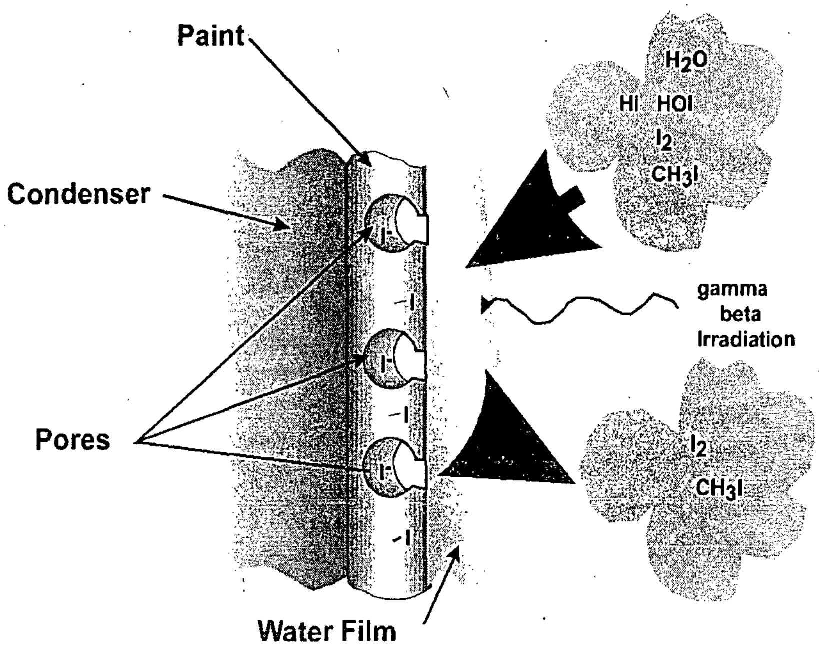


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## Scaling Issue

- How do observations of Phébus-FP tests translate to reactor accidents?
  - Original scaling of tests based on a different set of expectations
- Need a mechanistic understanding of the observed phenomenon of a steady-state concentration of gaseous iodine.

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*The Hypothesized Mechanism MUST Be tested and not just parameterized.*

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# Hypothesis Testing - 1

- Hypothesis:

  *"Interaction of iodine with paint is the source of the persistent gaseous iodine in the Phébus containment"*
- UK alternative:

  *"Low level iodine contamination of steel walls of containment is the source of the gaseous iodine"*
- Test:

  Conduct a test to show that gaseous iodine is the result of interactions with paint and not with steel.

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# Hypothesis Testing - 2

- Hypothesis:

  *"Iodine dissolved in water films on condensers will absorb onto paint fast relative to the rate water drains from the condenser"*
- Test:

  Test the rate of iodide absorption from water films onto paint – sensitivity to concentration, pH and radiation dose

  Determine where the iodine goes

  - water-filled pores in paint
  - reacts with residual paint functionalities
  - reacts with polymers

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# Hypothesis Testing -3

- Hypothesis:

  *"Iodine absorbed on paint from a water film desorbs in a radiation field as a volatile species whether a water film is present or not"*
- Test:

  Conduct desorption tests in EPICUR using coupons loaded with iodine from a water film rather than vapor from Dushman reaction.

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# Hypothesis Testing - 4

- Hypothesis:

  "Radiolytic processes destroy gas phase molecular iodine and organic iodide to form iodine oxide ($I_2O_5$) or iodine nitrogen oxide particles."
- Test:

  Conduct tests to prove the decomposition and characterize the properties of the product particles
- Alternative:

  Oceanographers have already done useful tests; find fractal particles; curious growth properties

  Saunders & Plane (2006); Hoffman *et al.* (2001)

  Jimenez *et al.* (2003); McFiggans *et al.* (2004)

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# Hypothesis Testing - 5

- Hypothesis:
  *"Iodine oxide and Iodine nitrogen oxide particles coagulate and sediment"*
- Alternative:
  *"Most particles diffuse or diffusiophoretically redeposit on wet surfaces and re-absorb on paint"*
- Test:
  Test the absorption of dissolved iodate etc. into irradiated paint

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# Hypothesis Testing - 6

- Hypothesis:
  *"Formation of gaseous organic iodides does not involve reaction with residual solvents in aged paint"*
- Alternative:
  *"Residual solvents are persistent in paint and key to the formation of volatile organic iodides from irradiated paint"*
- Test
  Conduct tests to demonstrate how organic iodide formed during the Phébus-FP tests.

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# Phébus-ST (EPICUR) and CSNI BIP Iodine Testing

- Examine the effects of prototypic sump debris on the partitioning of aqueous iodide back into the atmosphere as gaseous iodine
  - How well does debris inhibit gaseous iodine formation in a radiation field?
  - Use debris prescription from tests with paint chips and progressive development of corrosion products
- Validate the mechanism of gaseous iodine formation such that extrapolation to reactor accidents can be done confidently.
  - Formation of iodine oxide particulate
  - Nature of iodine interactions with paint
  - Characterization of reactive surface
  - Nature of radiolytic desorprtion of iodine from surfaces as a gaseous species

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# NRC Plan – Iodine Behavior

- Establish an analysis framework
  - Assemble mechanistic models to analyze iodine behavior in a containment
- Validate models
  - Obtain data to validate critical elements of models
- Understand the steady state gaseous iodine observed in the Phébus-FP tests
  - Analyze iodine behavior in Phébus-FP tests
- Scale to prototypic conditions
  - Analyze behavior to be expected in PWR containments
- Document
  - Publish
  - Peer Review
- Determine appropriate regulatory changes, if any (Regulatory Guidance, Rulemaking, etc.)

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# RESEARCH ON TRADITIONAL PROBABILISTIC RISK ASSESSMENT METHODS FOR DIGITAL SYSTEMS

Advisory Committee on Reactor Safeguards
May 08, 2008

**Alan S. Kuritzky**
Division of Risk Analysis
Office of Nuclear Regulatory Research
(301-415-6255, Alan.Kuritzky@nrc.gov)



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## Outline of Presentation

- Objective of traditional methods research
- Current status of research
- Preliminary insights from first benchmark study
- ACRS Digital I&C Subcommittee feedback
- Next steps in current project

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## Objective of Traditional Method Research

- To determine the existing capabilities and limitations of using traditional reliability modeling methods to develop and quantify digital system reliability models
  - Goal: Support the development of regulatory guidance for assessing risk evaluations involving digital systems and including digital system models into nuclear power plant probabilistic risk assessments (PRAs)



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## Status of Traditional Method Research

- NUREG/CR-6962 on initial project activities will be published soon.
  - Development of a list of desirable characteristics for reliability models of digital systems
  - Documentation of the process for using the event tree/fault tree (ET/FT) and Markov methods to develop and quantify a reliability model for a digital feedwater control system (DFWCS) – first of two benchmark studies
  - Preliminary identification of areas where limitations exist in the state-of-the-art using traditional PRA methods and where additional research and development are needed
  - No major advancements in the state-of-the-art (e.g., no detailed analysis and quantification of software reliability)
- Application of ET/FT and Markov methods to the DFWCS is almost complete.

## Development of a List of Desirable Characteristics for Reliability Models of Digital Systems

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- Characteristics were identified and grouped into nine broad categories covering the probabilistic model of a digital system and its documentation.
- The characteristics are based on knowledge and experience in PRA and analyzing digital systems, and on a literature review of digital systems.
- The characteristics were revised as the result of an external review panel meeting.
- As part of the review of the draft NUREG/CR, the revised characteristics were further reviewed by the NRC user offices, a set of external reviewers, and the public.
- The characteristics provided input to:
  - Interim staff guidance on review of digital system models in new reactor PRAs, and
  - The planning of a Nuclear Energy Agency meeting on digital system reliability to be held later this year.



## Process for Using ET/FT and Markov Methods for First Benchmark Study

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- The DFWCS was analyzed in detail, including its function, digital features, components, dependencies and interfaces.
- A failure modes and effects analysis (FMEA) was performed to determine the failure modes of the DFWCS components and the impact of each failure mode on system function.
- The relevant failure modes of the components and their impacts on the DFWCS were used in developing preliminary approaches for constructing and quantifying probabilistic models using the traditional ET/FT and Markov methods.
- Parameters needed for quantifying the probabilistic models were investigated for each digital component failure mode.
- Quantitative software reliability and human reliability analysis are beyond the current project scope.

## Capabilities of Traditional ET/FT and Markov Methods

- They are well established methods that are well understood by the reliability community.
- They are in general powerful methods that are capable of modeling many features of digital systems and capturing many important dependencies of these systems.
  - They must be supported by good engineering analyses, such as identifying failure modes and effects of digital components, and probabilistic data.
- ET/FT models can be easily integrated with an existing PRA.
- The Markov method is capable of explicitly treating some time dependencies and ordering of failures.



## Limitations of Traditional ET/FT and Markov Methods

- These methods do not explicitly account for the interactions between a plant system and the plant's physical processes (i.e., the values of the process variables), nor the timing of these interactions.
- The ET/FT method does not account for the order in which component failures occur.
- The Markov method is vulnerable to "state explosion."

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## Preliminary Areas of Additional Research Based on Current NUREG/CR

- Identifying the failure modes of the components of a digital system
- Determining the effects of a single failure mode or of combinations of failure modes on the system
- Failure parameter database
- Quantitative software reliability model
- Treatment of uncertainties
- Human reliability analysis associated with digital systems and human-system interfaces



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## Preliminary Insights from First Benchmark Study

- At the level of detail necessary to capture digital system design features that could affect system reliability, the models may be so complex that it may not be practical to use either the traditional fault tree or Markov methods to identify the component failure mode combinations that lead to system failure.
  - A simulation tool is needed to identify the system failure effects of combinations of component failure modes.
  - The output of the simulation tool is the set of the combinations of component failure modes that fail the system.
  - It was found that the order in which failures occur makes a difference.
  - The DFWCS in the benchmark study has a few hundred single failures, tens of thousands of double failures, and few million triple failures.
- The process of using the simulation tool is expected to be applicable to any complex system, though it is desirable to further simplify the process used.

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## ACRS Digital I&C Subcommittee Recommendations (Programmatic)

- The staff should explore the fundamental philosophical aspects of software failures and their use in developing a probabilistic model of a digital system.
- The staff should consider the relevant aspects of developing and evaluating a reliability model of a digital system that integrates hardware and software failures, based on the outcome of the work under item 1 above.
- Software failures can have an important contribution to the unreliability of a digital system. The work presented in the former Appendix C of NUREG/CR-6962 (now removed from this report) was a good first step in discussing the characteristics of this kind of failure, and should be taken into account in addressing items 1 and 2 above.
- The staff should explore the possibility of combining elements of the BNL work with elements of other methods, such as DFM, to better address the issues associated with developing digital system reliability models.
- BNL's task on integrating the digital system reliability models into the PRA of a nuclear power plant should be delayed until the work mentioned in items 1 and 2 above are completed.



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## Staff Response to Subcommittee Recommendations (Programmatic)

- The staff is undertaking (or will undertake) the following activities:
  - Reviewing draft former Appendix C of draft NUREG/CR-6962 and other various methods to assess software failures
  - Obtaining additional non-nuclear data sources to evaluate additional insights on software failures
  - Conduct internal discussions on the fundamental aspects of software failure modeling
  - Factor results of above efforts, and other Subcommittee programmatic recommendations, into the development of the new 5-year digital I&C research plan
  - Delay BNL's task on integrating the digital system reliability models into the PRA of a nuclear power plant

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## ACRS Digital I&C Subcommittee Recommendations (NUREG/CR-6962)

- The work on failure modes and their effects, and on developing and providing the theoretical basis for evaluating a traditional probabilistic model is valuable and should constitute the main content of the report.
- Because some of the criteria in Section 2 address issues for which current methods may not be available and others are somewhat vague, the staff should revisit these criteria.
- Due to the poor quality of the data available, it is not meaningful to quantitatively evaluate a probabilistic model. Hence, the NUREG/CR-6962 report should discuss the approaches for quantifying the model, but it should not suggest that a meaningful quantification can be carried out at this time.
- The fact that the report does not address software failures should be made very clear at the beginning of the report.



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## Staff Response to Subcommittee Recommendations (NUREG/CR-6962)

- The work on failure modes and their effects constitutes a significant portion of the report.
  - Developing and providing the theoretical basis for evaluating software failures probabilistically is out of the scope of the current project.
- The evaluation criteria in Section 2 have been revisited, and the principal change involves re-naming them as "desirable characteristics of digital system reliability models."
- The report discusses the approaches for quantifying the DFWCS model, but heavily caveats the data used, and specifies that the model is only being quantified to demonstrate the potential uses of the methods and models.
- The fact that the report does not address software failures is made more clear at the beginning of the report.

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# Next Steps in Current Project

- Complete the application of the two traditional methods to the DFWCS
  - Gain insights into reliability modeling of digital systems, and the major contributors to the failure of the system.
  - Further determine the capabilities and limitations of the methods.
  - Compare the results and insights with those from the parallel studies of the DFWCS using dynamic methods.
  - Prepare draft NUREG/CR by July 2008.
- Apply the two traditional methods to a RPS
  - The design requirements of safety-related systems are different from those of non-safety-related systems.
  - Modeling a protection system may be significantly different.