Eric T.

"Material Balance Report;" NUREG/BR–0007, "Instructions for the Preparation and Distribution of Material Status Reports;" and DOE/NRC Form 742C, "Physical Inventory Listing."

2. Current OMB approval numbers: 3150–0004 and 3150–0058.

3. How often the collection is required: DOE/NRC Forms 742 and 742C are submitted annually following a physical inventory of nuclear materials.

4. Who is required or asked to report: Persons licensed to possess specified quantities of special nuclear or source material.

5. The number of annual respondents:
DOE/NRC Form 742: 180 licensees.
DOE/NRC Form 742C: 180 licensees.

6. The number of hours needed annually to complete the requirement or request:
DOE/NRC Form 742: 900 hours.
DOE/NRC Form 742C: 1,080 hours.

7. Abstract: Each licensee authorized to possess special nuclear material totaling more than 350 grams of contained uranium-235, uranium-233, or plutonium, or any combination thereof, are required to submit DOE/NRC Forms 742 and 742C. In addition, any licensee authorized to possess 1,000 kilograms of source material is required to submit DOE/NRC Form 742. The information is used by NRC to fulfill its responsibilities as a participant in US/IAEA Safeguards Agreement and various bilateral agreements with other countries, and to satisfy its domestic safeguards responsibilities.

Submit, by June 5, 2006, comments that address the following questions:

1. Is the proposed collection of information necessary for the NRC to properly perform its functions? Does the information have practical utility?

2. Is the burden estimate accurate?

3. Is there a way to enhance the quality, utility, and clarity of the information to be collected?

4. How can the burden of the information collection be minimized, including the use of automated collection techniques or other forms of information technology?

A copy of the draft supporting statement may be viewed free of charge at the NRC Public Document Room, One White Flint North, 11555 Rockville Pike, Room O–1 F21, Rockville, MD 20852. OMB clearance requests are available at the NRC worldwide Web site: *http://www.nrc.gov/public-involve/doc-comment/omb/index.html*. The document will be available on the NRC home page site for 60 days after the signature date of this notice.

Comments and questions about the information collection requirements may be directed to the NRC Clearance Officer, Brenda Jo Shelton, U.S. Nuclear Regulatory Commission, T–5 F52, Washington, DC 20555–0001, by telephone at 301–415–7233, or by Internet electronic mail at *INFOCOLLECTS@NRC.GOV*.

Dated at Rockville, Maryland, this 28th day of March 2006.

For the Nuclear Regulatory Commission.

Brenda Jo Shelton,

NRC Clearance Officer, Office of Information Services.

[FR Doc. E6–4861 Filed 4–3–06; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

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

The ACRS Subcommittee on Reliability and Probabilistic Risk Assessment (PRA) will hold a meeting on April 20–21, 2006, Room T–2B1, 11545 Rockville Pike, Rockville, Maryland.

The entire meeting will be open to public attendance.

The agenda for the subject meeting shall be as follows:

Thursday, April 20, 2006—8:30 a.m. until the conclusion of business

Friday, April 21, 2006—8:30 a.m. until 12 Noon

The Subcommittee will review the PRA for General Electric's next generation simplified boiling water reactor, the ESBWR. The Subcommittee will hear presentations by and hold discussions with representatives of the NRC staff and industry regarding this matter. The Subcommittee will gather information, analyze relevant issues and facts, and formulate proposed positions and actions, as appropriate, for deliberation by the full Committee.

Members of the public desiring to provide oral statements and/or written comments should notify the Designated Federal Official, Mr. Eric A. Thornsbury, (Telephone: 301–415–8716) five days prior to the meeting, if possible, so that appropriate arrangements can be made. Electronic recordings will be permitted.

Further information regarding this meeting can be obtained by contacting the Designated Federal Official between 7:30 a.m. and 4:15 p.m.(ET). Persons planning to attend this meeting are urged to contact the above named individual at least two working days prior to the meeting to be advised of any potential changes to the agenda.

Dated: March 29, 2006.

Michael R. Snodderly,

Acting Branch Chief, ACRS/ACNW.

[FR Doc. E6–4860 Filed 4–3–06; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Sunshine Act Meeting

AGENCY HOLDING THE MEETINGS: Nuclear Regulatory Commission.

DATE: Weeks of April 3, 10, 17, 24, May 1, 8, 2006.

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

STATUS: Public and closed.

MATTERS TO BE CONSIDERED:

Week of April 3, 2006

Monday, April 3, 2006—

3:55 p.m. Affirmation Session (Public Meeting) (Tentative).

a. USEC, Inc. (American Centrifuge Plant); Geoffrey Sea appeal of LBP–05–28 (Tentative).

b. USEC, Inc. (American Centrifuge Plant)—Appeal of LBP–05–28 by Portsmouth/Piketon Residents for Environmental Safety and Security (PRESS) (Tentative).

c. Hydro Resources, Inc.—Petition for Review and Partial Initial Decision on Phase II Cultural Resource Challenges (Tentative).

Week of April 10, 2006—Tentative

There are no meetings scheduled for the Week of April 10, 2006.

Week of April 17, 2006—Tentative

There are no meetings scheduled for the Week of April 17, 2006.

Week of April 24, 2006—Tentative

Monday, April 24, 2006—

2 p.m. Meeting with Federal Energy Regulatory Commission (FERC), FERC Headquarters, 888 First St., NE., Washington, DC 20426, Room 2C (Public Meeting). (Contact: Mike Mayfield, (301) 415–3298).

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

Wednesday, April 26, 2006—

1 p.m. Discussion of Management Issues (Closed-Ex. 2).

Thursday, April 27, 2006—

1:30 p.m. Meeting with Department of Energy (DOE) on New Reactor Issues (Public Meeting).

Attachment 1

Advisory Committee on Reactor Safeguards
Reliability & Probabilistic Risk Assessment Subcommittee Meeting
Rockville, MD
20-21 April 2006

- Proposed Agenda -
Day 1
Cognizant Staff Engineer: Eric Thornsbury (301-415-8716, eat2@nrc.gov)

	Topic	Presenter(s)	Time	
	April 20			
	Opening Remarks and Objectives	G. Apostolakis, ACRS	8:30 - 8:35 am	
I	Introduction to Presentations	NRR/GE	8:35 - 8:45 am	
II	Overview: ESBWR Risk Management - Risk management goals - Scope of analysis - Defense in depth - PRA as a design tool	GE Wachowiak	8:45 - 9:15 am ~9:30	
III	Internal Events Risk Management & Severe Accident Prevention - Design features to prevent core damage - Analysis methodology - System success criteria - Results and conclusions	GE Wachowiak	930 ~~9:15~~ - ~~11:30 am~~ [Break 10:00-10:15] Break 10:12-10:27	12:00
	Lunch		~~11:30 am - 12:30 pm~~	12-1:08
IV	Severe Accident Mitigation - Design features to mitigate threats to containment - Analysis methodology - Assessment of containment integrity - Results and conclusions	GE (Theo fanous)	1:00 ~~12:30~~ - 4:00 ~~3:00~~ pm break 2:30-2:45	
	Break		~~3:00 - 3:15 pm~~	4:00-4:10
V	Containment Systems Performance - Design features to prevent long-term overpressurization - Analysis methodology - Long-term containment behavior - Results and conclusions	GE	~~3:15 - 4:15 pm~~ 4:10 - 445	
VI	Offsite Consequence Analysis - Radiological release assessment methodology - Results and conclusions	GE	~~4:15 - 4:45 pm~~ 445 - 5:15pm	
	Recess for the day		~~4:45 pm~~ 5:15pm	

Sud Basu - CD on presentation
~~copies of 1st presentation~~

Advisory Committee on Reactor Safeguards
Reliability & Probabilistic Risk Assessment Subcommittee Meeting
Rockville, MD
20-21 April 2006

- Proposed Agenda -
Day 2
Cognizant Staff Engineer: Eric Thornsbury (301-415-8716, eat2@nrc.gov)

	Topic	Presenter(s)	Time
	April 21		
	Reconvene		8:30 am
VII	External Events Risk Management - Fire analysis - Flood analysis - High wind analysis - Seismic analysis	GE	8:30 - ~~9:25~~ am 10:00
VIII	Shutdown Events Risk Management - Design features to prevent core damage - Analysis methodology - System success criteria - Results and conclusions	GE	10:15-10:40 ~~9:25~~ - ~~10:15~~ am
IX	ESBWR Risk Management Insights - Overall results and conclusions - Review of commitments from the meeting	GE	10:40-10:45 ~~10:15~~ - ~~10:30~~ am
	Break		~~10:30~~ 10:00 - ~~10:45~~ 10:15 am
X	Requests for Additional Information	NRR	10:45 - 11:15 am
XI	ESBWR Severe Accident Analyses	RES	11:15 am - ~~12:15~~ pm 11:30
	Adjourn		~~12:15~~ pm 1135

Notes:
- Presentation time should not exceed 50% of the total time allocated for a specific item.
- Number of copies of presentation materials to be provided to the ACRS - 35.

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

SUBCOMMITTEE MEETING ON RELIABILITY AND

PROBABILISTIC RISK ASSESSMENT

April 20, 2006
Date

NRC STAFF SIGN IN FOR ACRS MEETING

PLEASE PRINT

	NAME	NRC ORGANIZATION
1	Amy Cubbage	NRC
2	Jim Gaslevic	NRC
3	Larry Rossbach	NRC
4	Sud Basu	NRC
5	Allen Notafrancesco	NRC
6	Lynn Mrowca	NRC/NRR
7	Nick Saltos	NRC/NRR
8	BOB PALLA	NRC/NRR
9	Lauren Quinones	NRC/NRR/DNRL
10	Martha C. Barillas	NRC/NRR/DNRL
11		
12		
13		
14		
15		
16		
17		
18		
19		
20		

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

JOINT SUBCOMMITTEE MEETING ON

RELIABILITY AND PROBABILISTIC RISK ASSESSMENT

April 20, 2006
Date

PLEASE PRINT

	NAME	AFFILIATION
1	RICHARD WACHOWIAK	GE
2	JIM FULFORD	ISL
3	J. Alan Beard	GE Nuclear
4	Sid Bhatt	GE Nuclear
5	Richard Turcotte	AREVA NP, INC.
6	MIKE JONZEN	AREVA
7	Barbara Baron	Westinghouse
8		
9		
10		
11		
12		
13		
14		
15		
16		
17		
18		
19		
20		

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

SUBCOMMITTEE MEETING ON RELIABILITY AND

PROBABILISTIC RISK ASSESSMENT

April 21, 2006
Date

NRC STAFF SIGN IN FOR ACRS MEETING

PLEASE PRINT

	NAME	NRC ORGANIZATION
1	Larry Rossbach	NRR/DNRL/NESB
2	BOB PALLA	NRR/DRA
3	Nick Saltos	NRR/DRA
4	MARIE POHIDA	NRR\DRA\APOB
5	See Meng Wong	NRR/DRA/APOB
6	Jim Gaslevic	NRR/DNRL
7	Sud Basu	RES/DRASP
8	Amy Cubbage	NRR/DNRL
9		
10		
11		
12		
13		
14		
15		
16		
17		
18		
19		
20		

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

JOINT SUBCOMMITTEE MEETING ON

RELIABILITY AND PROBABILISTIC RISK ASSESSMENT

April 21, 2006
Date

PLEASE PRINT

	NAME	AFFILIATION
1	JIM FULFORD	ISL
2	Richard Turcotte	AREVA NP, INC.
3	MIKE JONZEN	AREVA
4	DAVID HINDS	GE
5	RICHARD WACHOWIAK	GE
6	J. Alan Beard	GE
7		
8		
9		
10		
11		
12		
13		
14		
15		
16		
17		
18		
19		
20		

ESBWR Design Certification

Amy Cubbage, Senior Project Manager
New Reactor Licensing Branch, NRR

ACRS Reliability and PRA Subcommittee Meeting
April 20 and 21, 2006

1

ESBWR Design Certification Status

- August 2005 - Design Certification Application submitted
- September and October 2005 – Application supplemented
- December 1, 2005 – Application docketed
- Revision 1 of the Design Control Document (DCD) submitted
  - Tier 2 Chapters 1 – 15, 18 and 18 – February 5, 2006
  - Tier 1 – March 13, 2006
  - Tier 2 Chapter 16 (Technical Specifications) – March 20, 2006

2

ESBWR Design Certification Status

- Preliminary Requests for additional information (RAIs) have been issued in the PRA and severe accident area
  - ESBWR RAI letter # 3, dated December 8, 2006
- PRA revision underway to address staff RAIs
- Revision 1 of the ESBWR PRA Report (NEDO-33201)
  - Chapter 21 – December 19, 2005
  - Chapters 2 through 6 – February 8, 2006
  - Chapter 7 – February 15, 2006
- Additional Chapters of the PRA report Revision 1 and DCD Tier 2 Chapter 19 Revision 1 to be submitted

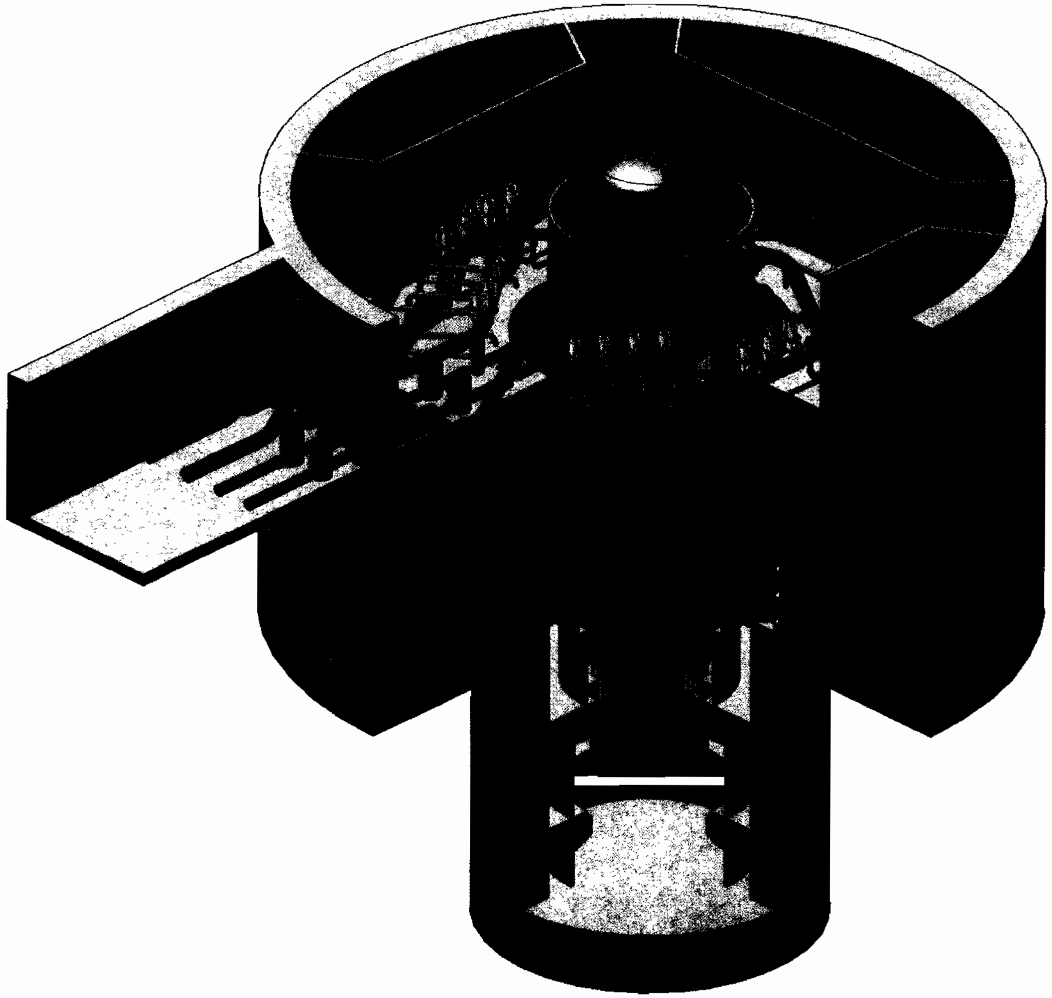
3

Design Certification Schedule

- NRC Issues RAIs through October 2006 (ongoing)
- GE responds to RAIs through November 2006 (ongoing)
- October 2007 – Planned Issuance of SER with Open Items
- Supplemental SER(s) issued – 15 months assumed
- Rulemaking - 12 months assumed

4

ESBWR Risk Management Overview



Presented By:
Rick Wachowiak
General Electric
April 20, 2006

imagination at work

GE Presentations in this Meeting

ESBWR Risk Management Overview

Severe Accident Prevention

Severe Accident Mitigation

Containment Systems Performance

Offsite Consequence Analysis

External Events Risk Management

Shutdown Events Risk Management

ESBWR Risk Management Insights

Purpose of this Meeting

Outline Strategy for Risk Management in ESBWR Design

Demonstrate the Robust Manner in which ESBWR Design Prevents and Mitigates Severe Accident Risk

Examine the Use of PRA Tools to Guide the Design and Licensing of New Nuclear Power Plants

Scope of DCD PRA

Internal Events - Full Power

> Levels 1, 2, and 3

Internal Events – Shutdown

> Level 1 and Simplified Level 2

External Events

> Internal Fire, Internal Flood, High Winds

> Seismic Margins

> Level 1

> Full Power and Shutdown

This Scope is Appropriate for ESBWR PRA Program Goals

Extended Defense – In - Depth

Classical Design / Analyses Provides DID using "Design Basis" Assumptions

ESBWR Adds Severe Accident Consideration

Main Objective is to Address Common Cause Failures

> Historically Addressed by Additional Requirements on SSCs
> ESBWR Adds Diversity to Design to Minimize Effect of CCF

Assessment of Non-Safety Equipment Performance Provided in Licensing Basis

PRA as a Design Tool

Overall Objective:
Eliminate Severe Accident Vulnerabilities

PRA Provides a Systematic Means for Finding and Eliminating These Vulnerabilities

Effectiveness May Be Limited By Information Availability Early in Design Phase

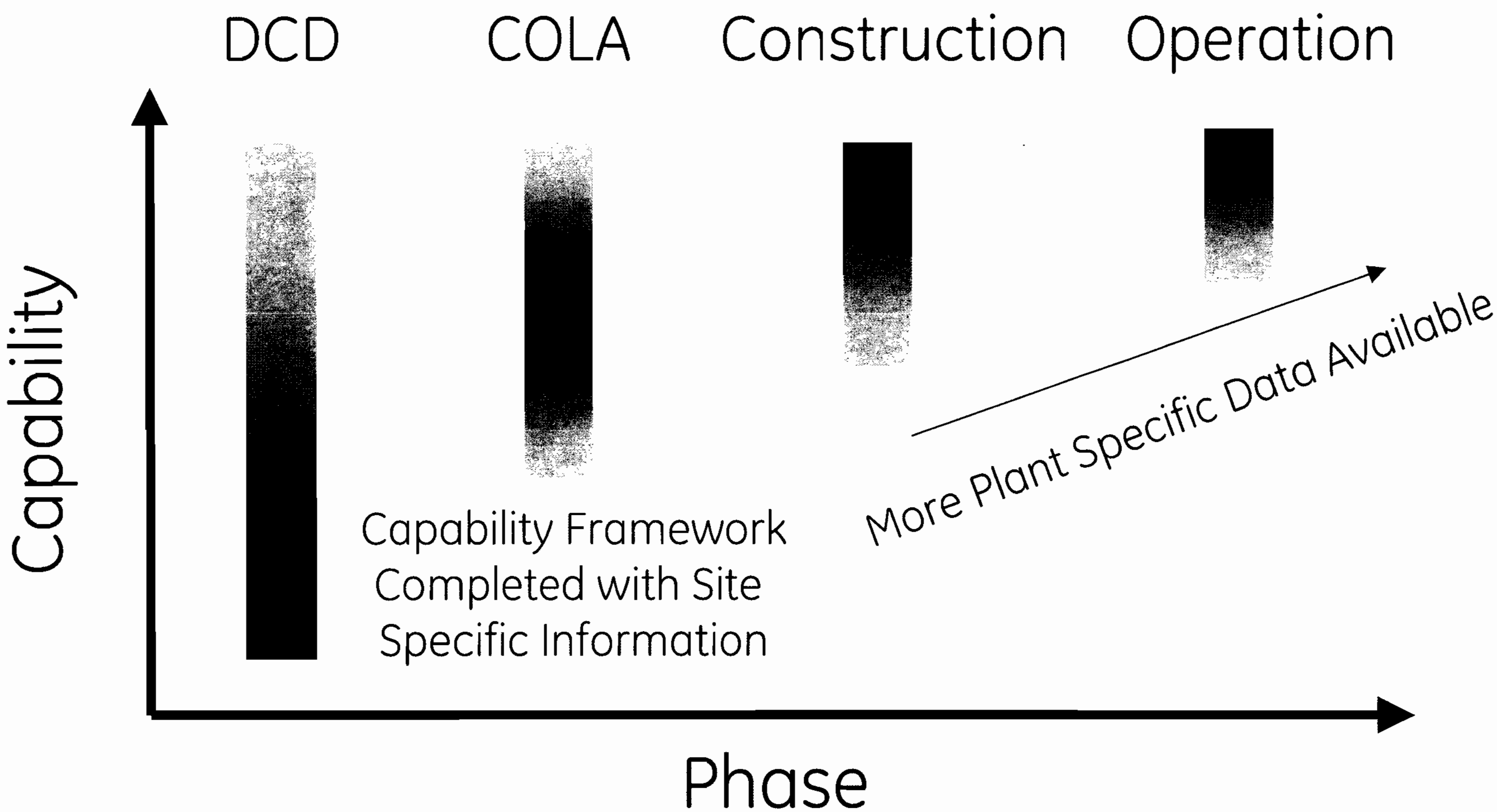
Easier to Make Corrections Earlier in Design Phase

Imperfect Tool is Better than None at All

Evolution of a Design and PRA

Conceptual Design	Design Base (DCD)	[redacted]	Construction Design	Plant in Operation
Is Design Feasible?	Can Design be Licensed?	[redacted]	Confirmation of Assumptions	Confirmation of Assumptions
Low Design Detail	Major Components Specified	[redacted]	All Components Described	All Components Described
Qualitative Risk Assessment	Qualitative & Quantitative PRA	[redacted]	Quantitative PRA with Fewer Gaps	As-Built As-Operated PRA
Defense-in-Depth Concepts	Defense-in-Depth Analyzed	[redacted]	No Defense-in-Depth Issues	No Defense-in-Depth Issues
Past Vulnerabilities Addressed	Sequence Level Vulnerabilities Eliminated	[redacted]	Component Level Vulnerabilities Eliminated	All Vulnerabilities Eliminated

Vision of ESBWR PRA



ESBWR Risk Management Program

Supports Desired Goals

Scope is Appropriate

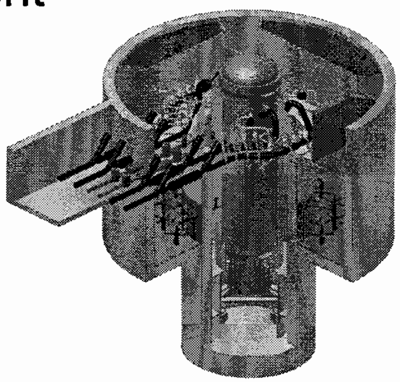
Enhanced Defense-in-Depth

PRA is a Valuable Design Tool

PRA Will Continue to Grow Through Plant Operation

ESBWR Internal Events Risk Management

Severe Accident Prevention



Presented By:
Rick Wachowiak
General Electric
April 20, 2006

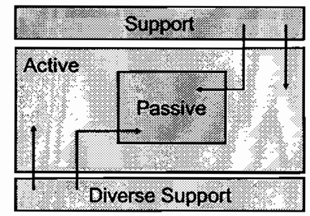
Key Features of ESBWR Risk Management

Passive Safety Systems

Active Asset Protection Systems

Support System Diversity

Target Configuration for Core Damage Prevention Functions



Functions for Core Damage Prevention

	Passive		Active
Reactivity Control	RPS SLCS		ARI FMCRD
Pressure Control	ICS SRV		Main Condenser
Inventory (High Press)	ICS		Feedwater CRD
Inventory (Low Press)		GDCS	FAPCS Fire Water Injection
Depressurization	DPV		SRV
Decay Heat Removal Condenser	ICS	PCCS	Main RWCU

Reactivity Control Function

Reactor Protection System - RPS
- > SCRAM function
- > Fail safe I&C
- > Control rod motion by stored energy

Alternate Rod Insertion - ARI
- > Provides backup to RPS I&C function

Fine Motion Control Rod Drive – FMCRD
- > Provides backup to hydraulic control rod motion

Standby Liquid Control System – SLCS
- > Negative reactivity by injection of boron solution
- > No pumps needed in this passive system

Extremely Reliable – ATWS is <1% of CDF

Pressure Control Function

Main Steam System
> Available in most transients
> Capable of handling 100% rated steam

Isolation Condenser System - ICS
> Provides decay heat removal if MSIVs close
> Prevents pressure from reaching SRV lift setpoint
> Sustains "Safe Shutdown" condition for at least 72 hours

Safety Relief Valves - SRV
> Provides backup steam relief function
> Discharges to suppression pool
> Does not lift for several minutes into a transient

Vessel Overpressure Sequences are Negligible

Inventory (High Pressure) Function

Feedwater System
> Available in most transients – requires Preferred Power
> Capable of handling any transient and small LOCA

Isolation Condenser System - ICS
> Provides closed loop cooling
> Condenses all reactor steam so additional makeup not needed
> Sustains "Safe Shutdown" condition for at least 72 hours

Control Rod Drive - CRD
> Provides backup high pressure injection function
> Power is backed by non-safety diesel generators
> Capable of handling any transient and most LOCAs

Helps Maintain ESBWR's Low CDF

Inventory (Low Pressure) Function

Gravity Driven Cooling System - GDCS
> Passive operation
> Necessary inventory is stored inside primary containment

Fuel and Auxiliary Pool Cooling System - FAPCS
> Has a LPCI mode of operation
> Transfers water from suppression pool to vessel
> Power is backed by non-safety diesel generators

Fire Water Injection
> Diverse, diesel driven fire pump can be aligned to the vessel
> Plant AC power is not required

Helps Maintain ESBWR's Low CDF

Depressurization Function

Depressurization Valves - DPV
> Passive operation
> Discharges directly to the drywell
> Provides complete depressurization for GDCS operation

Safety Relief Valves - SRV
> Active operation with manual backup
> Discharges into the suppression pool
> Depressurization sufficient for LPCI or Fire Water Injection

Very Reliable – HP Sequences <2% of CDF

Decay Heat Removal Function

Main Condenser
- > Available in most transients
- > Capable of handling 100% rated steam

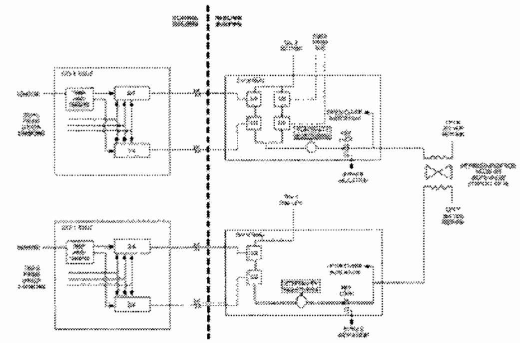
Isolation Condenser System - ICS
- > Sustains "Safe Shutdown" condition for at least 72 hours

Passive Containment Cooling System - PCCS
- > No support systems required for at least 24 hours
- > No credible failure modes

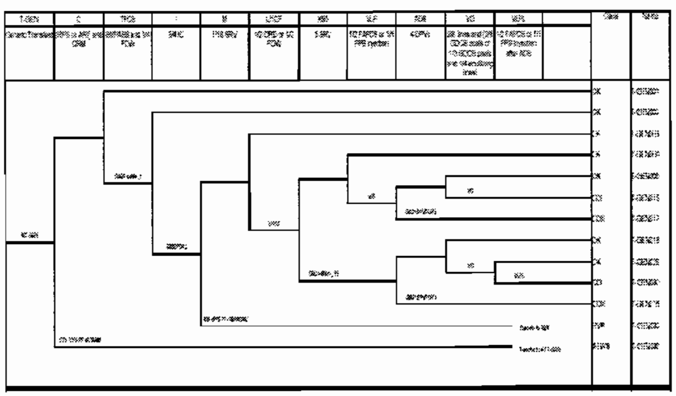
Reactor Water Cleanup – RWCU
- > Provides a shutdown cooling mode
- > Powered by non-safety diesel generators

Containment Heat Removal Not Needed For 24 Hours

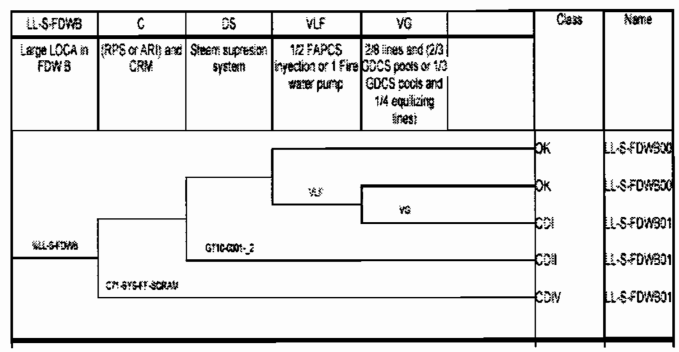
Example of Diverse Controls GDCS / DPV Simplified Logic



General Transient



Feedwater Line Break



Initiating Events for PRA

Transients	Loss of Coolant Accidents
> General	> Large Steam
> Loss of Condenser	> Medium / Small Steam
> Loss of Feedwater	> Medium Liquid
> IORV	> Small Liquid
> Loss of Offsite Power	> Break Below Core
	> Break Outside Containment

Special Initiators

Initiating Event Values

Relied on NUREG 5750

Considered Bounding Given Event Frequency Reduction Efforts for ESBWR

Only Eliminated Contributions that are N/A

PRA Demonstrates CDF is Low Due to Mitigating Capability

Event Frequency Reduction Efforts Add Margin

Basic Event Data

Generic Data Used

Generally from URD

Equipment in Harsh Environments Increased

> Example: GDCS Squib Valves

Failure Rates Increased for Components with Long Test Intervals

Low CDF Due to Design Rather than Data Valu

Human Actions

Pre-Accident

> e.g. Misposition of valves following maintenance

Post-Accident

> e.g. Backup of automatic actuation

Screening Values Used

No Repair Actions Credited

> Except for recovery of offsite power

Success Criteria

Based on One of the Following

> Hand Calculations (bounding)

> TRACG Results (design basis assumptions)

> MAAP Results

All Sequences Reviewed

The Limiting Sequences Were Used in Calculations

GE Will be Providing a Topical on This Process Later in 2006

Level 1 Internal Events Results

CDF = 3×10^{-8} per year

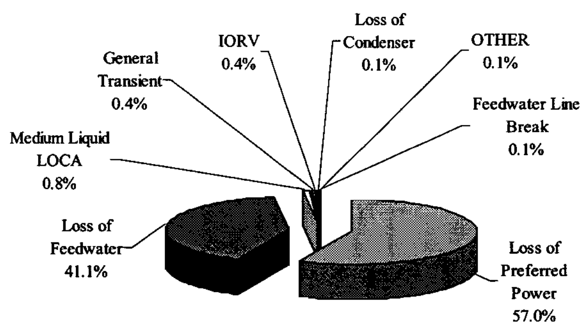
Highest Sequence = 1.6×10^{-8} per year

Highest Cutset = 5×10^{-10} per year

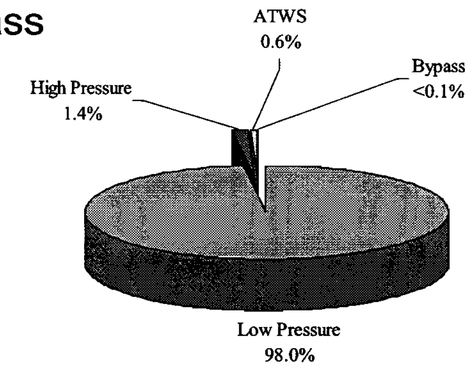
Combination of Active and Passive Failures in Top Cutsets

Passive Systems Fail by Common Cause in Top Cutsets

Breakdown of CDF by Initiating Event



Breakdown of CDF by Accident Class



Conclusions

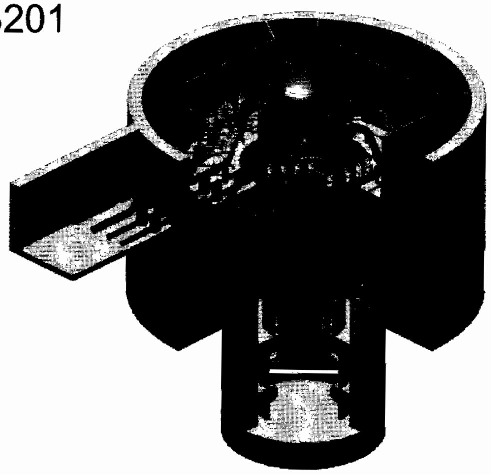
ESBWR Design is Robust

Probability of Severe Accident is Remote

Use of PRA as a Design Tool Ensured this Result

Combination of Passive Safety, Active Non-Safety Systems, and Diversity Leads to these Results

ESBWR
Severe Accident Treatment:
Ch. 21 of NEDO-33201
by Theofanous and Dinh



April 20, 2006
Theo Theofanous

Severe Accident Treatment

Included: Containment integrity threats due to severe accident phenomena
Not-Included: Containment decay heat removal system failures in the long term

Our Approach

Assessment ⟺ Management

We placed great emphasis on bounding, high-confidence evaluations

We employed new procedures and hardware to eliminate scenarios of concern

Conclusion:
Containment failure is physically unreasonable for all severe accident scenarios except postulated large Steam Explosions in very deeply-flooded LDW representing < 1% of the CDF

SA Threats and Failure Modes

- Direct Containment Heating (DCH)

  Energetic Failure of UDW, Liner (thermal) Failure

- Ex-Vessel Explosions (EVE)

  Pedestal/Liner Failure, BiMAC-Pipes Crushing

- Basemat Melt Penetration (BMP)

  BiMAC Thermal Failure (Burnout, Dryout, Melt Impingement)

Pivotal Issues and their Resolution

In-Vessel retention feasible but external supports for penetrations not agreeable to the designers

"Natural" ex-vessel (core on a flooded floor) coolability cannot be assured

Boundary-Internal Melt Arrest and Coolability (BiMAC) device

The pedestal cannot be shown to withstand arbitrarily large SE's

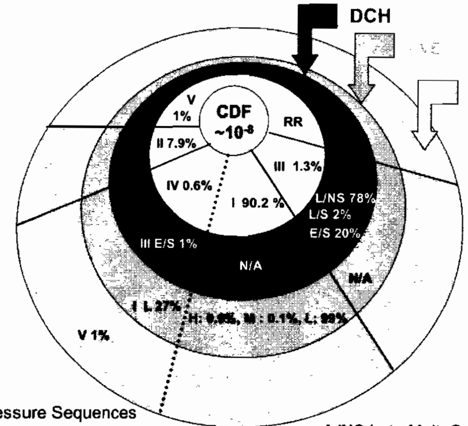
Deluge the LDW after lower head failure, eliminate pathways to LDW

Direct containment heating energetic containment loading

Bounding analysis-based resolution

I will follow the reverse order in this presentation
Same as in the report (Ch 21).

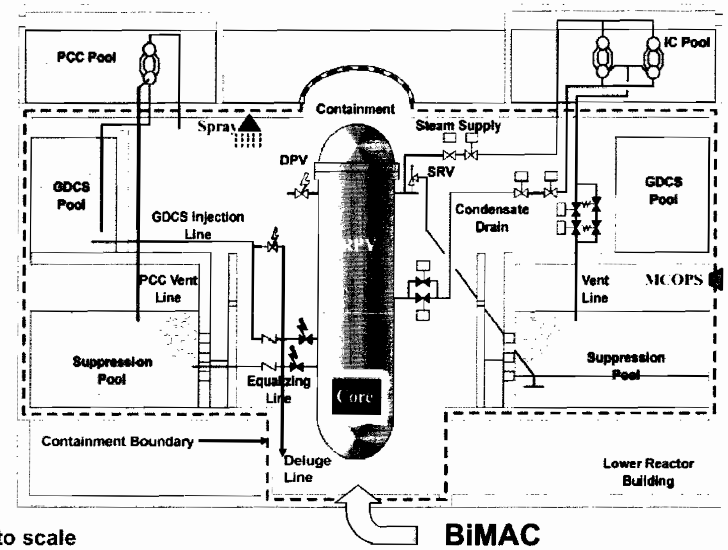
ESBWR SA Complexion



I Low Pressure Sequences
II Very Late Core Damage
III High Pressure Sequences
IV ATWS; 71% No RPV Failure
V Containment Bypass

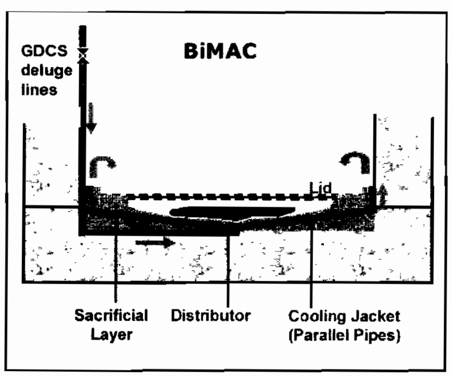
L/NS Late Melt, Sprays Fail
L/S Late Melt, Sprays Available
E/S Early Melt, Sprays Available

ESBWR SA Containment Highlights



Not to scale

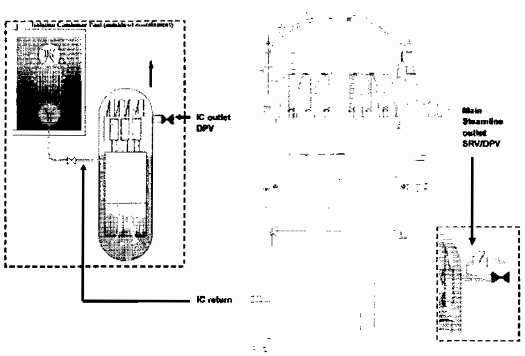
The Basemat-internal Melt Arrest and Coolability (BiMAC) device



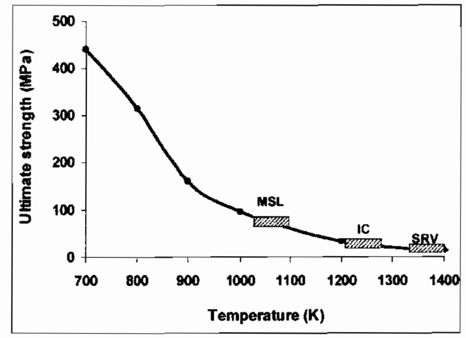
Direct Containment Heating (DCH)

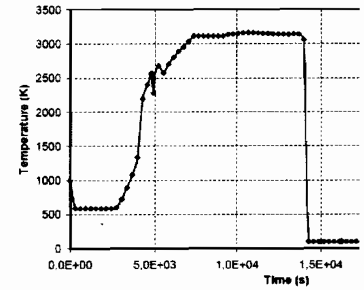
- Spontaneous depressurization
- Tools for DCH loads and verification approach
- Parameter range covered and results
- Thermal loads to liner
- Comparison to fragility (taken from NEDO-33201)
- Summary of bounding approach and conclusion

Preamble to DCH: there is a potential pressure relief path for spontaneous depressurization



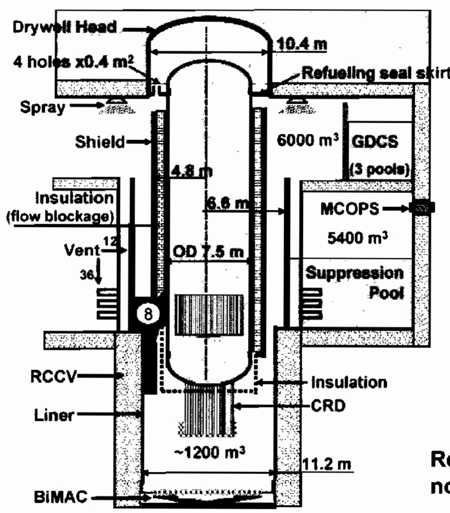
Potential for creep rupture





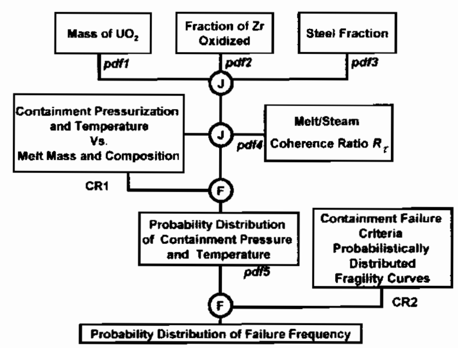
Core T transient during heat-up

DCH: Key features of the geometry



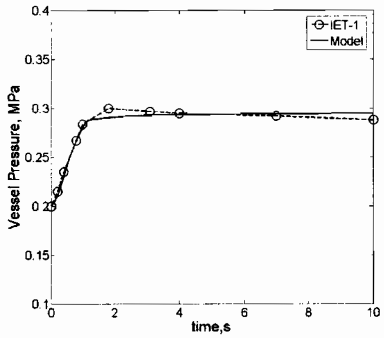
Representative but not to scale

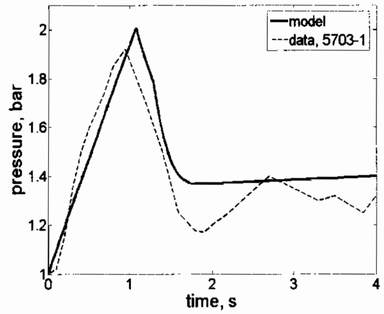
Framework for DCH issue resolution in Large Dry Containments. Such detail not necessary here



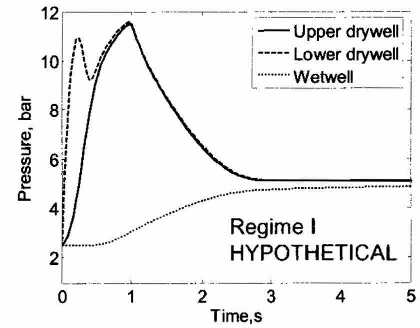
New: an extended CLCH model that couples to a vent clearing model

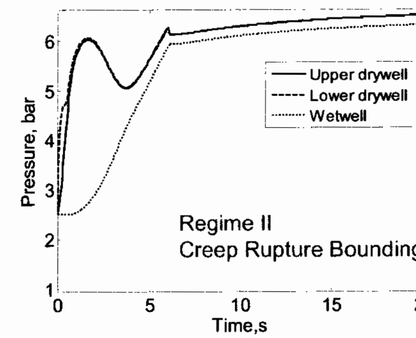
Validation Basis:
IET DCH Tests… GE PSTF Vent Clearing

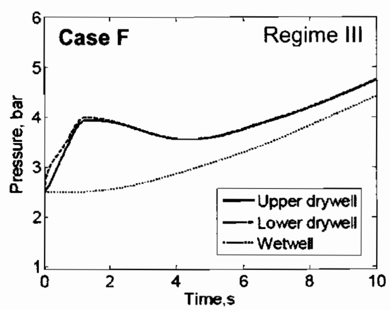




Quantification of Loads







More Dynamics

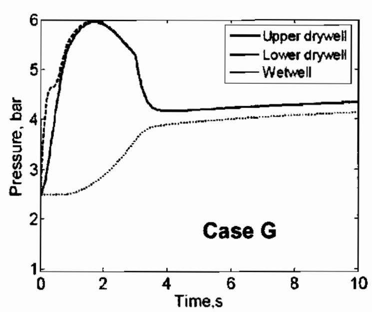


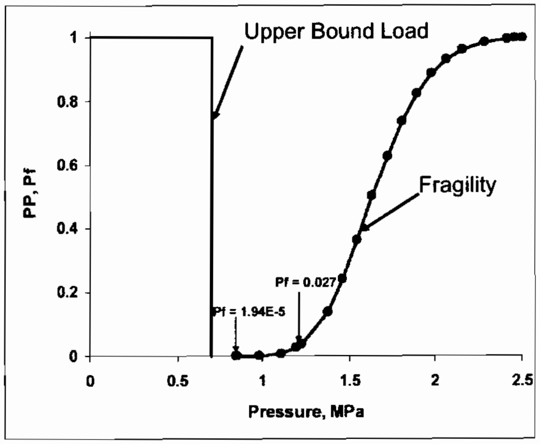
Table 1.4.3.5. Summary of Parameters and Variables used in Reactor Calculations.

Parameter	Parameter Definition	Reactor Case							
		A	B	C	D	E	F	G	H
m_m^0 (tons)	Initial mass of corium in the lower drywell	50	50	100	100	300	300	300	300
D_S, m	RPV hole size for steam blowdown	0.2	0.2	0.2	0.2	0.3	0.3	0.5	0.5
T_{RCS}^0 (K)	Initial temperature in the primary system	800	800	800	1500	800	800	800	800
τ_m (s)	Mixing time between melt and blowdown steam	7.8	3.6	10	10	7.8	10	3	6

Table 1.4.3.6. Summary of Results of Reactor Calculations.

Parameter	Parameter Definition	Reactor Case							
		A	B	C	D	E	F	G	H
τ_s (s)	Blowdown time scale	28.7	28.7	28.7	28.7	12.8	12.8	4.6	4.6
$R = \tau_m / \tau_s$	DCH scale	0.27	0.14	0.35	0.35	0.61	0.78	0.65	1.3
P_1 (bar)	First (before vent clearing) pressure peak	3.35	3.3	3.3	3.1	4.0	4.0	4.7	4.7
P_2 (bar)	Second pressure peak	3.2	3.1	3.5	3.0	4.2	4.8	6.0	6.0
P_∞ (bar)	Long-term pressure	3.3	2.8	3.5	3.2	4.5	5.1	4.3	6.5
T_{STAB} (K)	Stabilized temperature	600	500	750	800	900	1000	1000	1200

Minimum (bounding) Margins to Energetic DCH Failure

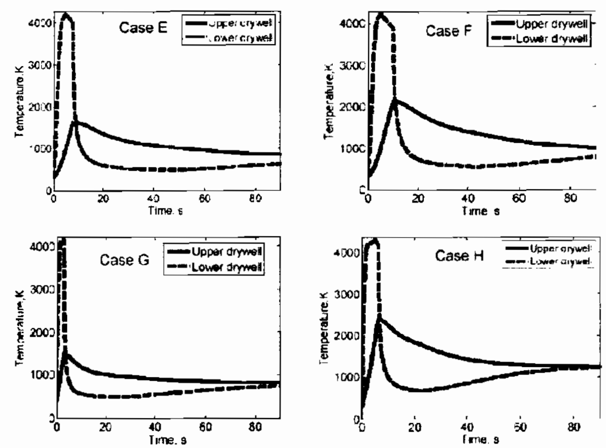


The key bounding Ingredients are:

- A conservative energy-release and transport model (CLCH) as used for PWR DCH-issue resolution,
- A creep-rupture RPV breach area that is at the upper end of the uncertainty range used for the most severe of the 4 scenarios considered for PWRs,
- Upper bound of available core materials participating in the ejection and dispersal process,
- No intersection to the lower bound of the DW fragility.

Conclusion: Failure is Physically Unreasonable

Thermal effects on liner were also considered



LDW liner would likely melt through. Liner isolation "lips" into the concrete and the basically impermeable pedestal wall should provide isolation to the outside. The liner of the UDW was shown to survive creep failure.

Ex-Vessel Explosions (EVE)

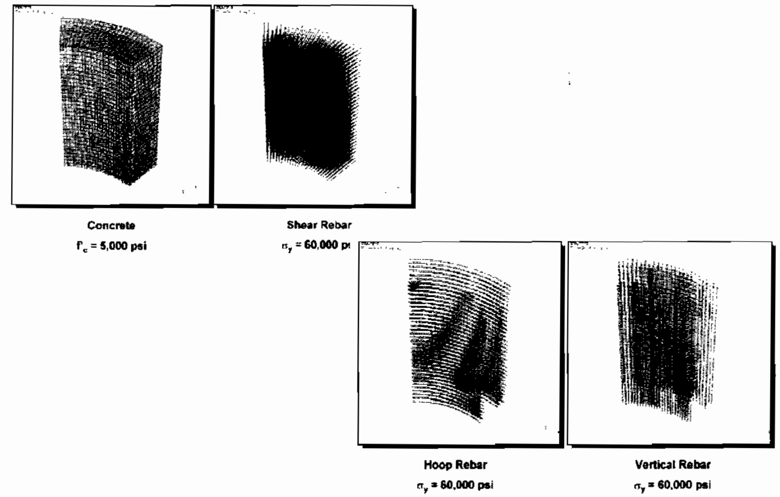
Pedestal/Liner Failure, BiMAC-Pipes Crushing

Energetic impulses that could potentially damage the reactor pedestal and BiMAC pipes cannot be conservatively excluded if there are deep, subcooled water pools on the LDW at the time of vessel breach.

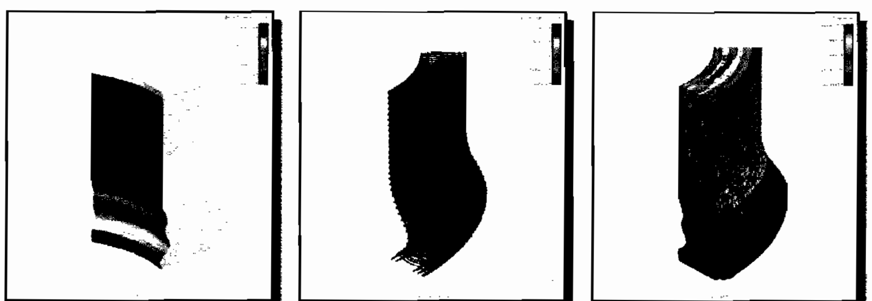
Our approach relies in prohibiting the formation of such pools by design changes in containment layout/systems, and placing a high reliability requirement on the operation of the LDW deluge system

According to bounding estimates of impulses and fragilities (both pedestal and BiMAC) there are additional margins even for subcooled 1 to 2 meter pools.

Pedestal model in DYNA3D



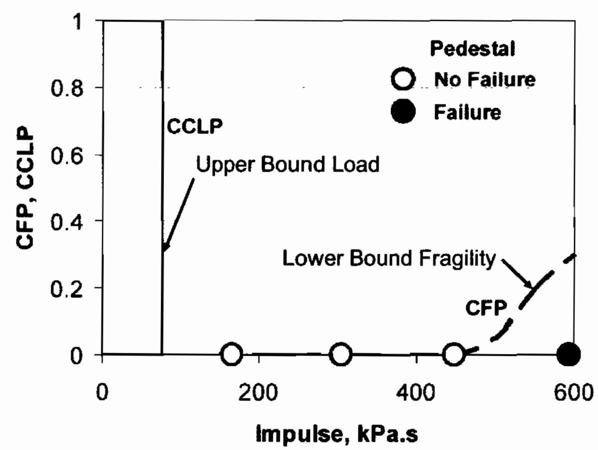
Pedestal damage in DYNA 3D



■ eps > 30% ■ eps > 20% ■ Concrete damage

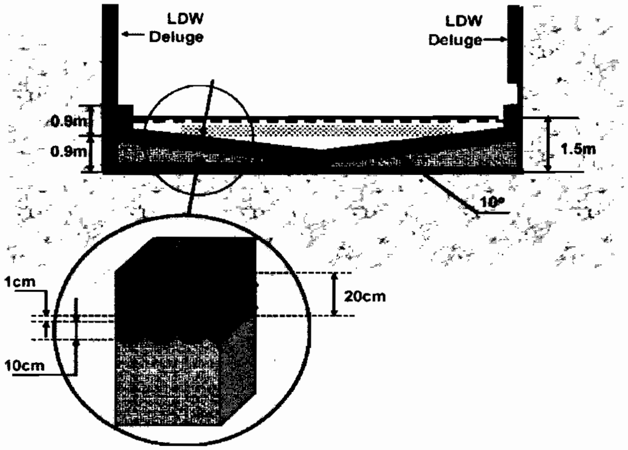
Pedestal Failure Margins to EVE

1 to 2 m Subcooled Pools

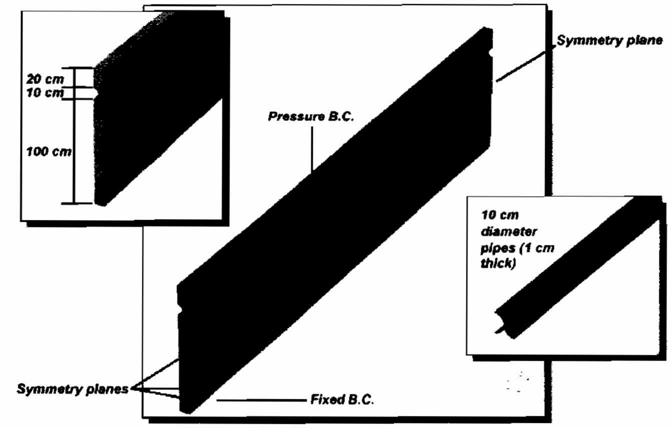


Significant upwards revision of previously used failure criteria

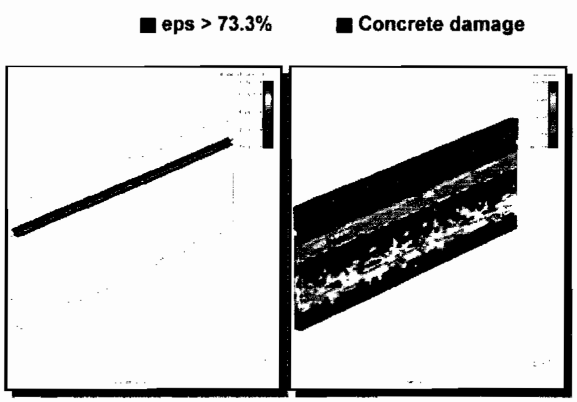
BiMAC Structural Configuration



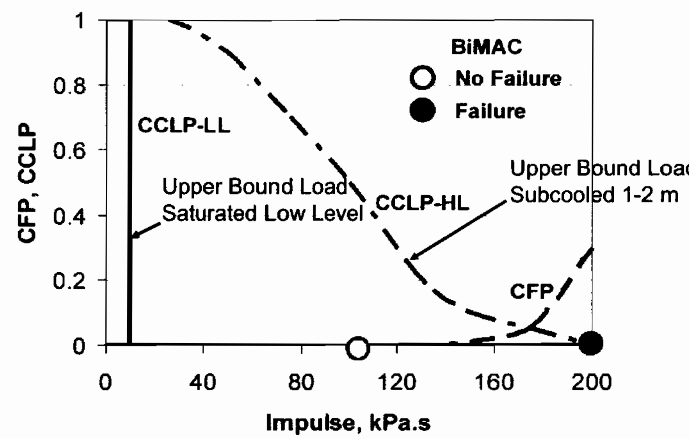
DYNA3D model of BiMAC



BiMAC damage in DYNA3D



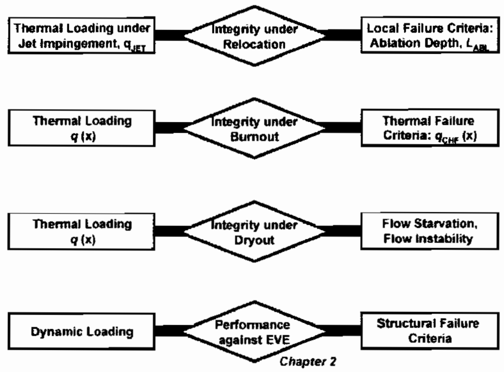
BiMAC Failure Margins Due to EVE
1-2 m subcooled pools



Basemat Melt Penetration (BMP)

BiMAC Thermal Failure (Burnout, Dryout)

The scope of work



Basemat Melt Penetration (BMP)

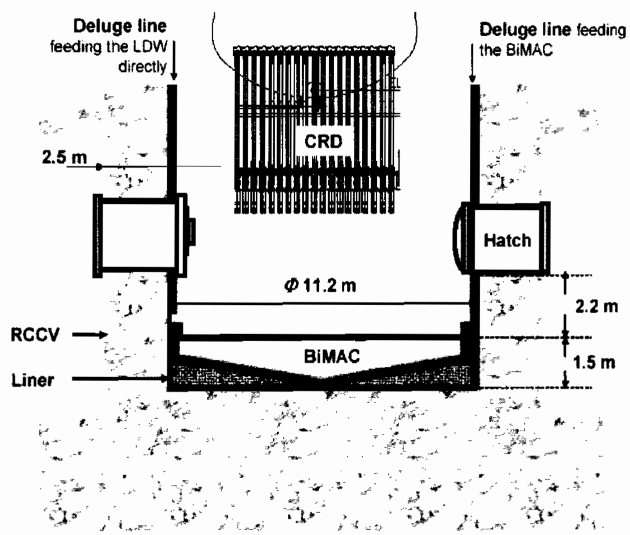
BiMAC Thermal Failure (Burnout, Dryout)

The key bounding Ingredients are:

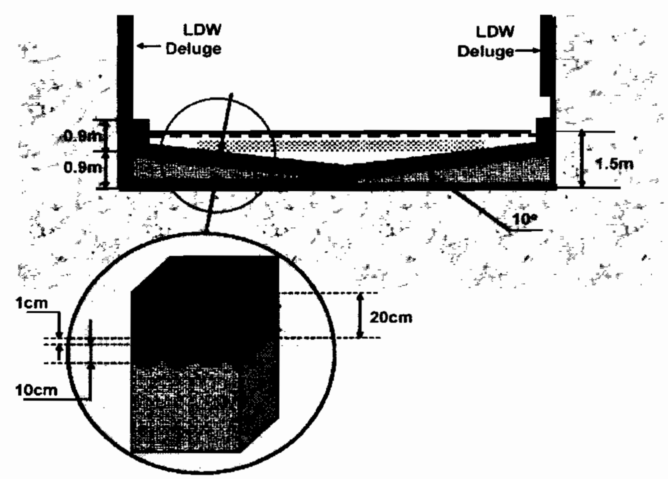
1. Average thermal loads from full-core pools at bounding decay power levels,
2. Bounding local peaking of loads from verified CFD calculations,
3. Lower bounds of CHF from ULPU in pool boiling (to be verified by full-scale experiments at the COL stage)
4. No flow-stability, or boil-off issues, found using a two-phase flow model verified with inclined-channel data from the SULTAN experiments
5. Full floor area coverage—the melt has no other place to go but inside the BiMAC.

Failure is Physically Unreasonable

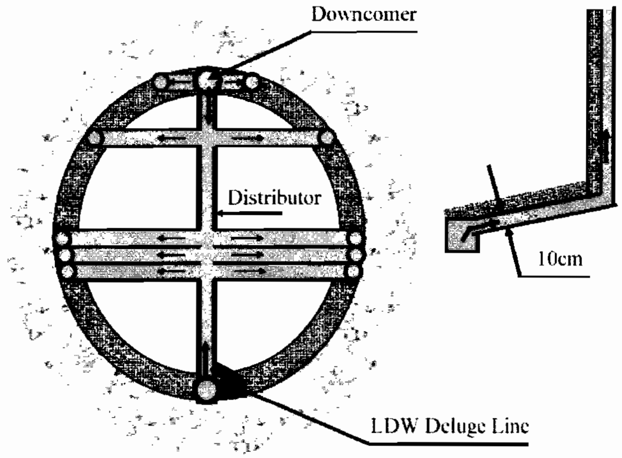
Lower Drywell



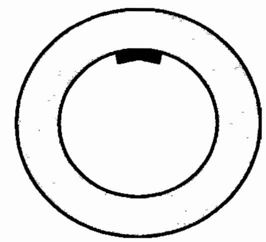
BiMAC Detail

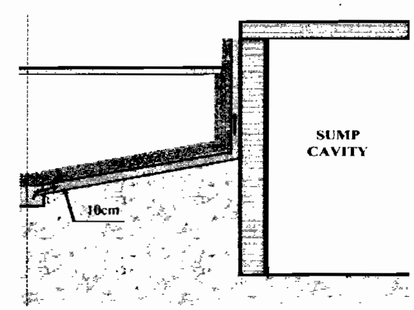


BiMAC Flow Path



Sump Protection too

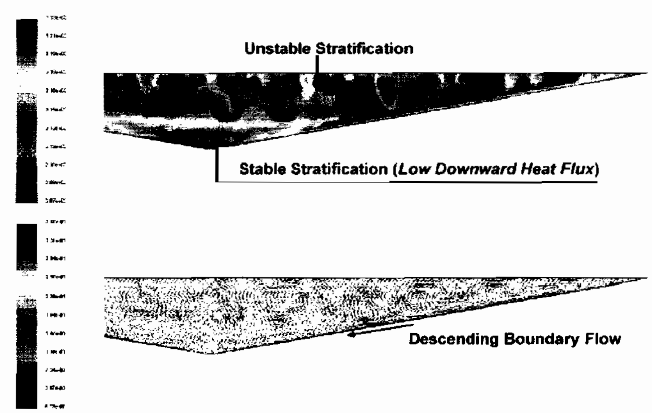




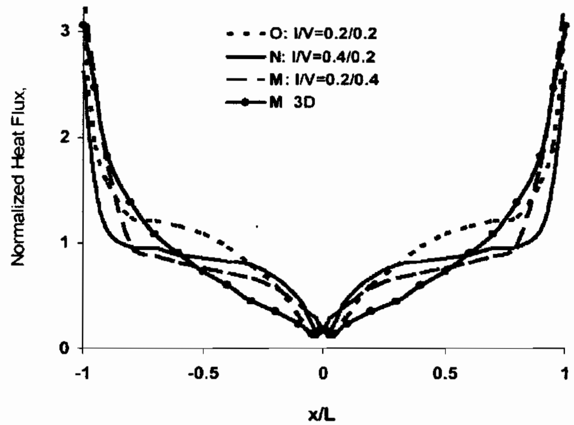
BiMAC capacity as a function of melt pool height, and resulting average heat fluxes.

H_melt, m	0.2	0.4	0.6	0.8	1.0
V_melt[a], m^3	2.2	9.	20.5	35.8	53.8
Mass, tons	18	72.5	164	287	431
i_vertical[b]	51	47	41	29	1
V_sump, m^3	0.3	0.85	1.4	2	2.6
M_sacrificial layer, tons	7.6	15	21.7	27.3	30.7
Top Boundary, m^2	25	49	70.5	87.7	95.8
Bottom Boundary, m^2	25.4	49.7	71.5	88	97.3
Side Boundary, m^2	0	~0	0.8	2.1	5.1
	All melt assumed to be Fuel			All oxides + 20 tons of metal	All oxides + 160 tons of metal
Decay power, MW	1.5	8.6	21.5	36.4	36.4
Upward heat flux, kW/m^2	45	132	226	305	271
Downward heat flux, kW/m^2	15	43	74	100	89
Sideward heat flux, kW/m^2	-	-	300	320	350

Natural convection patterns



The Peaking at the Edge of Near-Edge Channels is the most Limiting



Summary of Power Split and Peaking Factor Results from the Direct Numerical Simulations (all fluxes in kW/m2)

Case No.	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
A	63	30	N/A	2.1	1.25
B	120	54	N/A	2.2	1.25
C	178	80	N/A	2.2	1.25
C-3D	238	68	N/A	3.5	1.2
M-3D	286	85	280	3.4	3.0 / 1.4
M	255	125	330	2.0	3.0 / 1.4
N	238	126	340	1.9	3.0 / 1.2
O	168	83	245	2.0	3.0 / 1.2

The 3D results were confirmed with further calculations that included refined meshes, and a 10-fold increase in viscosity due to addition of the sacrificial concrete.

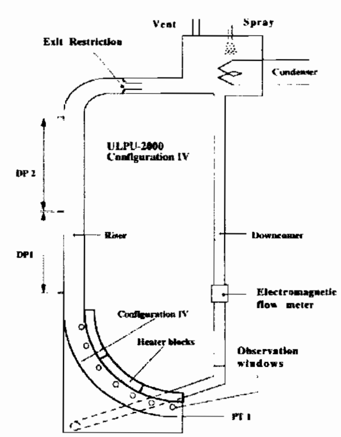
Bounding estimates of thermal loads

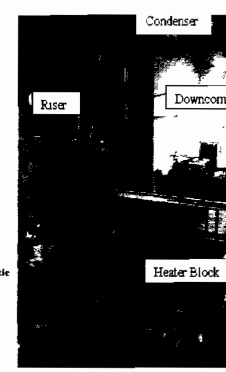
Central Channels: $q_{dn} = 100\ kw/m^2 \quad q_{\max,dn} = 125\ kw/m^2$

$$q_{dn} = 100\ kw/m^2 \quad q_{\max,dn} = 300\ kw/m^2$$

Near-Edge Channels: $q_v = 320\ kw/m^2 \quad q_{\max,v} = 450\ kw/m^2$

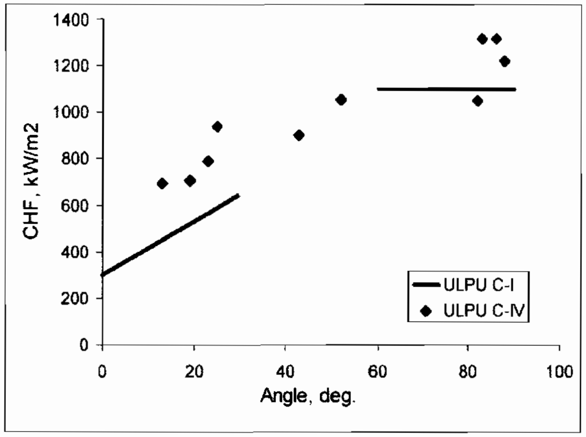
The ULPU facility



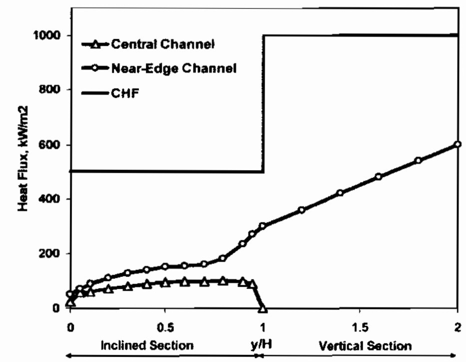


Coolability Limits for BiMAC

Applicability based on similarity of geometries and flow/heating regimes

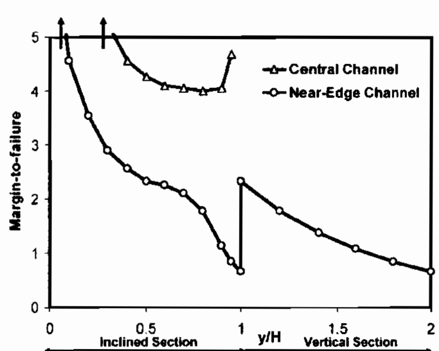


Thermal Loads against Coolability Limits in BiMAC Channels

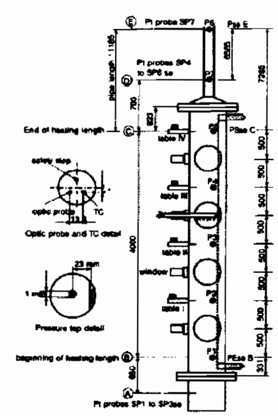


Thermal Margins for BiMAC

Local Burnout

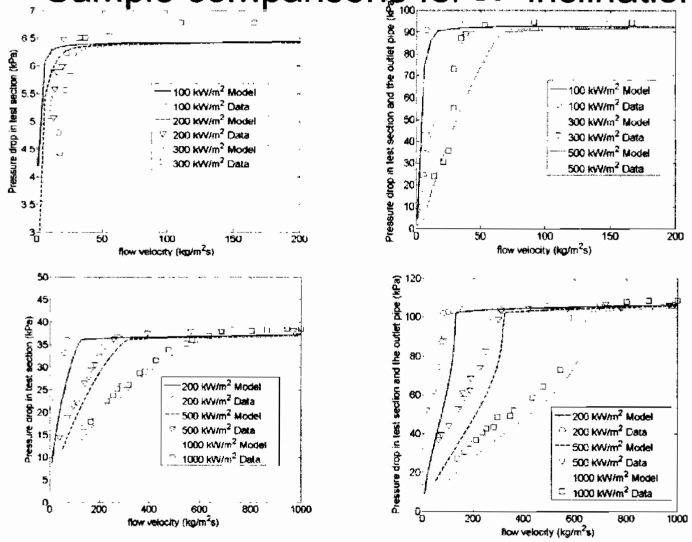


Natural convection boiling in inclined channels: the SULTAN facility

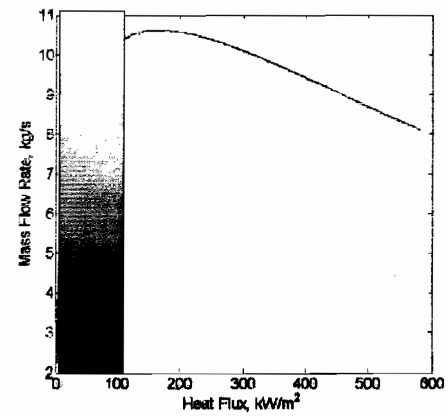


- Vertical and 10 degrees inclination
- Characteristic length: 3 and 15 cm
- Channel length: 4 m
- Pressure: 0.5 MPa
- Power levels 100 to 500 kw/m2
- Detailed pressure drop data

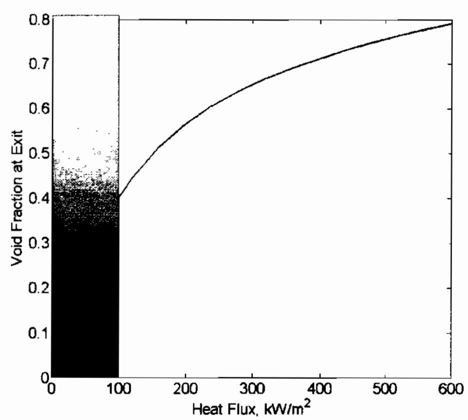
Boiling in inclined channels:
Sample comparisons for 10^o inclination



Natural convection in BiMAC: stable, self-adjusting flow



Thermal Margins for BiMAC
no-Dryout due to water depletion or flow starvation



BiMAC needs to be at least RTNSS

- Qualification of function in the as-designed state

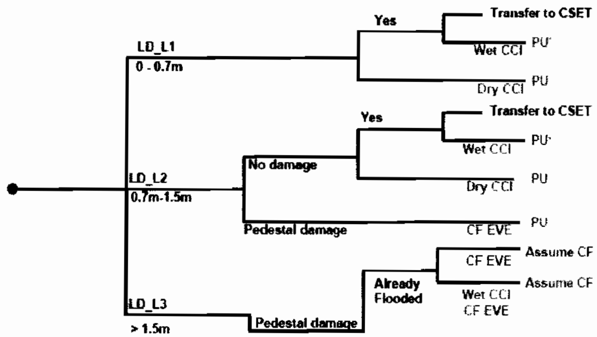
  This is shown now in terms of principle and available experimental knowledge. It will verified by full-scale tests. These tests are of the engineering practice type so they belong to the COL stage of the review.

- Verification of continuing ability to function as designed through-out the operating life.

  This will require some periodic testing for the I&C features of the BiMAC system

Conclusion (1): The Low Pressure CPET

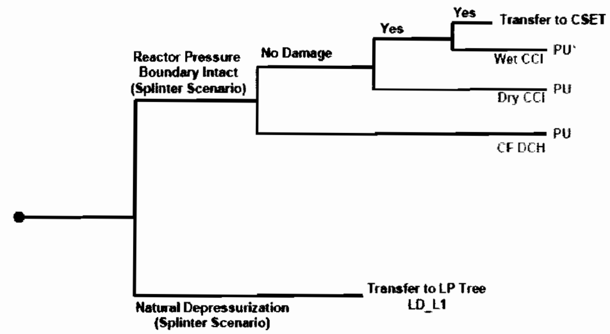
Class I: LP	LD_LVL	EVE_DAM	BI_SP	BI_FN	Probability
RPV Failure at Low Pressure (< 1 MPa)	Water Level prior to RPV Failure	*Pedestal Intact*	GDCS Deluge Supply to BiMAC Successful	*Debris is Successfully Cooled*	



PU is for Physically Unreasonable; PU* is Pending Experimental Verification at COL
The LD_L3 branch represents less than 1% of the CDF. See also Ch.9 of NEDO-33201

Conclusion (2): The High Pressure CPET

Class III: HP	RCB_I	DCH	BI_SP	BI_FN	Probability
RPV Failure at High Pressure (> 1 MPa)	Reactor Coolant Boundary Intact	*Containment Intact Insignificant DCH*	GDCS Deluge Supply to BiMAC Successful	*Debris is Successfully Cooled*	



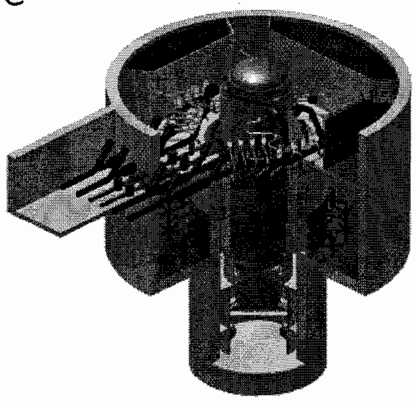
PU is for Physically Unreasonable; PU* is Pending Experimental Verification at COL

Conclusion (3): Summary of containment threats and mitigative mechanisms or systems in place for responding to them

Threat	*Failure Mode*	*Mitigation*
DCH	*Energetic DW Failure*	*Pressure Suppression Vents* *Reinforced Concrete Support*
	UDW Liner Thermal Failure	*Liner Anchoring System*
	LDW Liner Thermal Failure	*Reinforced Concrete Barrier* *Gap Separation from UDW*
EVE	*Pedestal/Liner Failure*	*Dimensions and Reinforcement*
	BiMAC Failure	*Pipe Size and Thickness* *Pipes Embedded into Concrete*
BMP & CCI	*BiMAC Activation Failure*	*Sensing & Actuation Instrumentation* *Diverse/Passive Valve Action*
	Local Burnout	*Natural Circulation*
	Water Depletion	*Natural Circulation*
	Local Melt-Through	*Refractory Protective Layer*

ESBWR Containment Systems Performance

Severe Accident Mitigation



Presented By:
Rick Wachowiak
General Electric
April 20, 2006

imagination at work

Overview of Containment Systems Performance

ESBWR Containment Robust for Severe Accident Phenomena – BMP, EVE, DCH

Need to Address

> Containment Bypass

> Containment Overpressurization

Containment Systems Provided to Address These

Bypass

Can Only Occur if Large Penetrations are Open to the Environment

All Penetrations in DCD Were Dispositioned as:

> Normally Closed During Operation

> Connected to Closed System Inside Containment

> Connected to Closed System Outside Containment

> Already Addressed in Level 1 Break Outside Containment Analysis

Containment Bypass is not Credible in ESBWR

Overpressure Protection

Function is Provided By:

> Passive Containment Cooling System

> Fuel and Auxiliary Pool Cooling System

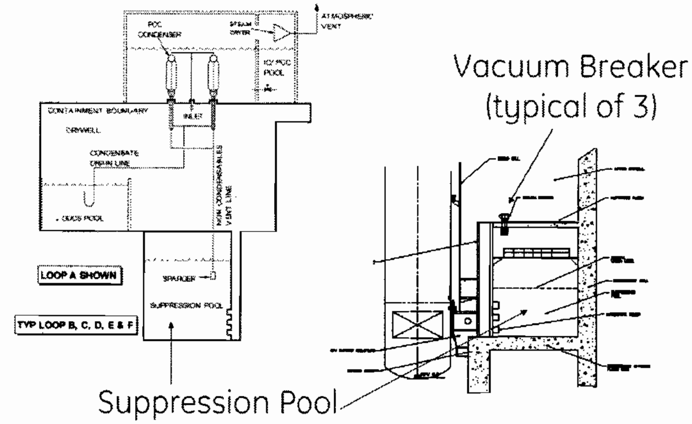
> Manual Venting

Just as in Level 1, Passive Function is Backed Up by Redundant Active Functions

PCCS Operation During Severe Accidents

- No Active Components (24 hours)
- Steam in Drywell is Condensed and Returned to the Drywell – Closed System
- Aerosols are Carried Along With the Condensate
- The Only Issue is Build-Up of Non-Condensable Gas
  - > This reduces the effectiveness of the system
  - > Vent line provided to address non-condensables
  - > Requires vacuum breaker to be seated

PCCS (typical of 6)



Vacuum Breaker Design

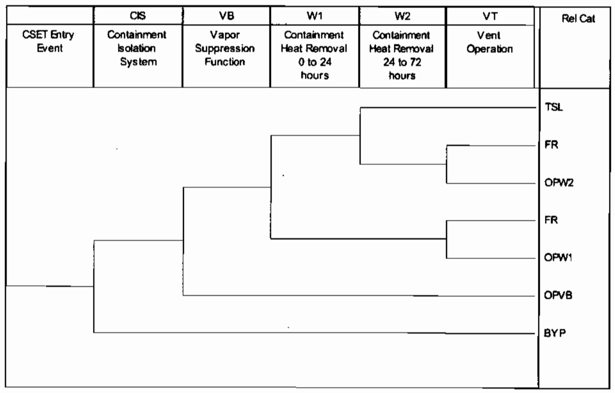
- Vacuum Breakers are Passive Components
- Gravity Holds Them In Place
- Positive Indication of Closure
- Internal Valve Can Isolate Failed VB
- Operator Can Also Isolate VB Based on Containment Conditions

PCCS Reliability

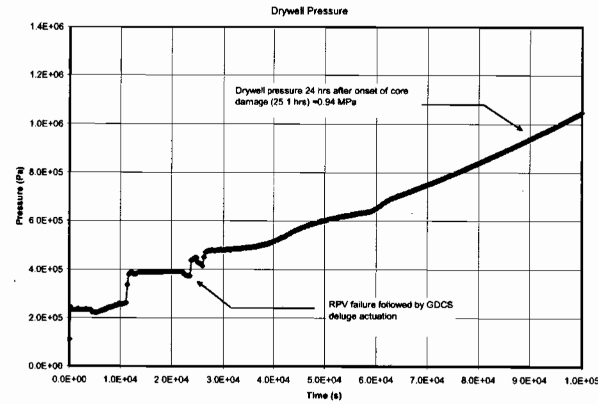
- Unreasonable to Consider PCCS Failure Within 24 Hours of Initiating Event
- Water Makeup Needed for 24 – 72 Hour Period
- Automatic Makeup Considered – Requires DC Power
- Backup Water Addition Also Considered – Requires Manual Action

PCCS Failure Extremely Unlikely in 99% of Core Damage Sequences

Containment System Event Tree



Long Term Containment Behavior Hypothetical Early Loss of PCCS



Containment Systems Results

Bypass is Negligible

Overpressure within 24 Hours is Negligible

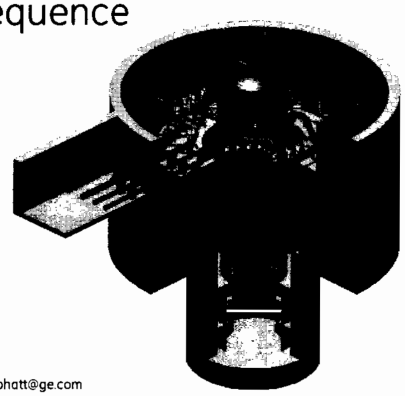
Overpressure later than 24 Hours Can Occur in Some High Pressure Sequences

> Mitigated by venting – but still a release

> Release does not occur for more than 24 hours

Overall Containment Systems Reliability is 99%

Offsite Consequence Analysis



April 20, 2006
Sid Bhatt
Phone: 408-925-5251, **email:** sid.bhatt@ge.com

Offsite Consequence Analysis

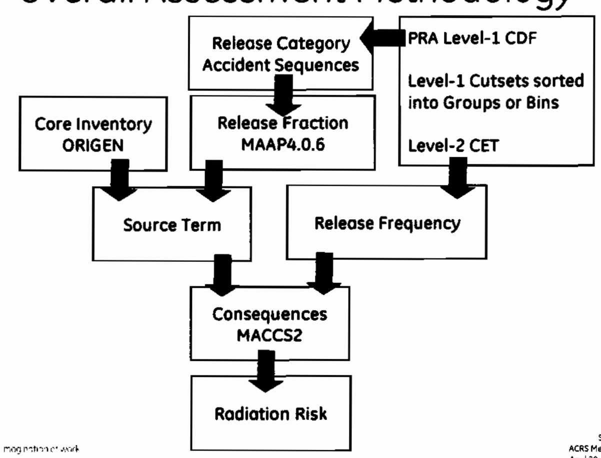
- Goals
- Radiological release assessment methodology
- Results and conclusions

Offsite Consequence Analysis - Goals

1. Individual Risk: Risk to average individual in the "vicinity" of a nuclear power plant of prompt fatalities that might result from nuclear accidents should not exceed 0.1% of the sum of "prompt fatality risks" resulting from other accidents to which the U.S. Population are generally exposed. (<0.1% * 39 deaths/100,000 people per year[Ref 1]) *Ref.1 Accident Facts, 1988, National Safety Council*
2. Societal Risk: Risk to the population in the area "near" a nuclear power plant of cancer fatalities that might result from nuclear power plant operation should not exceed 0.1% of the sum of the "cancer fatality risks" resulting from all other causes. (<0.1%*169 deaths/100,000 people per year [Ref 2])

   Ref. 2 1986 Cancer Facts and Figures, American Cancer Society, 90 Park Ave, New York, NY 10016.
3. Radiation Dose: Probability of exceeding a whole body dose of 0.25 Sv at a distance of 0.5 mile from the reactor shall be less than one in a million per reactor year (<10E-6)

Overall Assessment Methodology



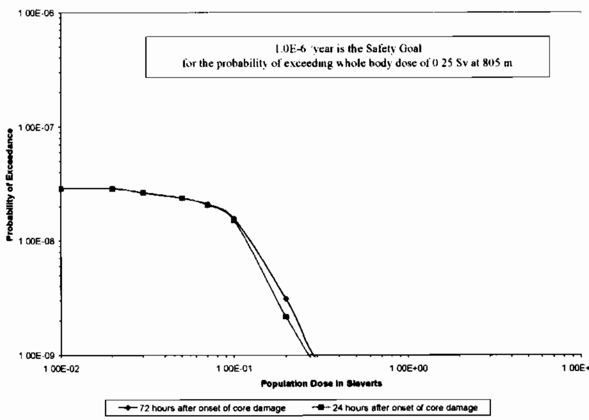
MAAP Simulation of Representative Sequence

- Source terms associated with each release category were developed using MAAP simulations of a representative sequence.
- Each representative MAAP sequence provided release fractions for 12 radionuclide groups (Xe/Kr, CsI, TeO2, SrO, MoO2, CsOH, BaO, La2O3, CeO2, Sb, Te2, and UO2) for the period 24 hours and 72 hours after onset of core damage.
- Source terms and associated release category frequencies are used in the offsite consequence analysis

MACCS2 Application to ESBWR

- 24hr and 72 hrs after onset of core damage
- Core inventory: ORIGEN – Equilibrium Core
- Release fraction calculation: MAAP4.0.6
- Meteorology – ALWR URD
- Population – SANDIA Siting Study Used
  (more bounding 0-10 mile population density that in ALWR URD)
- No Evacuation or relocation scenario used
  (public continue normal activity during reactor accident)
- Other Input Parameters
  - Building data for Wake Effect model
  - Release height: ground level
  - Heat content of plume: 0 W

Whole Body Dose at (0.5 Mile) as Probability of Exceedance



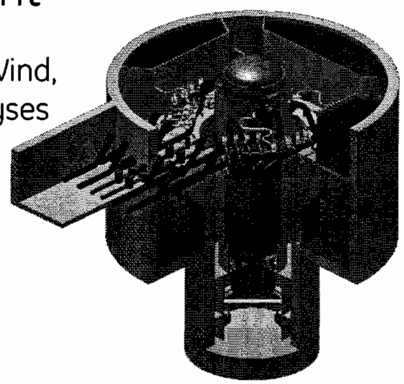
Whole Body Dose at 805 m (0.5 Mile) as Probability of Exceedance
*The goal of a maximum probability of 1E-6 is well above the entire dose range at 0.5 mile

Results & Conclusion

Goal	Numerical Goal	24 Hours After the Onset of Core Damage	Safety Goal Achieved (24 Hours)	72 Hours After the Onset of Core Damage	Safety Goal Achieved (72 Hours)
Individual Risk (0 – 1 Mile)	$<3.9\times10^{-7}$ (0.1%)	2.6E-11	Yes	3.7E-11	Yes
Societal Risk (0 – 10 Mile)	<1.7x10-6 (0.1%)	4.8E-12	Yes	6.0E-12	Yes
Radiation Dose Probability - Whole Body Dose of 0.25 Sv (0 – 0.5 Mile)	<10-6	<2.2E-9	Yes	<3.1E-9	Yes

ESBWR External Events Risk Management

Fire, Flood, High Wind, and Seismic Analyses



Presented By:
Rick Wachowiak
General Electric
April 21, 2006

Probabilistic Fire Analysis

FIVE Methodology Provides the Bases for:

> Identifying fire compartments
> Defining fire ignition frequencies
> Performing quantitative screening analyses of fire risk

Risk of Core Damage due to Fire in Each of the Area Groups Should be Lower than the Risk of Core Damage due to Internal Events

Scope of Analysis

Fire scenarios in:

> Reactor Building
> Control Building
> Fuel Building
> Turbine Building
> Electrical Building
> Service Water Building

Full Power and Shutdown Modes of plant operation

Bounding Assumptions

Fires Grow Within A Building to Non-Mechanistically Affect All Equipment In A Division

> Any Fire in a Division I Room In the Reactor Building is Assumed to Damage All Division I Equipment in the Reactor Building

Fire Protection is Not Credited

Worst Case Spurious Actuation is Postulated

Fire Results

All Fire Scenarios But One Have CDF < 3×10^{-10}

Turbine Building Considered One Fire Area

Turbine Building Fire Treated as Loss of Feedwater

This Sequence Has a CDF of 1×10^{-8}

Similar to Loss Of Feedwater in Internal Events

Shutdown Fire Results

Still Under Development

Probabilistic Flooding Analysis

Initiation frequency based upon BWR experience

Flood scenarios in:

> Reactor Building

> Control Building

> Fuel Building

> Turbine Building

> Electrical Building

> Service Water Building

Full Power and Shutdown Modes of plant operation

Flooding Frequencies

At-Power Flooding Frequencies

Based on the general information contained in NUREG/CR-5750 and NUREG/CR-2300

Shutdown Flooding Frequencies

Shutdown frequencies are calculated based on the data for BWR plants for the years 1980-1996. Provided in NUREG/CR-5496.

The ESBWR flooding frequency values also accounts for a 24 month refueling outage cycle.

Reactor Building

Major water sources:

(1) Fuel Auxiliary Pool Cooling System (FAPCS)
(2) Reactor water Cleanup / Shutdown Cooling (RWCU/SDC)
(3) Reactor Component Cooling Water System: (RCCWS)
(4) Fire Protection System: (FPS)
(5) Feedwater System: FW pipe breaks are LOCA initiators. FW lines outside containment are located in the steam tunnel. A FW line break in the steam tunnel flood progression into the Turbine Building.

Control Building

Major water sources:

(1) Chilled Water System: Limited volume of water is not sufficient to cause an initiating event.

(2) Potable Water and Sanitary Waste System (PWSWS): Small water volume released.

(3) Fire Protection System (FPS): Pipes are of short length and small diameters (2-1/2 inches). The frequency and the impact of the water released is small.

Fuel Building

Potential flooding sources:

(1) Fuel and Auxiliary Pools Cooling System (FAPCS): Check valves and vacuum breaker valves eliminate potential siphon effect discharge from the fuel pool. Flooding requires a system pipe break and failure of at least one vacuum breaker valve.
(2) Reactor Component Cooling Water (RCCW):
(3) Fire Protection System (FPS): Larger FPS pipes. Water released to the FB lower floor can progress through an open doorway. Large volume of released water could cause the loss of the RWCU/SDC.

Turbine Building

Flooding sources considered are:

(1) Circulating Water System (CWS)

Flooding from a break in the circulating water system is the bounding scenario in the TB

(2) Condensate and Feedwater System (C&FS)
(3) Reactor Component Cooling Water (RCCWS)
(4) Plant Service Water System (PSWS)
(5) Fire Protection System (FPS)

Electrical Building

Flooding sources considered are:

(1) FPS system: FPS flow rate is low.

(2) RCCW: Flooding due to diesel generator cooling water system leak in a single diesel generator room is considered to be a negligible risk. Flooding in one diesel generator room would not affect the other diesel generator, and flooding in a DG room would not affect external power supplies, or cause an initiating event.

Service Water Building

The Plant Service Water System (PSWS) is the primary flood source in the Service Water Building.

The loss of service water scenario is included and analyzed in the ESBWR PRA internal events analysis by use of the Complete Loss of PSWS initiator.

The frequency of service water floods in the Service Water Building is inherently included in the Complete Loss of PSWS initiator frequency.

At Power Flooding Scenarios

AP-1: Reactor Building Outside Containment - CRDS pipe breaks outside containment.

AP-2: Reactor Building Outside Containment - FPS pipe breaks.

AP-3: Reactor Building Outside Containment - RWCU/SDCS line break outside of containment.

AP-4: Reactor Building Outside Containment - FPS line break and general transient

AP-5: Turbine Building - Complete loss of feedwater

AP-6: Turbine Building - Loss of Plant Service Water

AP-7: Electrical Building - Loss of Power Conversion System

AP-8: Diesel Generator Room – General Transient with Loss of One Diesel Generator

Shutdown Flooding Scenarios

SD-1 and SD-3: Reactor Building - CRDS pipe breaks outside containment

SD-2 and SD-4: Reactor Building Outside Containment Shutdown Flooding Scenario

SD-5 and SD-6: Reactor Coolant System Inventory Control

SD-7 an SD-8: Fuel Building Shutdown Flooding Scenarios

CDF for internal flooding

CDF for internal flooding is a not a dominant contributor to the overall plant CDF.

The contribution due to flood to the CDF is one order of magnitude less than the CDF due to internal events:

Contribution Description	CDF (per calendar year)
Internal Events	2.9E-08
Flood At-Power	3.7E-09
Flood Shutdown	1.6E-09

Key Features Important to Flood Results

- Layout and safety design features
- Safety system redundancy and physical separation provide protection from flooding by large water sources
- Alternate safe shutdown features in buildings separated from flooding of safety systems
- Watertight doors on the Control and Reactor Buildings
- Floor drains in the Reactor and Control Buildings
- Automatic CWS pump trip and valve closure on high water level in the condenser pit

High Wind Risk - Tornado

Treated as Loss of Preferred Power with No Recovery within 24 Hours

Condensate Storage Tank is Assumed Failed

Initiating Event Frequency is Much Lower than LOPP without Recovery

Risk Due to this Scenario is Very Small ~ 10^{-12}

Seismic Margins

Addresses the Capability of Safety Systems for Seismic Response

Determined Fragility for All Safety Systems

Assigned That Fragility To Each Branch of The Event Tree

> Non-Safety assigned 0.0*SSE fragility value

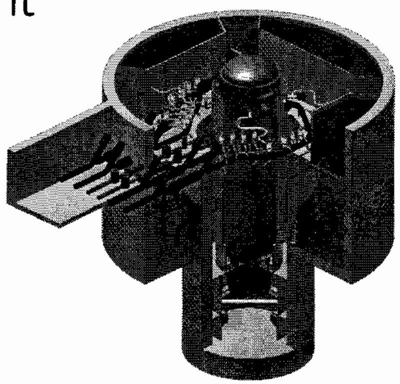
Fragility for the Sequence is the Maximum of Each Branch

Total Fragility is the Minimum of All the Sequences

Seismic Margins Results

All Sequences Show At Least 2 * SSE Capability
Full Power and Shutdown
Unlikely the Seismic Will Be a Vulnerability

ESBWR Shutdown Risk Management



Presented By:
Rick Wachowiak
General Electric
April 21, 2006

imagination at work

Scope of Shutdown Analysis

- Internal Events & External Events
- Seismic Margins
- Mode 5 (Cold Shutdown)
- Mode 6 (Refuel)
- Same Level of Detail as Power Operation PRAs

Initiating Events During Shutdown

- Manual Shutdown
- LOCA – Mode 6 Only
- Loss of Power
- Loss of Shutdown Cooling
- Fires
- Floods

Not Applicable for Mode 6 With Reactor Cavity Flooded

There is More than 72 Hours to Recover DHR

Maintenance Activities During Shutdown

- Multiple Pumps / Trains of Feedwater And Condensate May be Unavailable
- Some Fire and Flood Barriers May Be Open
- ICS Out of Service in Mode 6
- 1 GDCS Pool Allowed Unavailable in Mode 6
- PCCS Unavailable in Mode 6
- SRVs and DPVs Unavailable in Mode 6

Recovery Actions During Shutdown

Shutdown Events Tend to Move Slower
More Time to Recover Initiating Event
Recovery Events Added to Shutdown Model
- > Recovery of Shutdown Cooling
- > Recovery of Offsite Power
- > Recovery of Service Water

Approximately 5 Hours to Recover
Non-Recovery Value Based on Industry Events that Have Occurred During Shutdown

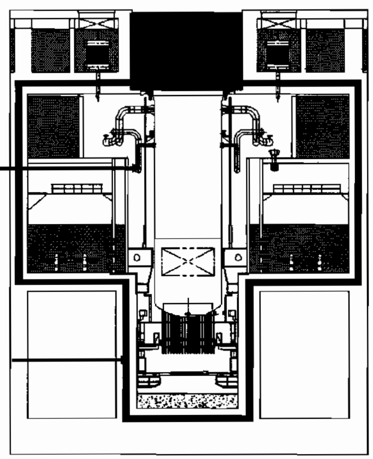
Shutdown CDF Results

Manual Shutdown	2×10^{-12}
Loss of DHR	$< 10^{-12}$
Loss of Service Water	3×10^{-12}
Loss of Preferred Power	4×10^{-10}
LOCA	4×10^{-9}

Containment Water Capacity During Shutdown LOCA

Approximate Water Level With Hatch Closed

Elevation of Hatch



Shutdown PRA As A Design Tool Example

LOCA Dominated by Pipes Connected Below the Core
PRA Assumes Hatches are Open During Mode 5
PRA Assumes Containment is Open During Mode 5

GE is Considering Options to Address This Scenario

Shutdown PRA Input to Operational Programs Example

Fire Barriers Should Be Controlled During Shutdown

Remaining Intact is Best Option

Compensatory Measures (e.g. Fire Watch) are Adequate

Detailed Layout / Routing and Fire Modeling Needed to Relax This Requirement

Final Remarks on Shutdown

Iterative Process with Design Still in Progress for Shutdown

Fire and Flood Models For Shutdown Still Under Development

ESBWR Risk Management Insights

ESBWR Risk Management Program

Supports Desired Goals

Scope is Appropriate

Enhanced Defense-in-Depth

PRA is a Valuable Design Tool

PRA Will Continue to Grow Through Plant Operation

Overall Results and Observations

ESBWR Robust Design Results in Low CDF and LRF

We Are Testing The Limits of Current PRA Techniques

> Unknowns may be as important as the known

Some Screening Methods Not as Effective

> Thresholds too low to screen anything

Relative Risk Ranking Could Be A Significant Issue

> Also a threshold problem

When Compared to Other Plants, Using the Same Methods, ESBWR Provides the Best Level of Safety Available

ESBWR DESIGN CERTIFICATION PRA AND SEVERE ACCIDENTS OVERVIEW

ACRS - Reliability and Probabilistic Risk Assessment Subcommittee

April 20 & 21, 2006

PRA and Severe Accident RAIs

RAI 19.0.0-1: Requested peer review results for ROAAM methodology used to support the assessment of direct containment heating, steam explosions, and core concrete interactions for ESBWR.

RAI 19.2.3-1: Requested equipment survivability assessment.

RAI 19.2.4-1: Requested information regarding the accident management program under which guidance and training would be provided on the use of such features as containment venting, drywell sprays, and fire pumps for isolation condenser make-up.

RAI 19.4.0-1: Requested more rigorous evaluation of Severe Accident Mitigation Design Alternatives (SAMDAs).

2

PRA and Severe Accident RAIs (cont.)

RAI 19.1.0-3: Requested submittal of additional cutsets and a discussion on the use of uncertainty, sensitivity and importance analyses.

RAI 19.1.0-4: Requested that GE Identify design requirements based on PRA insights and assumptions.

RAI 19.1.0-5: Requested references for component reliability data base.

RAI 19.1.0-6: Requested detailed evaluations of important human actions and their associated human error probabilities.

RAI 19.1.0-7: Requested additional details regarding GE's fire risk analysis.

5

PRA and Severe Accident RAIs (cont.)

RAI 19.3.0-1: Requested risk assessment for fires and floods during shutdown.

RAI 19.3.0-2: Requested discussion of large release frequency (LRF) risk during shutdown.

Additional issues identified during meetings:

- Assessment of potential RCS draindown paths through the RWCU/SDC system and risk of using freeze seals
- LRF contribution from cold shutdown operations when the containment can be open
- Impact on level 2 PRA results if BiMAC is not credited
- Effect of impingement of molten core debris on the lower drywell equipment/personnel hatch
- Drywell water level at time of vessel breach
- Modeling of the digital I&C system in the PRA

6

CONFIRMATORY ANALYSIS OF SEVERE ACCIDENTS FOR ESBWR

Advisory Committee on Reactor Safeguards
U. S. Nuclear Regulatory Commission
April 21, 2006
by:

M. Khatib-Rahbar, Z. Yuan, M. Zavisca, A. Krall and H. Esmaili
Energy Research, Inc.
6167 Executive Blvd.
Rockville, Maryland 20852

ERI
Energy Research, Inc.

OUTLINE

- Objectives
- MELCOR Modeling of ESBWR
- Preliminary Results
- Planned Analyses

ERI
Energy Research, Inc.

2

OBJECTIVES

- To support the design certification review of severe accident risk by NRC in
  - Independent assessment of severe accident response
  - Confirmatory assessment of representative radiological release estimates
  - Development of uncertainties in the initial and boundary conditions for analysis of selected severe accident issues
  - Confirmatory analysis of selected severe accident issues (e.g., ex-vessel steam explosion, MCCI, etc.)

ERI
Energy Research, Inc.

3

MELCOR Model Development

- Developed initial input decks for MELCOR 1.8.6 using GE design data
- MELCOR 1.8.6 deck subjected to an independent QA and review (Purdue & SNL/JTA)
- Review comments factored in the modification to the MELCOR 1.8.6 deck
- Due to code performance issues, the deck was finalized for MELCOR 1.8.5
- The initial baseline calculations were performed with MELCOR 1.8.5 & work is underway to finalize MELCOR 1.8.6 deck as performance issues are being resolved by SNL

ERI
Energy Research, Inc.

4

OTHER FEATURES

- Containment spray system and the venting system included
- Refill of PCC/IC pool included.
- BiMAC system not explicitly modeled
- Pre-accident steady state calculation performed prior to simulation of accidents

ERI
Energy Research, Inc.

MELCOR Steady-State Results vs. GE DCD Values

Parameters	Design value	Simulated value
Steam flow rate (kg/s)	2433	2436
Feedwater flow rate (kg/s)	2451	2452
Core coolant flow rate (kg/s)	9034-10584	9452
Control Rod Drive flow rate (kg/s)	5.9	5.9
Cleanup demineralizer system flow rate (kg/s)	24.3	24.3
System pressure, nominal in steam dome (kPa)	7171	7177
System pressure, nominal core design (kPa)	7240	7243
Core inlet temperature (°C)	543-545	543
Total core pressure drop (from bottom of the core support plate to top of the core) (kPa)	70.0	47.0
Core plate pressure drop (kPa)	41.3	31.5
Core maximum exit void fraction	0.916	0.90
Downcomer liquid level (m)	17.27	17.6

ERI
Energy Research, Inc.

MELCOR-Simulated Accident Scenario

- Transient event initiated by a loss of feedwater (i.e., scenario T_DP_nIN of the ESBWR PRA):
  - Short or long-term coolant injection to RPV not available (i.e., GDCS injection to RPV & wetwell injection through equalization lines not available).
  - ADS is assumed to be actuated if downcomer water level drops below 11 m.
  - Heat removal by ICs not credited.
  - PCC & PCC/IC pool makeup available (thereby allowing long-term containment heat removal).
  - GDCS deluge system is also available for injection onto the lower drywell floor.

ERI
Energy Research, Inc.
7

MELCOR-Simulated Accident Scenario (Cont.)

- Two cases considered:
  - ❖ Case 1: MCCI suppressed (Perfect BiMAC)
  - ❖ Case 2: MCCI allowed to occur (assuming MELCOR standard basaltic concrete composition).

ERI
Energy Research, Inc.
8

Comparison With MAAP Results (Case 1: Without MCCI)

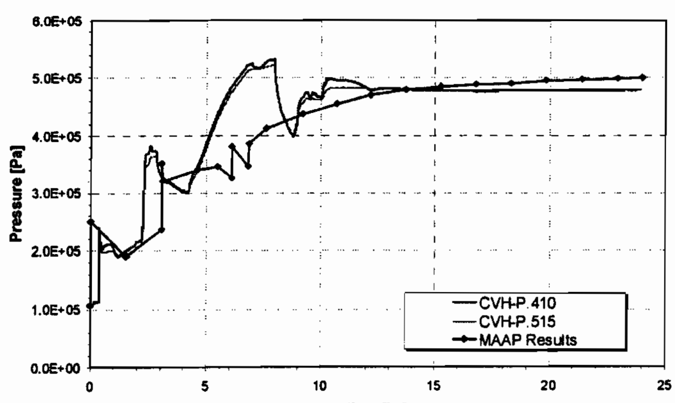
Event	MAAP*	MELCOR
RPV depressurization starts (DPVs open), hour	8.6×10^{-3}	0.33
Start of core uncovery, hour	0.36	0.86
Onset of core damage (i.e., fuel temperature exceeds 2500 K), hour	0.97	1.69
RPV lower head penetration failure, hour	6.3	3.91
Deluge system actuated, hour	6.3	7.9
Containment (upper drywell) pressure at 24 hours, bar-abs	5.0	4.8
Containment (lower drywell) temperature at 24 hours, K	425	427
Containment fail/vent, hour	N/A	N/A
PCCS heat removal at 24 hours, MW	18.5	22.7
Water level in drywell at 24 hours (relative to bottom of the RPV), m	13.1	12.5
Axial concrete erosion in 24 hours, m	0.07	0.0
Mass fraction of noble gases released to environment	9.0×10^{-4}	8.7×10^{-4}
Mass fraction of CsI released to environment	7.4×10^{-5}	1.8×10^{-5}

*MAAP results taken from NEDC-33201P (Rev 0)

ERI Energy Research, Inc.

9

Upper Drywell and Wetwell Pressure (Case 1: Without MCCI)

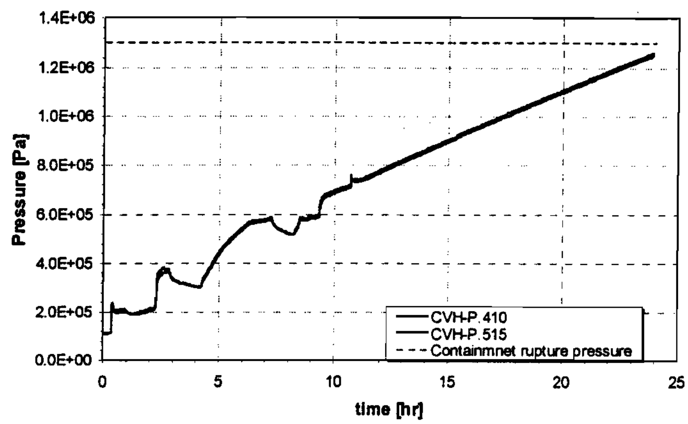


MAAP results taken from NEDC-33201P (Rev 0)

ERI Energy Research, Inc.

10

Upper Drywell and Wetwell Pressure (Case 2: with MCCI)



ERI
Energy Research, Inc.

11

Planned Calculations

- Rationale for selection of scenarios:
  - To provide initial & boundary conditions for NRC confirmatory analyses (e.g., FCI)
  - To enable limited comparison to MAAP predictions
  - To assess sensitivity to design/operational aspects (e.g., sprays)
  - To support other NRC objectives
- "Risk-dominant", "frequency-dominant", and "consequence-dominant" scenarios will be examined, together with influence of various assumptions and sensitivity cases

ERI
Energy Research, Inc.

12

STATUS

- MELCOR model completed and initial confirmatory calculations are underway.
  - Results of a representative accident scenario with limited comparisons to the GE submittal completed
  - Identified representative scenarios to be analyzed
  - Baseline MELCOR calculations for the most part, have been completed with the available data; however, final calculations await the receipt of requested data from GE.

ERI
Energy Research, Inc.

13

STATUS (Cont.)

- Ex-Vessel FCI analyses have been started:
  - Initial calculations aimed at confirming the GE calculations under identical conditions
  - Will formulate initial conditions for ex-vessel analyses:
    - Lower head failure size and location
    - Debris mass, composition and temperature
    - Etc.
  - Perform analyses to span a wide range of conditions and parameters (similar to those of AP1000)

ERI
Energy Research, Inc.

14