ACCIDENT PROGRESSION ANALYSIS
P-300

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United States
Nuclear Regulatory Commission

Idaho National Laboratory

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Accident Progression Analysis (P-300)

Course Presented by

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Course Objective

- To understand the basics of severe accident progression, from the onset of core damage to the release of a radioactive source term to the environment
  - Onset of core damage (for PWRs) often defined as the uncovering of the top of active fuel (TAF)
    - Temperature criteria also used
  - Two phases: core degradation and containment challenge
    - In-vessel and ex-vessel
  - Release to the environment often characterized in terms of Large Early Release Frequency (LERF)

Course Outline

1. Risk-Informed Regulation and Review of PRA Basic concepts
2. Overview of Level-1/2/3 PRA
3. LWR Containment Designs
4. Phenomena Affecting Vessel Integrity
5. Phenomena Affecting Containment Integrity
6. Containment Event Tree Development
7. Phenomenological Modeling Capabilities
8. Radionuclide Release and Transport
9. Level-2 PRA Integration and Quantification
10. Example Level-2 Analysis
11. NUREG/CR-6595
12. Review
13. Exam

Annotated Bibliography

- WASH-1400, Reactor Safety Study: An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants, October 1975
  - Original Level-2 analysis.
- NUREG/CR-4551, Volumes 1 - 7, Evaluation of Severe Accident Risks, Dates: varied (1990 - 1993)
  - Most comprehensive Level-2 analysis, developed Accident Progression Event Tree (APET) method of modeling containment performance (i.e., event tree with 75 - 125 top events).
- NUREG/CR-6595, Rev.1, An Approach for Estimating the Frequencies of Various Containment Failure Modes and Bypass Events, September 2004.
  - Developed simple LERF models to support Reg. Guide 1.174.
- NUREG-1560, Volumes 1, 2 & 3, Individual Plant Examination Program: Perspectives on Reactor Safety and Plant Performance, December 1997
  - Extracted and summarizes highlights and insights from the collective IPE results (75 IPEs covering 108 NPP units), including containment performance issues.

Annotated Bibliography (cont.)

- NUREG/CR-6338, Resolution of the Direct Containment Heating Issue for All Westinghouse Plants With Large Dry Containments or Subatmospheric Containments, February 1996
  - Comprehensive analysis of all referenced plants, includes PWR containment design details extracted from IPEs, including fragility curves.
- NUREG/CR-6475, Resolution of the Direct Containment Heating Issue for Combustion Engineering Plants and Babcock & Wilcox Plants, November 1998.
  - Comprehensive analysis of all referenced plants, includes PWR containment design details extracted from IPEs, including fragility curves.
- NUREG/CR-5423, The Probability of Liner Failure in a Mark-I Containment, August 1991.
  - Detailed analysis of issue, benefited from a public workshop and an extensive peer review process.

Annotated Bibliography (cont.)

- EPRI NP-6260-M, Criteria and Guidelines for Predicting Concrete Containment Leakage, April 1989.
  - EPRI developed method for predicting containment failure mechanisms and leakage locations.
- NUREG-1037, Draft Report for Comment, Containment Performance Working Group Report, May 1985.
  - Analyzed potential leakage of containment penetrations as a result of conditions beyond design basis.
- IDCOR T-10.1, Containment Structural Capacity of Light Water Nuclear Power Plants, July 1983
  - Analyzes ultimate containment capacity of several PWR and BWR containment structures. Appendix B describes the method used to generate containment fragility curves.

Annotated Bibliography (cont.)

- NUREG/CR-4242, Survey of Light Water Reactor Containment Systems, Dominant Failure Modes, and Mitigation Opportunities, January 1988
  - Detailed descriptions of various containment designs, rest of information somewhat dated.
- NUREG-1570, Risk Assessment of Severe Accident-Induced Steam Generator Tube Rupture, March 1998.
  - Latest information available on induced SGTRs.
- NUREG-1150, Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants, December 1990.
  - Summary report on the five full-scope PRAs performed and documented in the NUREG/CR-4550, Vol. 1-7; and NUREG/CR-4551, Vol. 1-7.

Acronyms

ACRS	Advisory Committee on Reactor Safeguards
ADS	Automatic Depressurization System
AFW	Auxiliary Feedwater System
AM	Accident Management
AP-600	Westinghouse Advanced PWR (600 MWe)
APB	Accident Progression Bin
APET	Accident Progression Event Tree
ASP	Accident Sequence Precursor
AST	Accident Source Term
ATWS	Anticipated Transient Without SCRAM
B&W	Babcock & Wilcox
BWR	Boiling Water Reactor
CCFP	Conditional (on core damage) Containment Failure Probability
CCI	Core Concrete Interaction
CD	Core Damage
CDF	Core Damage Frequency
CE	Combustion Engineering
CET	Containment Event Tree
CFD	Computational Fluid Dynamics
CFF	Containment Failure Frequency
CHF	Critical Heat Flux
CHR	Containment Heat Removal
CRD	Control Rod Drive
CS	Cut Set
CSR	Containment Spray Recirculation
CSS	Containment Spray System
DCH	Direct Containment Heating
DW	Drywell (BWR)
ECCS	Emergency Core Cooling System
ECI	Emergency Coolant Injection
ECR	Emergency Coolant Recirculation
ERVC	External Reactor Vessel Cooling
FAI	Fauske Associates, Incorporated
FCI	Fuel-Coolant Interaction
FEM	Finite Element Method
FIBS	Final Bounding State
H2	Hydrogen
HPIS	High Pressure Injection Systems
HPME	High Pressure Melt Ejection
IPE	Individual Plant Examination
ISLOCA	Interfacing System Loss of Coolant Accident
IVR	In-Vessel Retention

Acronyms (cont.)

JAERI	Japan Atomic Energy Research Institute
KAERI	Korea Atomic Energy Research Institute
LERF	Large Early Release Frequency
LHF	Lower Head Failure
LOCA	Loss of Coolant Accident
LPIS	Low Pressure Injection System
LWR	Light Water Reactor
MAAP	Modular Accident Analysis Program
MACCS	MELCOR Accident Consequence Code System
MCCI	Molten Core Concrete Interaction
MSSV	Main Steam Safety Valve
OECD	Organization for Economic Cooperation and Development
OTSG	Once-Through Steam Generator
PCS	Power Conversion System
PDF	Probability Density Function
PDS	Plant Damage State
PORV	Power (or Pilot) Operated Relief Valves
PST	Parametric Source Term
PWR	Pressurized Water Reactor
QHO	Quantitative Health Objective
RCP	Reactor Coolant Pump
RCS	Reactor Coolant system
ROAAM	Risk Oriented Accident Analysis Methodology
RPS	Reactor Protection System
RPV	Reactor Pressure Vessel
RSGPS	Reactor Safety Goal Policy Statement
RST	Revised Source Term
RWST	Refueling Water Storage Tank
SAMG	Severe Accident Management Guidelines
SBLOCA	Small Break LOCA
SBO	Station Blackout
SERG	Steam Explosion Review Group
SG	Steam Generator
SGTR	Steam Generator Tube Rupture
SNL	Sandia National Laboratory
SRV	Safety Relief Valve
TAF	Top of Active Fuel (in reactor core)
TEDE	Total Effective Dose Equivalent
TMI-2	Three Mile Island Unit 2
UCSB	University of Santa Barbara
UHI	Upper Head Injection
VB	(Reactor Pressure) Vessel Breach
WW	Wetwell (BWR)

Pressure																
1	bar	14.5	psi	100	kPa	0.10	MPa		14.5	psi	1.0	bar	100.0	kPa	0.10	MPa
2	bar	29	psi	200	kPa	0.20	MPa		100	psi	6.9	bar	689.7	kPa	0.69	MPa
3	bar	43.5	psi	300	kPa	0.30	MPa		150	psi	10.3	bar	1,034.5	kPa	1.03	MPa
4	bar	58	psi	400	kPa	0.40	MPa		200	psi	13.8	bar	1,379.3	kPa	1.38	MPa
5	bar	72.5	psi	500	kPa	0.50	MPa		250	psi	17.2	bar	1,724.1	kPa	1.72	MPa
6	bar	87	psi	600	kPa	0.60	MPa		300	psi	20.7	bar	2,069.0	kPa	2.07	MPa
7	bar	101.5	psi	700	kPa	0.70	MPa		350	psi	24.1	bar	2,413.8	kPa	2.41	MPa
8	bar	116	psi	800	kPa	0.80	MPa		400	psi	27.6	bar	2,758.6	kPa	2.76	MPa
9	bar	130.5	psi	900	kPa	0.90	MPa		450	psi	31.0	bar	3,103.4	kPa	3.10	MPa
10	bar	145	psi	1,000	kPa	1.00	MPa		500	psi	34.5	bar	3,448.3	kPa	3.45	MPa
11	bar	159.5	psi	1,100	kPa	1.10	MPa		550	psi	37.9	bar	3,793.1	kPa	3.79	MPa
12	bar	174	psi	1,200	kPa	1.20	MPa		600	psi	41.4	bar	4,137.9	kPa	4.14	MPa
13	bar	188.5	psi	1,300	kPa	1.30	MPa		650	psi	44.8	bar	4,482.8	kPa	4.48	MPa
14	bar	203	psi	1,400	kPa	1.40	MPa		700	psi	48.3	bar	4,827.6	kPa	4.83	MPa
15	bar	217.5	psi	1,500	kPa	1.50	MPa		750	psi	51.7	bar	5,172.4	kPa	5.17	MPa
16	bar	232	psi	1,600	kPa	1.60	MPa		800	psi	55.2	bar	5,517.2	kPa	5.52	MPa
17	bar	246.5	psi	1,700	kPa	1.70	MPa		850	psi	58.6	bar	5,862.1	kPa	5.86	MPa
18	bar	261	psi	1,800	kPa	1.80	MPa		900	psi	62.1	bar	6,206.9	kPa	6.21	MPa
19	bar	275.5	psi	1,900	kPa	1.90	MPa		950	psi	65.5	bar	6,551.7	kPa	6.55	MPa
20	bar	290	psi	2,000	kPa	2.00	MPa		1000	psi	69.0	bar	6,896.6	kPa	6.90	MPa
21	bar	304.5	psi	2,100	kPa	2.10	MPa		1100	psi	75.9	bar	7,586.2	kPa	7.59	MPa
22	bar	319	psi	2,200	kPa	2.20	MPa		1200	psi	82.8	bar	8,275.9	kPa	8.28	MPa
23	bar	333.5	psi	2,300	kPa	2.30	MPa		1300	psi	89.7	bar	8,965.5	kPa	8.97	MPa
24	bar	348	psi	2,400	kPa	2.40	MPa		1400	psi	96.6	bar	9,655.2	kPa	9.66	MPa
25	bar	362.5	psi	2,500	kPa	2.50	MPa		1500	psi	103.4	bar	10,344.8	kPa	10.34	MPa
50	bar	725	psi	5,000	kPa	5.00	MPa		1600	psi	110.3	bar	11,034.5	kPa	11.03	MPa
75	bar	1087.5	psi	7,500	kPa	7.50	MPa		1700	psi	117.2	bar	11,724.1	kPa	11.72	MPa
100	bar	1450	psi	10,000	kPa	10.00	MPa		1800	psi	124.1	bar	12,413.8	kPa	12.41	MPa
125	bar	1812.5	psi	12,500	kPa	12.50	MPa		1900	psi	131.0	bar	13,103.4	kPa	13.10	MPa
150	bar	2175	psi	15,000	kPa	15.00	MPa		2000	psi	137.9	bar	13,793.1	kPa	13.79	MPa
175	bar	2537.5	psi	17,500	kPa	17.50	MPa		2100	psi	144.8	bar	14,482.8	kPa	14.48	MPa
200	bar	2900	psi	20,000	kPa	20.00	MPa		2200	psi	151.7	bar	15,172.4	kPa	15.17	MPa
225	bar	3262.5	psi	22,500	kPa	22.50	MPa		2250	psi	155.2	bar	15,517.2	kPa	15.52	MPa
250	bar	3625	psi	25,000	kPa	25.00	MPa									

Temperature													
	0	C	32	F	273	K		32	F	0	C	273	K
	100	C	212	F	373	K		100	F	38	C	311	K
	200	C	392	F	473	K		200	F	93	C	366	K
	300	C	572	F	573	K		212	F	100	C	373	K
	400	C	752	F	673	K		300	F	149	C	422	K
	500	C	932	F	773	K		400	F	204	C	478	K
	600	C	1112	F	873	K		500	F	260	C	533	K
	700	C	1292	F	973	K		600	F	316	C	589	K
	800	C	1472	F	1073	K		700	F	371	C	644	K
	900	C	1652	F	1173	K		800	F	427	C	700	K
	1000	C	1832	F	1273	K		900	F	482	C	755	K
	1100	C	2012	F	1373	K		1000	F	538	C	811	K
	1200	C	2192	F	1473	K		1100	F	593	C	866	K
	1300	C	2372	F	1573	K		1200	F	649	C	922	K
	1400	C	2552	F	1673	K		1300	F	704	C	978	K
	1500	C	2732	F	1773	K		1400	F	760	C	1033	K
	1600	C	2912	F	1873	K		1500	F	816	C	1089	K
	1700	C	3092	F	1973	K		1600	F	871	C	1144	K
	1800	C	3272	F	2073	K		1700	F	927	C	1200	K
	1900	C	3452	F	2173	K		1800	F	982	C	1255	K
	2000	C	3632	F	2273	K		1900	F	1038	C	1311	K
	2250	C	4082	F	2523	K		2000	F	1093	C	1366	K
	2500	C	4532	F	2773	K		2250	F	1232	C	1505	K
	2750	C	4982	F	3023	K		2500	F	1371	C	1644	K
	3000	C	5432	F	3273	K		2750	F	1510	C	1783	K
								3000	F	1649	C	1922	K
								3250	F	1788	C	2061	K
								3500	F	1927	C	2200	K
								3750	F	2066	C	2339	K
								4000	F	2204	C	2478	K

Accident Progression Analysis (P-300)

1. Risk-Informed Regulatory Background and Review of PRA Basic Concepts

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Session Objectives

- To understand the motivation for Level-2 PRA
  - NRC regulatory philosophy
    - PRA Policy Statement
    - Reactor Safety Goal Policy Statement
    - Regulatory Guide 1.174
- To understand some of the basic PRA concepts
  - Risk
  - Large Early Release Frequency (LERF)

PRA Policy Statement

- The Nuclear Regulatory Commission's (NRC's) policy for implementing risk-informed regulation was expressed in the 1995 policy statement on the use of probabilistic risk assessment (PRA) methods in nuclear regulatory activities. The policy statement states:
  - The use of PRA technology should be **increased** in all regulatory matters to the extent supported by the **state-of-the-art** in PRA methods and data and in a manner that **complements the NRC's deterministic approach** and supports the NRC's traditional defense-in-depth philosophy.
  - PRA and associated analyses (e.g., sensitivity studies, uncertainty analyses, and importance measures) should be used in regulatory matters, where practical within the bounds of the state-of-the-art, to **reduce unnecessary conservatism** associated with current regulatory requirements, regulatory guides, license commitments, and staff practices. Where appropriate, PRA should be used to support the proposal of additional regulatory requirements in accordance with 10 CFR 50.109 (Backfit Rule). Appropriate procedures for including PRA in the process for changing regulatory requirements should be developed and followed. It is, of course, understood that the intent of this policy is that **existing rules and regulations shall be complied with** unless these rules and regulations are revised.

PRA Policy Statement (Continued)

- PRA evaluations in support of regulatory decisions should be as **realistic** as practicable and appropriate **supporting data** should be publicly available for review.
- The Commission's safety goals for nuclear power plants and subsidiary numerical objectives are to be used with appropriate consideration of **uncertainties** in making regulatory judgements on the need for proposing and backfitting new generic requirements on nuclear power plants licensees.

Reactor Safety Goal Policy Statement

- Originally issued in 1986
- Expressed Commission's policy as:
  - …consequences of nuclear power operations such that individual bear no significant additional risk to life and health.
  - Societal risks...from NPP…should be comparable or less than the risks of generating electricity by viable competing technologies and should not be a significant addition to other societal risk.

RSGPS (continued)

- Established Quantitative Health Objectives (QHOs)
  - Early fatality risk (0.1% of total accident risk) and latent cancer risk (0.1% from all causes)
    - For an individual living in the vicinity of a NPP
  - Based on the risk of accidental death in the U.S., this implies a prompt fatality QHO of 5E-7 per year
  - Based on the occurrence of cancer fatalities, this implies a latent cancer fatality QHO of 2E-6 per year

RSGPS (concluded)

- Update proposed by NRC staff - March 30, 2000 (SECY-00-0077)
- Commission approved (with exceptions) - June 27, 2000
  - Emphasize safety goals are “goals” not limits
- Nine issues addressed, including:
  - Maintained core damage frequency subsidiary goal of 10^{-4} per reactor-year
  - Incorporated Large Early Release Frequency (LERF) subsidiary goal of 10^{-5} per reactor-year
    - Consistent with Reg. Guide 1.174

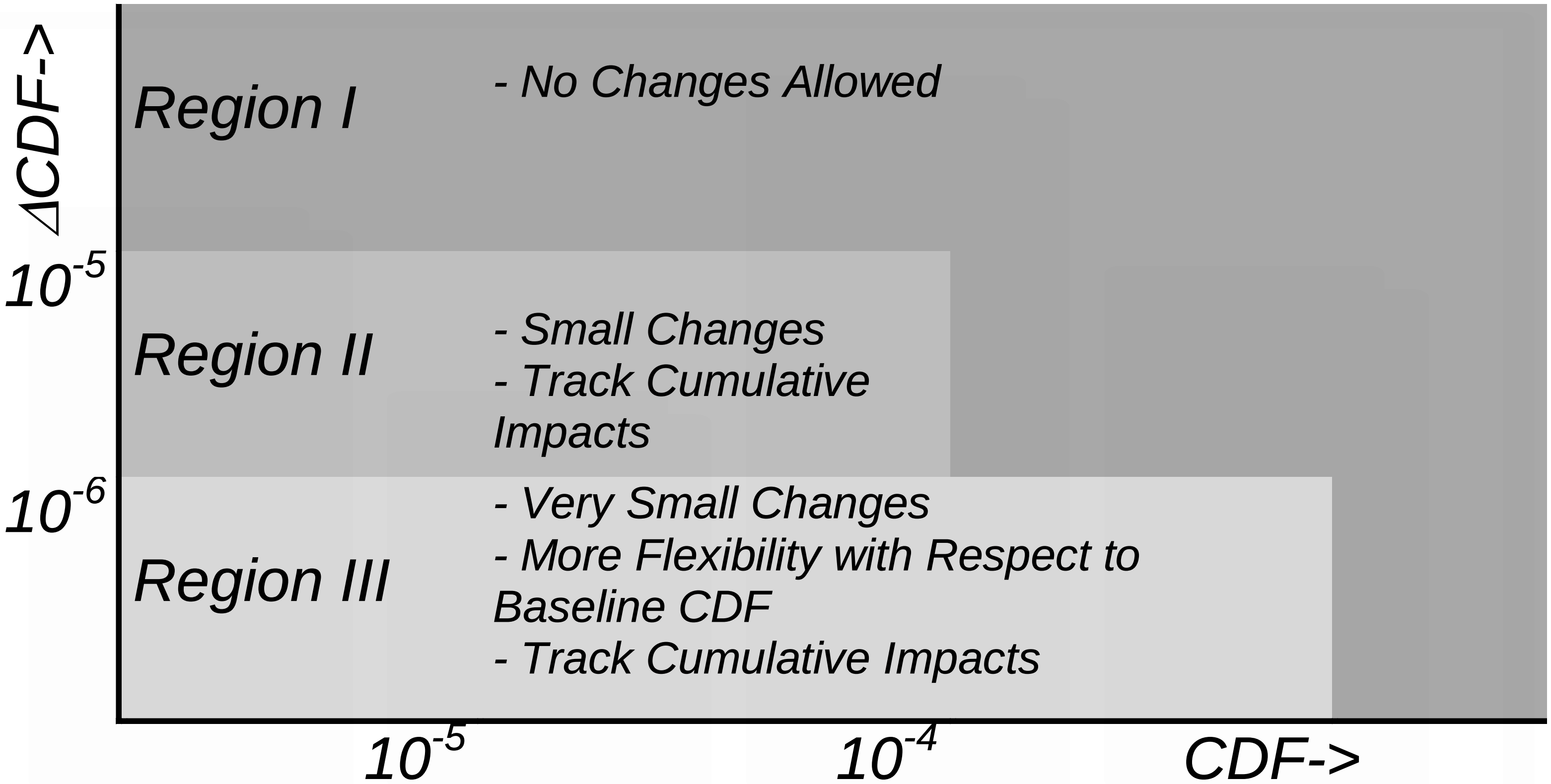
Regulatory Guide 1.174

- An Approach for Using Probabilistic Risk Assessment In Risk-Informed Decisions On Plant-Specific Changes to the Licensing Basis
- Defines the five principles of risk-informed integrated decision-making
  - #4. Proposed increases in CDF or risk are small and consistent with Commission’s Safety Goal Policy Statement
    - Use of CDF and LERF as bases for PRA acceptance guidelines is an acceptable approach to addressing Principle 4.

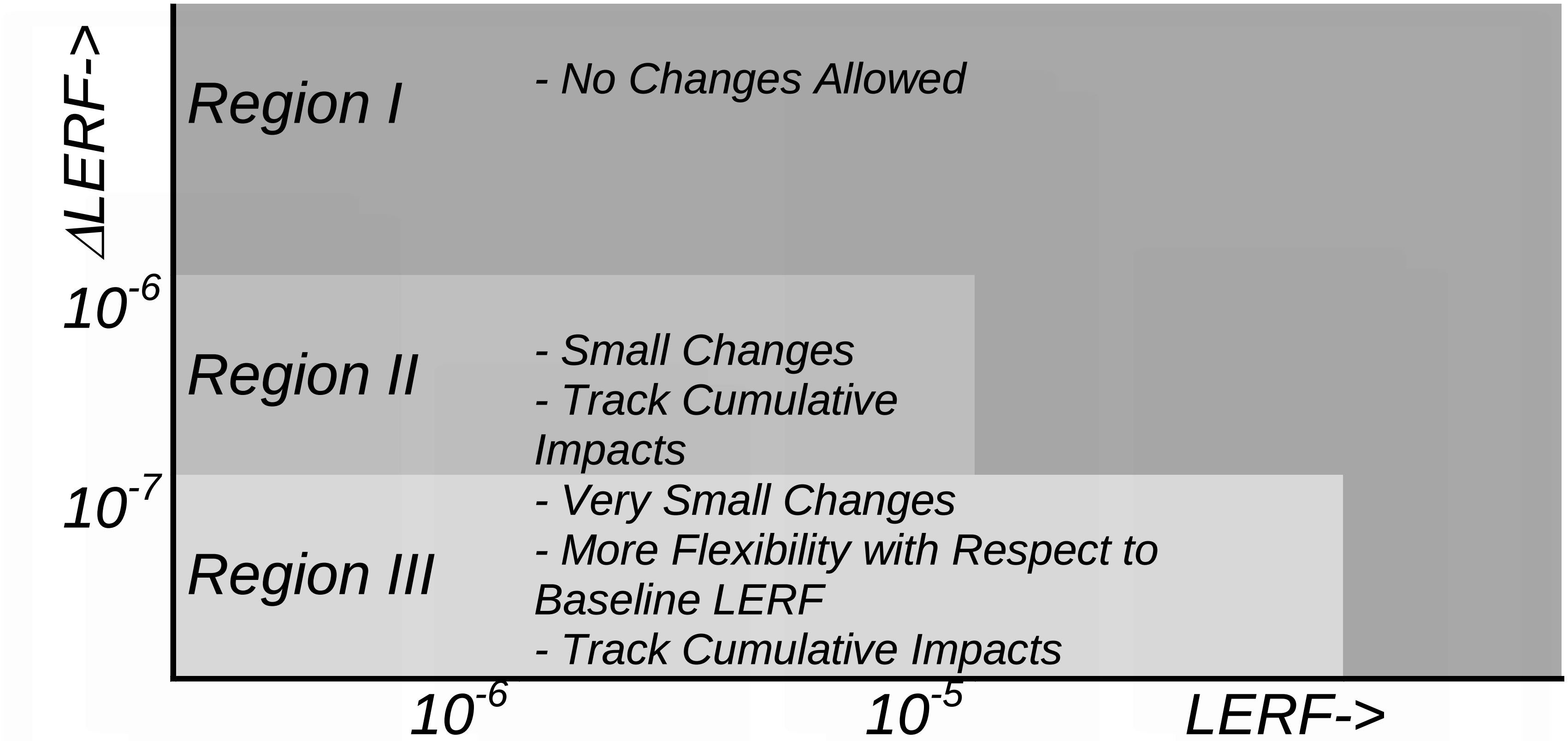
Large Early Release Frequency (LERF)

- In the context of Reg Guide 1.174, LERF is used as a surrogate for the early fatality QHO
- Defined as: the frequency of those accidents leading to significant, unmitigated releases from containment in a time frame prior to effective evacuation of the close-in population such that there is a potential for early health effects
  - No quantitative definition (w.r.t. timing or magnitude)
  - By definition, late releases would result in no early fatalities

RG-1.174 Acceptance Guidelines for Core Damage Frequency



RG-1.174 Acceptance Guidelines for Large Early Release Frequency



Common PRA Terms

- Probability - likelihood of the occurrence of a specific event (unitless)
- Frequency - The occurrence rate of an event (typically expressed in number of events per unit of time)
- Conditional probability - probability of an event given the occurrence of another preceding event upon which the succeeding event has some dependence on
- Core damage - beginning of core degradation, (uncovery of top of active fuel, UTAF – common PWR definition, but not universal)
- Plant Damage State (PDS) - Identifies the status of specified plant systems and functions during a core damage event (typically includes information on containment systems)
- Large early release - significant, unmitigated release from containment in a time frame prior to effective evacuation of the close-in population such that there is a potential for early health effects.

Probabilistic Risk Assessment (PRA) Basic Concepts

- Risk involves both likelihood and consequences of an event
- PRA attempts to answer three specific questions:
  - What can go wrong?
  - How likely is it?
  - What are the consequences?

Risk Can be Defined in Different Ways

- Vector Definition
  - Risk Triplet: Risk = $\{S_i, F_i, C_i\}$,
    - where: S_i = Accident sequence i,
      F_i = Frequency of sequence i,
      C_i = Consequence of sequence i.
- Scalar Definition
  - Risk = $\Sigma_{i=1,n} F_i \times C_i$
  - Sometimes called aggregated risk

Sequence Frequency Quantified by Combining Challenges and Failures

- Initiating events (IE) challenge plant systems to response to upset conditions
- Plant safety systems are barriers between initiating events and core damage
- Sequence frequency combines IE frequency and safety system failure probabilities (reliabilities)

  $CDF = \lambda\varphi$

  where: λ = Initiating event frequency

  φ = Failure probability of safety barriers (systems)

PRAs Characterized as Level-1, Level-2 or Level-3

- Level 1: Core damage risk
  - Quantifies the frequency of accidents that result in core damage
- Level 2: Radioactive material release risk
  - Core damage frequency combined with the conditional probability the containment structure fails to prevent the release
- Level 3: Health consequence risk
  - Combines radioactive material release frequency with the health consequences associated with each release

Full Scope PRA Process/Structure

LEVEL 1

SYSTEMS ANALYSIS

Plant System Models, and Equipment and Operator Failure Data

LEVEL 2

ACCIDENT PROGRESSION ANALYSIS

Models Progression of Severe Accident (APET or CET)

SOURCE TERM ANALYSIS

Parametric Information About Fission Product Transport and Removal

LEVEL 3

CONSEQUENCE ANALYSIS

Demographic and Meteorological Data, and Radiological Consequences (Health Effects and Costs)

RISK INTEGRATION

Combines core damage accident sequence frequency with the consequences associated with that particular accident sequence

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Frequency of accident sequences that result in the uncovering the top of active fuel

Frequency of containment failure and release of radioactive material

Risk (frequency of public consequences) - e.g., fatalities/year, cost-of-accidents/year

Uncertainty is a Vital and Integral Component in Any PRA

- RG-1.174 Section 2.2.5 discusses the importance of considering uncertainty in the decision-making process
  - Cited in proposed modifications to RSGPS
- Accurate representation of uncertainty in Level-2 results requires reflection of Level-1 uncertainties
- Fully integrated uncertainty analysis usually impractical
- Typically, intermediate (Level-1 output) results generated in the form of histograms on PDS frequencies, which serve as input to Level-2 analysis

Session Review

- Why is Level-2 PRA important?
- What are some basic PRA concepts?

Accident Progression Analysis (P-300)

2. Overview of PRA

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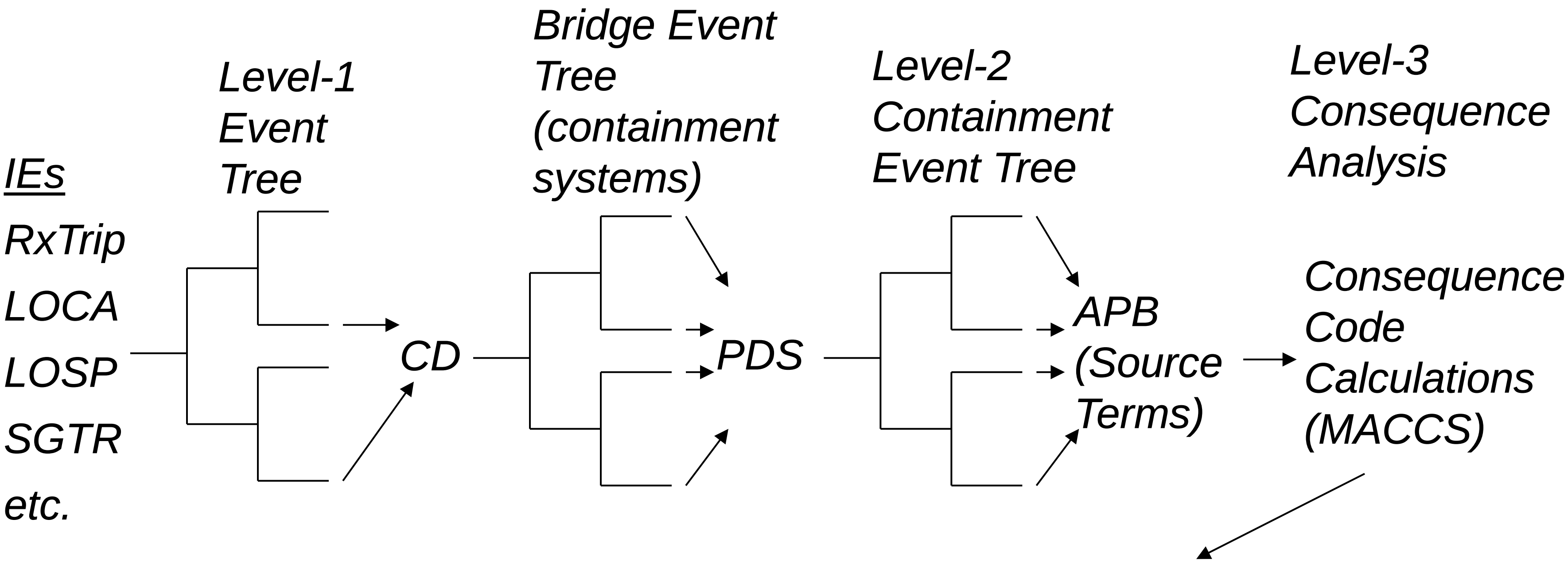
Session Objectives

- To understand the PRA framework
  - Level-1, Level-2 and Level-3 PRA
  - Results of each phase of the PRA

Overview of Level-1/2/3 PRA



CD - Core Damage
PDS - Plant Damage States
APB - Accident Progression Bins

Public Consequence Risk

- *Early Fatalities/year*
- *Latent Cancers/year*
- *Population Dose/year*
- *cost/year*
- *etc.*

Purpose of Level 1 PRA Analysis

- Estimate core damage accident risk (frequency)
  - Typical definition of core damage: Uncovering of top of active fuel
- Total CD risk (or CD frequency) is sum of the frequencies of the different ways core damage can occur
  - Distinctions made among:
    - accidents initiated by site-centered events (internal events analysis) during plant power operations
    - accidents initiating by offsite-centered events (external events)
    - accidents initiated while plant is in a shutdown (non-power producing) state (shutdown/low-power PRA)

Level 1 PRA Analysis Approach

- Potential initiating events identified
- Plant response modeled as a sequence of events (system failures)
  - Accident Sequence = IE combined with set of system failures that leads to undesired consequence (i.e., CD)
- Integrated analysis of plant system reliability
  - Includes consideration of human actions, support system dependencies, common cause failure dependencies
- Core Damage Frequency comprises set of accident sequence frequencies
- Each accident sequence comprises set of accident scenarios (cutsets)

Level-1 PRA (Internal Events Analysis)

IE

Plant Systems and Operator Actions (i.e., plant response to IE)

Typically quantified using fault trees or some other detailed system analysis technique

<u>IEs</u>

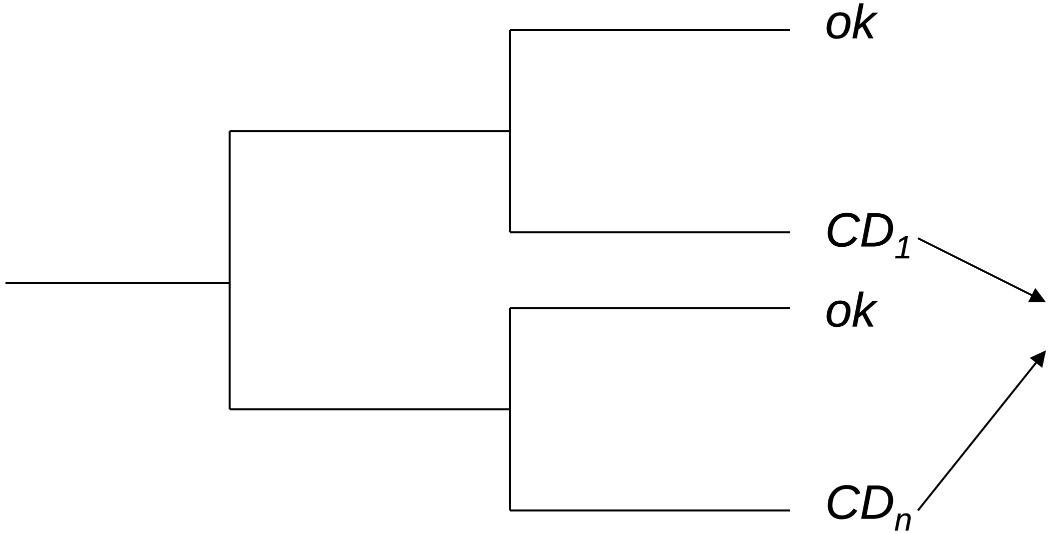
RxTrip

LOCA

LOSP

SGTR

etc.



Total CDF = $\Sigma_{i=1,n}\ CDF_i$

Purpose of Level-2 PRA Analysis

- Extend the severe accident analysis beyond the occurrence of core damage
  - Core damage accident sequences vary in timing and severity
- Issues addressed in Level-2 include:
  - Does fuel damage actually occur? (Remember, Level-1 only analyzes up to the point where CD nominally starts)
  - Does accident progress to RPV failure, and how?
  - How does the containment respond?
  - Is radioactive material released into the environment?

Level-2 PRA Analysis Approach

- Characterize challenges to containment resulting from various core damage sequences
  - e.g., core degradation produces H2, which can burn
- Estimate strength of containment
- Identify probable containment failure mode (e.g., failure due to hydrogen detonation or steam explosion, melt through, leakage)
- Describe radioactive source term released into the environment
  - Including the energy associated with containment failure and radioactive material release

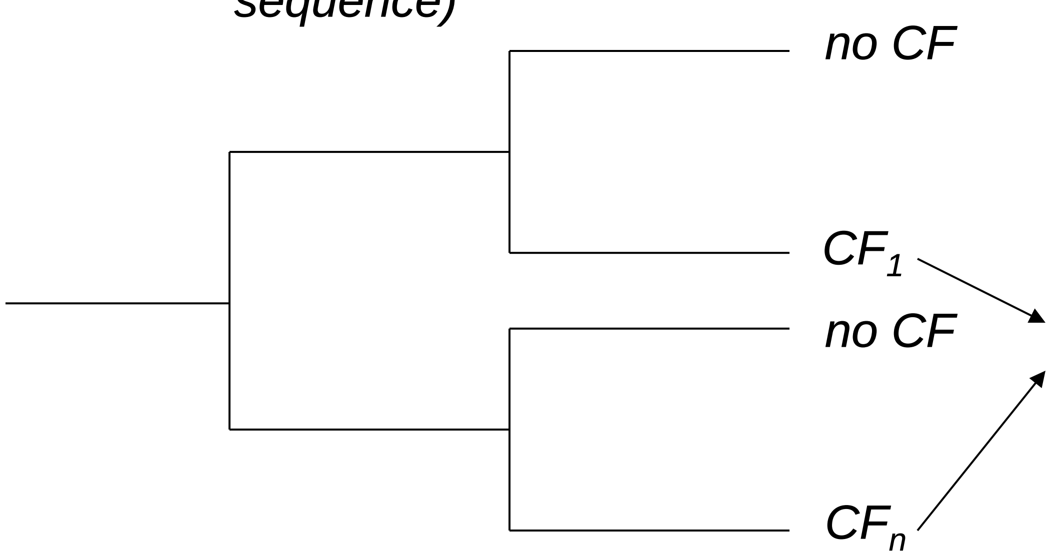
Level-2 PRA (Containment Event Tree)

CD

Containment Systems and physical phenomena (i.e., containment response to core damage sequence)

Typically quantified using fault trees (for cont. systems), and detailed code analyses and experimental results (for physical phenomena)

Core Damage (Plant Damage State)



Total CF = $\Sigma_{i=1,n} CF_i$

Note that this example focuses on Containment Failure (CF), some Level-2 analyses estimate releases (i.e. source terms) or Large Early Release Frequency (LERF)

Purpose of Level-3 PRA Analysis

- Estimate the public consequences (mostly health) of a severe accident
  - Person-rem (individual and population), early fatalities, latent cancers, financial cost, etc.
- Site-specific calculation
  - Considers local demographics, weather, emergency plan

Level 3 PRA Analysis Approach

- Source term information from Level-2 analysis result used as input to Level-3 consequence analysis
- Source term information includes:
  - radionuclide composition, energy associated with release, timing and duration of release, etc.
- Source term transport and offsite consequences (both health and economic) modeled using consequence code
  - MACCS2 (1998)
  - MACCS (1987 - NUREG-1150)
  - CRAC2 (1982)
  - CRAC (1975 - WASH-1400)

Level-3 Analysis Combines Source Term Frequencies and Consequences

Source Terms
(for each STG)
Demographics
Weather data

MACCS Code → *Public Consequences for each Source Term Group*

Frequency of each Source Term Group (from Level-2) → *Risk Integration*

Public Risk (both health and financial)

Level 1/2/3 PRA Integration Issues

- Level 1 Accident sequence analysis quantifies core damage frequency
  - However, not all CD accident sequences are equal (with respect to potential consequences)
- Containment analysis (Level 2) and consequence analysis (Level 3) usually performed “separate” from CDF analysis
  - Different areas of expertise, therefore different analysts
  - Because of size and complexity of Level 1/2/3 PRA, difficult to fully integrate analysis, therefore usually performed in pieces or steps
- Special methods used to link accident sequence analysis to containment analysis

Level-1 Result (CDF) Not Sufficient for Level-2 Analysis

- Specific details on core damage sequence are needed to model containment response to the severe accident
- Typical Level-1 PRA produces 10,000’s of core damage sequences, each of which can comprise 100’s of individual scenarios (cut sets)
- Containment systems usually do not impact CDF, therefore often not included in Level-1 systems analyses
  - Containment systems analysis must be integrated with Level-1 analysis (need to account for dependencies)

Dependencies Often Dominate Risk

- Multiple system failures required for radioactive release to environment
- Failure of multiple systems caused by independent mechanism very incredible probability
- Only by failing multiple barriers (systems) by the same mechanism will the likelihood of the sequence be significant
- Level-2 analysis must account for dependencies between the Level-1 and Level-2 models
- Probabilistic definition of dependency:
  - $P(a|b) \neq P(a)$

Systems Analyses Needs to Include Containment Systems

- Dependencies between Level-1 modeled systems and containment systems must be considered
  - Support system dependencies
  - Shared equipment dependencies
  - Human action dependencies
  - Common cause failure dependencies
- Inclusion of containment systems can be accomplished two ways
  - Expand Level-1 event trees
  - Bridge trees

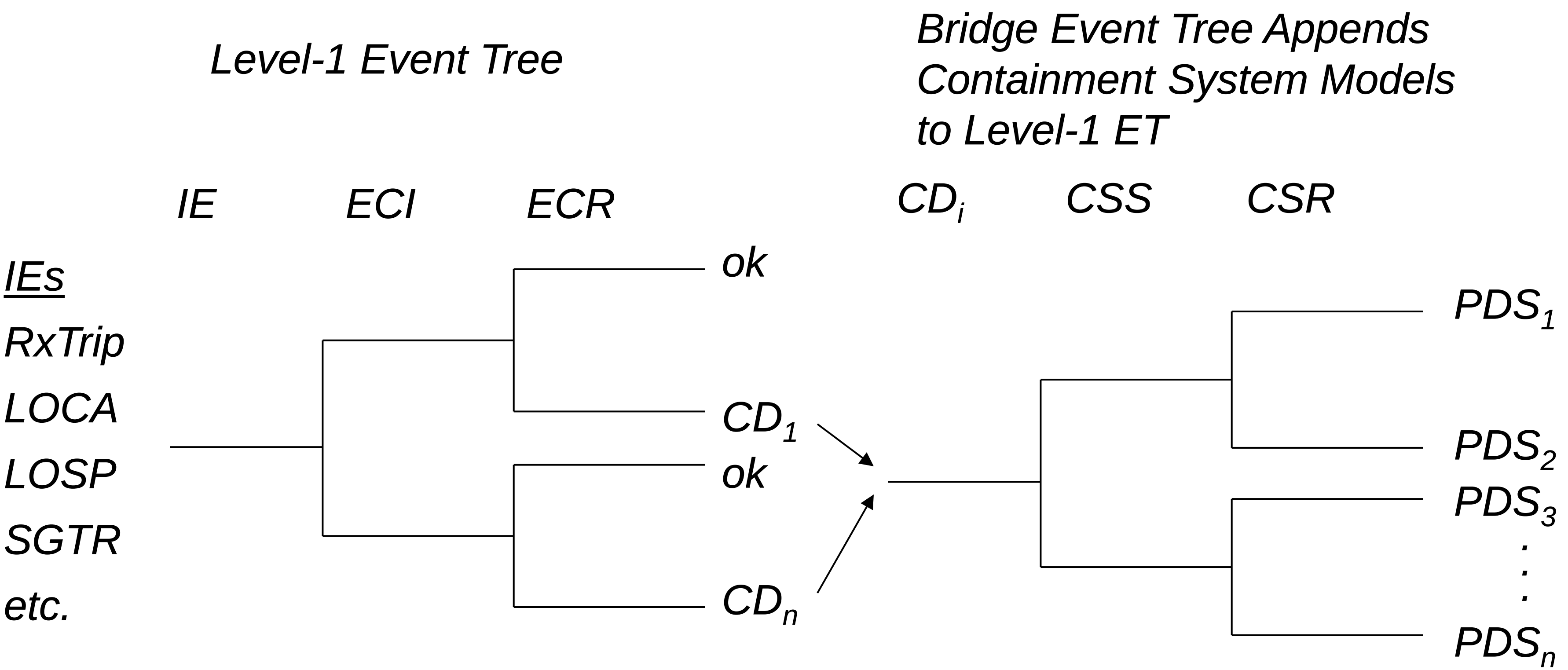
Bridge Event Trees

- Additional system models and analyses needed before containment analysis can be performed
  - “Core Damage” result from Level-1 is not adequate for starting containment analysis
  - Some containment systems not relevant to CDF are important for containment response
  - Containment system models need to be integrated with Level 1 system analysis (i.e., need to account for dependencies)
  - Bridge Event Tree (BET) used to model additional systems/phenomena, linked to Level 1 event trees

Plant Damage States (PDS) Framework Used As Input to Level-2 (from Level-1)

- Output (end states) of BET defined in terms of specific details about CD accident sequence
- Method utilizes a vector identifier
  - Each character position of the vector identifies the status of a particular system or event
    - e.g., ACCBABDC
  - Vector is “read” by the Level 2 analysis

Expanded Systems Analysis Needed to Support Level-2 Model



Each Plant Damage State Represents a Unique Plant Response/Condition

- Direct link between expanded Level-1 sequence analysis and Level-2 models usually not feasible
- Process includes collapsing the sometimes millions of Level-1 sequences into a manageable number of PDS
  - Often referred to as “binning”
- Each unique PDS vector serves as an initiating event for Level-2 analysis
- PDS vector transmits necessary information from Level-1 to Level-2 analyses

Example Plant Damage State (PDS) Vector

Character	PWR	BWR
1	Status of RCS at onset of core damage	Status of RPS
2	Status of ECCS	Status of electric power
3	Status of containment heat removal	RPV integrity
4	Status of electric power	RPV pressure
5	Status of contents of RWST	Status of HPI
6	Status of heat removal from S/Gs	Status of LPI
7	Status of cooling for RCP seals	Status of containment heat removal
8	Status of containment fan coolers	Status of containment venting
9		Level of pre-existing leakage from containment
10		Time to core damage

Example PDS Scheme - Grand Gulf (NUREG-1150)

Character #	*Description*
1	*Initiating event*
2	*Reactor vessel pressure*
3	*Status of both high and low pressure injection*
4	*Status of containment spray and suppression pool cooling*
5	*Status of containment and containment systems as start of core damage*
6	*Time of core damage (early or late)*

PDS Scheme from NUREG-1150 (Grand Gulf)

#	ID	Description
1	B1	Station blackout (SBO) transient has occurred. Offsite power is not recoverable because there is no emergency DC power.
	B2	SBO transient has occurred. Offsite power is recoverable.
	T2	Loss of PCS transient has occurred. Offsite or onsite power is available.
	TC	ATWS has occurred. Offsite or onsite power is available.
2	P1	The reactor vessel (RV) is at high pressure (HP) at the onset of core damage (CD) and depressurization is not possible.
	P2	The RV is at HP at the onset of CD because the operator failed to depressurize; depressurization is possible.
	P3	The RV could be at HP at the onset of CD. The operator depressurizing the vessel (which is possible) was not included in the model.
	P4	The RV is at low pressure (LP)

PDS Scheme from NUREG-1150 (Grand Gulf) - cont.

#	ID	Description
3	I1	Injection to the RV is not available after the onset of CD.
	I2	Injection with the Firewater system is available before and after the onset of CD.
	I3	Injection with the Condensate system is recoverable with the restoration of offsite power.
	I4	Injection with the LP systems [core spray (LPCS) and coolant injection (LPCI)] is recoverable with the restoration of offsite power (or RV depressurization).
	I5	Injection with both the HP and LP systems is recoverable with the restoration of offsite power.
	I6	Injection with the HP systems (reactor core isolation cooling and control rod drive) and LP systems (LPCS and LPCI) is recoverable with the restoration of offsite power (or RV depressurization).
4	H1	Containment Spray (CS) is not available at the onset of CD, neither is it recoverable.
	H2	At least on train of CS is recoverable with the restoration of offsite power
	H3	At least one train of CS is available at the onset of CD.
5	M1	Miscellaneous systems (Venting, SBGT, CI, H2I) are not available at the onset of CD.
	M2	Miscellaneous systems (Venting, SBGT, CI, H2I) are recoverable with the restoration of offsite power.
	M3	Miscellaneous systems (Venting, SBGT, CI, H2I) are available at the onset of CD.
6	ST	CD occurs in the short term (at ~1 hour).
	LT	CD occurs in the long term (at >12 hours).

List of PDS from NUREG-1150 (Grand Gulf)

PDS	PDS Character Vector	Accident Sequence
PDS-1	B2-P3-I5-H2-M2-ST	T1B-16 T1B-17 T1B-21
PDS-2	B2-P3-I5-H1-M2-ST	T1B-16 T1B-17 T1B-21
PDS-3	B2-P3-I3-H1-M2-ST	T1B-16 T1B-17 T1B-21
PDS-4	B2-P4-I5-H2-M2-LT	T1B-14
PDS-5	B2-P4-I5-H1-M2-LT	T1B-14
PDS-6	B2-P4-I2-H1-M2-LT	T1B-14
PDS-7	B1-P1-I1-H1-M1-ST	T1B-16 T1B-17 T1B-21
PDS-8	B1-P1-I1-H1-M1-LT	T1B-13
PDS-9	TC-P2-I6-H3-M3-ST	TC-74
PDS-10	TC-P2-I4-H3-M3-LT	TC-74
PDS-11	T2-P2-I5-H3-M3-ST	T2-56
PDS-12	T2-P2-I5-H3-M3-LT	T2-56

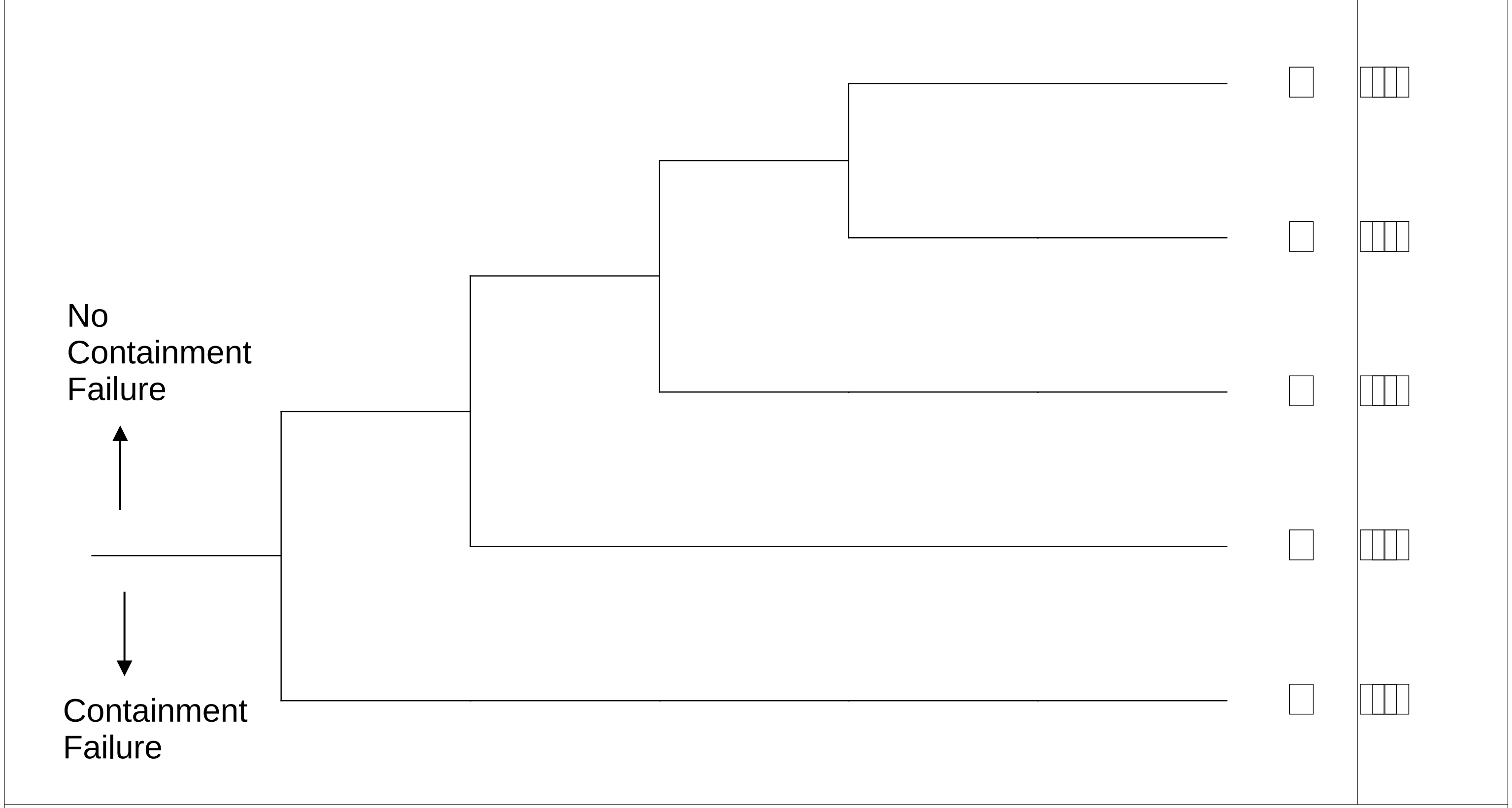
Level-2 Analysis Assesses Containment Response to Each PDS

- Each PDS represents a unique (by design) challenge to containment integrity
- Containment strength (actual, not design) estimated through a detailed engineering evaluation
- Challenge presented by PDS compared to estimated pressure capacity of containment
- Conditional probability of containment failure then calculated
- CET (or APET) provides the framework for this analysis

Two General Techniques for Level-2 Modeling

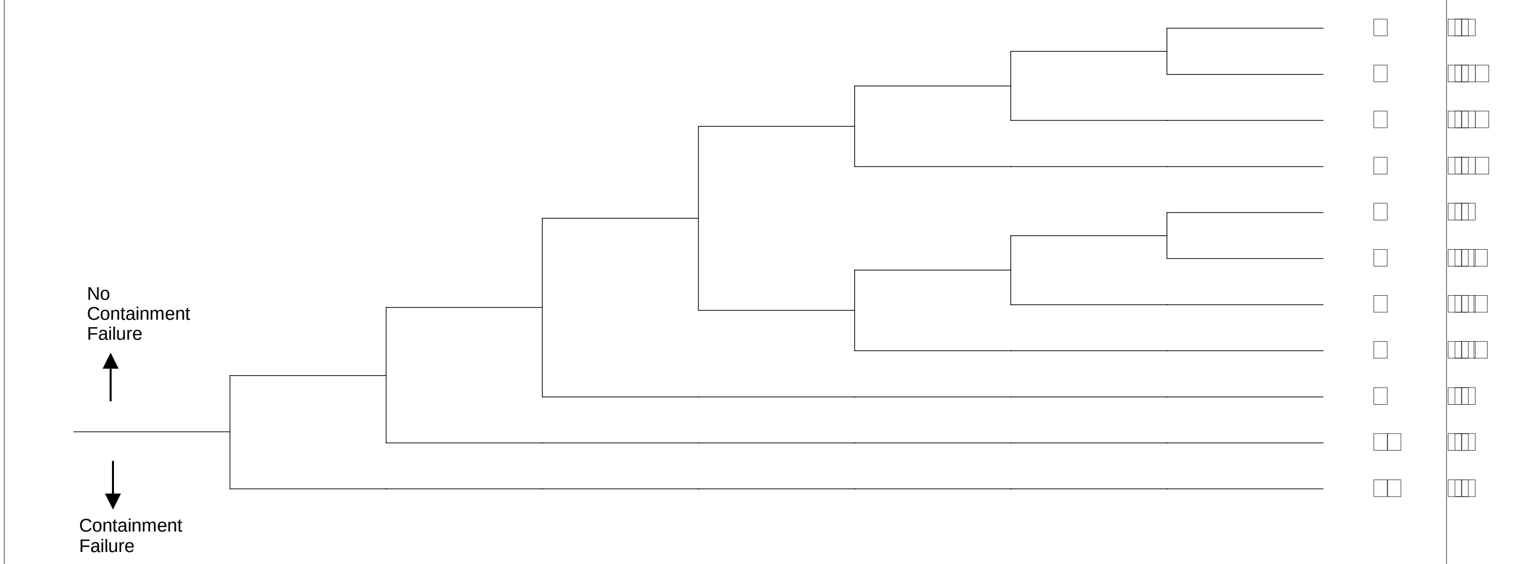
- Containment Event Trees (CETs)
  - Typically displayed in graphical form
  - Comprising 8-15 top events (major summary events with underlying detailed models)
  - Original example: WASH-1400
- Accident Progression Event Trees (APETs)
  - No graphical representation
  - All details explicitly modeled
    - 75-125 top events, many with multiple (more than 2) branches
  - example: NUREG-1150
- Terms often used interchangeably

Core Damage Sequence	Containment Rupture due to a Reactor Vessel Steam Explosion	Containment Leakage	Containment Rupture due to Hydrogen Burning	Containment Rupture by Overpress-urization	Containment Rupture by Meltthrough		
CD	CRVSE	CL	CR-B	CR-OP	CR-MT	#	CF-Mode



WASH-1400-PWR-CET - PWR Containment Event Tree (WASH-1400) 2009/02/09 Page 7

Core Damage Sequence	Containment failure due to a steam explosion in the reactor vessel	Containment failure due to a steam explosion in containment	Containment failure by overpressure	Containment isolation failure in drywell vs wetwell	Containment leakage greater than 2400% per day	Secondary Containment Failure	Standby Gas Treatment System failure		
CD	VSE	CSE	OP	DW_VS_WW	LCL	SCF	SGTS	#	CF-Mode



WASH-1400-BWR-CET - BWR Containment Event Tree (WASH-1400) 2009/02/09 Page 8

Zion APET from NUREG-1150

- Zion - PWR with large dry containment
- APET comprises 72 top events questions (most with multiple branches)
  - 10 determined by Plant Damage State (from Level-1)
  - 5 determined by systems or data analyses
  - 14 determined by expert elicitation
  - 19 determined from severe accident research
  - 21 summary question (i.e., determined by answers to previous questions in the APET)
  - 3 determined through internal calculations

Zion APET - Example Questions

- Size/location of RCS break when the core uncovers?
- Initial containment leak or isolation failure?
- Temperature-induced hot leg or surge line break?
- Vessel pressure just before vessel breach?
- Amount of Zr oxidized in-vessel during core degradation?
- Adding H2 produced by core concrete interaction to H2 already in containment.

CET/APET Outputs Source Term

- Containment failure details
  - Size of containment failure
  - Timing of failure
  - Energy associated with failure
- In-containment transport of radioactive material also modeled in CET/APET
  - Quality and quantity of radioactive material escaping containment

Level-3 Analysis Estimates Health Consequences for Each Release Event

- Output of Level-2 analysis (i.e., details of the radioactive material source term release) provide one input to the Level-3 analysis
- Each source term combined with site-specific information on demographics, weather, emergency planning, etc. to calculate health and economic consequences to the surrounding population
- MACCS code used to perform consequence calculations

MACCS2 Code Features

- Atmospheric transport and deposition under time-variant meteorology
- Short- and long-term mitigative actions and exposure pathways
  - evacuation, sheltering and relocation of people
  - interdiction of milk and crops
  - decontamination or interdiction of land and buildings
- Deterministic and stochastic health effects, and economic costs
  - Includes Direct (cloudshine, inhalation, groundshine, and skin deposition) and indirect (ingestion) radiation dose pathway

MACCS2 Available Since 1998

- Improvements over MACCS include:
  - More flexible emergency-response model
  - Expanded library of radionuclides
  - Semidynamic food-chain model
  - Improved phenomenological modeling
  - New output options

Typical Consequence Measures

- From NUREG-1150 (MACCS)
  - Early fatalities
  - Total latent cancer fatalities
  - Population dose within 50 miles
  - Population dose within entire region
  - Individual early fatality risk within 1 mile (used for QHO comparison)
  - Individual latent cancer fatality risk within 10 miles (used for QHO comparison)

Session Review

- PRA structure and outputs
  - Level-1 PRA
  - Level-2 PRA
  - Level-3 PRA

Accident Progression Analysis (P-300)

3. LWR Containment Designs

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Session Objectives

- To understand the various LWR containment designs
  - Features important to severe accident response

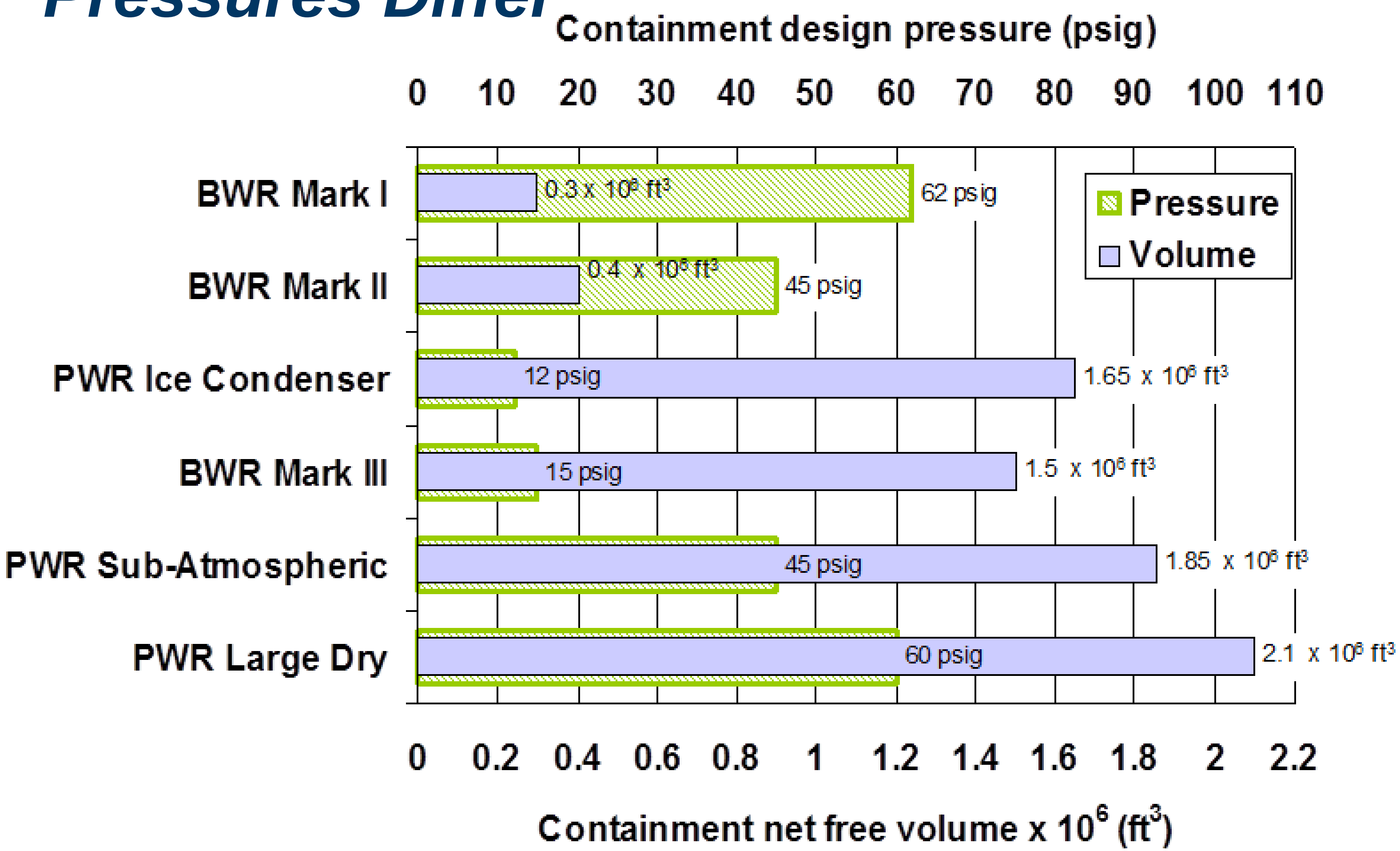
Seven Major Types of LWR Containment Designs

- Boiling Water Reactors (BWRs)
  - Mark I (e.g., Peach Bottom 2 & 3, Cooper and Fukushima Daiichi 1-5)
  - Mark II (e.g., Limerick 1 & 2, Columbia)
  - Mark III (e.g., Clinton, Grand Gulf)
- Pressurized Water Reactors (PWRs)
  - Large Dry (e.g., ANO 1 & 2, Indian Point 2 & 3)
  - Subatmospheric (e.g., Surry 1 & 2, Millstone 3)
    - Subatmospheric usually grouped with Large Dry
  - Ice Condensers (e.g., Sequoyah 1 & 2, D. C. Cook 1 & 2)
  - AP1000 (e.g. Vogtle 3 & 4)
- Design variations within each group

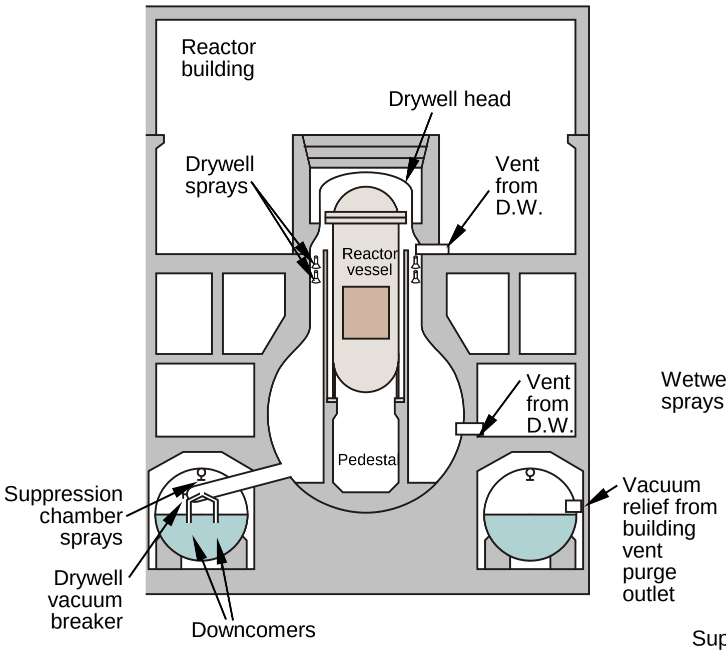
Significantly Larger Number of Dry Containments

Containment Type	*Number*
Large dry *- ANO 1 & 2, Indian Point 2 & 3*	*58*
Subatmospheric *- Surry 1 & 2, Millstone 3*	*7*
Ice Condenser *- Sequoyah 1 & 2, D.C. Cook 1 & 2*	*9*
Mark I *- Peach Bottom 2 & 3, Cooper*	*24*
Mark II *- Limerick 1 & 2, Columbia*	*8*
Mark III *- Clinton, Grand Gulf*	*4*

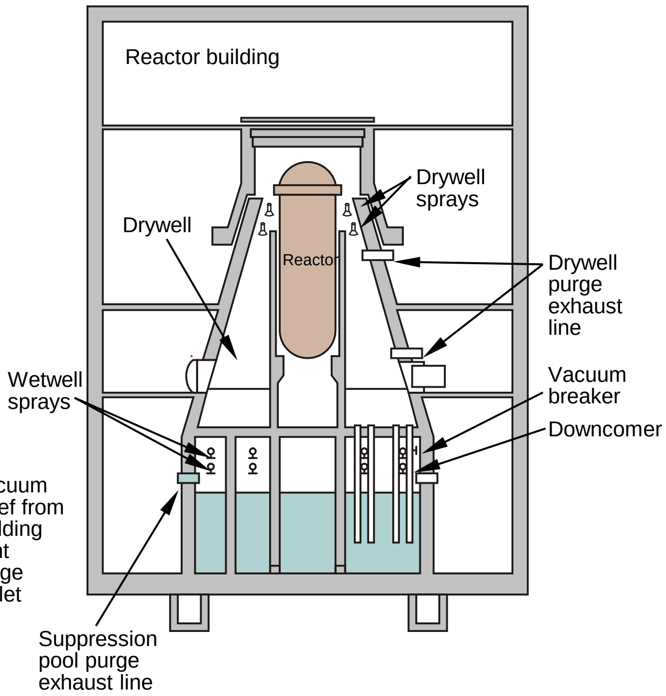
Containment Free Volumes and Design Pressures Differ



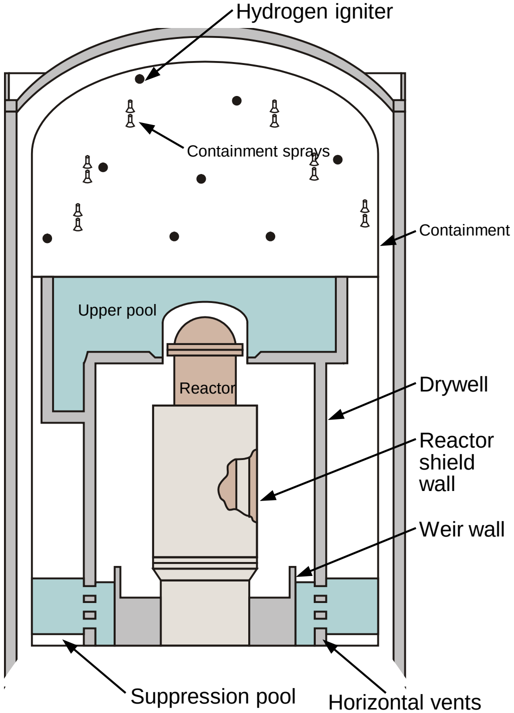
BWR Containment Designs Differ



Mark I

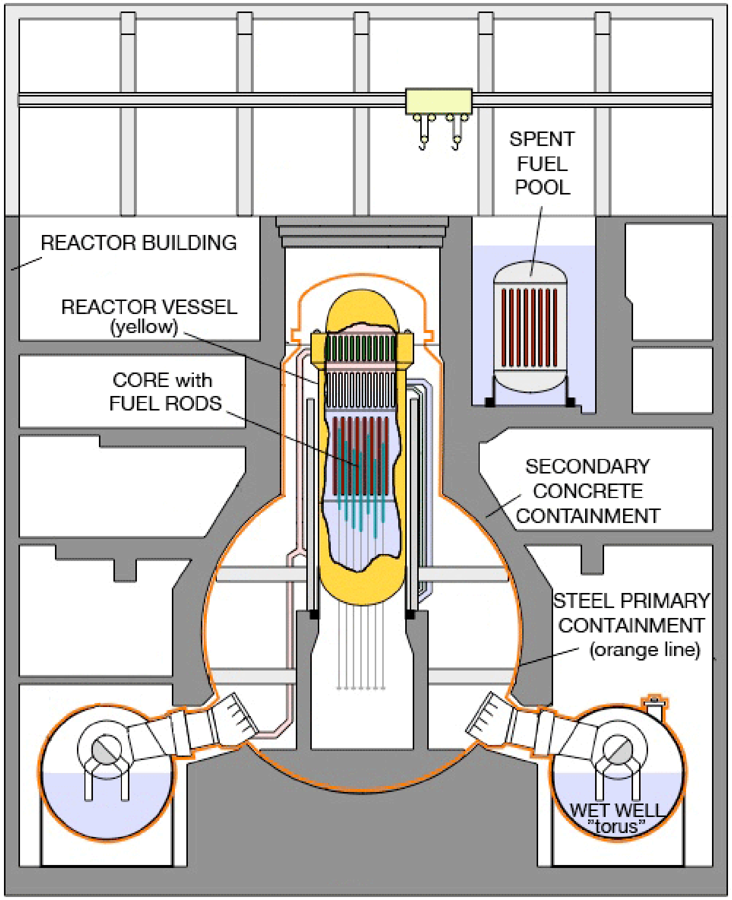


Mark II



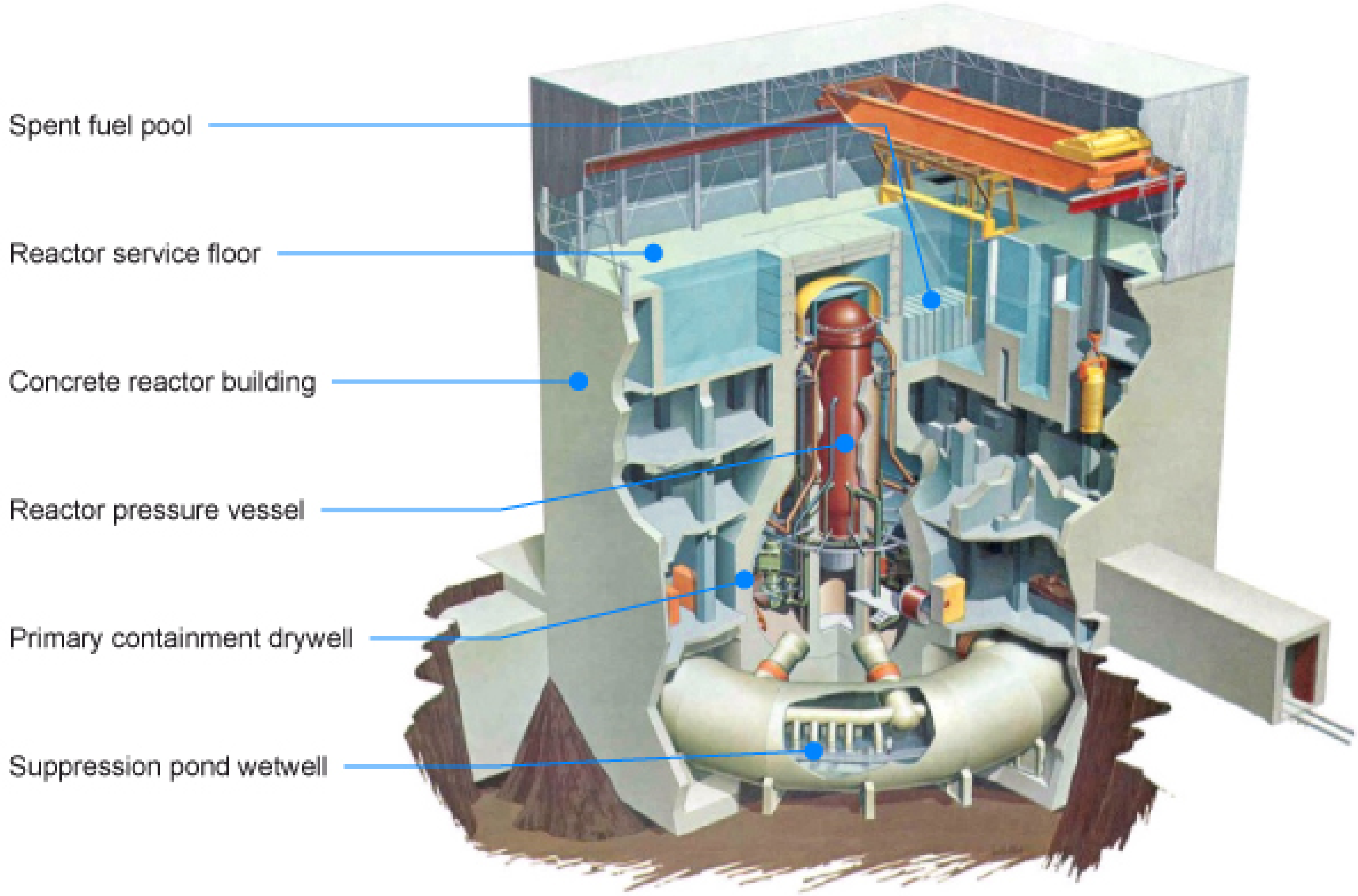
Mark III

Mark I Design Used in Older BWRs

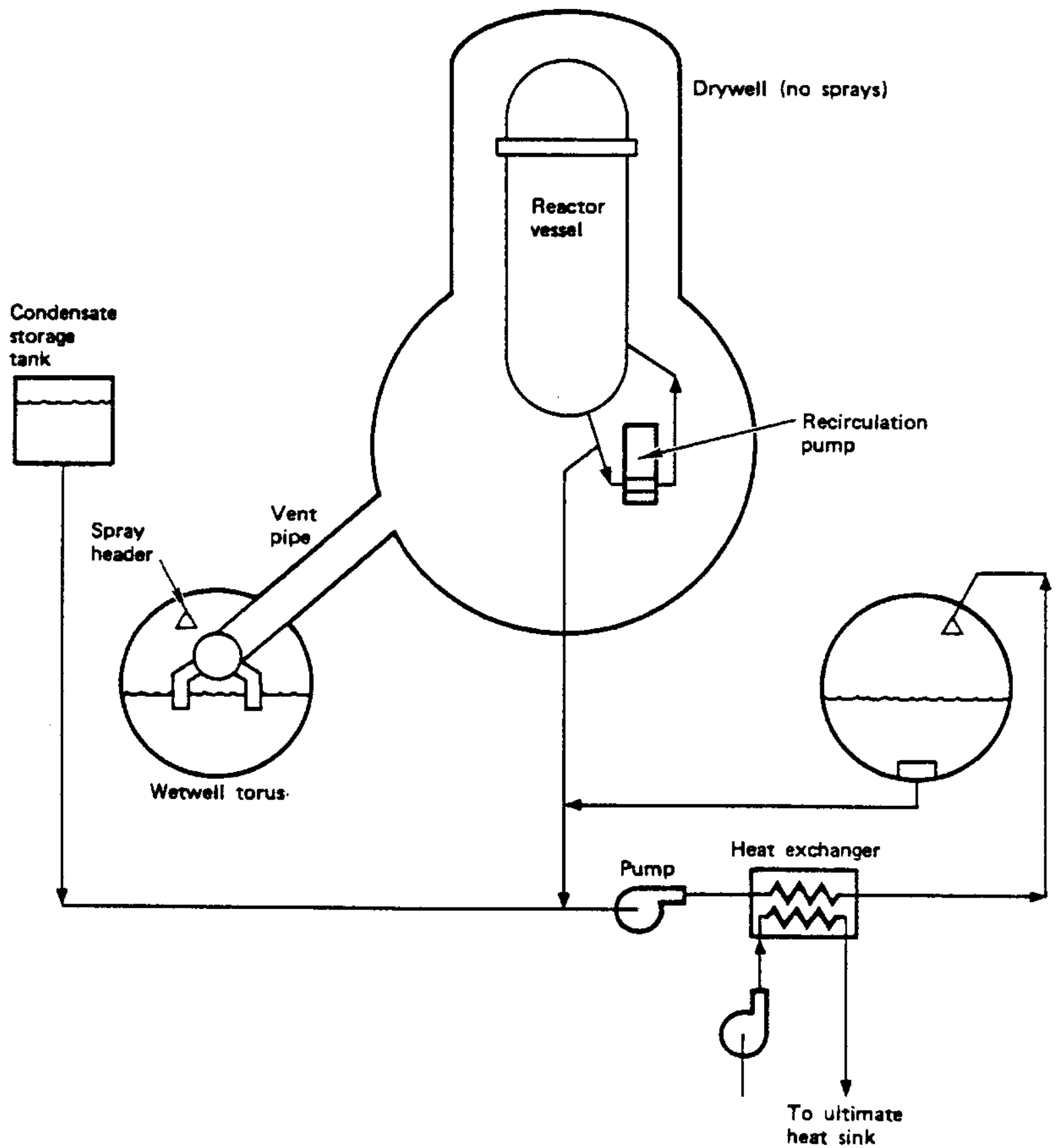


- Two structures/volumes connected by large diameter pipes
  - Drywell: reactor vessel and primary system
  - Wetwell: torus containing large volume of water used for pressure suppression and heat sink
- Containment atmosphere inerted to prevent hydrogen (H2) combustion

Mark I cutaway



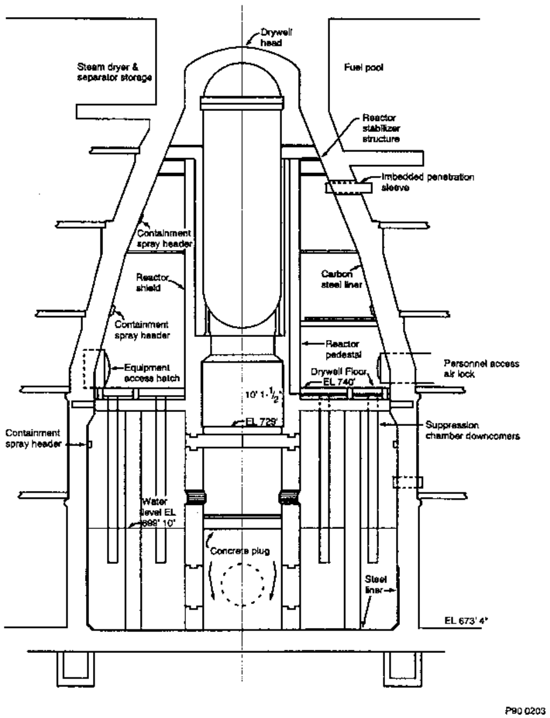
Mark I Containment Heat Removal Relies Primarily on Suppression Pool Water

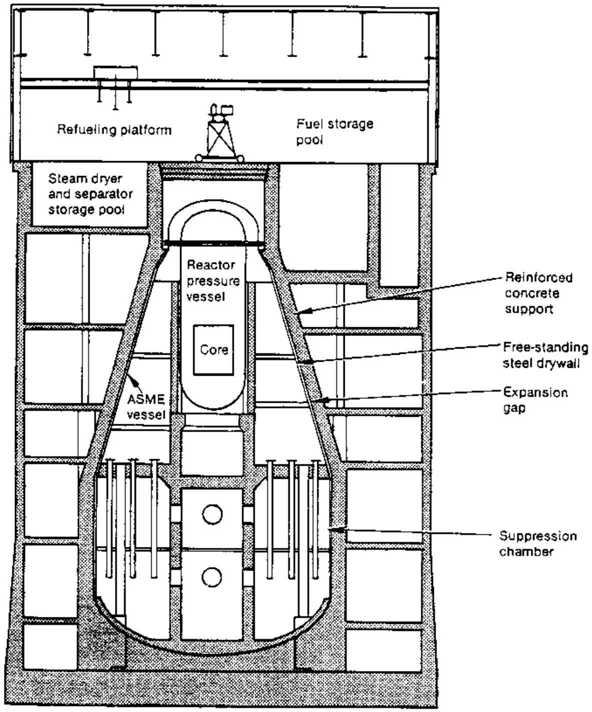


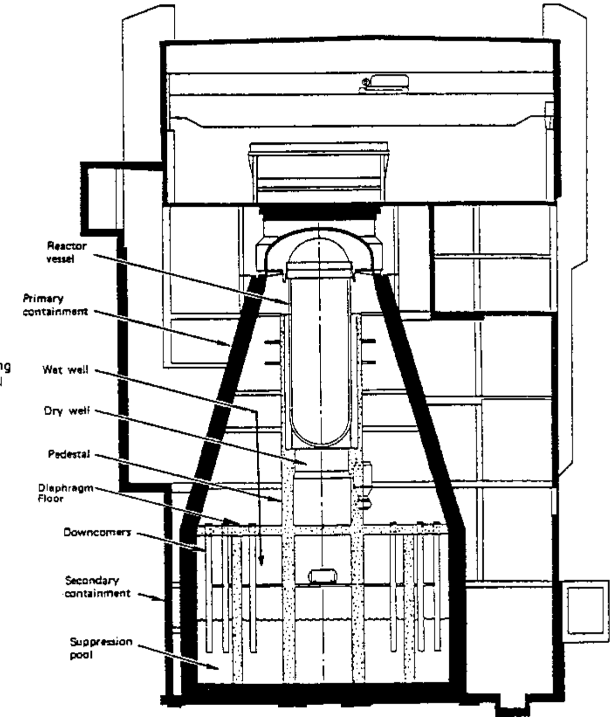
Mark II Design More Unified than Mark I Design

- Single structure divided into two volumes by concrete floor
  - Drywell is directly above wetwell
  - Drywell and wetwell connected by vertical pipes
- Reinforced or post-tensioned concrete structures with steel liner (Columbia is exception - free-standing steel)
- Containment atmosphere inerted to prevent H2 combustion

Mark II Design More Unified than Mark I Design (continued)

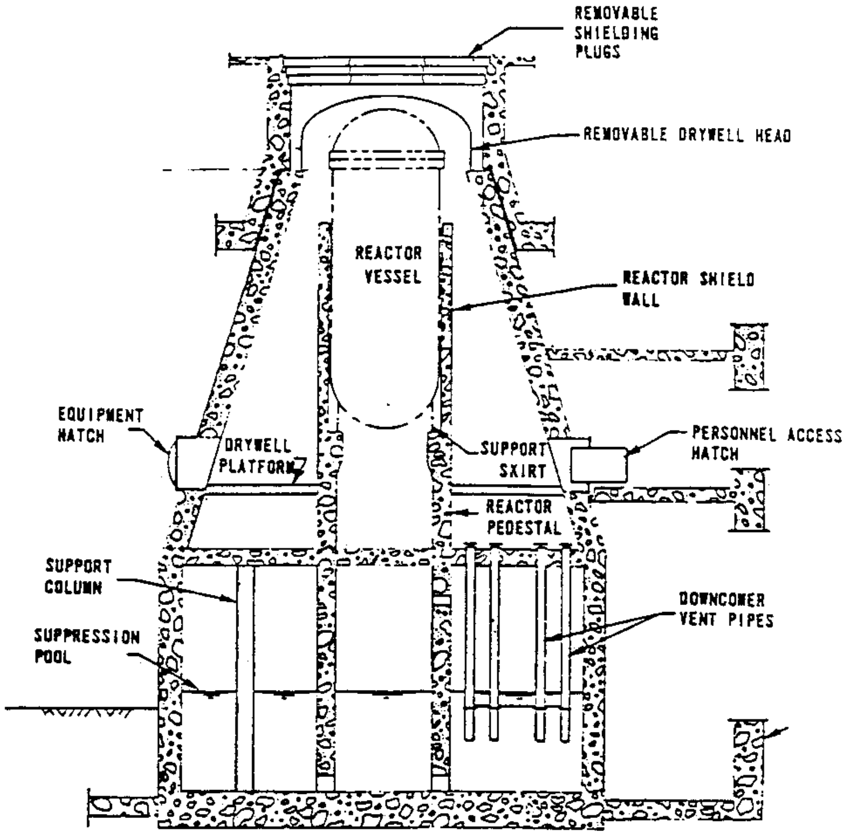




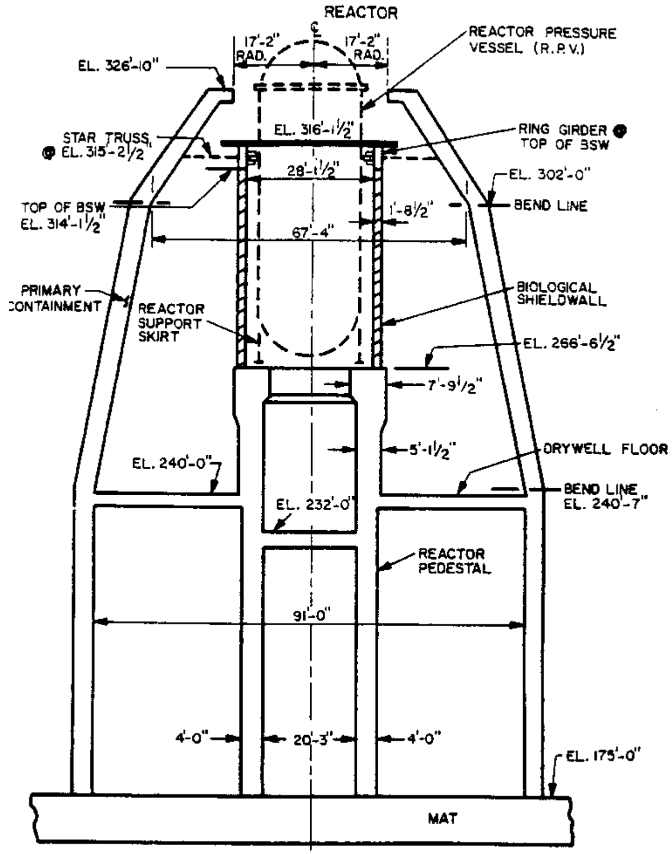


LaSalle Units 1 & 2 **Columbia (WNP-2)** **Limerick 1 & 2**

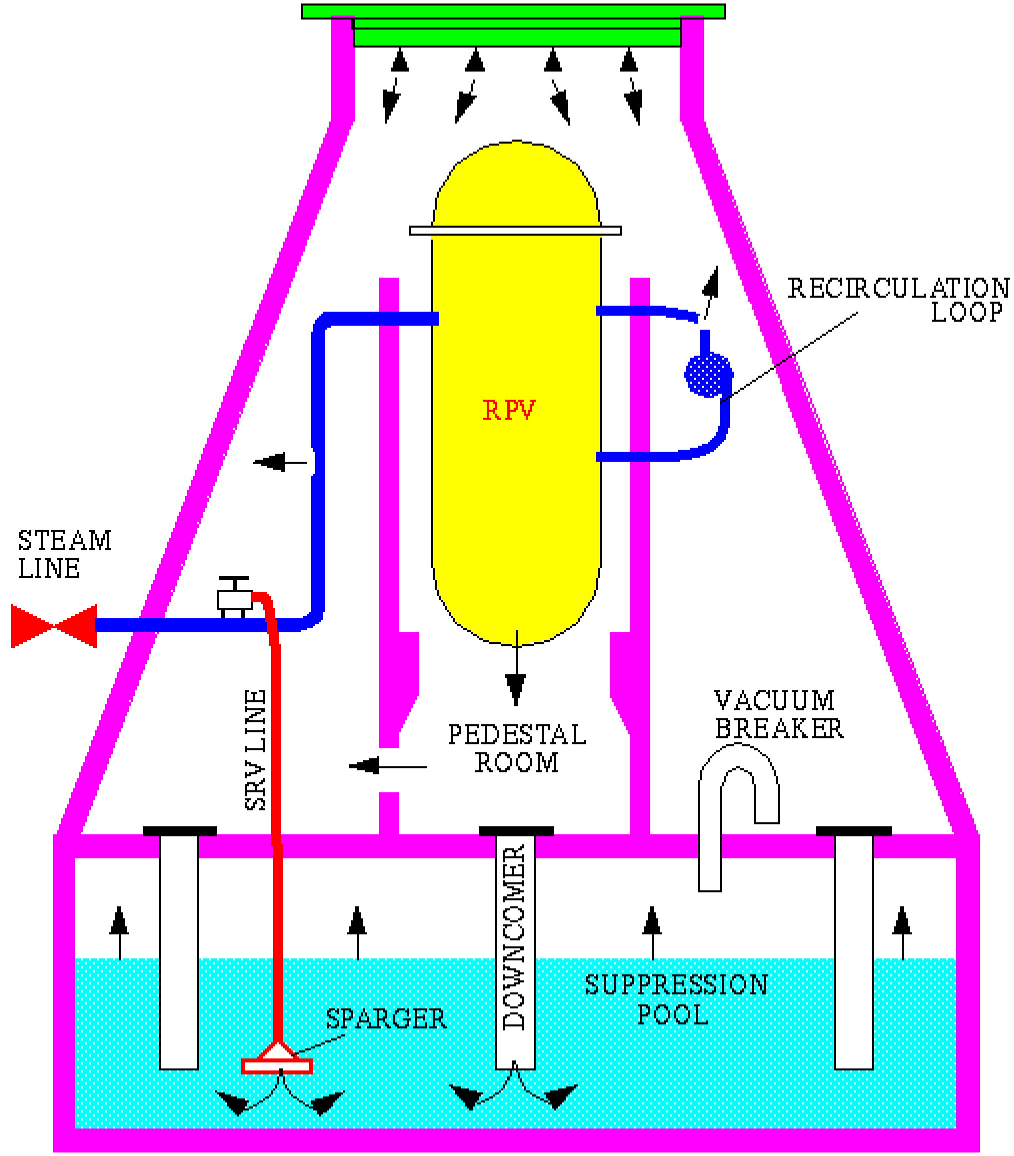
Mark II Design More Unified than Mark I Design (continued)



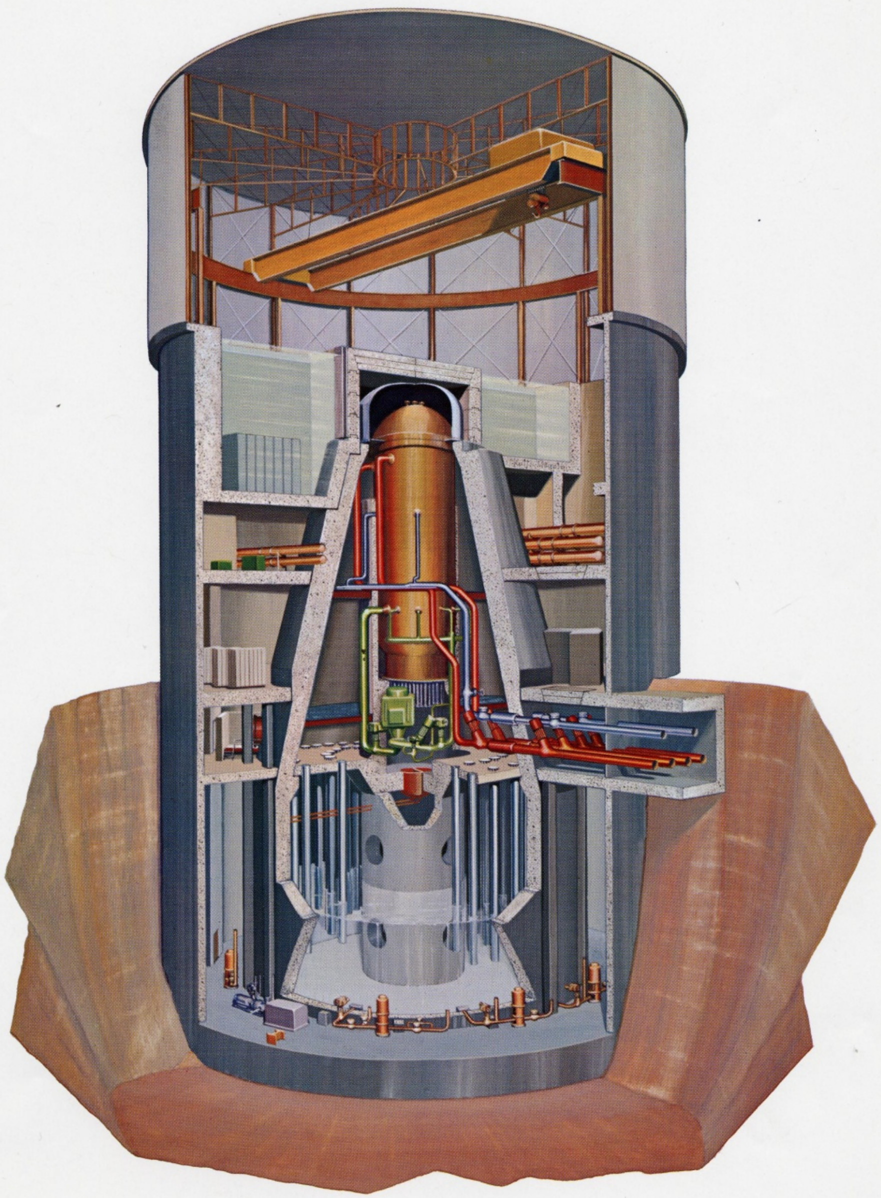
Susquehanna Units 1 & 2



Nine Mile Point 2

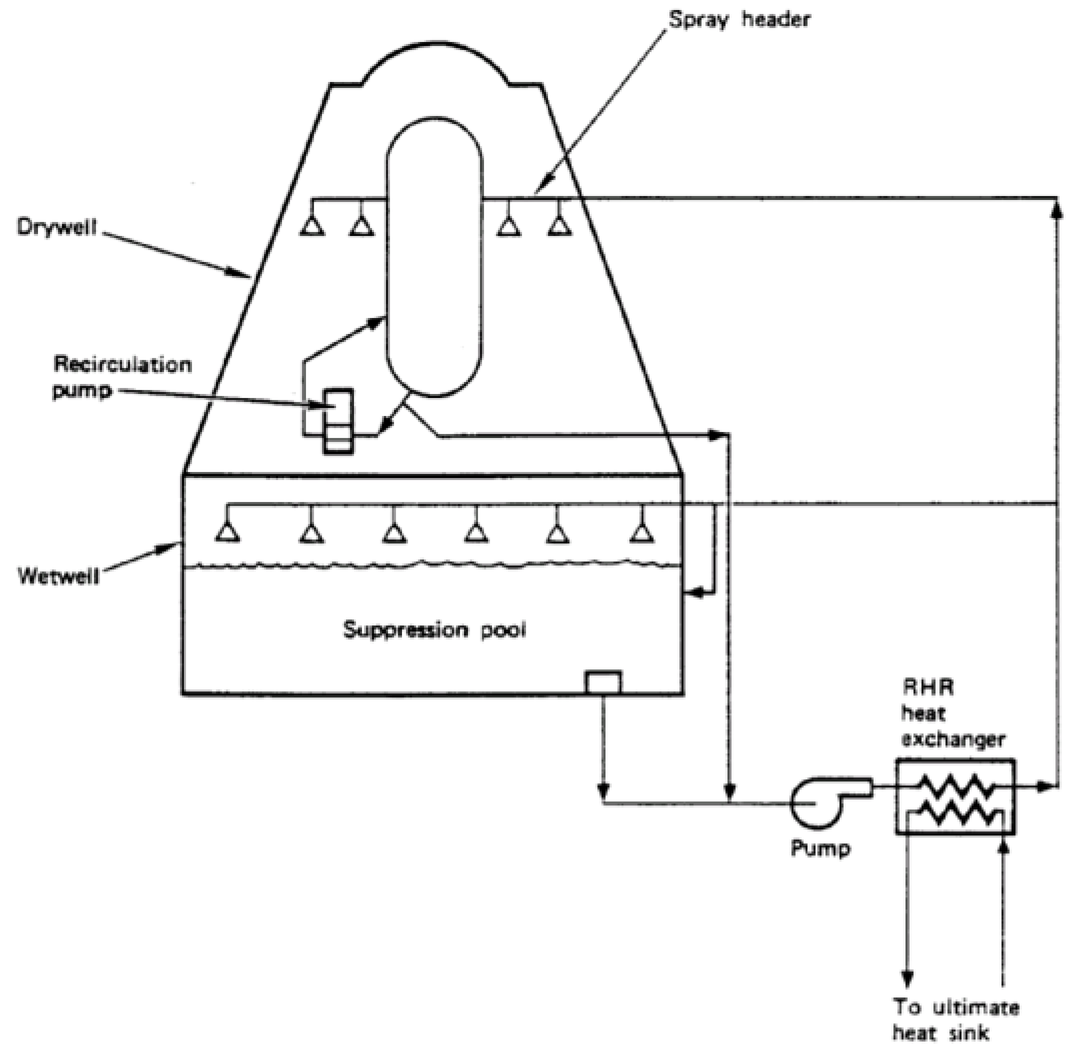

RECIRCULATION
LOOP
RPV
STEAM
LINE
SRV LINE
PEDESTAL
ROOM
VACUUM
BREAKER
DOWNCOMER
SUPPRESSION
POOL
SPARGER

BWR CONCRETE CONTAINMENT



GENERAL ELECTRIC

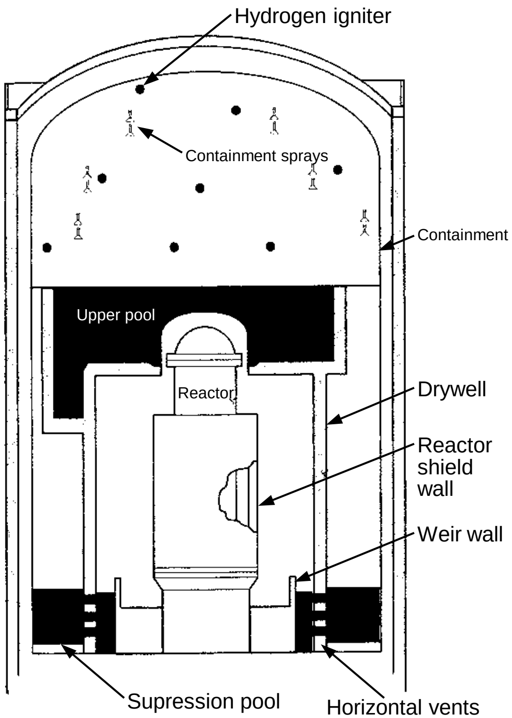
Containment Heat Removal for Mark II Containment

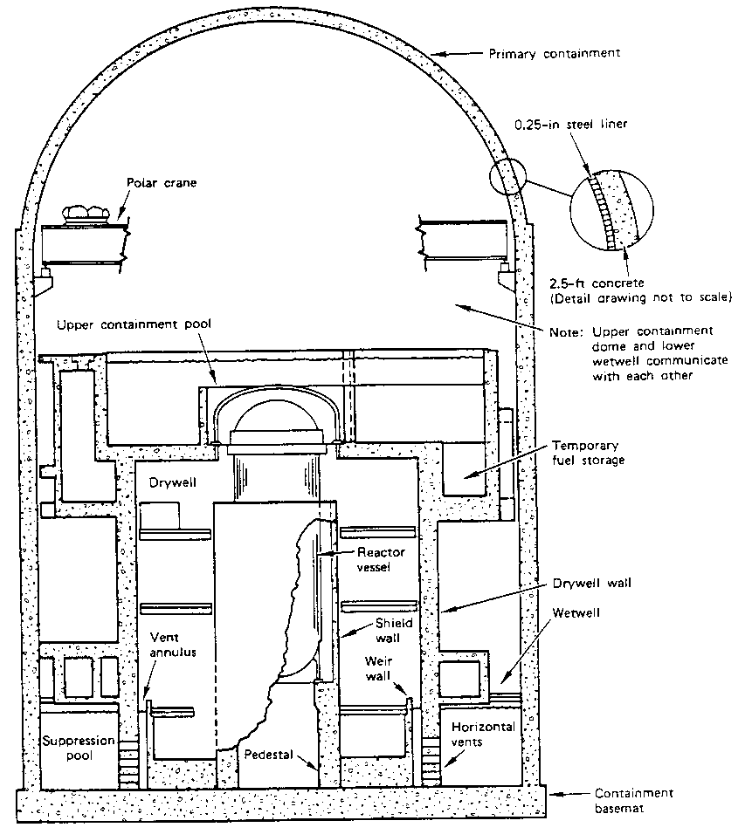


Mark III Dramatically Differs from Mark I and II Designs

- Two volumes (drywell and wetwell) connected by horizontal vents
- Significantly larger volume than Mark I and Mark II designs
  - but lower design pressure
- Containment atmosphere NOT inerted
  - relies on hydrogen igniters
- Two types of primary containment designs
  - free-standing steel structure (Perry & River Bend)
  - reinforced concrete with steel liner (Clinton & Grand Gulf)

Two Types of Mark III Primary Containments

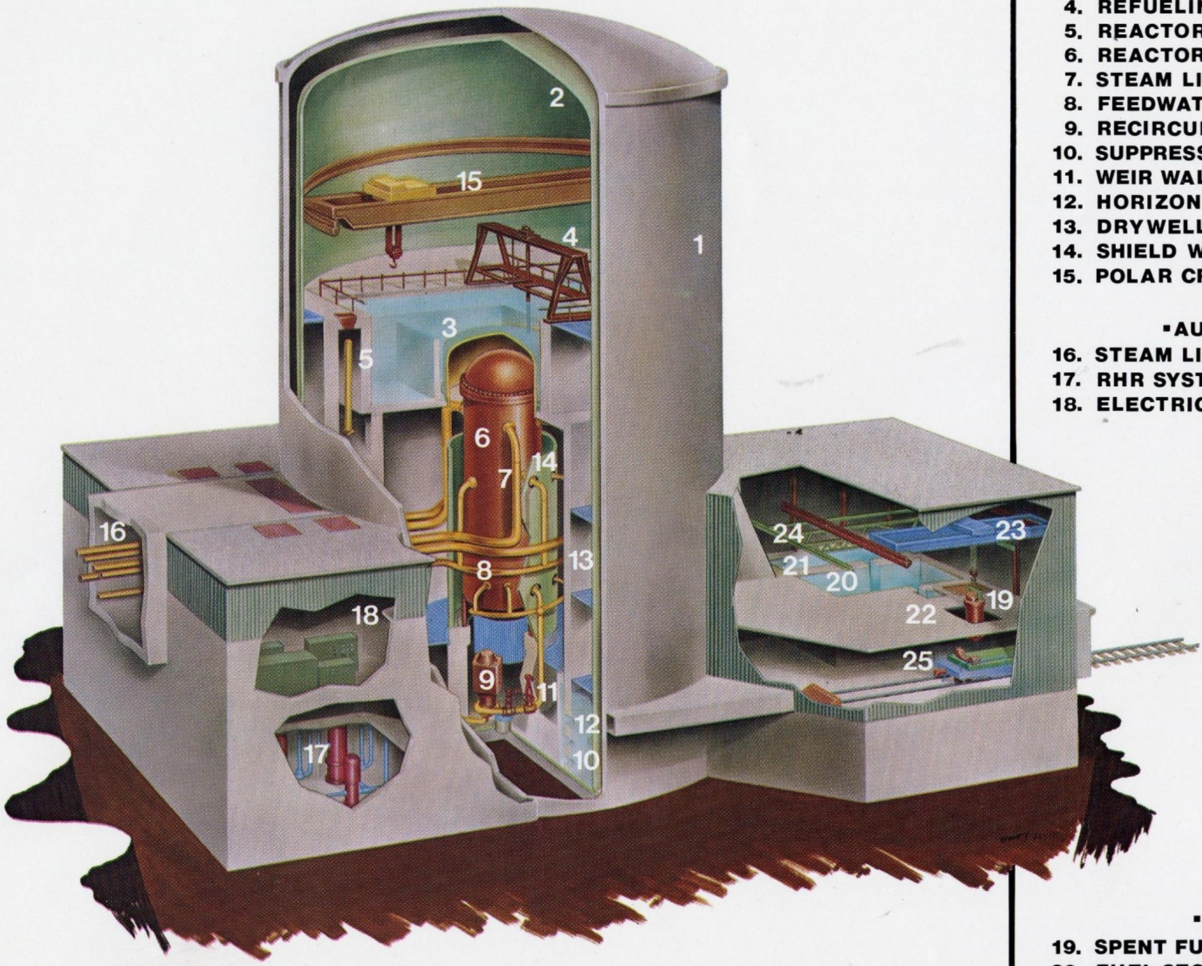




Free standing steel structure

Reinforced concrete

MARK III CONTAINMENT



▪REACTOR BUILDING▪

1. SHIELD BUILDING
2. FREESTANDING STEEL CONTAINMENT
3. UPPER POOL
4. REFUELING PLATFORM
5. REACTOR WATER CLEANUP
6. REACTOR VESSEL
7. STEAM LINE
8. FEEDWATER LINE
9. RECIRCULATION LOOP
10. SUPPRESSION POOL
11. WEIR WALL
12. HORIZONTAL VENT
13. DRYWELL
14. SHIELD WALL
15. POLAR CRANE

▪AUXILIARY BUILDING▪

16. STEAM LINE TUNNEL
17. RHR SYSTEM
18. ELECTRICAL EQUIPMENT ROOM

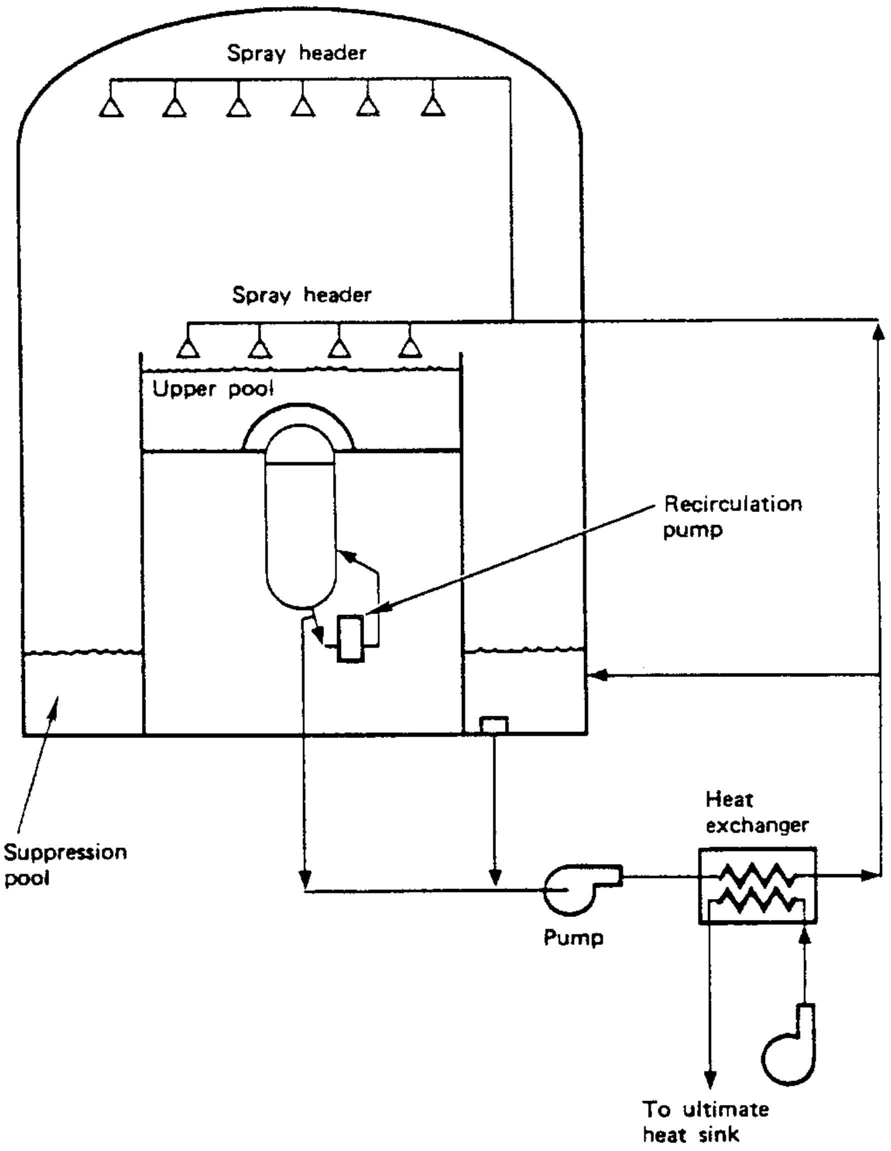
▪FUEL BUILDING▪

19. SPENT FUEL SHIPPING CASK
20. FUEL STORAGE POOL
21. FUEL TRANSFER POOL
22. CASK LOADING POOL
23. CASK HANDLING CRANE
24. FUEL TRANSFER BRIDGE
25. FUEL CASK SKID ON RAILROAD CAR

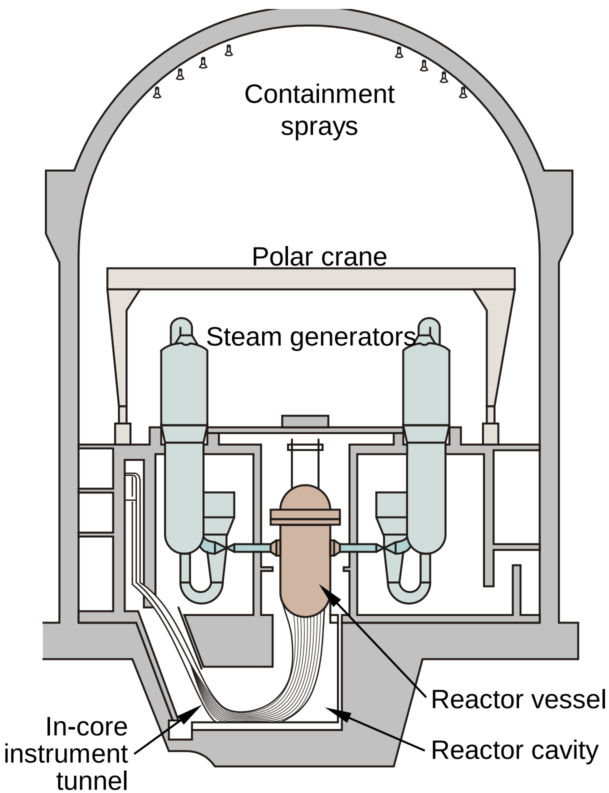
GENERAL ELECTRIC

GEZ-4385.1

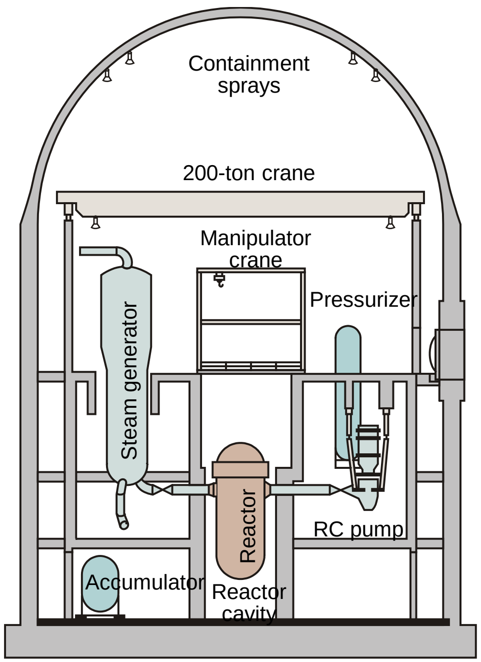
Mark III Containment Heat Removal Accomplished via Sprays and Suppression Pool



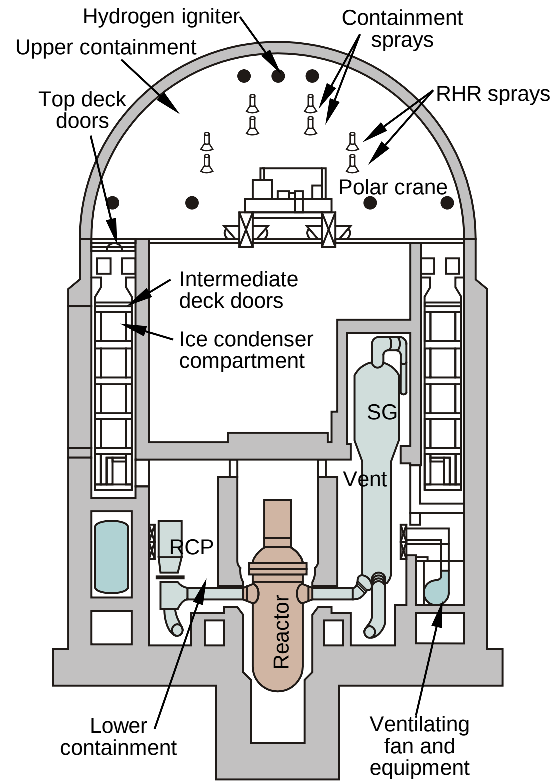
PWR Containment Designs Differ



Large dry



Subatmospheric



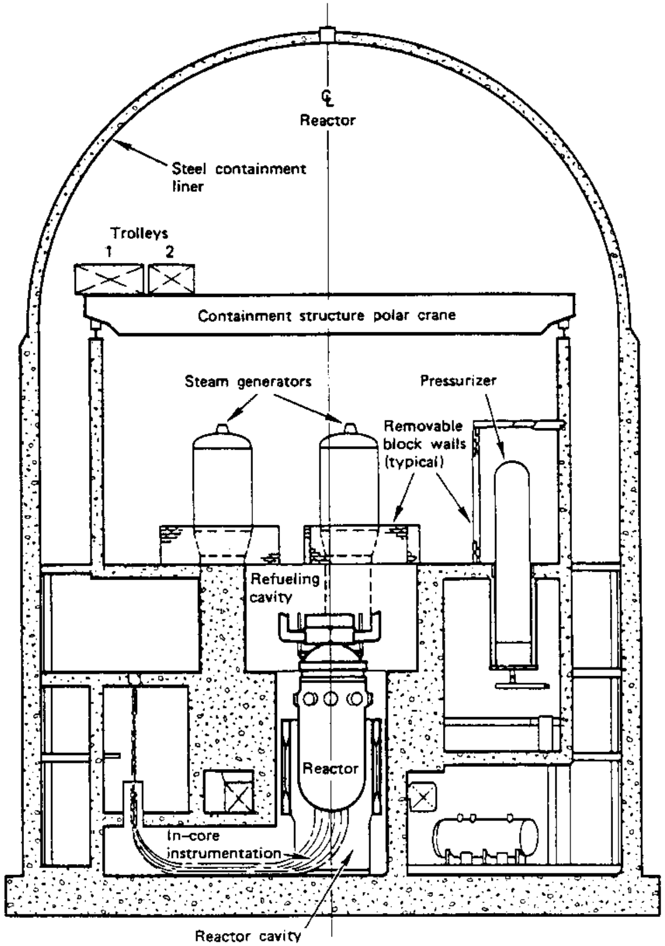
Ice condenser

Diverse Types of Large Dry Containments

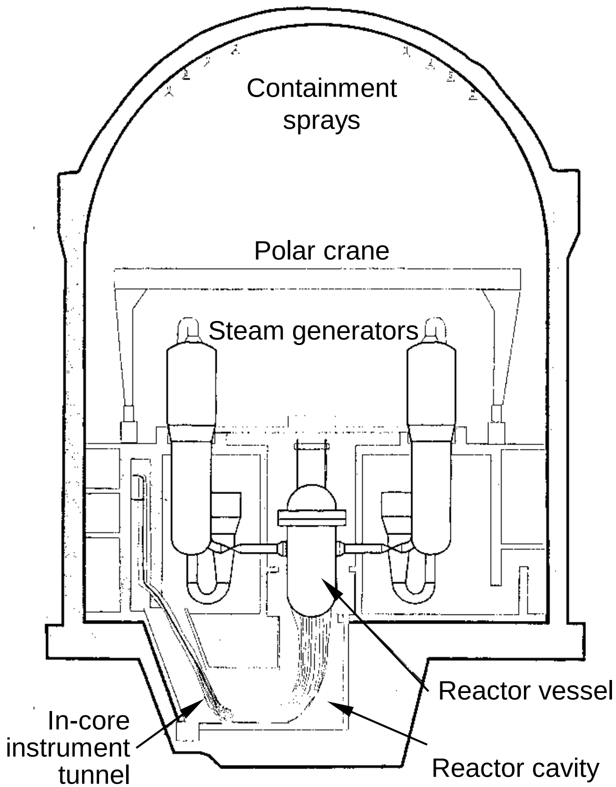
- Rely on large internal volume and structural strength (i.e., no passive pressure suppression system)
  - greater diversity of designs compared to other types
- Represents largest containment design group
  - includes a small subset (about 7) subatmospheric containment designs
- Most use reinforced or post-tensioned concrete with steel liner
  - few are of steel construction with reinforced concrete secondary containment

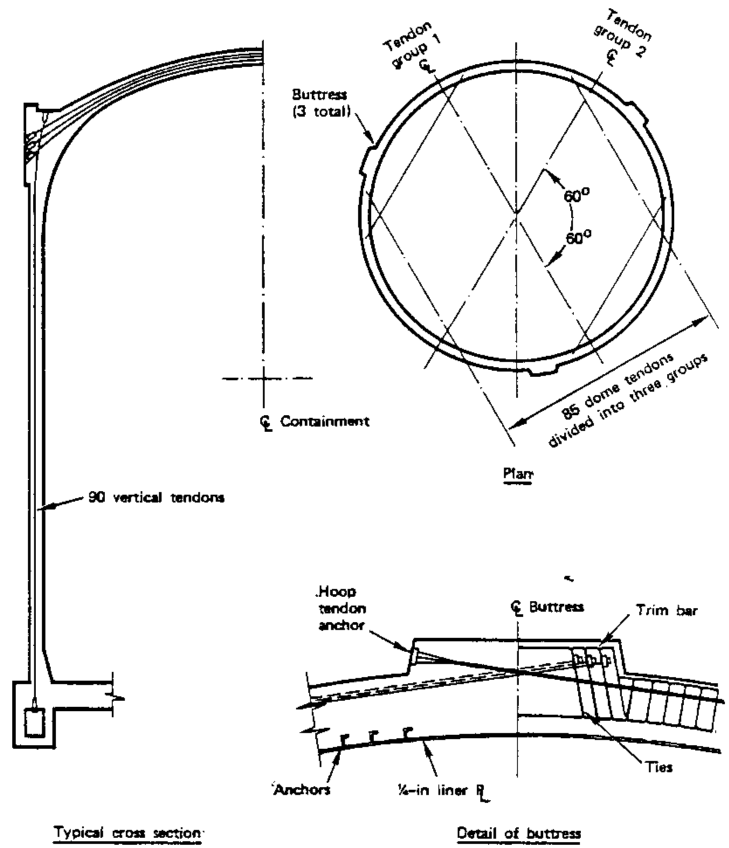
Diverse Types of Large Dry Containments (continued)

Large dry reinforced concrete
e.g., Diablo Canyon
(Most subatmospheric designs are of this type)



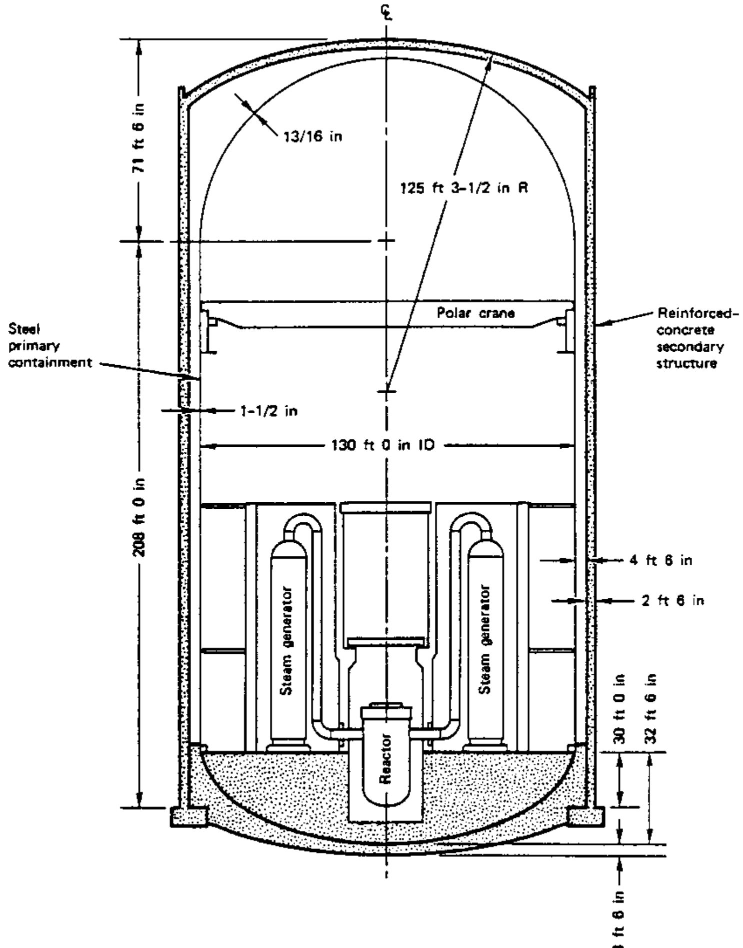
Diverse Types of Large Dry Containments (continued)





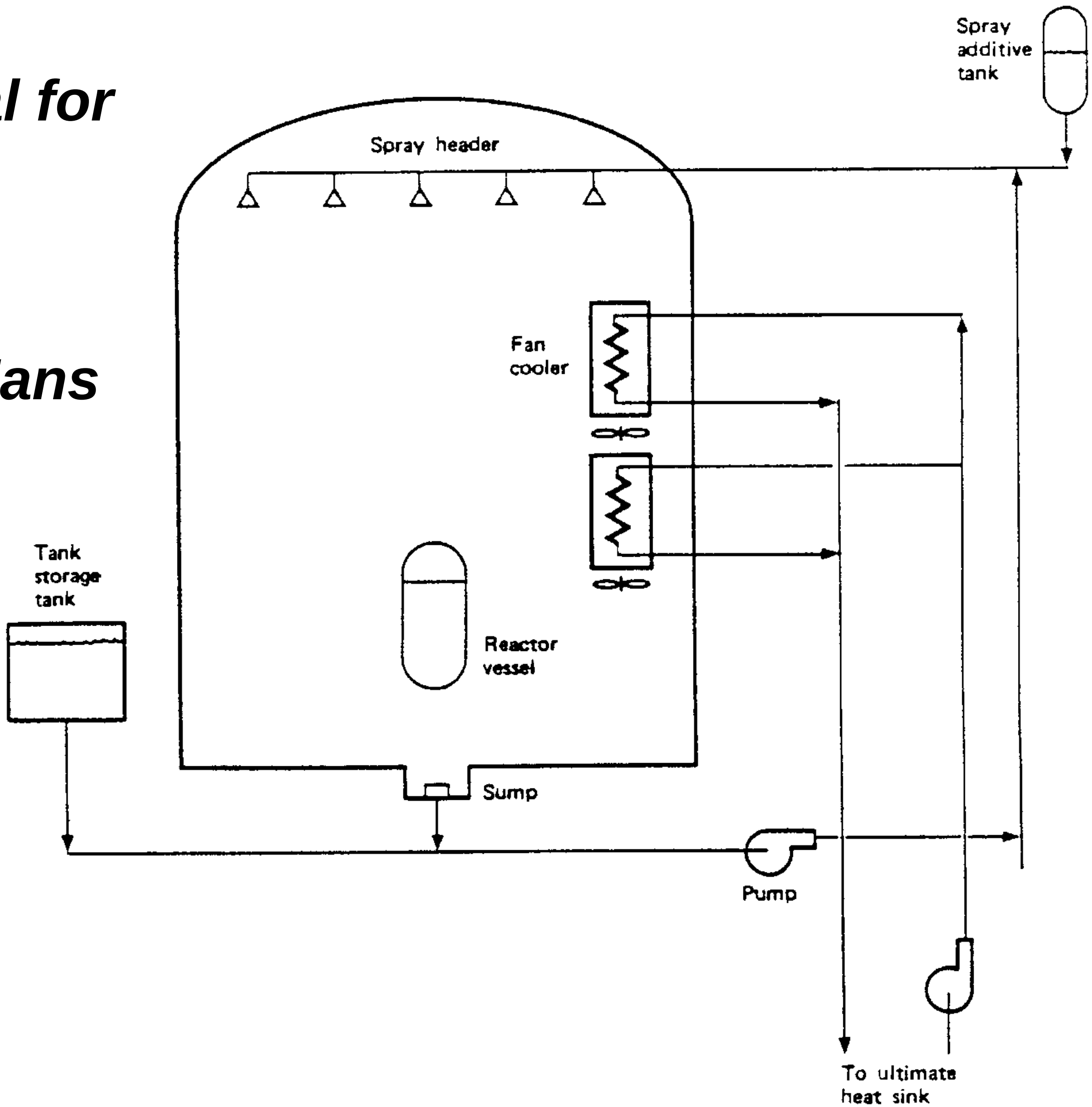
Large Dry Pre-stressed (or Post-tensioned) Concrete
e.g., Palisades
(This is the most common containment design)

Diverse Types of Large Dry Containments (continued)



Large dry steel containment with reinforced concrete secondary containment e.g., Davis Besse

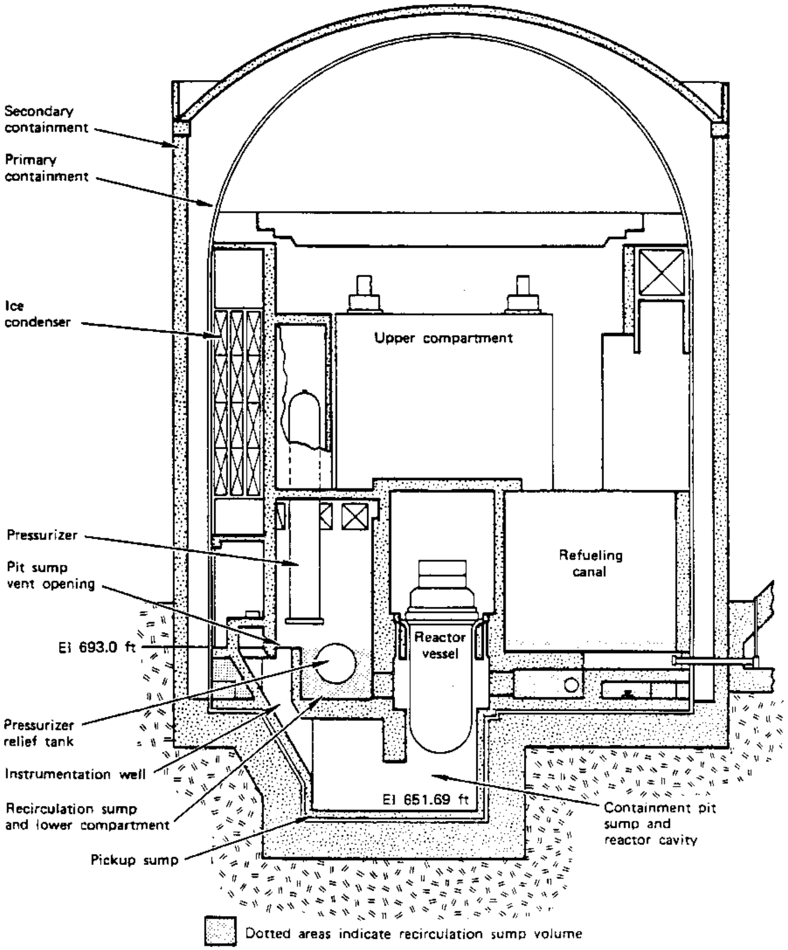
Containment Heat Removal for Large Dry Containment Design Uses Sprays and Fans Coolers

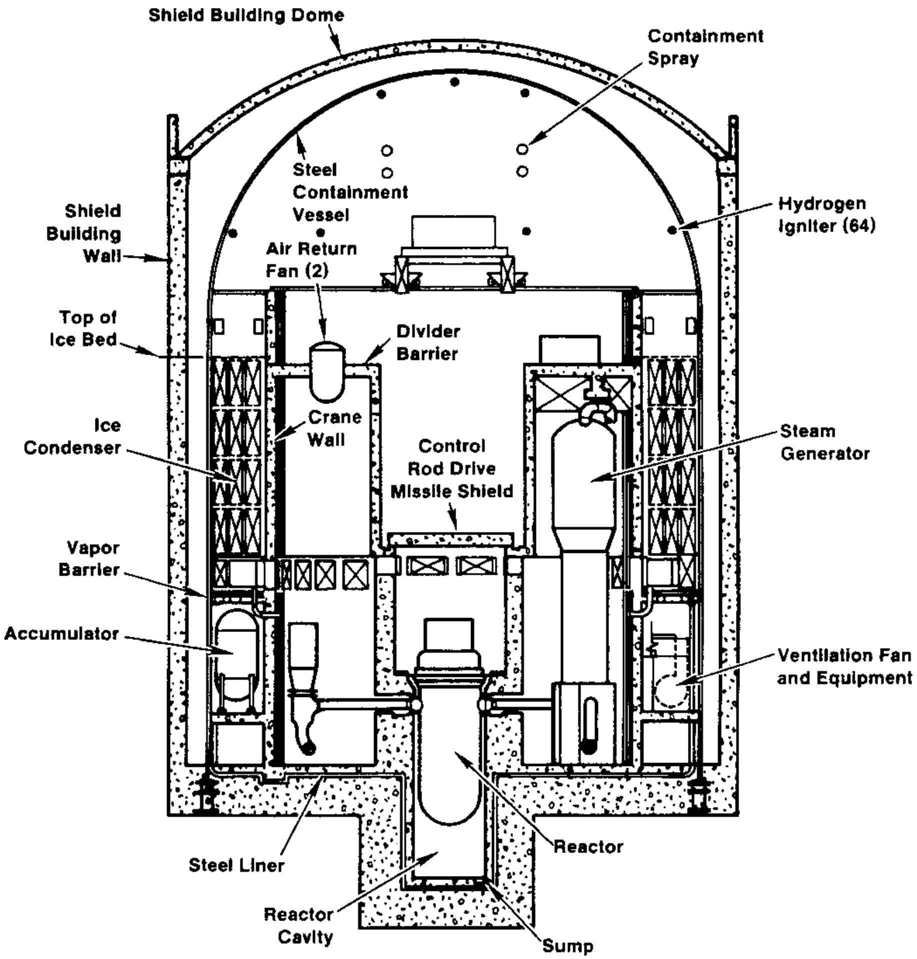


Ice Condenser Containments

- Three volumes: lower compartment, upper compartment, ice condenser
  - Ice condenser connects lower compartment containing RPV and RCS to upper compartment
  - Ice condenser holds approximately 2,300,000 lb. of borated ice in perforated metal baskets
- Relies on igniters for hydrogen control
- Most have cylindrical steel containment surrounded by concrete secondary containment
  - D. C. Cook: concrete containment with steel liner

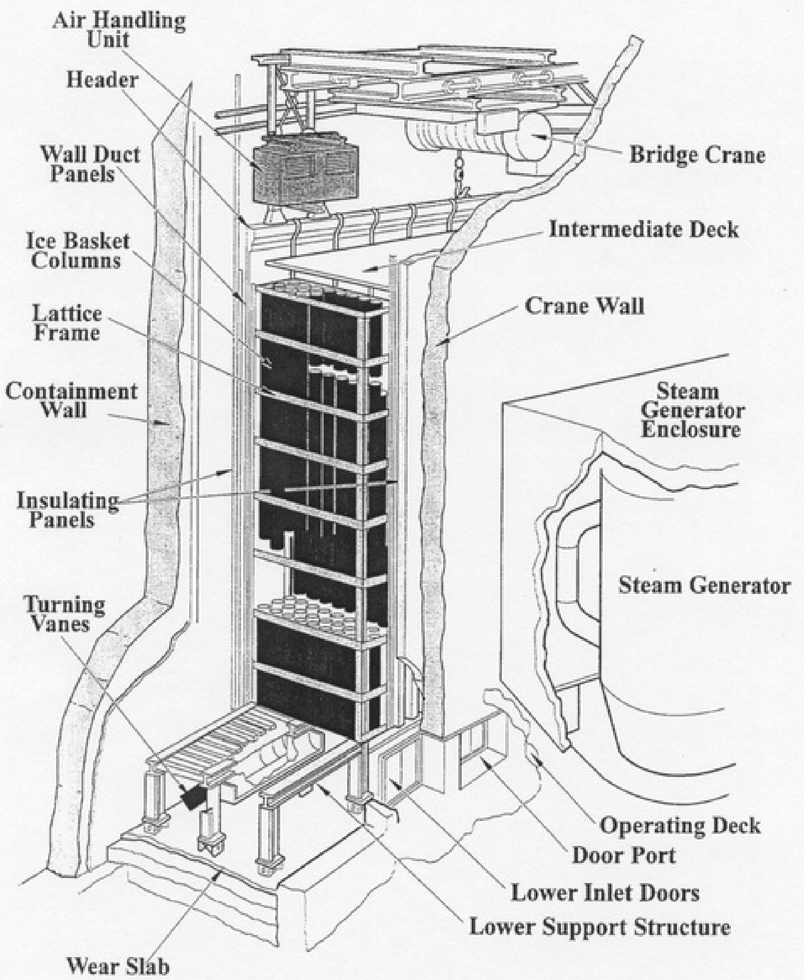
Ice Condenser Containments (continued)





Sequoyah

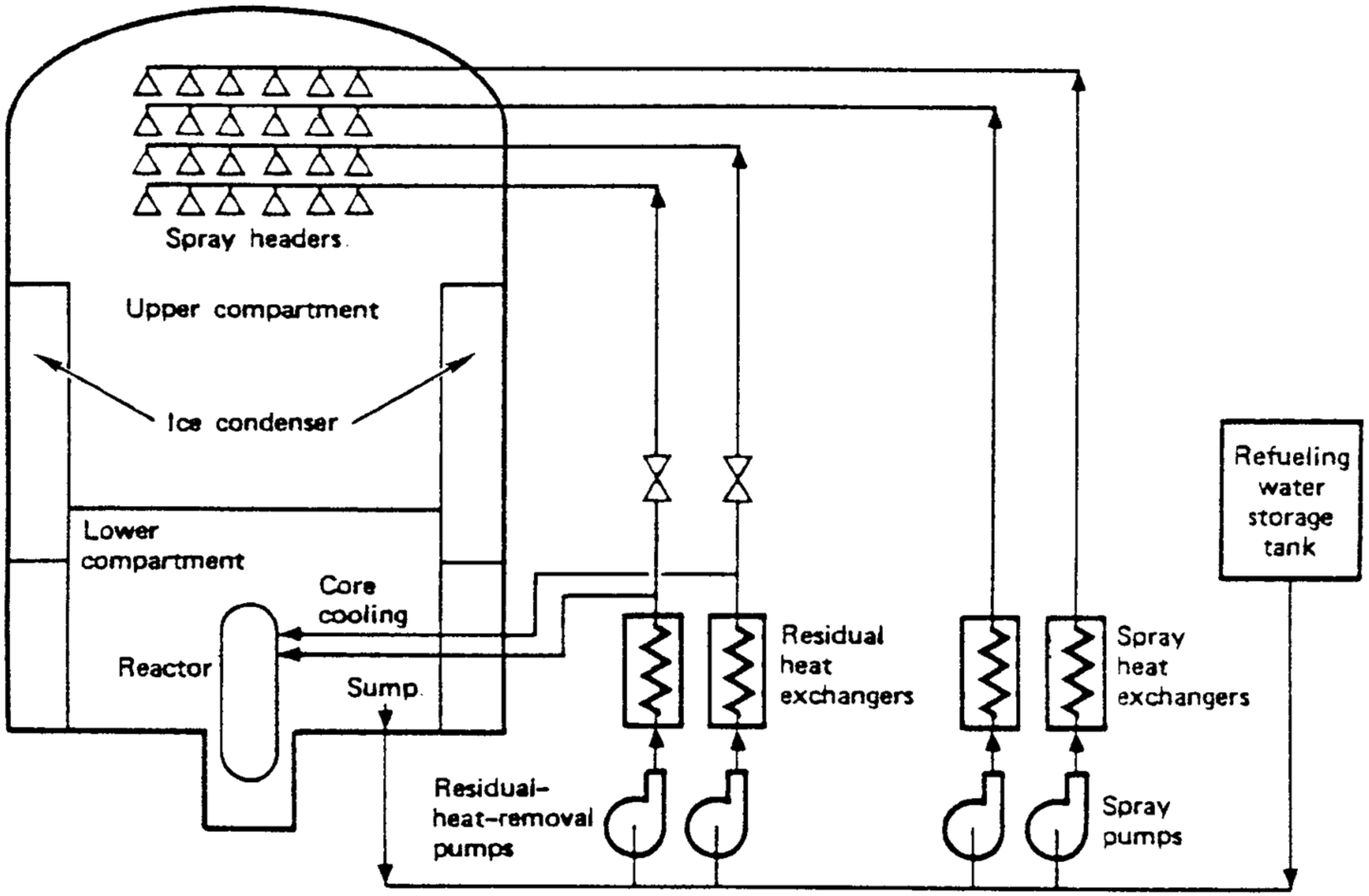
Ice Condenser Containments (continued)



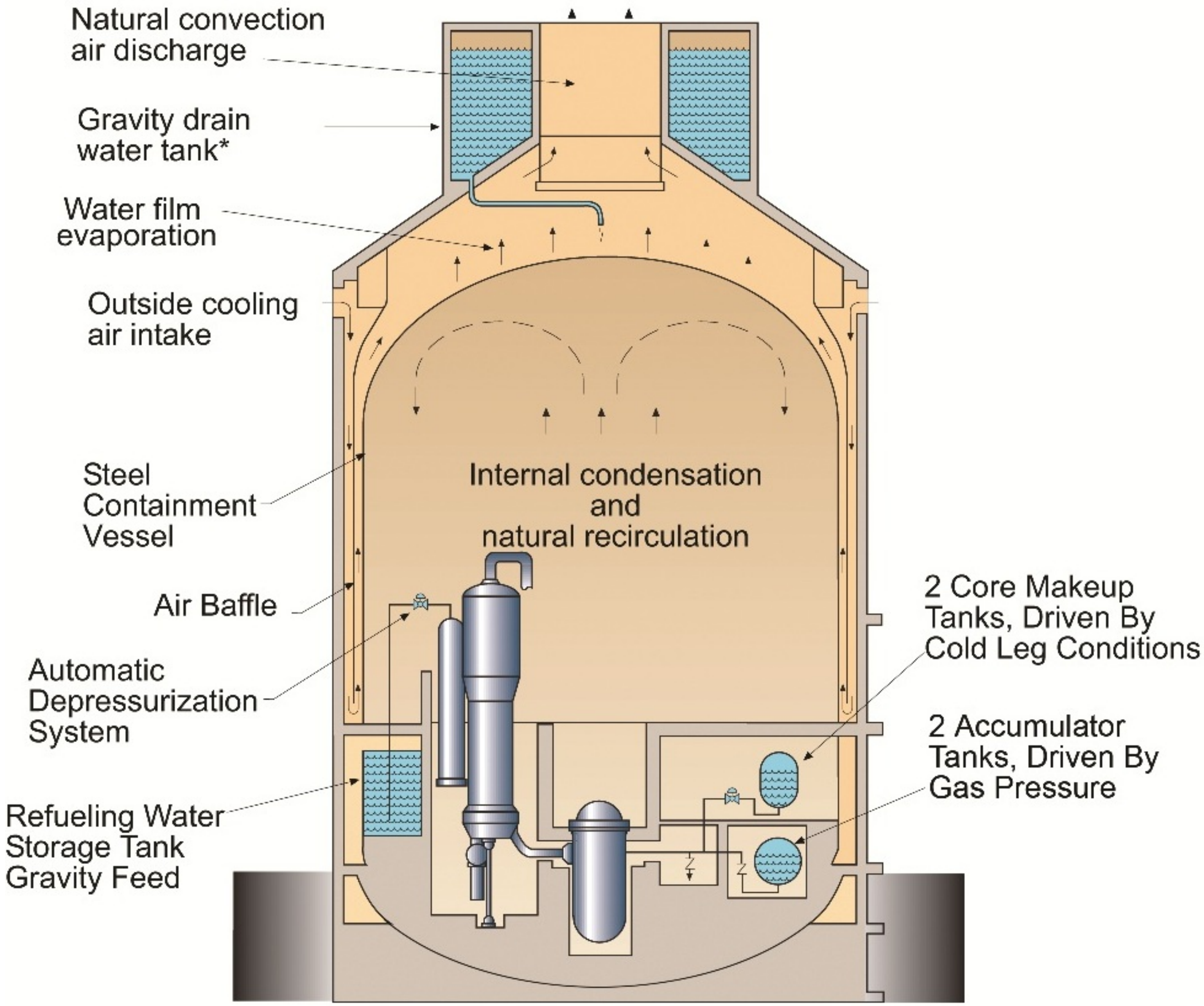
Ice condenser cutaway

Figure 2

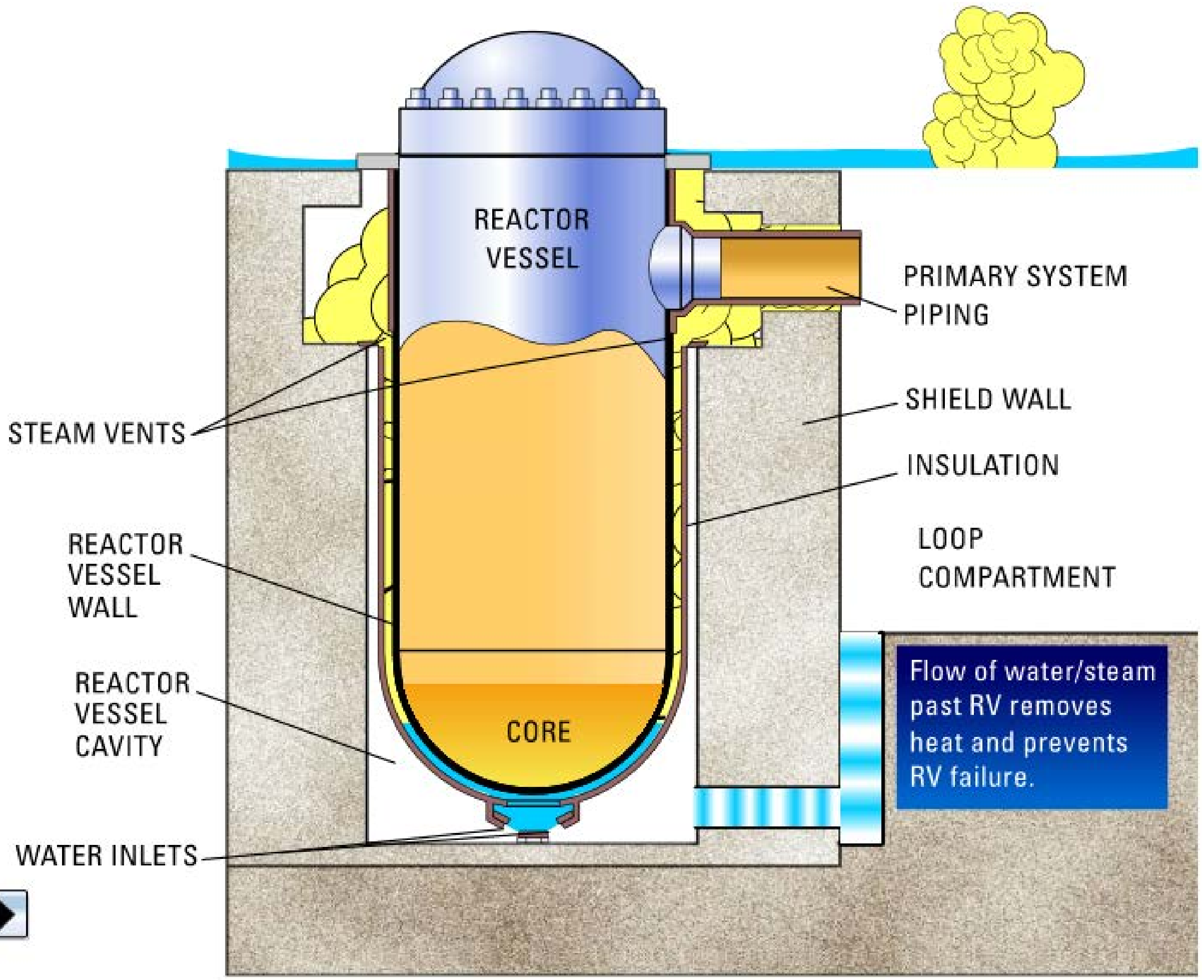
CHR for IC Design Uses Sprays and Ice Condensers



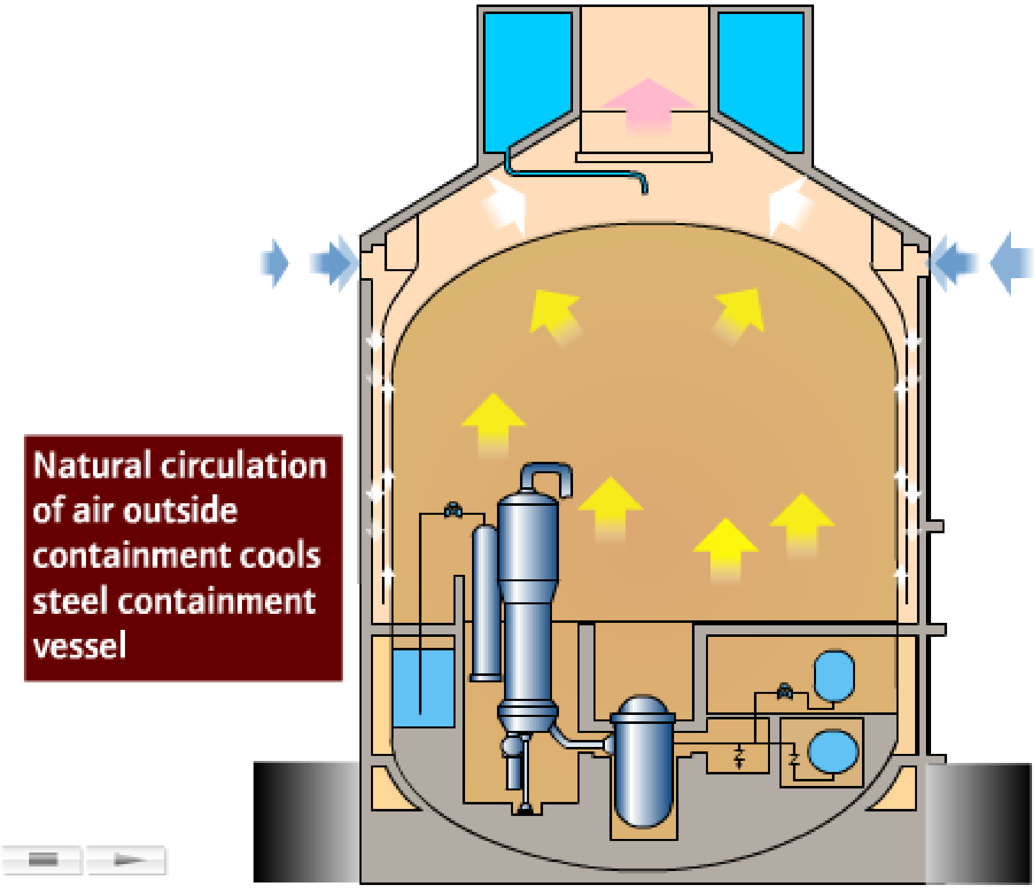
AP1000 Containment Design



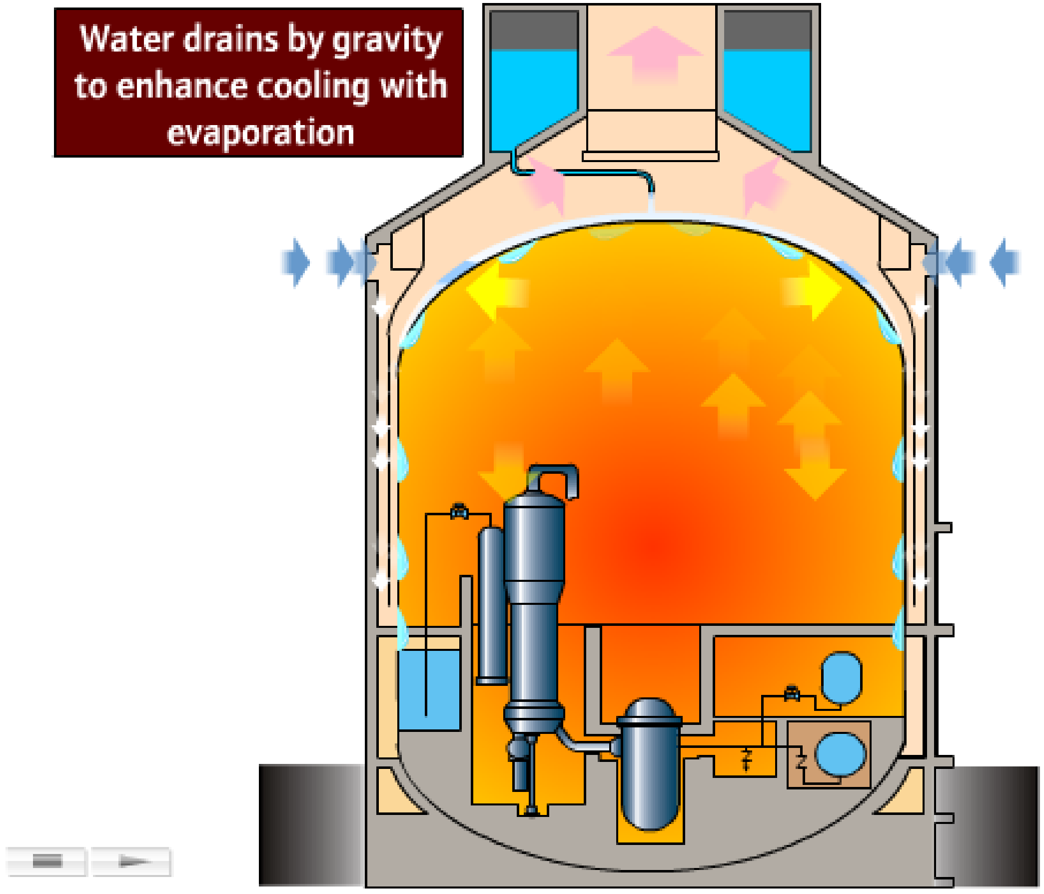
AP1000 Containment Utilizes In-Vessel Core Damage Retention



AP1000 Passive Containment Cooling System



AP1000 Containment Cooling

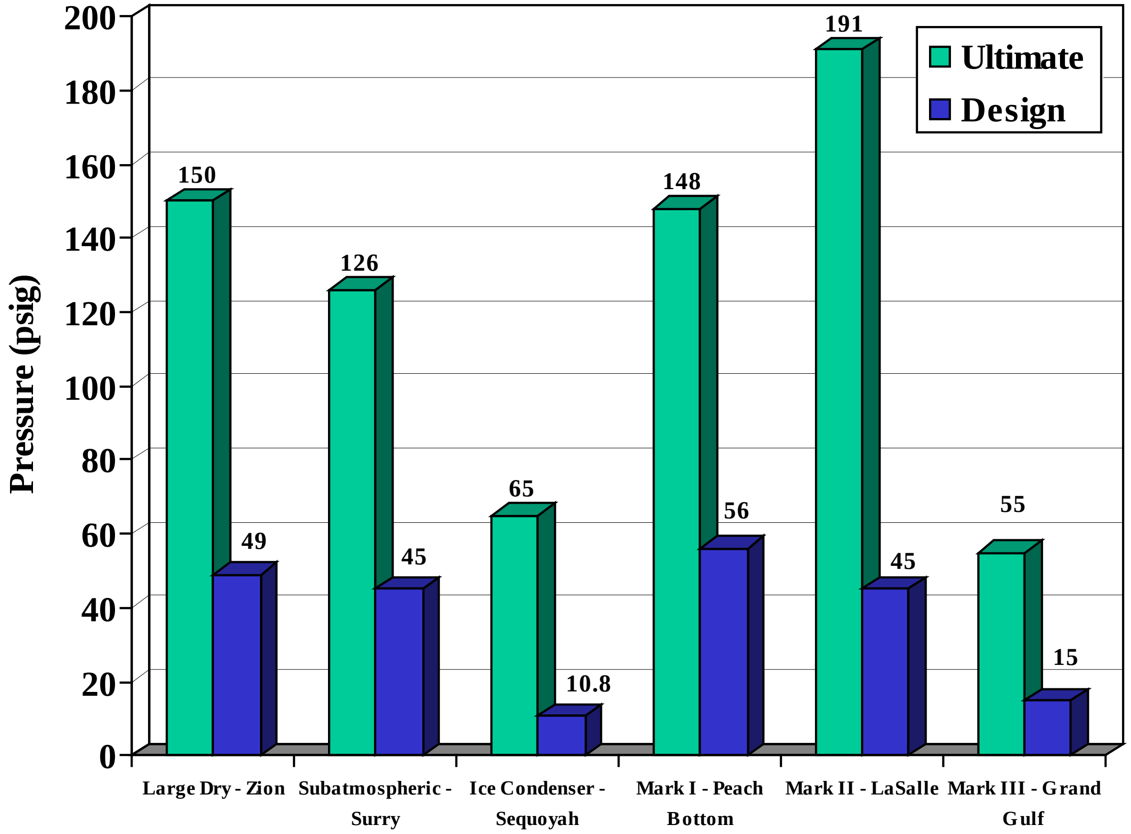


Severe Accidents Pose Several Challenges to Containment Integrity

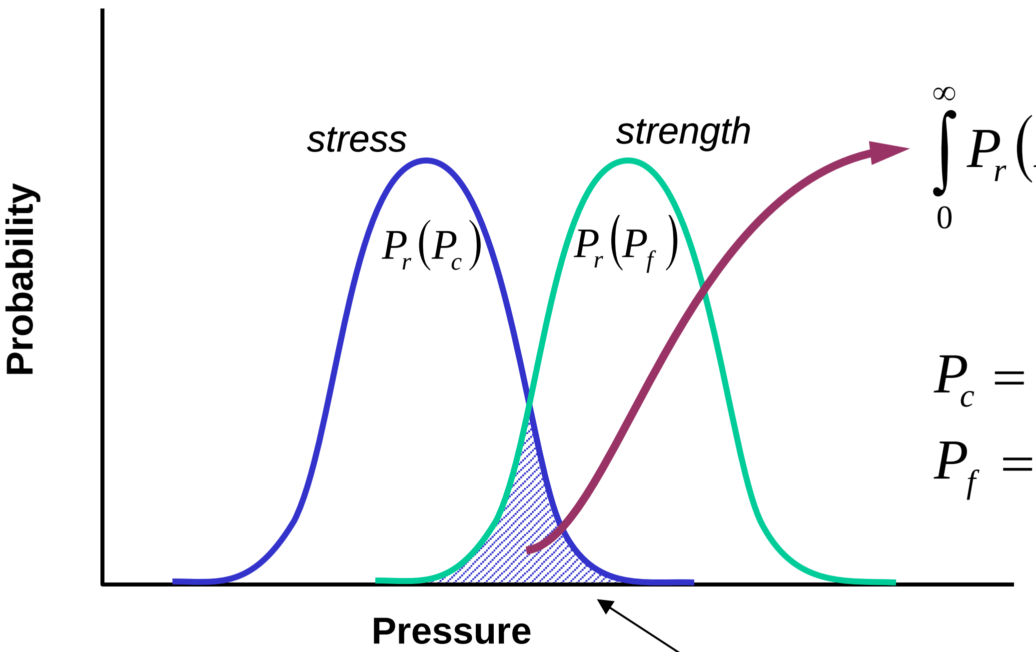
- Overpressure
- Dynamic pressure (shock wave)
- Missiles generated by steam explosions
- Melt-through (containment liner or basemat)
- Bypass
  - ISLOCA and SGTR
- Isolation failures

- (Note: These will be discussed in detail in Chapter 5)

Containment Failure Pressures Significantly Higher than Design Pressures



Conditional Probability for Containment Failure for Each Sequence Calculated Probabilistically



Conditional containment failure probability (CCFP) =

$$\int_0^{\infty} P_r(P_c = p)\left\{\int_0^{p} P_p(P_f = p')dp'\right\}dp$$

$$P_c = Peak\ containment\ pressure$$

$$P_f = Containment\ failure\ pressure$$

overlap of two curves represents the probability of containment failing

Containment Structural Response and Failure Characterization

- Objective is to develop a probabilistic description of the internal pressure capacity of the containment structure
- Typically expressed in the form of a fragility curve
  - Cumulative probability of failure as a function of internal pressure
    - Internal pressure assumed to be static and uniform
  - Composite fragility curve combines the individual fragility curves for different failure mechanisms
- Mathematical model treats containment pressure capacity as a random variable because of:
  - Variability in material properties and manufacturing, and lack of knowledge uncertainties

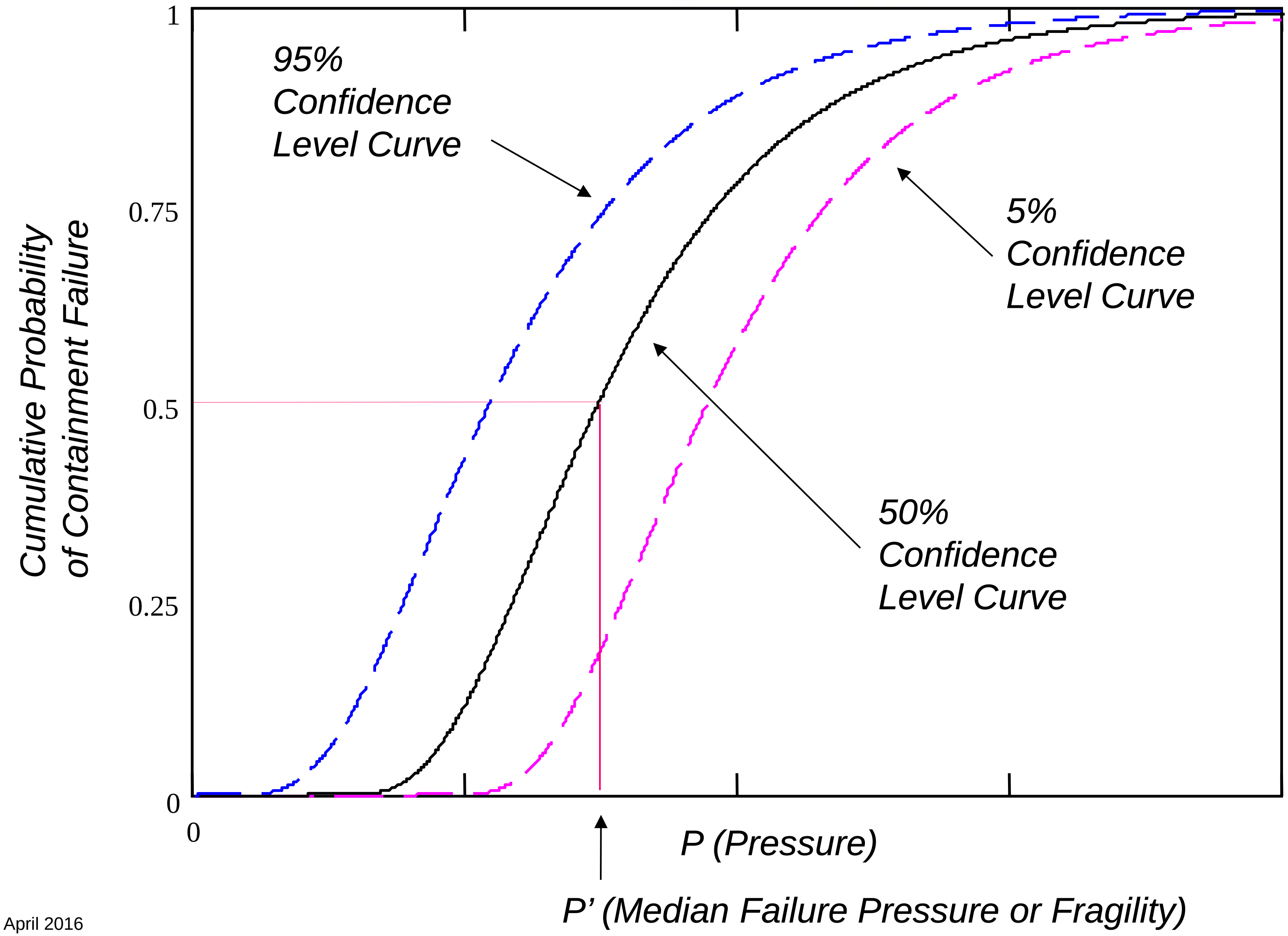
Static Uniform Internal Pressures Can Lead to a Number of Different Failure Modes

- Membrane failure in the hoop direction in the cylinder or dome
- Membrane failure in the meridial direction in the cylinder or dome
- Radial shear failure at cylinder to basemat or dome to cylinder discontinuity
- Bending failure in basemat
- Shear failure in basemat
- Shear failure in the containment shell at penetrations
- Membrane, bending or shear failure in penetrations

Pressure Fragility Model Similar to Seismic Fragility Model

- Fragility curve and uncertainty is expressed in terms of median pressure capacity (fragility) times the product of two random variables
- Pressure capacity (fragility) P is given by:
  - $P = P' * \varepsilon_R * \varepsilon_U$. Where: P' = median fragility, and
- ε_R and ε_U are random variable with unit medians that represent the inherent randomness (variability or aleatory uncertainty) and uncertainty (epistemic uncertainty) in the estimate of P'
- ε_R and ε_U are assumed to lognormally distributed with logarithmic standard deviations of β_R and β_U, respectively

Containment Fragility Curves at Different Confidence Levels



Since Containment Can Fail in Several Ways, Need to Combine Fragilities

- Referred to as the "Composite Fragility"
- Probability that containment will fail in at least one failure mode at a given internal pressure is:

  $PrF(p) = 1-\Pi_{i=1,n}[1-PrF_i(p)]$

  where:

  $PrF_i(p)$ = probability of failure mode i at pressure p

  n = total number of failure modes
- Note that this formulation assumes independence among the different failure modes
  - Assumption of independence in this case, is conservative

Containment Fragility and Severe Accident Loads are Integrated in CET

- Plant Damage States (PDS) provide the boundary conditions for the accident progression analysis performed in the containment event tree (CET)
  - Phenomena affecting vessel and containment integrity are the topics of the next two sections
- Containment fragility curve establishes the failure criteria for containment integrity
- CET models the progression of the severe accident with respect to the containment failure criteria

CET Tracks Probabilities or Loads

- Event tree branch probabilities/values can be either
  - Simplistic: track phenomena, then document failure probability estimate
    - When DCH happens (given certain conditions), then Containment fails early with probability = 0.1
  - Complex: estimate likelihood of phenomena, incrementally track loads on containment
    - DCH happens 20% of the time, increases containment pressure 100 psi
    - Running total of containment pressure then tracked
      - Contributions from various phenomena

Session Review

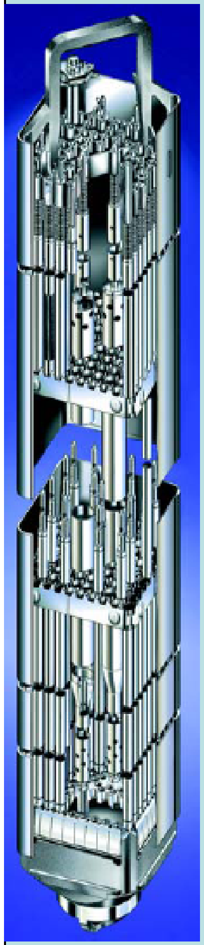
- What are the major containment designs?
- What are some of the characteristic features of each?

4. Phenomena Affecting Vessel Integrity

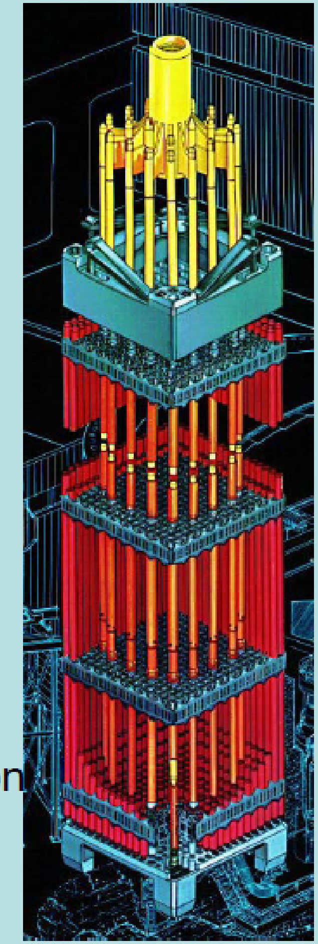
- Introduction
- Reactor Fuels
- Design, RIA and LOCA accidents for PWRS and BWRS
- Failure Modes
- Debris Heat Loads
- Failure Mitigation Measures
- Case Study and Problems
- Study Questions
- References
- Special Summary TMI Damage Implications for Fukushima

Objectives

- Define reactor fuels effects on PWR, and BWR Reactors – and RIA and LOCA Behavior
- Identify various vessel failure modes and understand their likelihood in various reactor designs and accident scenarios.
- Describe possible end states for debris that relocates to vessel lower head.
- Discuss various mechanisms or actions that may prevent vessel failure.

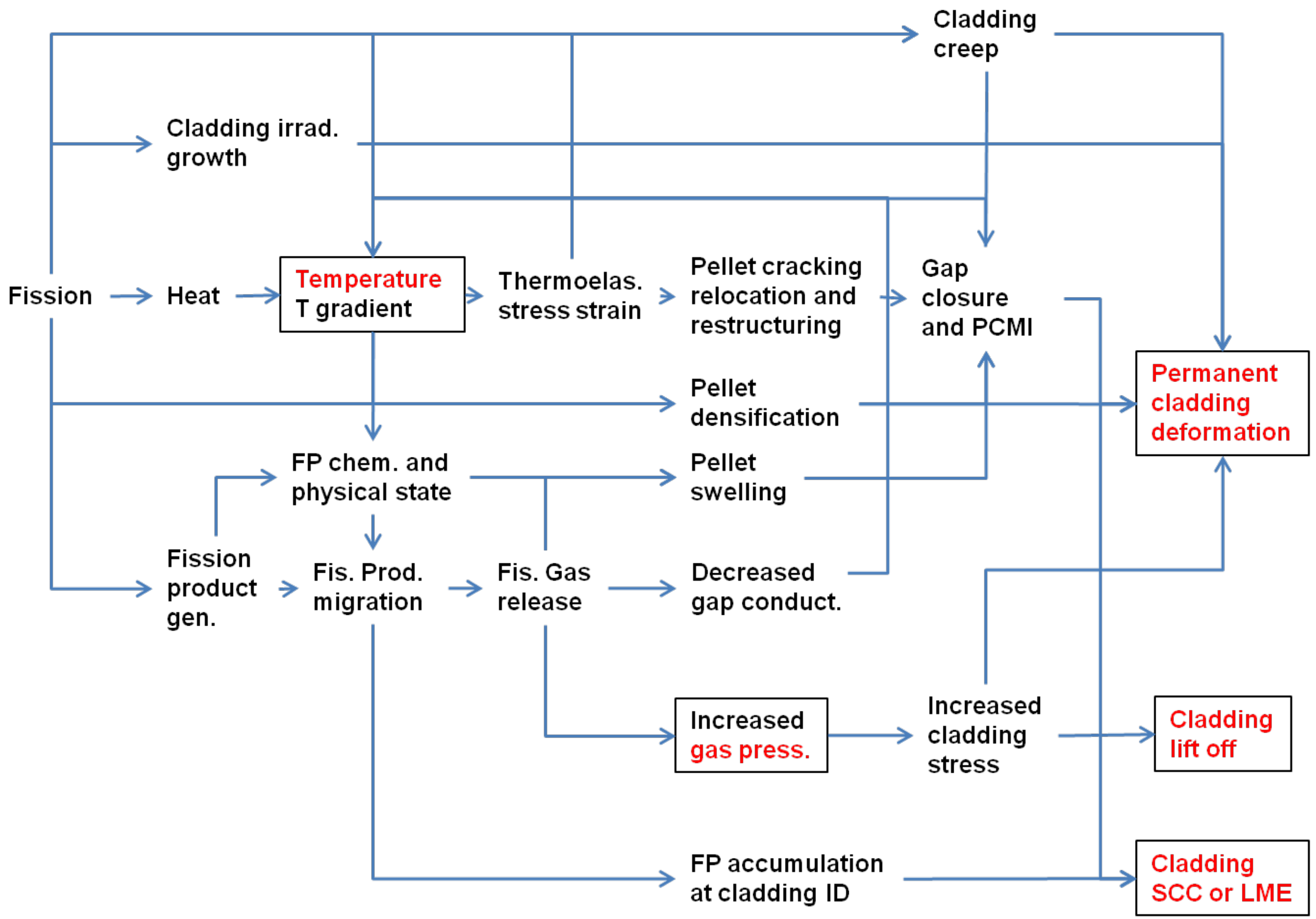


	BWR	PWR
Lattice	10x10	14x14 – 18x18
Lattice size	~5.3”	~9”
Height	120”-150”	144”-168”
Fuel	UO_2/MOx	UO_2/MOx
Fuel rods	~92	176-300
Part length rods	~14	0
Non-fueled rods	~2	20-25
Control	Ext. control rod	Int. control cluster
Cladding	Zr2 for PCI, nodular corrosion	Zr4/Zirlo/M5 for uniform corrosion & hydrogen
Channels	Yes	No
Fuel mass	~180 kgU	~600 kgU



[Crawford, 2009]

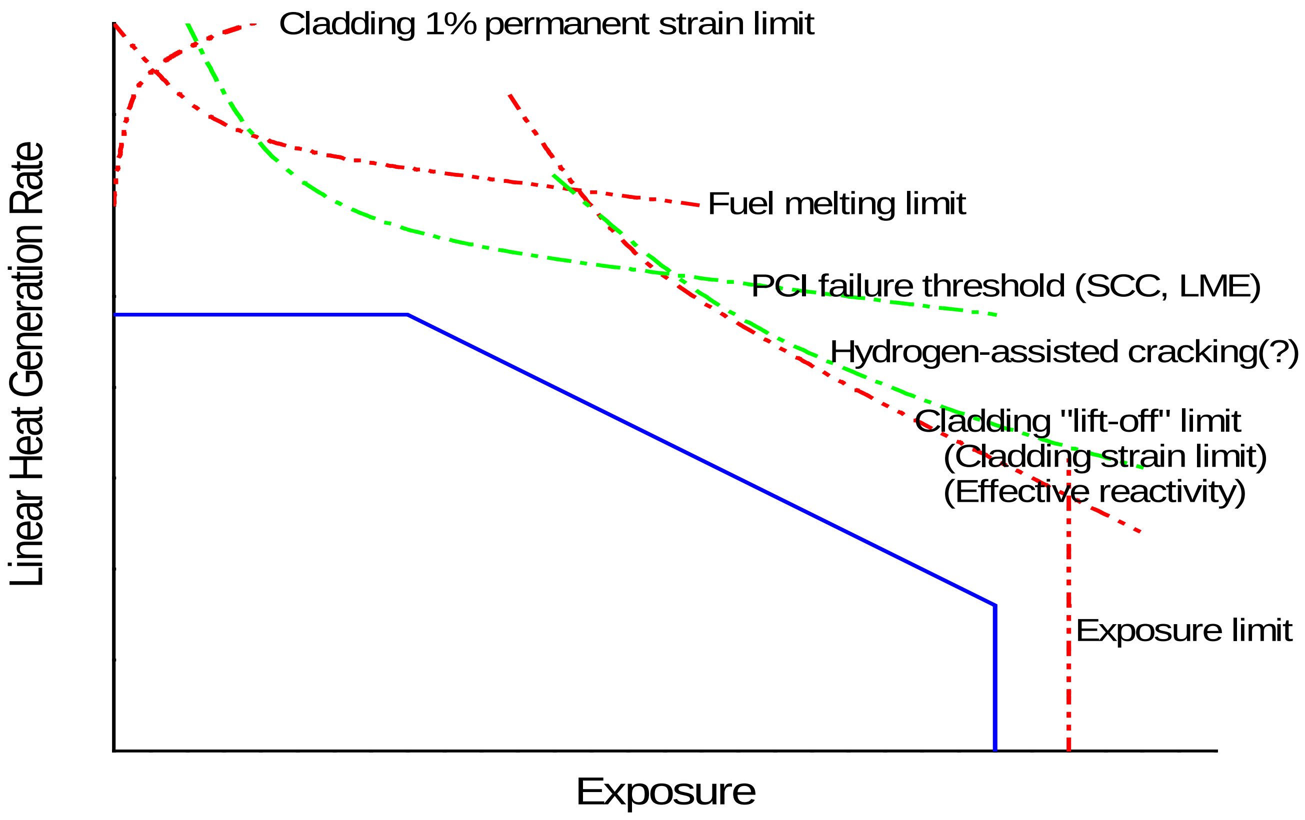
Key Design Criteria for Fuels -



Principal internal, fuel rod processes and their primary interactions (Key design criteria are enclosed with a box and shown in red)

[after Mohr, et al. 1976 – Rudling and Patterson -2012]

Fuel Designs Limited Heat Generation Rate and Total Exposure



Constraints based on MATPRO/FRAPCON; [Hagrman 1993- Rudling and Patterson 2012]

UO2 has ideal Fuel Properties Although New Accident Tolerant Fuels are Under Development

Property	Uranium	UO_2	UC	UN
A. Chemical				
Free energy of formation at 1000 °K (Kcal/mole)	-	-218.2	-25.2	-47
Corrosion resistance in water	Very poor	Excellent	Very poor	Poor
Compatibility with clad materials	Reacts with normal clad	Excellent	Variable	Variable
Thermal stability	Phase change at 665 and 770 °C	Good	Good in reducing atmosphere	Good, decomposes at 2600 °C
B. Physical				
Uranium (metal) density (g/cm³)	19.04	9.65	12.97	13.52
Theoretical Density (T.D.) (g/cm³)		10.96		
Melting point (°C)	1132	2865	2850	2850
Thermal conductivity W/cm/K	0.28 at 430 °C	0.03 at 1000 °C	0.25 at 100-700 °C	0.2 at 750 °C

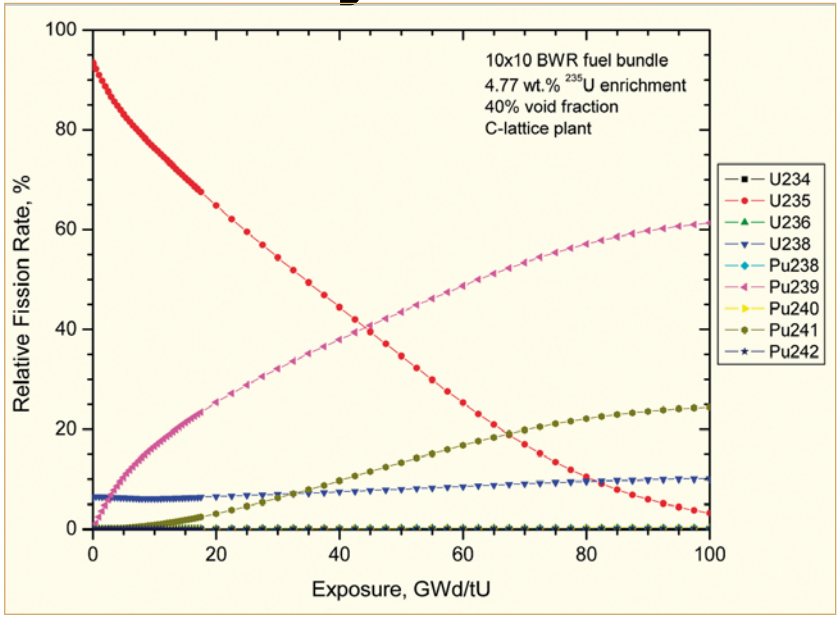
after Garzarolli in [Rudling, et al. 2007]

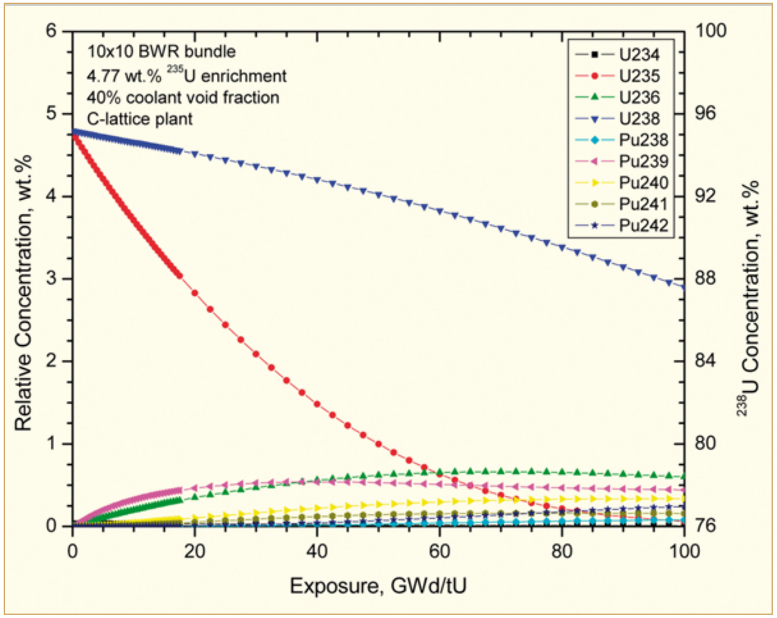
Primary Fission Sources are Uranium and activation Produce Plutonium

	Elastic Scattering $\sigma(nn)$	Inelastic Scattering $\sigma(nn')$	Radiative Capture $\sigma(n\gamma)$	Fission $\sigma(nf)$	Average neutron yield (ν-bar)
Fissile materials	Average over thermal spectrum (barns)[1]				
^{235}U	15.98		86.70	504.81	2.433
^{239}Pu	7.90		274.32	699.34	2.882
^{241}Pu	12.19		334.11	936.65	2.946
Fertile materials					
^{238}U	9.37		2.41	1.05E-05	2.489
^{240}Pu	1.39		262.65	6.13E-02	2.784
Fissile materials	Slowing-down region resonance integrals (barns)[2]				
^{235}U	152.82		131.97	271.53	2.438
^{239}Pu	155.87		184.06	289.36	2.876
^{241}Pu	148.68		169.13	570.66	2.933
Fertile materials					
^{238}U	319.06		277.70	2.16E-03	2.490
^{240}Pu	913.76		8448.70	3.74	2.785

[Kaye & Laby 2005] – Rudling and Patterson 2012

Plutonium is Significant Contributor to Fission by end of Life

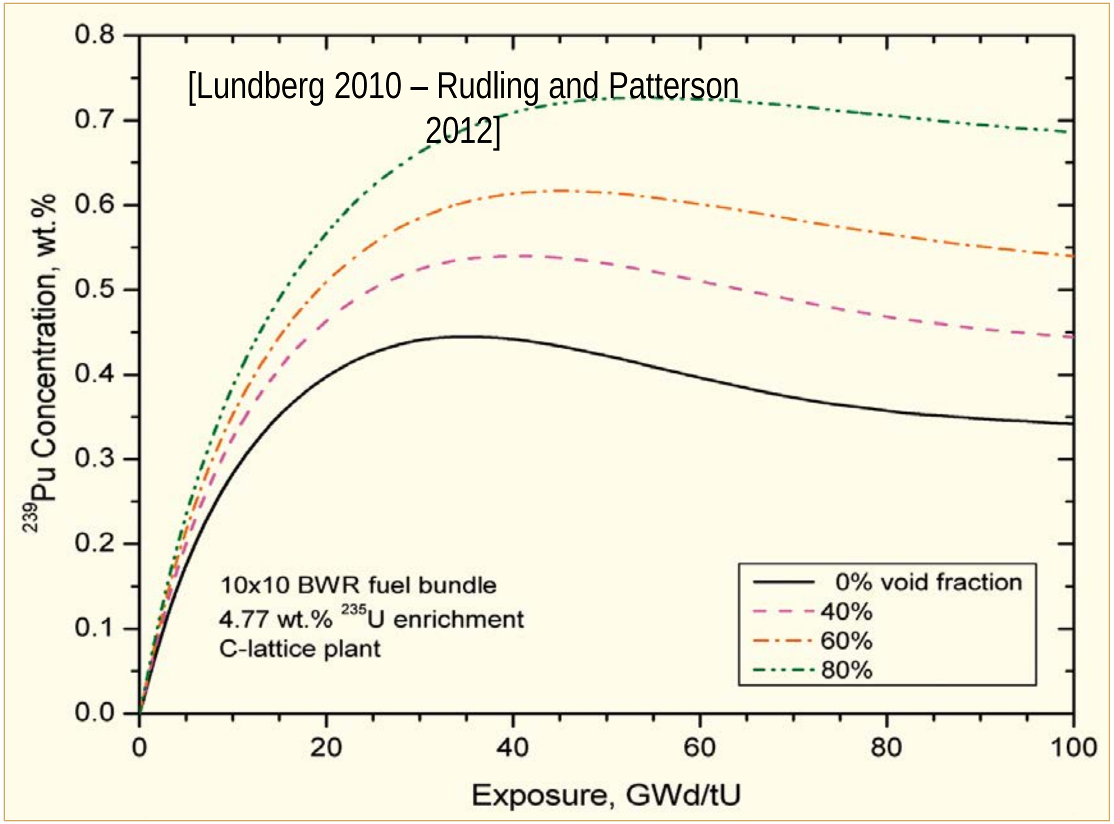




[Lundberg 2010, Rudling and Patterson 2012]

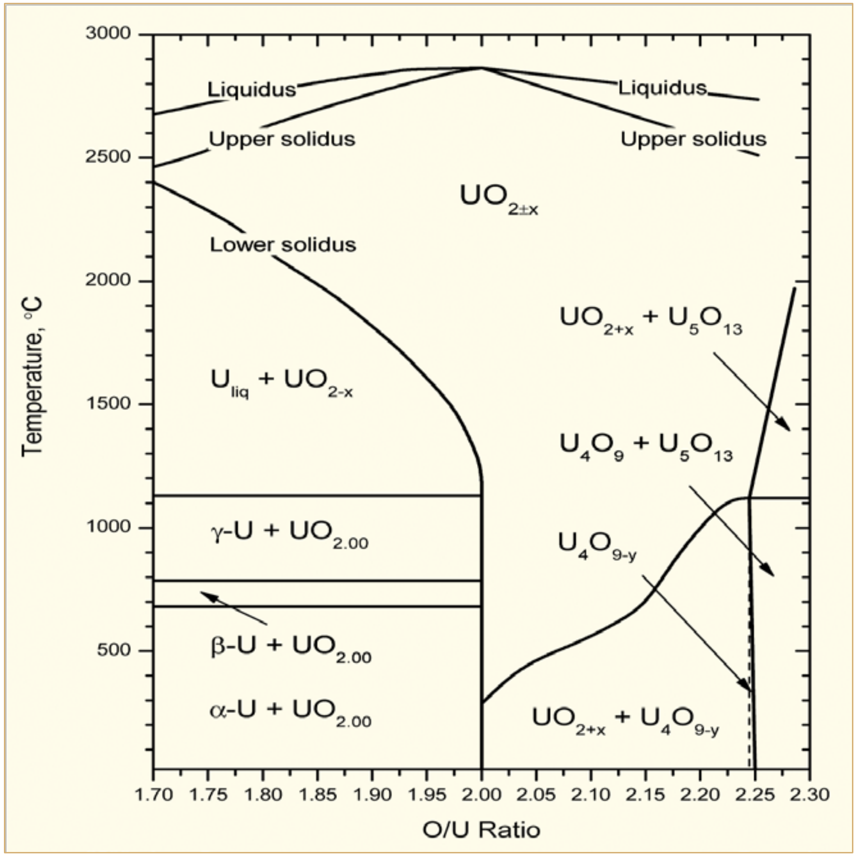
- Relative fission rate changes with exposure
- Plutonium becomes a significant source by mid-life and the dominant source by end of life
- Pu production balanced by loss due to fission
- (Pu concentration by EOL typically <1%)

Plutonium Content Varies With Void Fraction



- Plutonium production varies with fuel design and core conditions
- Rate increases with flux of higher energy (epithermal) neutrons
- Production and consumption reach equilibrium for given set of conditions

Fuel Chemistry Affects Accident Behavior



- Fuel fabricated to be nearly stoichiometric; i.e., $UO_{2.00\pm}$
  - Structure stable to T_{melt}
  - Maximum T_{melt}
- O/M ratio varies slightly during irradiation
- Large deviations from stoichiometry relevant to
  - Fabrication
  - Defected fuel behavior
  - Reprocessing

[Levin & McMurdie, 1975], [Olander, 1976], [Kim, 2000], [Guéneau et al, 2002], [Baichi et al, 2006], [Rudling et al, 2007] Rudling and Patterson 2012

RIA and LOCA – Severe Accident

- During a Reactivity Initiated Accident and a Loss of Coolant Accident no fuel dispersal is allowed
- RIA -> Severe Accident
  - Chernobyl
- LOCA -> Severe Accident
  - TMI
  - Fukushima

LWR Design Affects Severe Accident Response

Design Feature		Impact
Masses **Uranium Dioxide** **Zirconium** **Steel**	 BWRs have at least 50% more. BWRs have at least 100% -200%. BWRs have at least 20% more.	 Potential for larger relocation masses. Potential for more hydrogen production. Relocated materials have higher steel content.
Power Distribution	Average power factors in peripheral regions of BWRs significantly lower.	Significant time lag between heatup in central and peripheral core regions.
Coolant Volume	Much larger volume of coolant (relative to core structural volume) beneath BWR core.	Higher potential to quench relocated materials for longer time periods.

Reactor Kinetics

- About 99.4% of all neutrons are born directly in fission (*prompt neutrons*), with a very short lifetime. However, approximately 0.64% of the neutrons are delayed (*delayed neutrons*) – fraction of delayed neutrons= β
- In a system reached *criticality*, just as many neutrons are produced by fission as are lost by absorption and leakage from the reactor in a given time
  - k_{eff} = Rate of neutron production/(Rate of neutron absorption + Rate of neutron leakage)=1
- The fractional departure of a system from criticality is often expressed by the reactivity, ρ, and defined by:

$$\rho \equiv \frac{k_{eff} - 1}{k_{eff}}$$

[Rudling and Patterson 2012}

Reactor Kinetics

- The conditions for a reactor power transient like RIA to be of a concern are:
  - It must be very fast.
  - The reactivity added must be larger than 0.006.
- However, water reactors are designed so that a power increase will generate negative *reactivity* feedback
  - where a fuel temperature increase gives a fast negative feedback (Doppler effect).
  - an increase in the moderator temperature and steam (void) fraction, gives a slower negative feedback.
  - A slow reactivity increase may not cause any harm even if it is larger than 0.006 because of the negative feedback mechanisms.

[Rudling and Patterson 2012}

RIA Kinetics

- PWR
  - The most severe RIA scenario is the control rod ejection accident (CREA).
  - The CREA is caused by mechanical failure of a control rod mechanism housing, such that the coolant pressure ejects a control rod assembly completely out of the core.
    - Reactivity increase to the core occurs within about 0.1 s in the worst possible scenario.
    - The actual time depends on reactor coolant pressure and the severity of the mechanical failure.
    - With respect to reactivity addition, the most severe CREA would occur at hot zero power (HZP) conditions, i.e. at normal coolant temperature and pressure, but with nearly zero reactor power.

[Rudling and Patterson 2012}

RIA Pulse

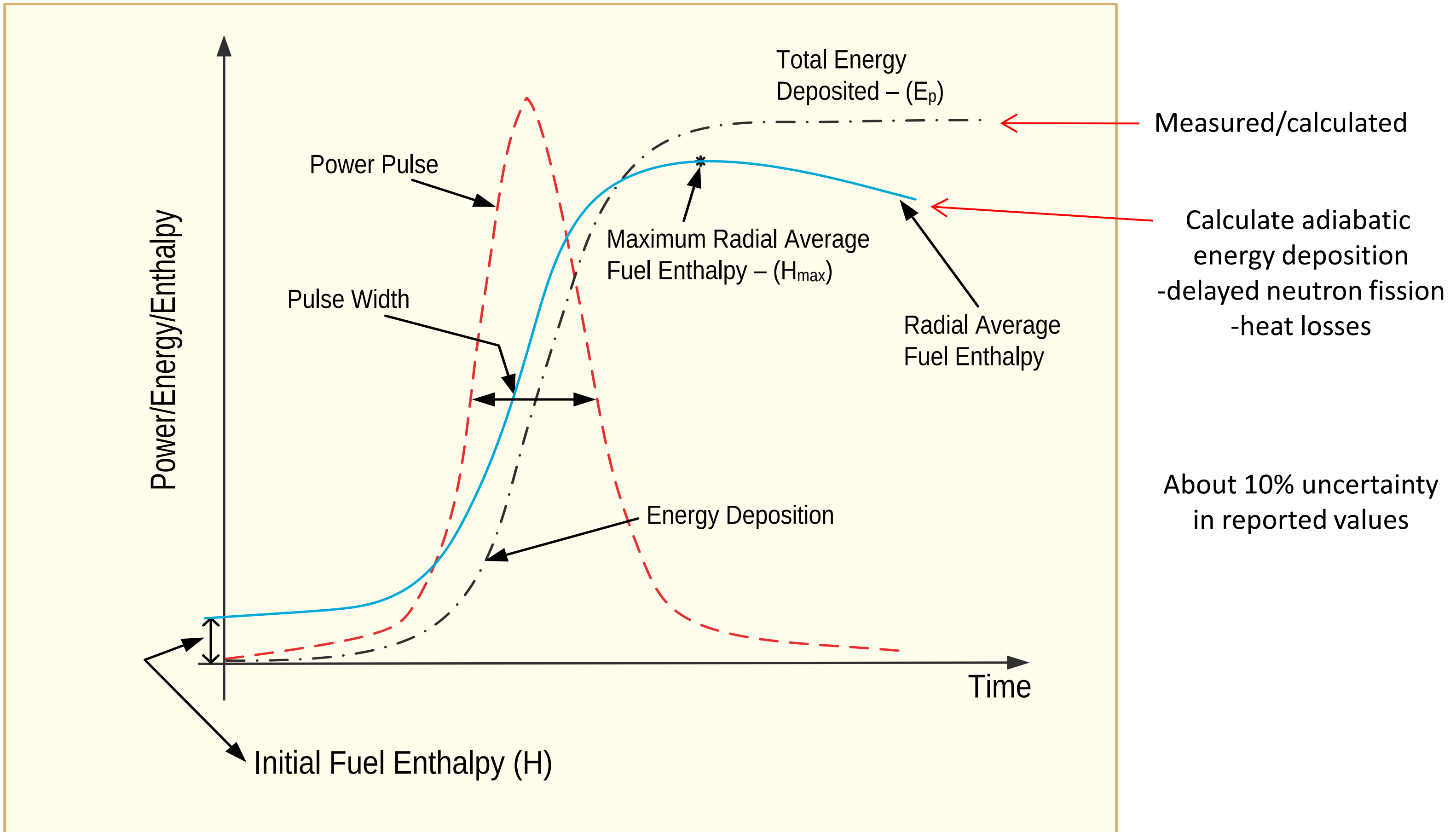


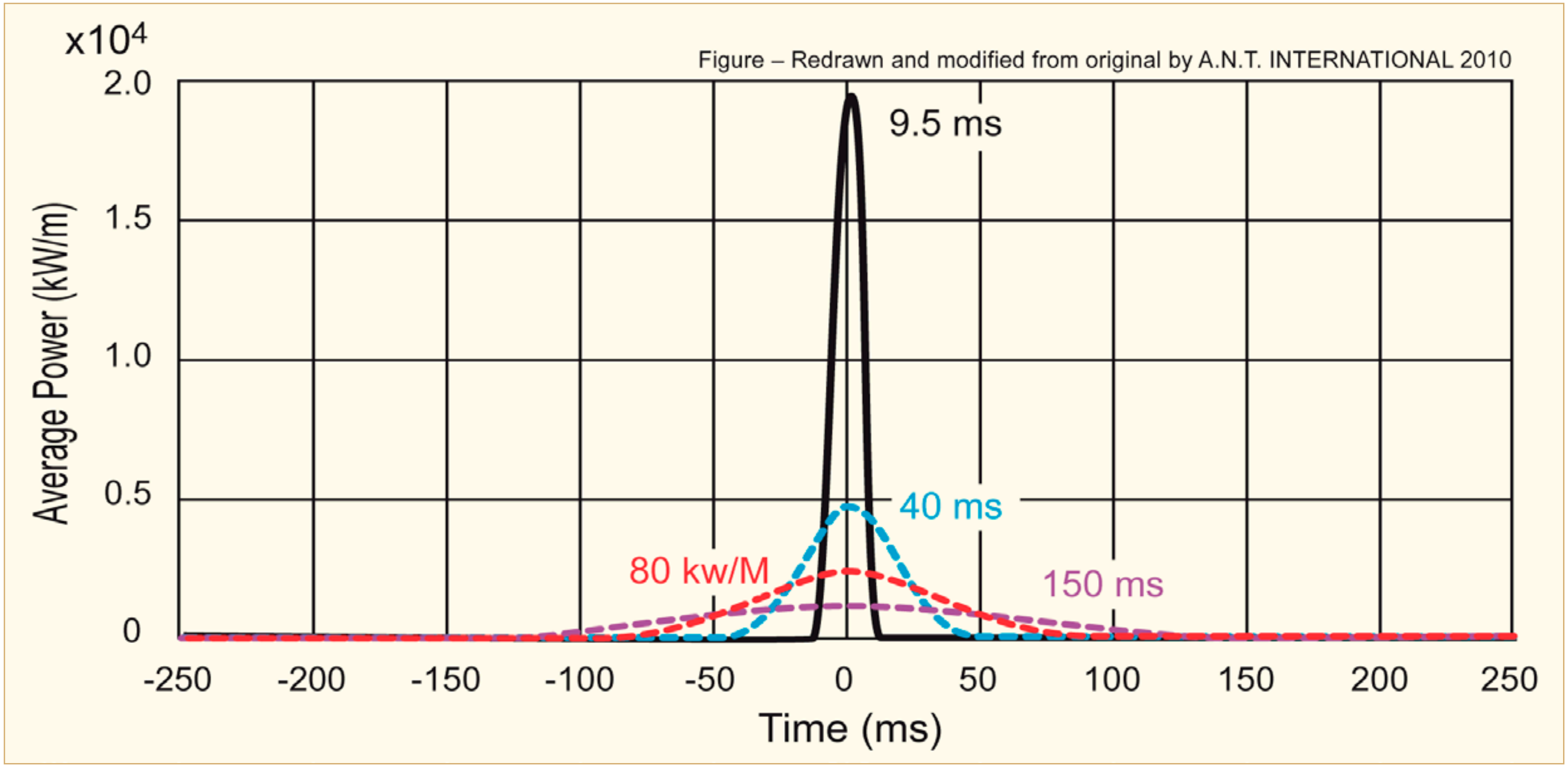
Figure - Redrawn and modified from original by A.N.T. INTERNATIONAL 2007

About 10% uncertainty in reported values

[Rudling and Patterson 2012}

4/19/2016

Pulse Characteristics



[Rudling and Patterson 2012}

[Montgomery et al, 2003]

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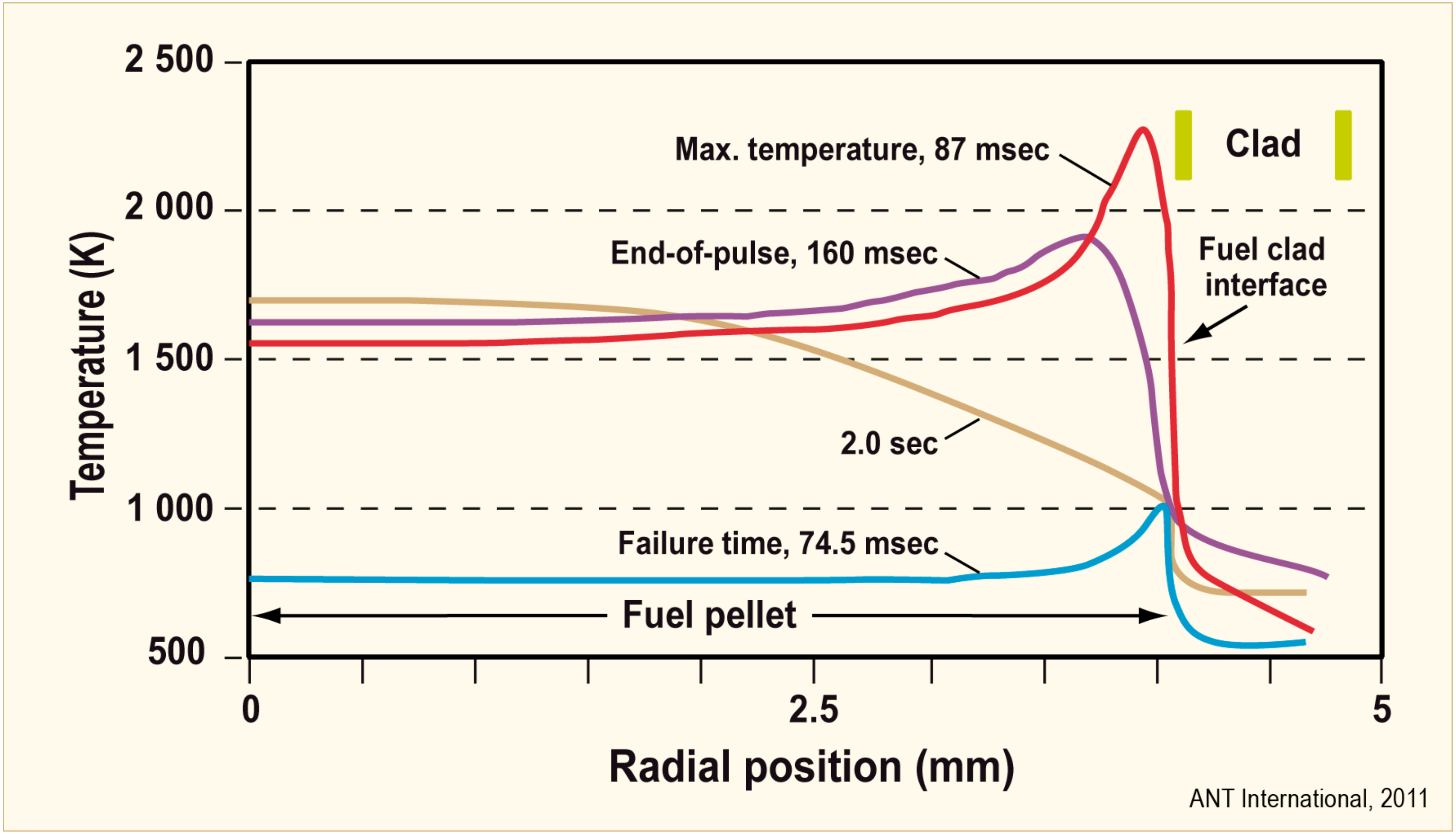
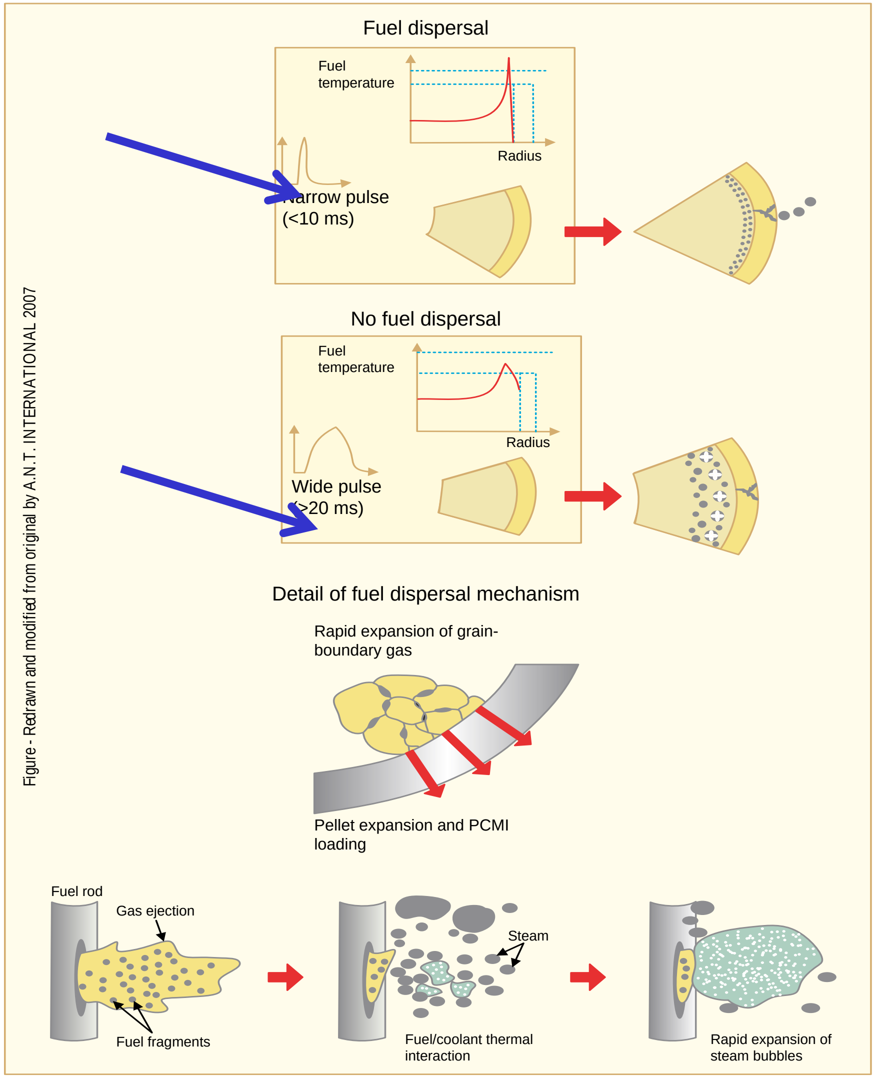
RIA



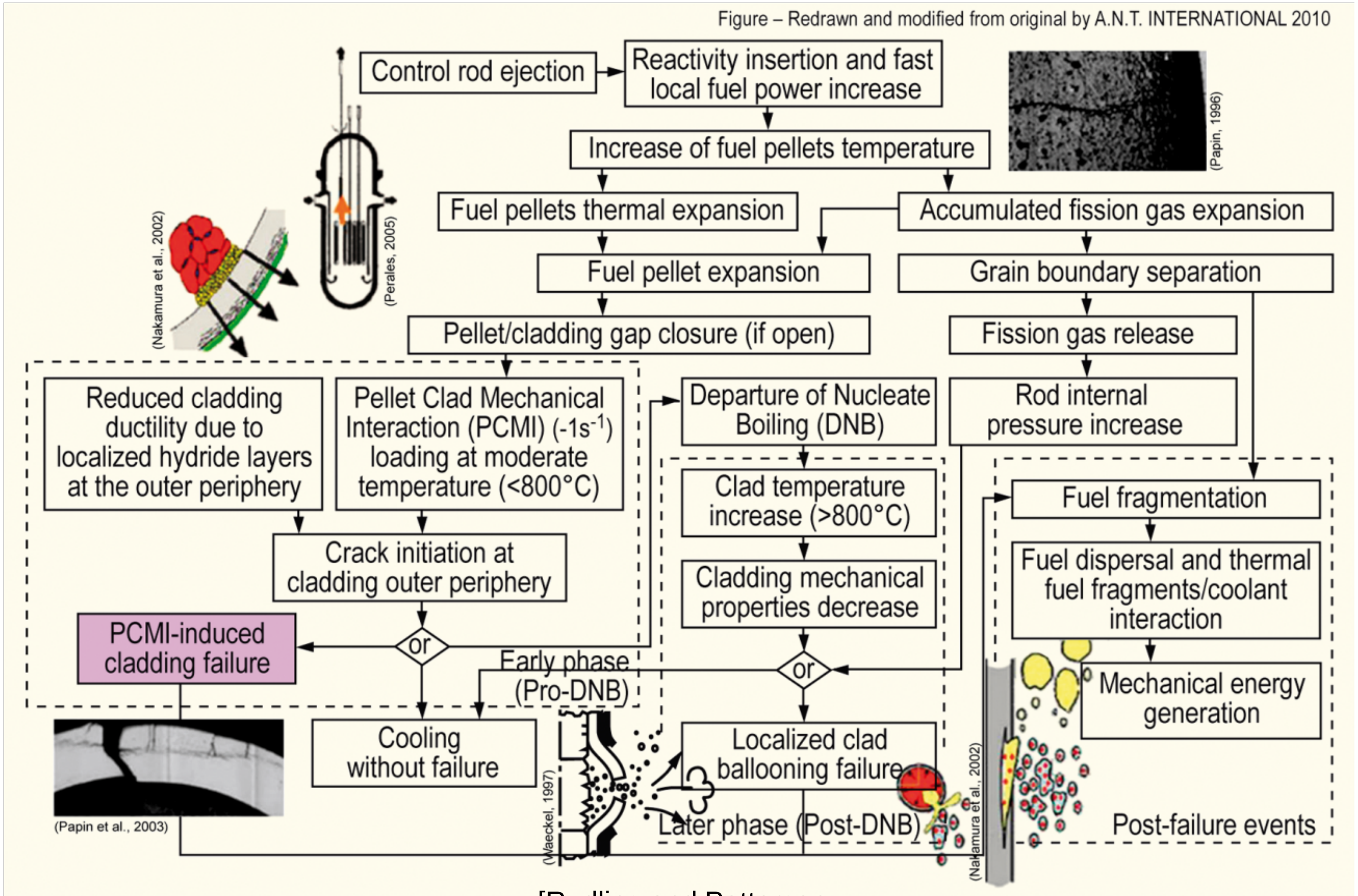
[Rudling and Patterson 2012}

[Montgomery & Rashid, 1997]



[Rudling and Patterson 2012}

RIA Effect on Fuel



[Rudling and Patterson 2012}

[Le Saux et al, 2007]

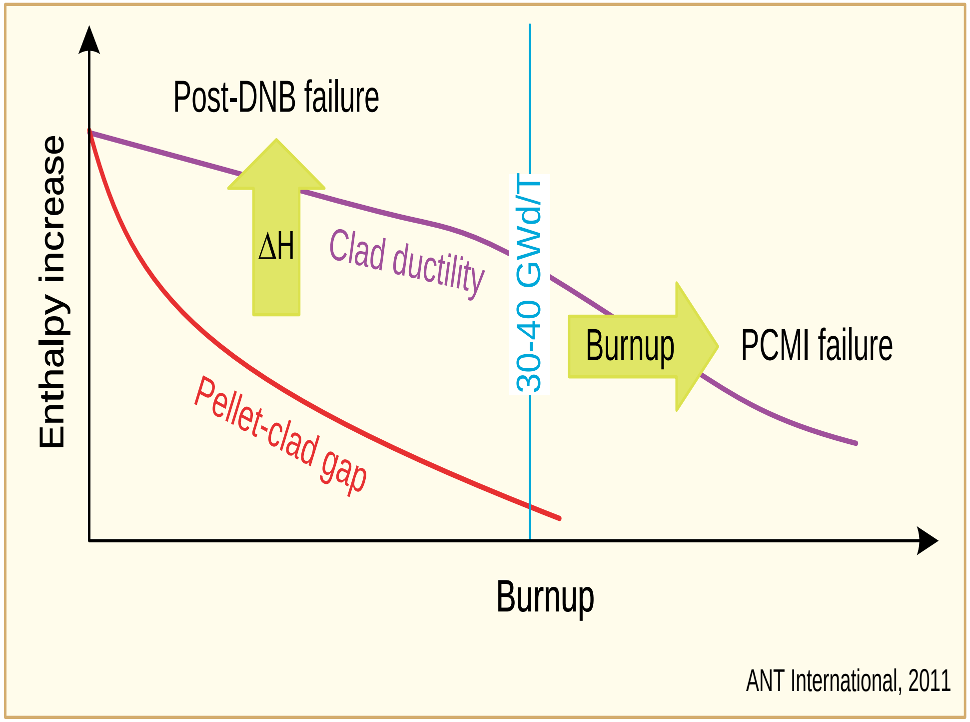
Clad failure mechanism

- During a RIA event, the fuel may survive or fail due to:
  - Post-DNB fracture of oxidised embrittled cladding at all burnup levels.
  - Melting of fuel cladding.
  - PCMI failures at higher burnups.
  - Post-DNB ballooning and creep burst at higher burnups for fuel rods with an internal overpressure.

[Rudling and Patterson 2012}

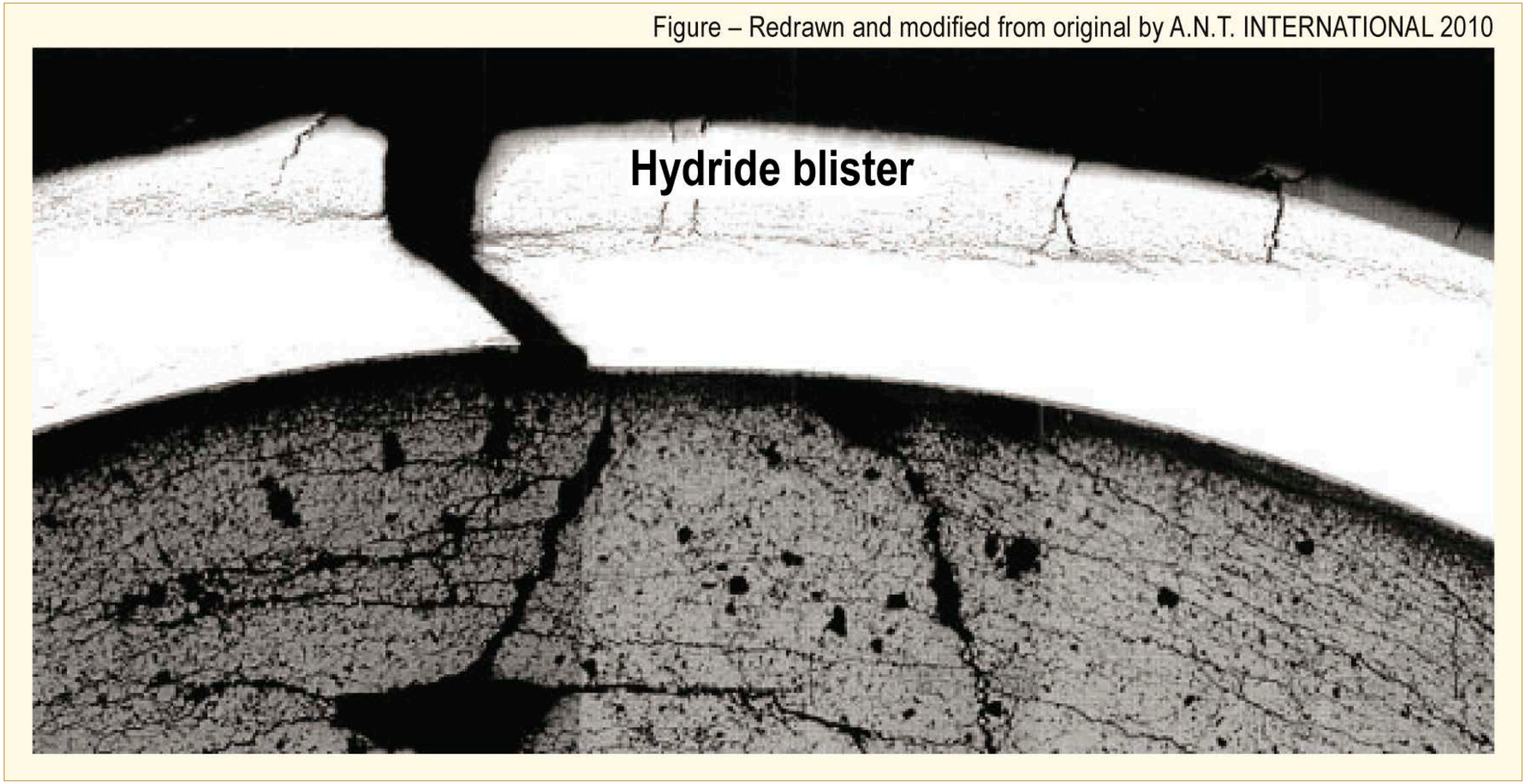
RIA

- Fuel Failure Modes
  - Low Burnups
    - Post-DNB failures
  - High Burnups
    - PCMI
    - Creep Rupture
- Fuel Dispersal



[Rudling and Patterson 2012}

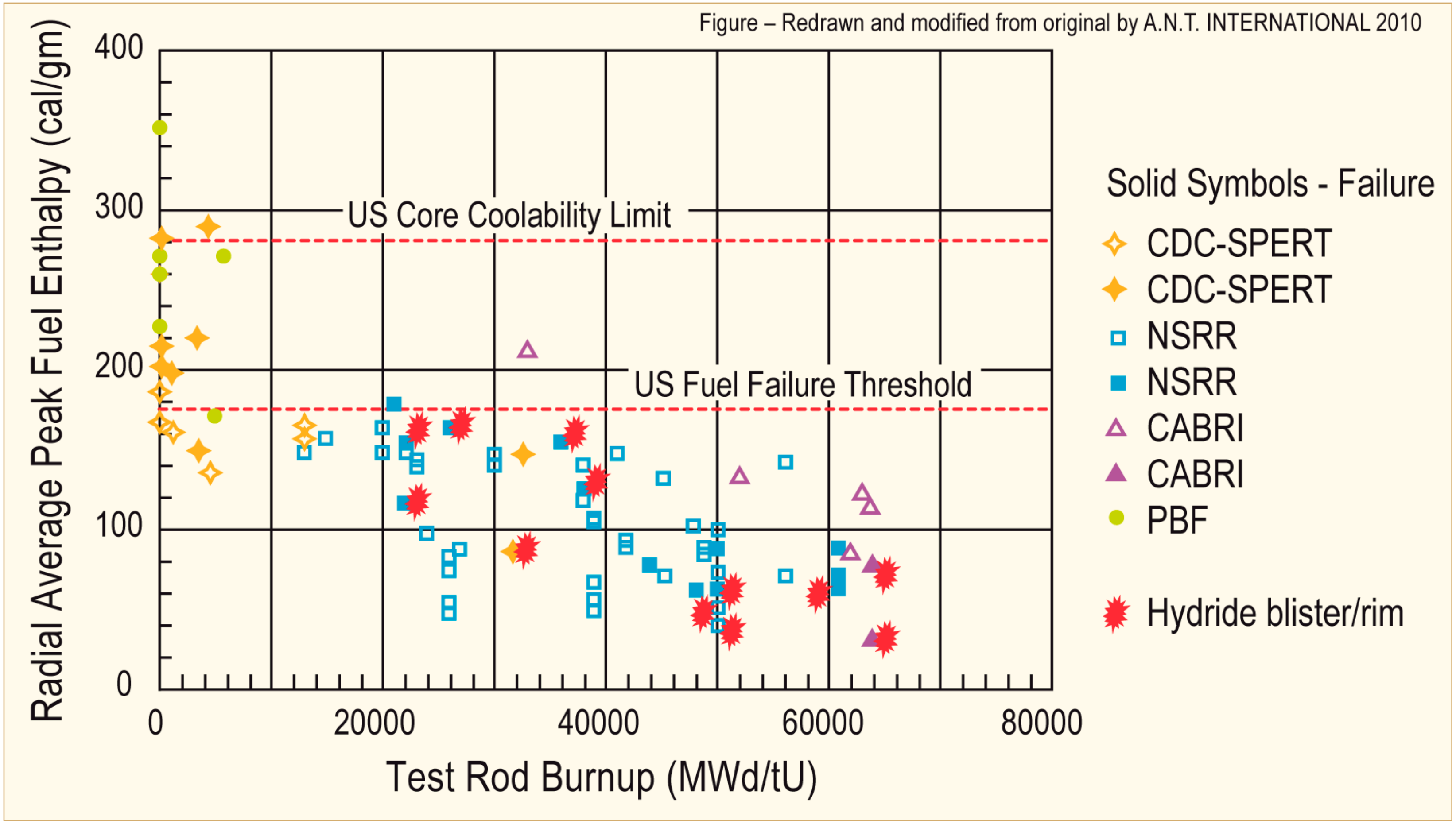
PCMI Failure



[Rudling and Patterson 2012}

[Papin et al, 2003] and [Garde et al, 1996]

BWR and PWR RIA Tests



[Rudling and Patterson 2012}

[Rudling et al, 2004/05].

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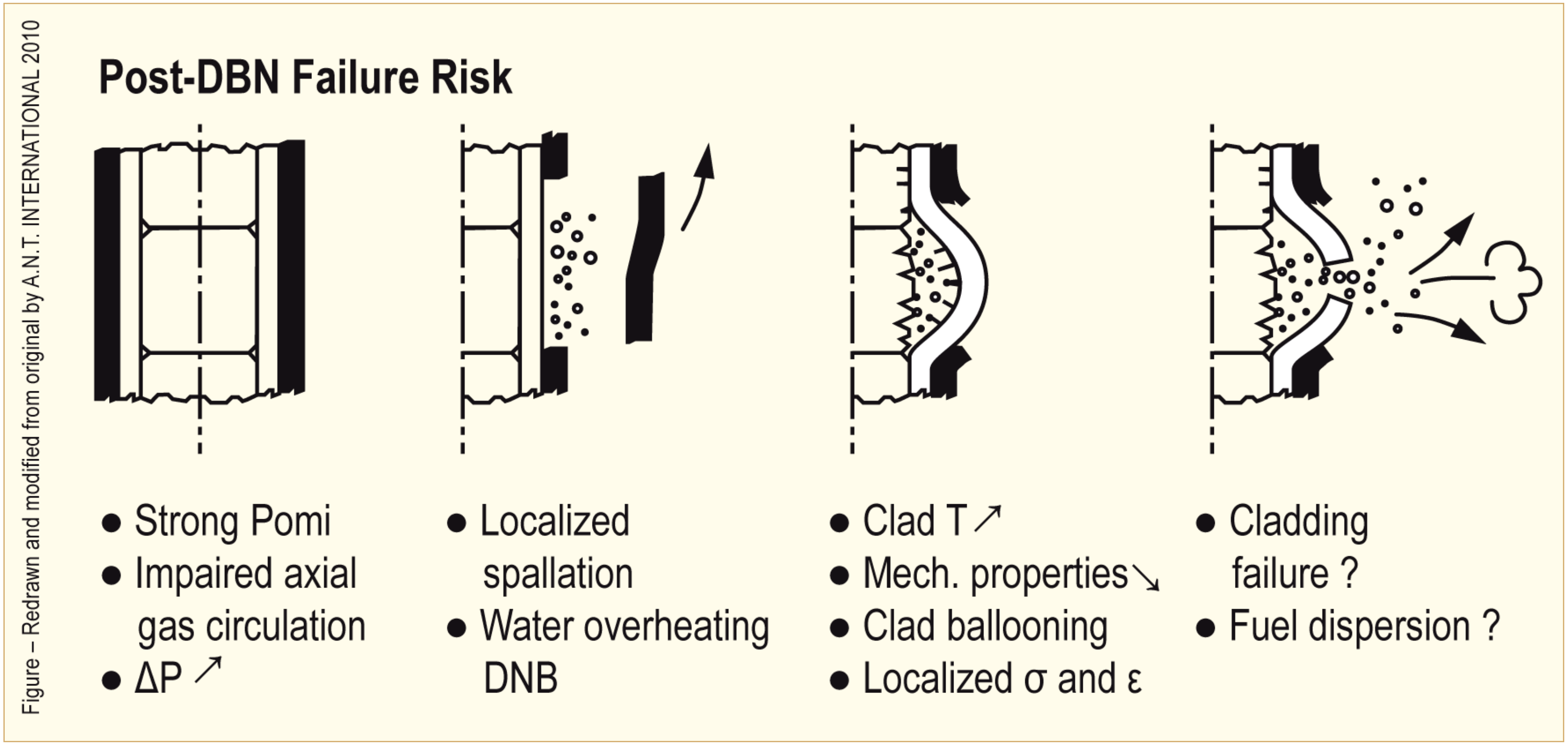
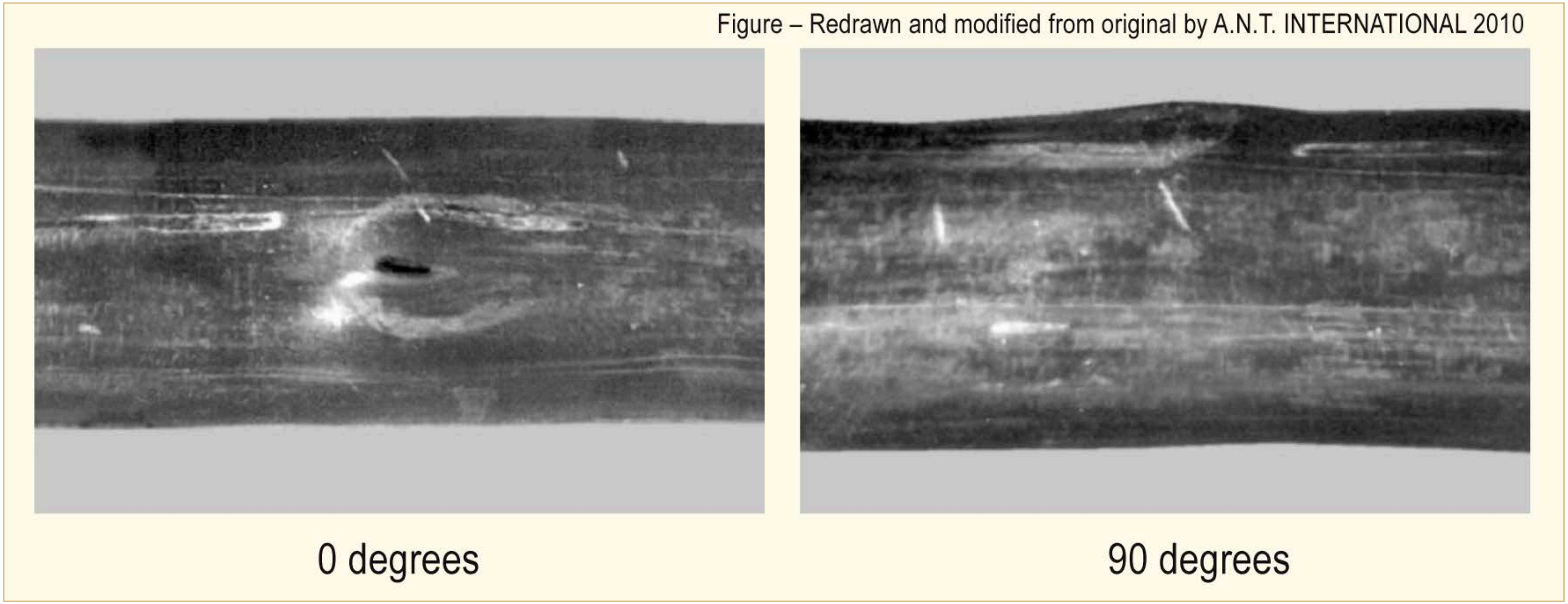
Ballooning and Creep Burst Failure



[Rudling and Patterson 2012}

[Waeckel, 1997]

Ballooning and Creep Burst Failure in VVER



[Rudling and Patterson 2012}

[Yegorova et al, 2006]

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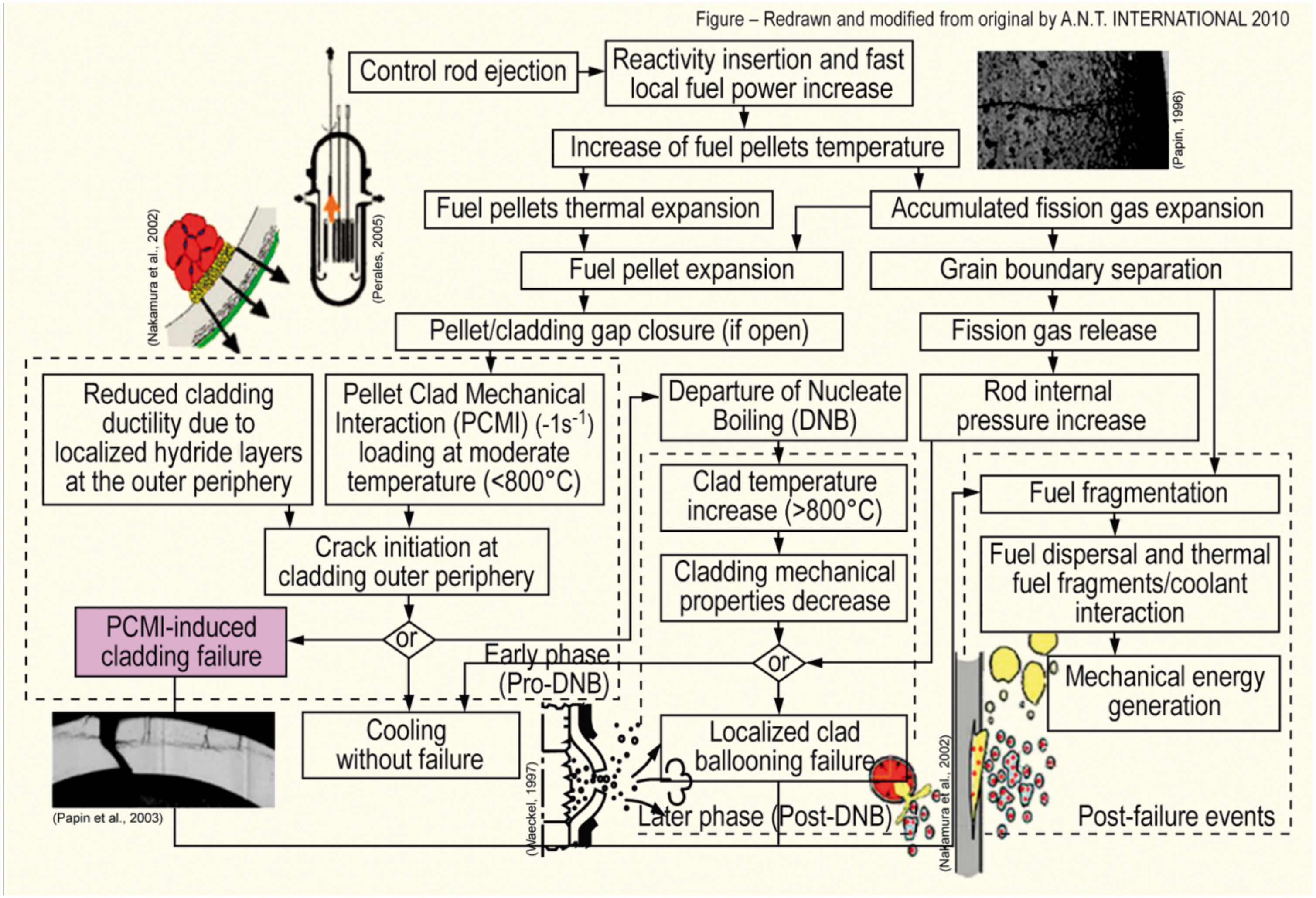
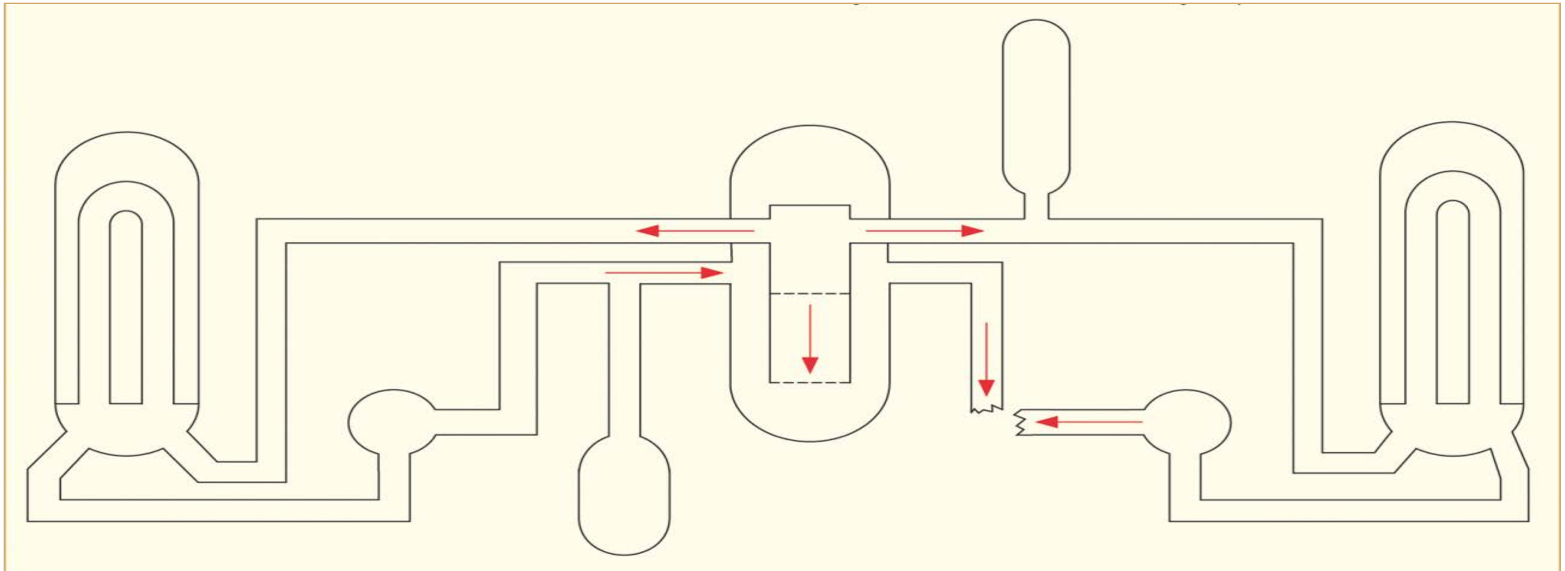
RIA Effect on Fuel



[Le Saux et al, 2007]

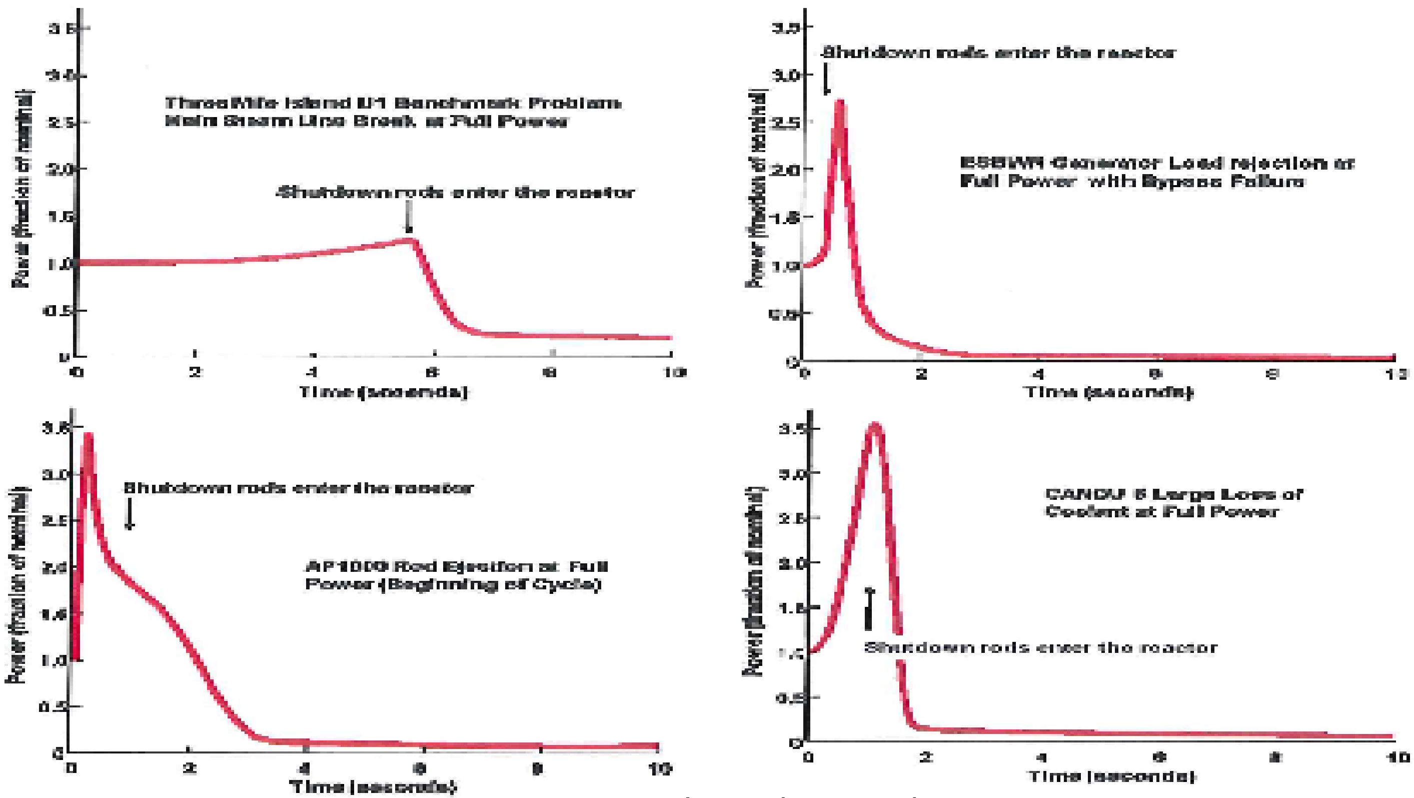
Introduction to LOCA

Cold-leg break in PWR – blowdown phase



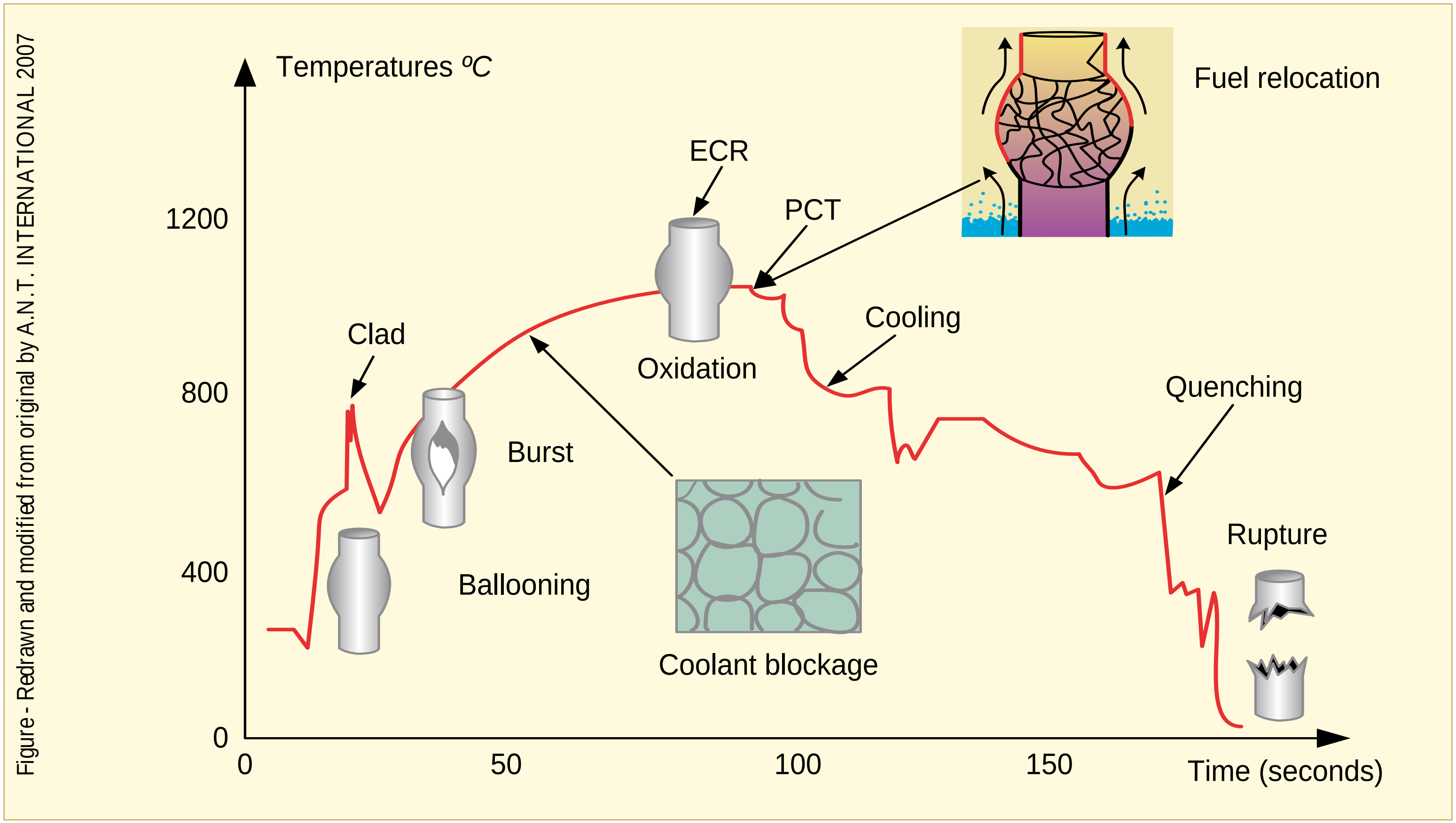
[Rudling and Patterson
2012}

Representative Shutdown Response to a LOCA



Meneley and Muzumdar

Introduction to LOCA



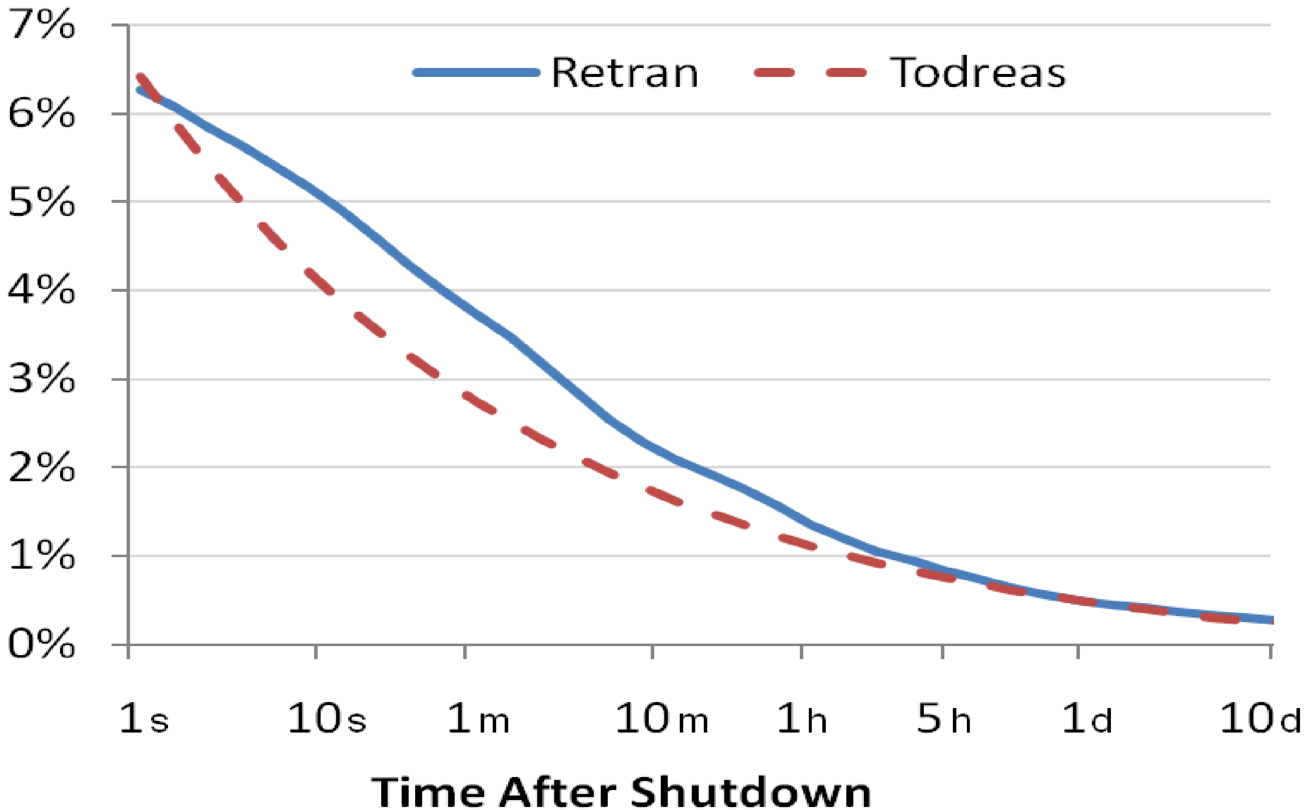
[Rudling and Patterson 2012}

LOCA, Decay Heat

- The removal of the decay heat is a significant reactor safety concern, especially shortly after normal shutdown or following a loss-of-coolant accident.
- Failure to remove decay heat may cause the reactor core temperature to rise to dangerous levels and has caused nuclear accidents, including the nuclear accidents at Three Mile Island and Fukushima I.
- The heat removal is usually achieved through several redundant and diverse systems, from which heat is removed via heat exchangers.

[Rudling and Patterson 2012}

Decay Heat



From Wikipedia, the free encyclopedia

[Rudling and Patterson 2012}

Fuel Temperature Profile

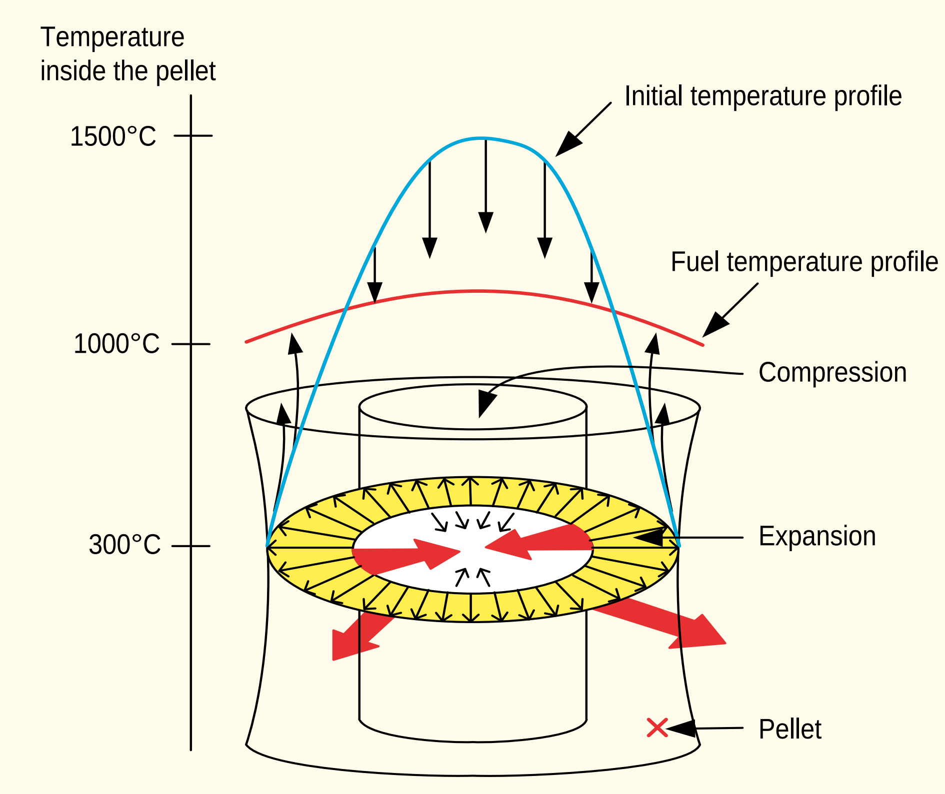


Figure - Redrawn and modified from original by A.N.T. INTERNATIONAL 2010

[Maillat et al, 2003]

[Rudling and Patterson 2012}

Summary

- Increased burnup may impact LOCA fuel performance:
  - Development of rim zone - high inventory of fission gases, contained mainly in large over-pressurized pores => TFGR may result in fuel dispersal in rods failed through burst.
  - Increased rod internal pressure (FGR) and TFGR increase ballooning, more rods failed through burst
  - The rod internal pressure at burst (the FGR-prior to the LOCA- and TFGR-during the LOCA-) constitutes the parts of the source term
  - Fuel relocation in ballooned area not a concern
  - Pre-LOCA H-pickup
    - Increase fraction of rods failed through burst (concern in Germany)
    - Less margins to fuel rod fracture through clad embrittlement
      - Increase O solubility and diffusivity in the prior-beta Zr phase => reduce allowable LOCA ECR
  - Fuel-clad bonding may increase embrittlement => may reduce allowable LOCA ECR

[Rudling and Patterson 2012}

Fuel Performance During LOCA

- During base irradiation and during the LOCA event several changes of the UO_2 pellet and within the fuel rod can occur which are significant for LOCA performance. These are:
  - FGR from the pellet during base irradiation increases the inner pressure, which affects the ballooning behaviour of the cladding and the probability and time of a burst during LOCA.
  - Degradation of the thermal conductivity of the UO_2 pellet resulting in increased fuel temperature (at constant rod power) which in turn will increase the FGR.
  - A high burnup rim zone is formed at the pellet periphery during the base irradiation at high burnups. This high burnup rim zone has a high inventory of fission gases, contained mainly in large over-pressurized pores. During a LOCA the outer rim experiences a temperature increase, e.g. from 400 to 1100 ° C, which may lead to a pronounced transient fission gar release TFGR during the LOCA.
  - The rod internal pressure at burst (the FGR-prior to the LOCA- and TFGR-during the LOCA-) constitutes the parts of the source term
  - Pellet-Clad Bonding

[Rudling and Patterson 2012}

Ballooning and Burst

- The basic parameters controlling fuel clad deformation and ballooning are:
  - Stress,
  - temperature and
  - creep strength, which is affected by oxidation, grain size and anisotropy.
- Burst leads to release of noble gases, iodine, caesium and other species released by the fuel (source term)
- Burst is facilitated by Hydrogen

[Rudling and Patterson 2012}

Effect of Decay Heat

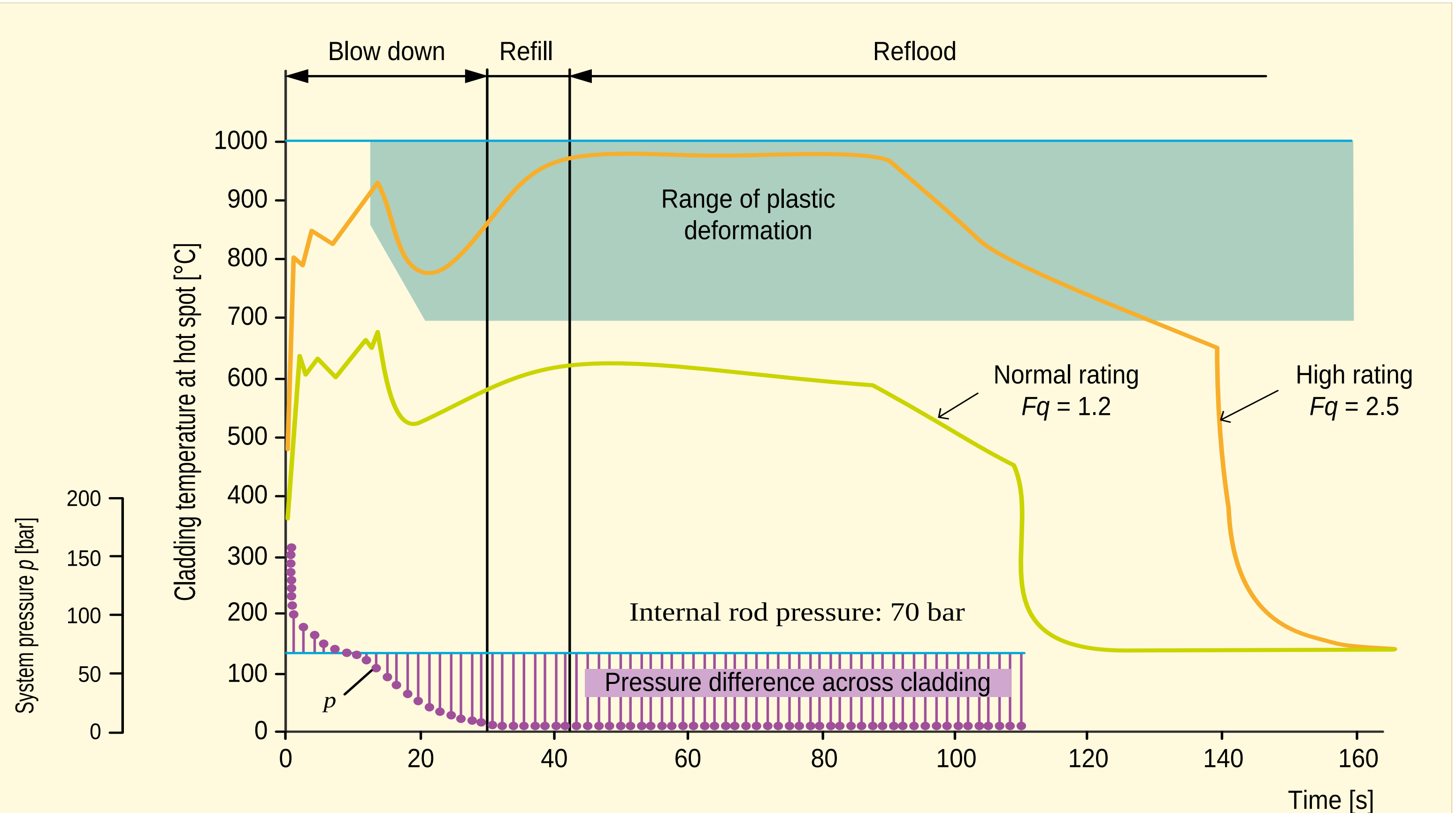
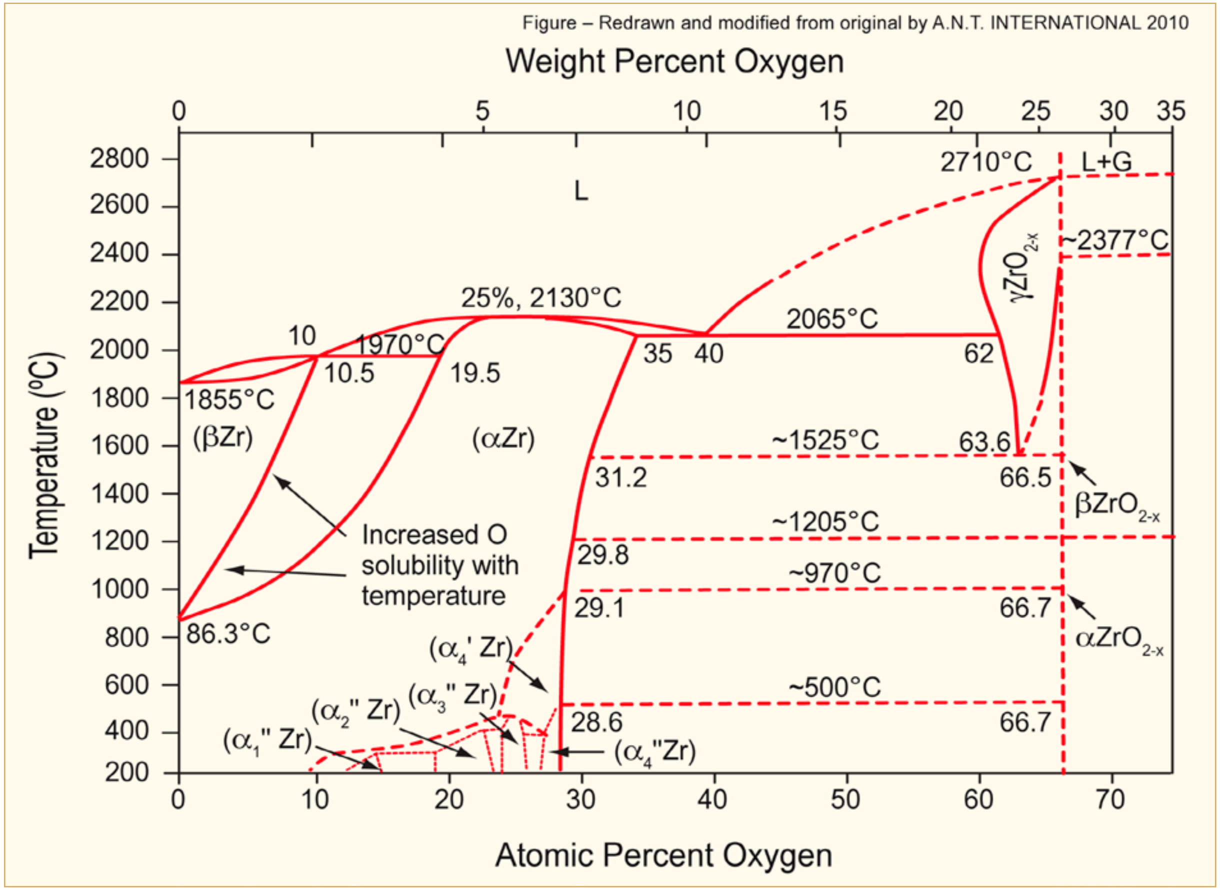


Figure - Redrawn and modified from original by A.N.T. INTERNATIONAL 2007

[Rudling and Patterson 2012}

Zr-O phase diagram



[Abriata et al, 1986]

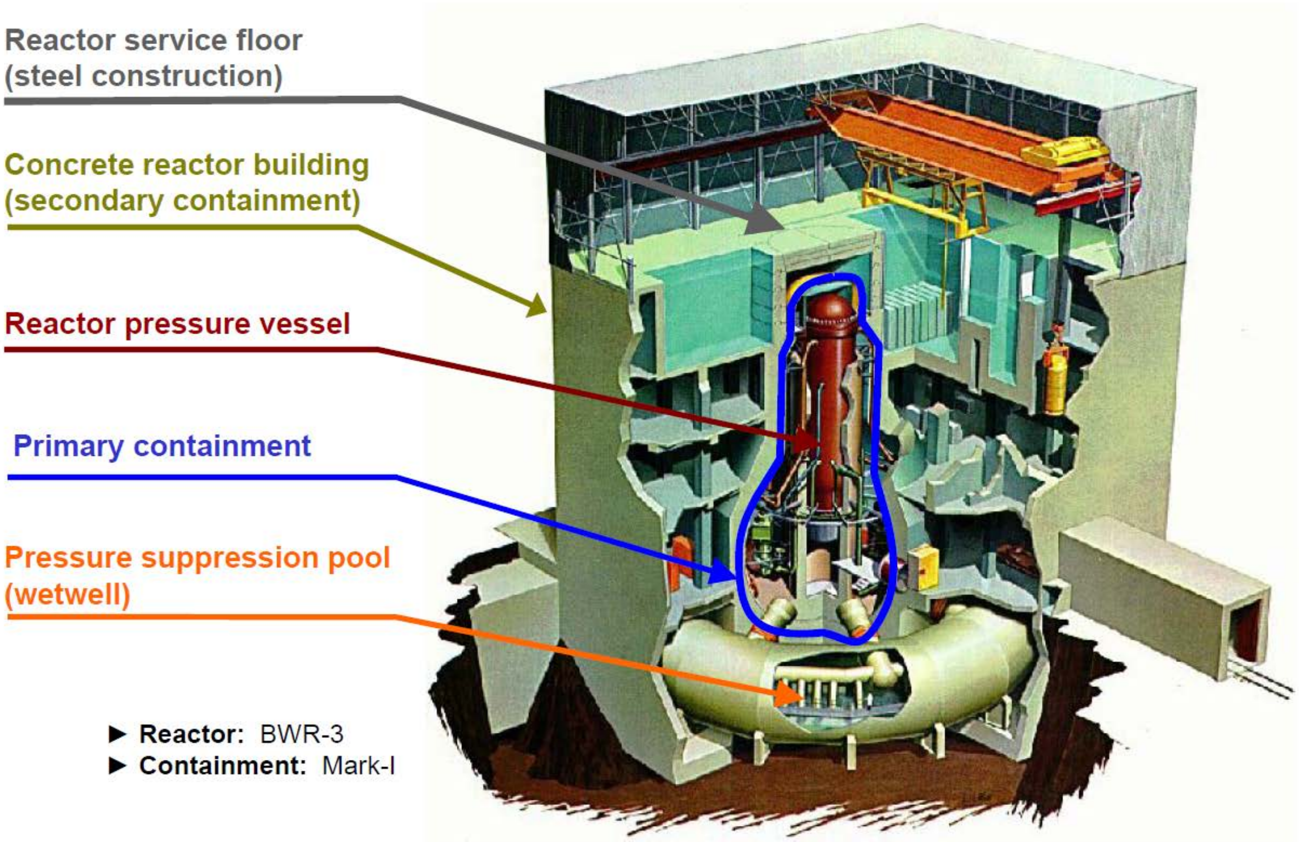
[Rudling and Patterson 2012}

4/19/2016

In-vessel Severe Accident Progression

- Thermal-hydraulic and fuel rod degradation
- Hydrogen generation
- Degradation of core structure
- In-vessel fuel-coolant interaction
- Oxide/metal separation
- In-vessel debris formation
- RPV failure w/ or w/o high pressure melt ejection

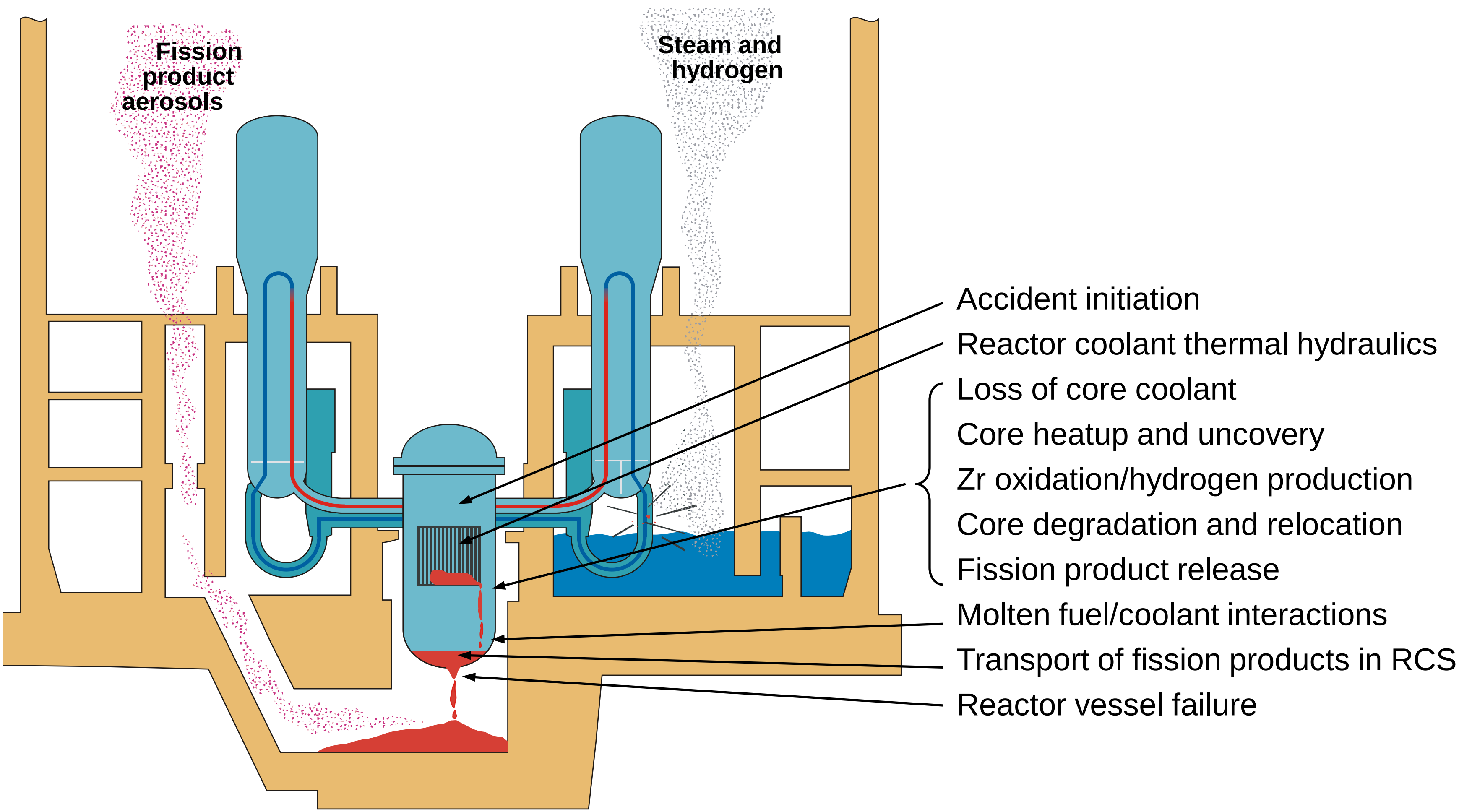
Design of Fukushima-Daiichi-1 Provides Primary Containment Around Vessel



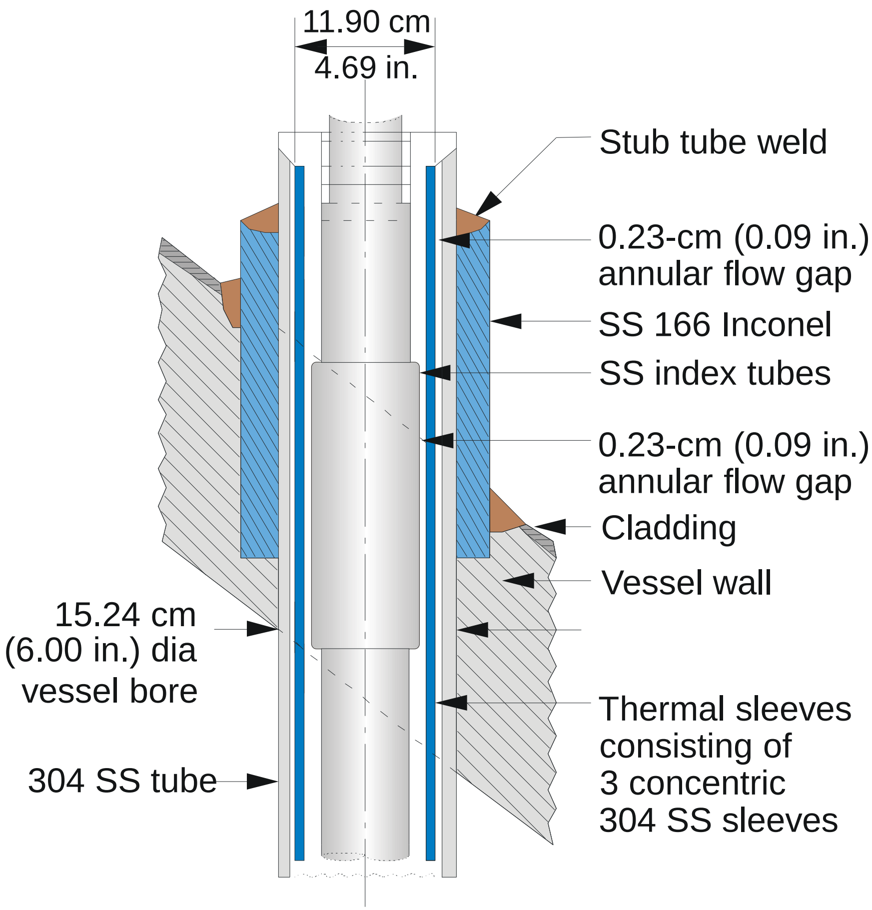
- **Reactor:** BWR-3
- **Containment:** Mark-I

Sources: NRC, General Electric, www.nucleartourist.com

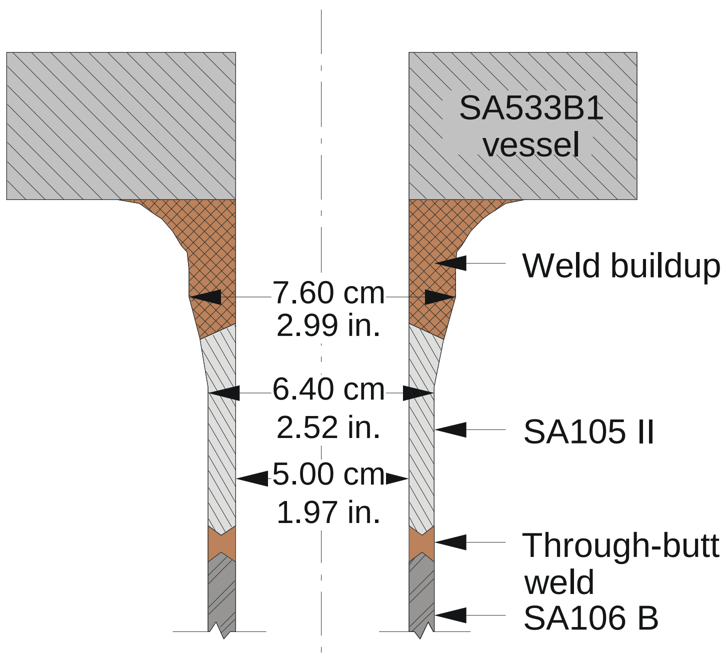
Range of Melt Progression Phenomena Affects Vessel Failure Mode and Timing For All Reactor Designs



BWR Vessels Also Penetrated by CRD Assemblies and Drain Line

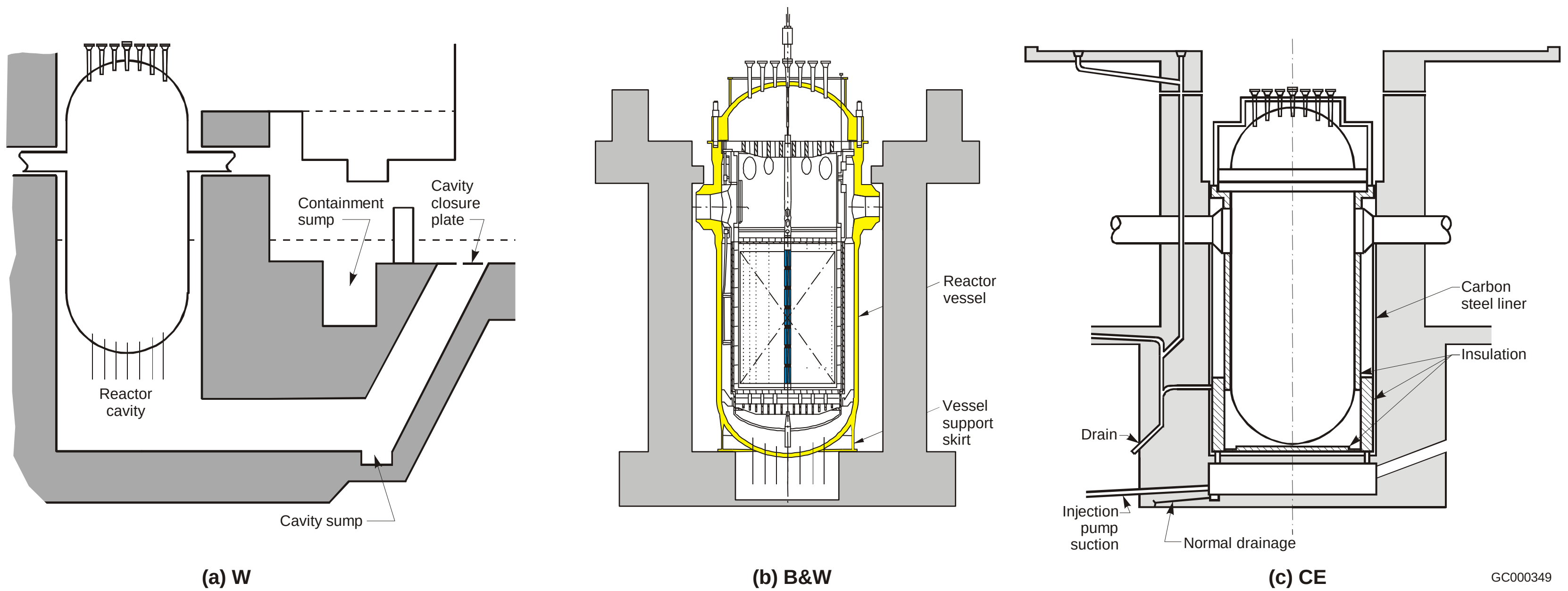


Typical GE CRD Assembly Penetration

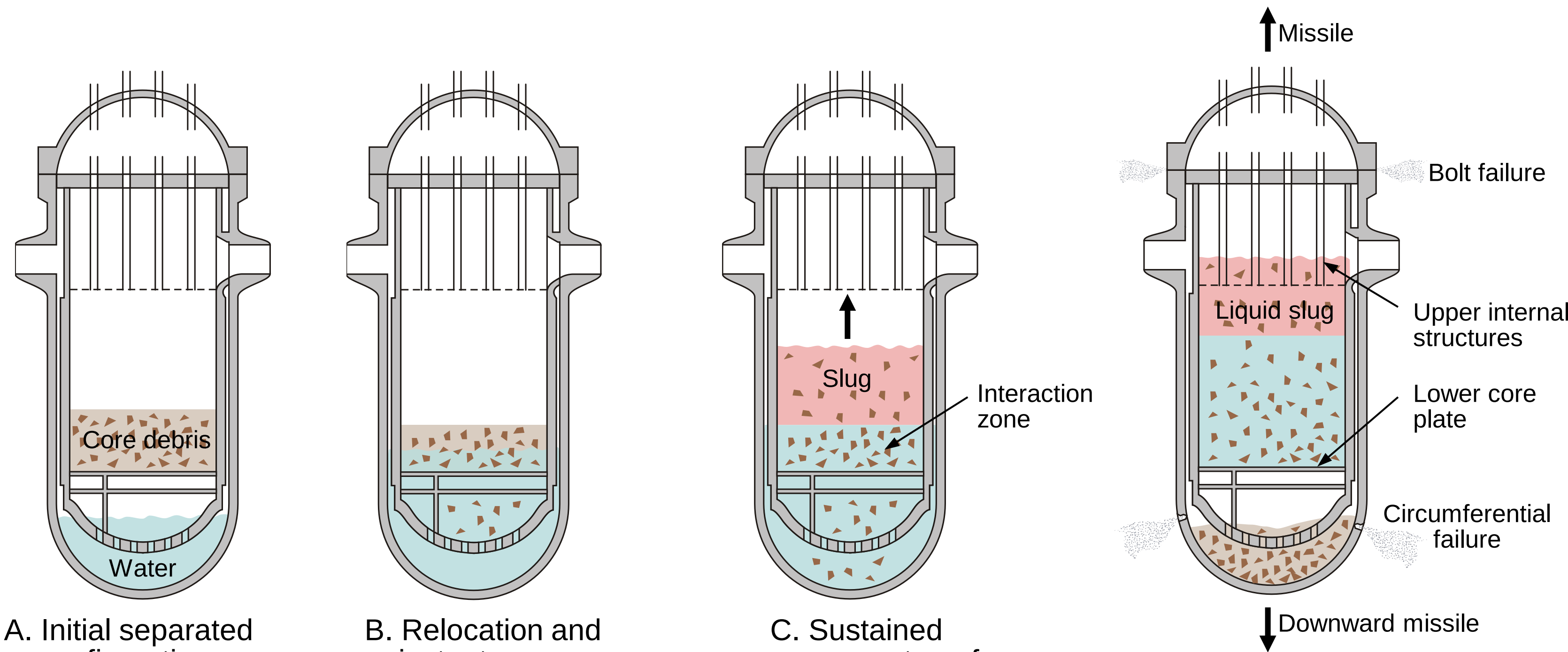


Typical GE Drain Line Nozzle Penetration

Insulation, Supports, and Cavities for Lower Heads Differ



In-vessel Steam Explosion Issues



A. Initial separated configuration

B. Relocation and instantaneous mixing

C. Sustained energy transfer and slug acceleration

D. Slug impact

- Will in-vessel fuel/water interactions cause energetic reactions?
- Are such reactions sufficient to accelerate a slug that fails the vessel and/or create a missile that causes early containment failure?

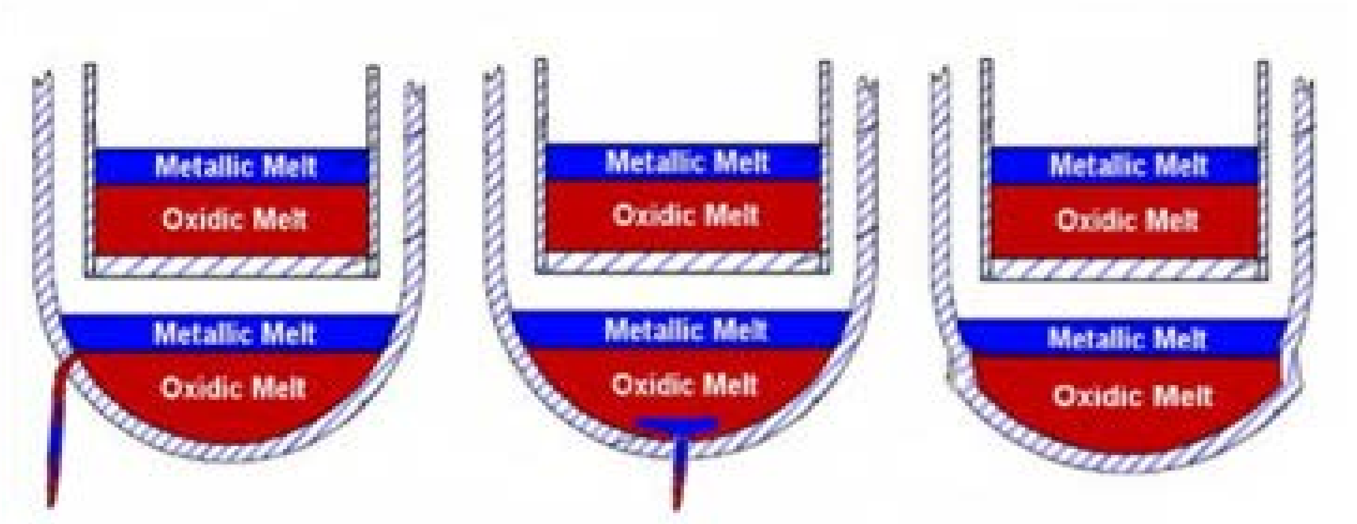
Additional Data obtained since NUREG-1150 Evaluations

- Issues so controversial at time NUREG-1150 completed, expert panel refused to address.
- SNL staff internally developed distribution based on opinions expressed by Steam Explosion Review Group (SERG) in NUREG-1116.
- More recent experimental results indicate:
  - At low pressure [< 0.1 MPa (14.7 psi)], limited fuel mass expected to participate in energetic FCI
  - At higher pressures [> 1 MPa (147 psi)], explosion difficult to trigger
- All eleven SERG-2 experts estimated low probabilities for energetic in-vessel steam explosion

In-vessel Core Debris Coolability

- **Initial conditions for stabilization are subject to the uncertainties of in-vessel melt progression**

- **Event progression through RPV failure represents the largest source of uncertainty for SA mitigation**



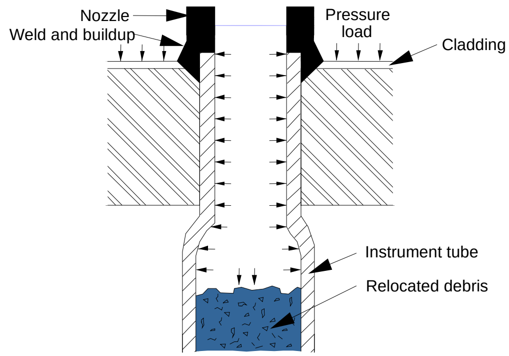
Large Uncertainties Associated with Early Methods for Quantifying Vessel Lower Head Failure Potential

- Codes typically assumed early penetration failure (with subsequent depressurization) or global vessel failure based on temperature criterion
- NUREG-1150 developed aggregate distributions derived from uncertainty models provided by three experts
  - Several cases considered (varied pressure, availability of upper head injection, and accumulator injection)
  - Expert review based on calculation results, TMI-2 data, and severe fuel damage test data
  - Wide variation in expert opinion
- Singled out as area with major uncertainty in Special Committee Review for NUREG-1150 due to importance of vessel failure mode and timing on subsequent accident progression.

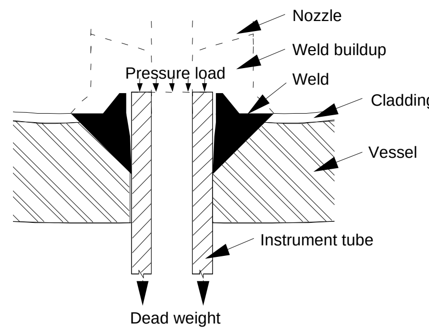
Several research programs provide data and improved tools for predicting vessel failure.

Program	Focus	Heat Loads	Vessel	Pressure
NRC Lower Head Failure Program (INL)	Models and material data for evaluating vessel and penetration failure	Wide range of well-defined localized and global heat loads	Wide range (with and without penetrations)	Wide range (0.1 to 15 MPa / 14.5-2175 psi)
OECD TMI-2 Vessel Investigation Program	Data to assess tools for predicting vessel and penetration failure	Localized and global heat loads (but not well defined)	B&W PWR SS-lined SA533 vessel with penetrations	High (3-15 MPa / 435-2175 psi)
NRC and OECD Lower Head Failure Tests (SNL)	Failure data for well-defined heat loads	Localized and global heat loads	1/5th scale SA533 (with and without penetrations); OLHF - ½ scale wall	High (2-10 MPa / 30 -1450 psi)

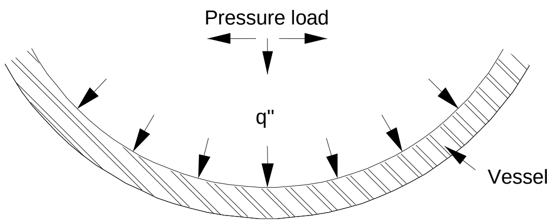
INL Lower Head Failure Program First Comprehensive Study of Vessel Failure Mechanisms



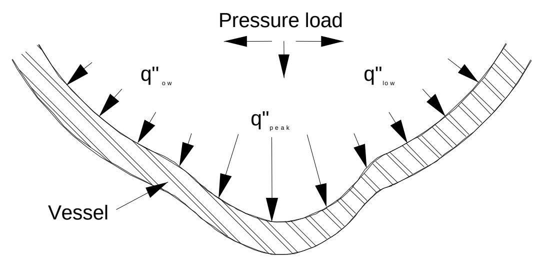
Tube rupture



Weld failure/tube ejection



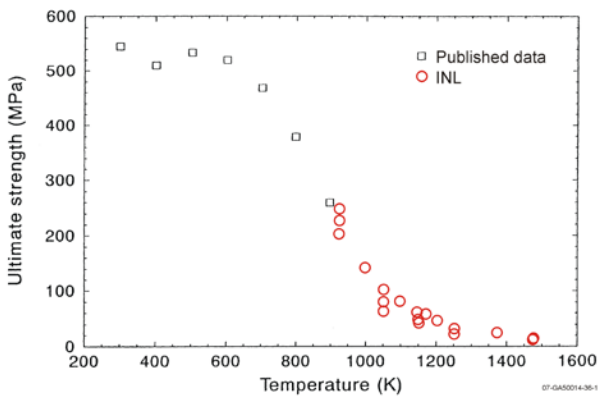
Global vessel failure



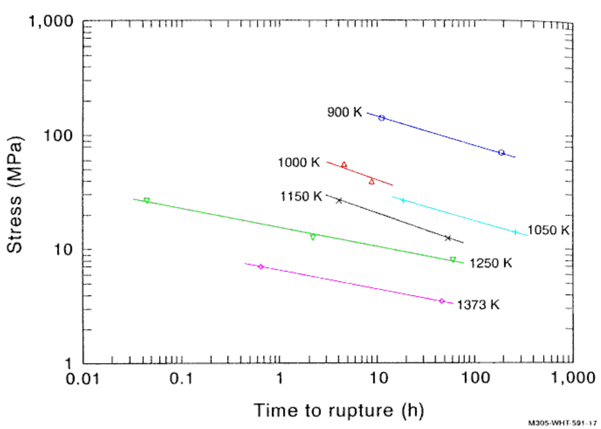
Localized vessel failure

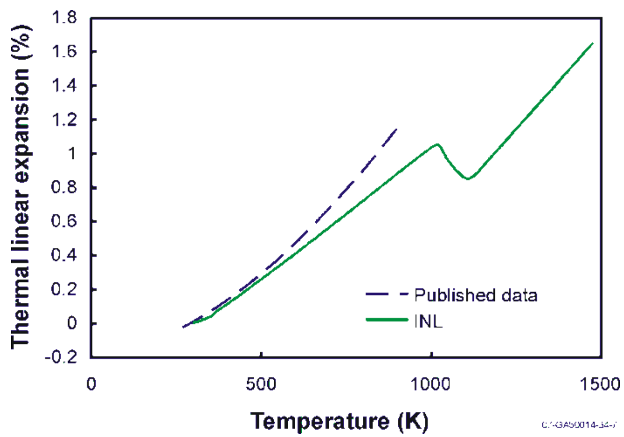
- **Identified and developed models for each failure mechanism**
- **Obtained high temperature creep and tensile data for vessel and penetration materials**
- **Applied methods to obtain insights for range of accident conditions and reactor designs**

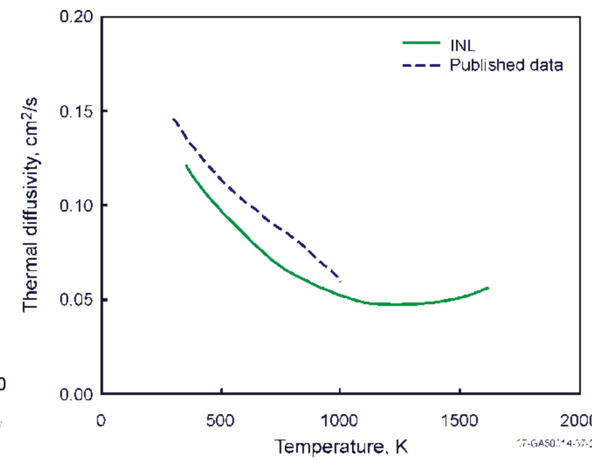
High-temperature Tensile and Creep Data Obtained for Vessel and Penetration Materials



- **Data for penetration materials (SS304, Inconel 600 and SA105/106) also available**
- **For SA533B1:**
  - **Significant reduction in SA533B1 yield strength at temperatures above 1000 K**
  - **Stress versus time to rupture only moderately sensitive to phase transformation**
  - **Higher temperature thermal diffusivity and thermal expansion data smaller than extrapolated published values**
- ***Calculations needed to assess impact of new data!***







Summary

- Research results suggest energetic in-vessel steam explosions not important from risk perspective
- Recent assessments and experiments provide key insights about potential for other failure modes:
  - Importance of RCS pressure and relocated debris mass, composition, decay heat distribution and melt fraction, and vessel material and fabrication
  - Experimental data and analyses suggest localized and global vessel failures more likely than penetration failures at high pressures

Debris Heat Loads Impact Quantification of Several Events

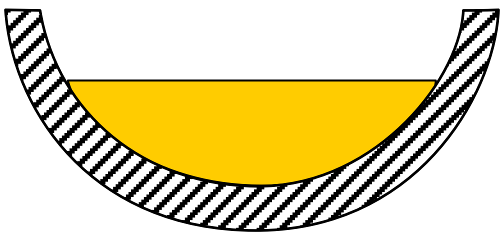
- Debris heat loads impact mode and timing of vessel failure and potential for containment failure.
- Information needed to address key questions:
  – What type of debris endstates may occur?
  – How does debris endstate affect vessel heat loads?
  – What phenomena affect debris coolability?

Debris Heat Load Considered by NUREG-1150 Experts Evaluating Vessel Failure Mode

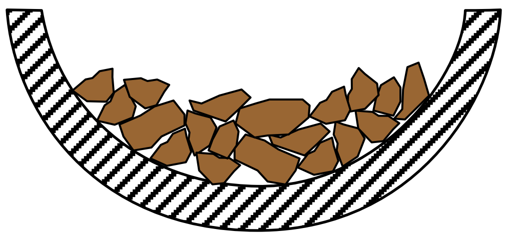
- Three experts asked to evaluate several cases (medium to high pressure, with and without injection)
- Available code calculations, TMI-2 post-accident examinations, and severe fuel damage tests used to derive
  – mass ejection rate
  – melt temperature
  – oxidation fraction of released melt
  – molten fraction of released melt
- Wide variation in expert opinion (due to limited data).

Debris Endstate Configurations Key in Assessing Vessel Response

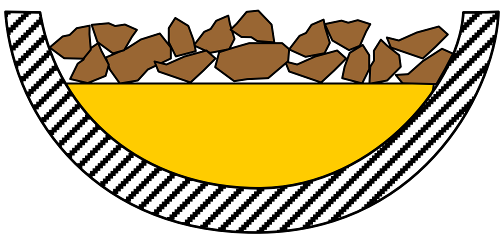
Molten pool



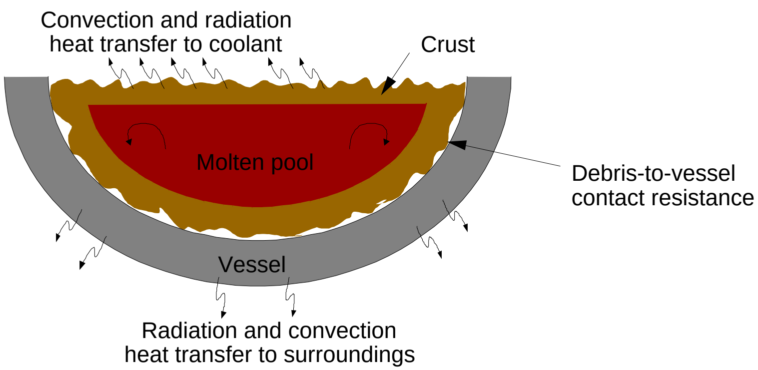
Fragmented rubble



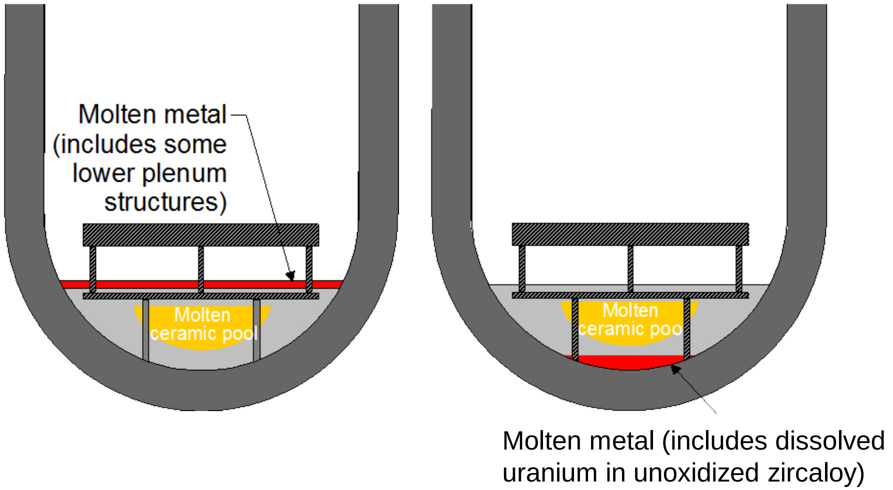
Molten pool beneath fragmented rubble



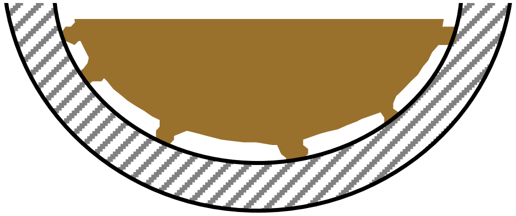
Homogeneous



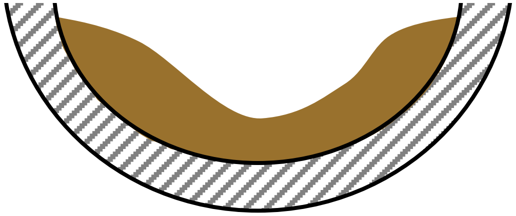
Stratified



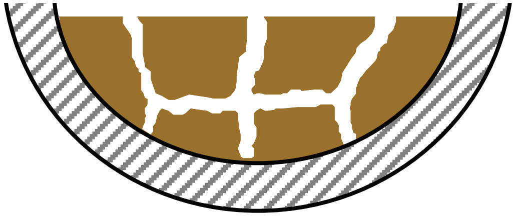
Enhanced Cooling Possible As Relocated Core Material Solidifies



Intermittent debris-to-vessel gap



Enhanced upper surface corium surface area

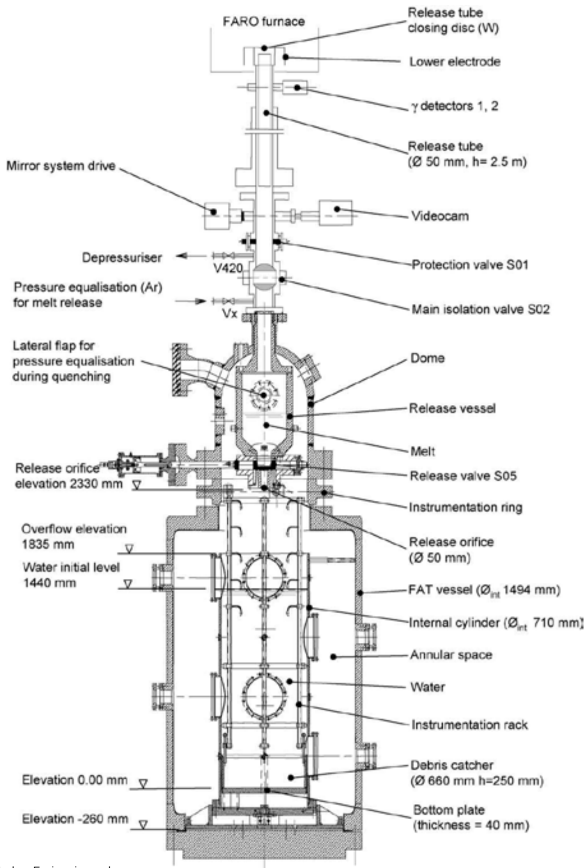


Interconnected corium cracks

Wide Range of Investigations Provide Insights about Heat Load from Relocated Corium

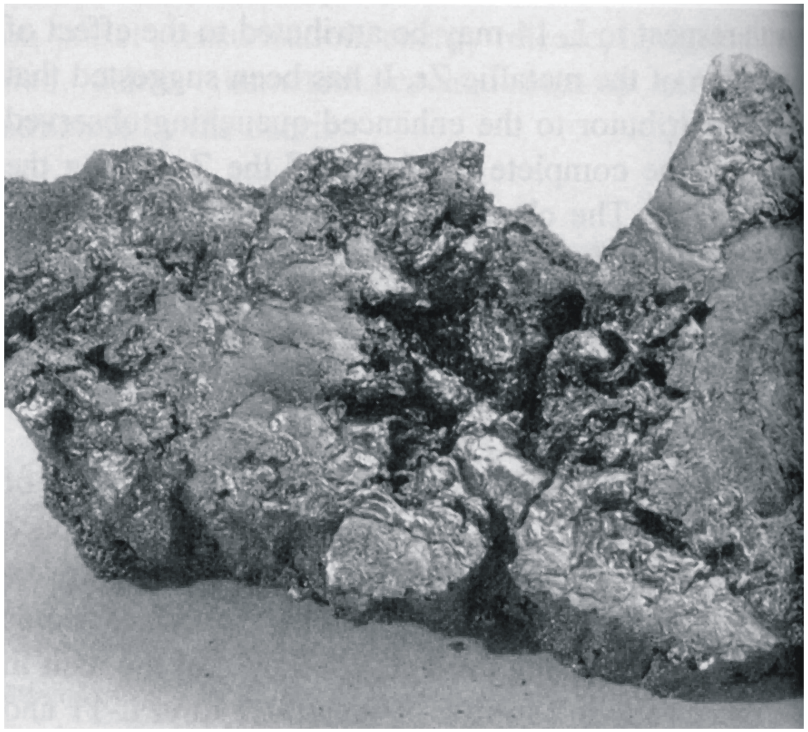
Program	Insight	Materials			Pressure
		Corium	Vessel	Coolant	
RRC/OECD RASPLAV	Natural convection heat fluxes, corium stratification	UO_2, ZrO_2, Zr, C, FeO, LaO	W/Ta protected graphite in slice geometry	None	Low (0.1 MPa / 14.7 psi)
JRC/ISPRA FARO	Melt/water interactions, debris cooling, morphology, interactions with structures	UO_2, ZrO_2, Zr,	Flat plate	Water	High (0.5 to 5 MPa / 72.5-725 psi)
OECD TMI-2 Vessel Investigation Program	Debris cooling, morphology, and interactions with structures	UO_2, ZrO_2, FeO_2, Ag, SS-304	SS-lined carbon steel vessel with penetrations	Water	High (3-15 MPa / 435-2175 psi)

FARO Provides Insights about Relocating Debris Initial Condition, Morphology, and Heat Transfer



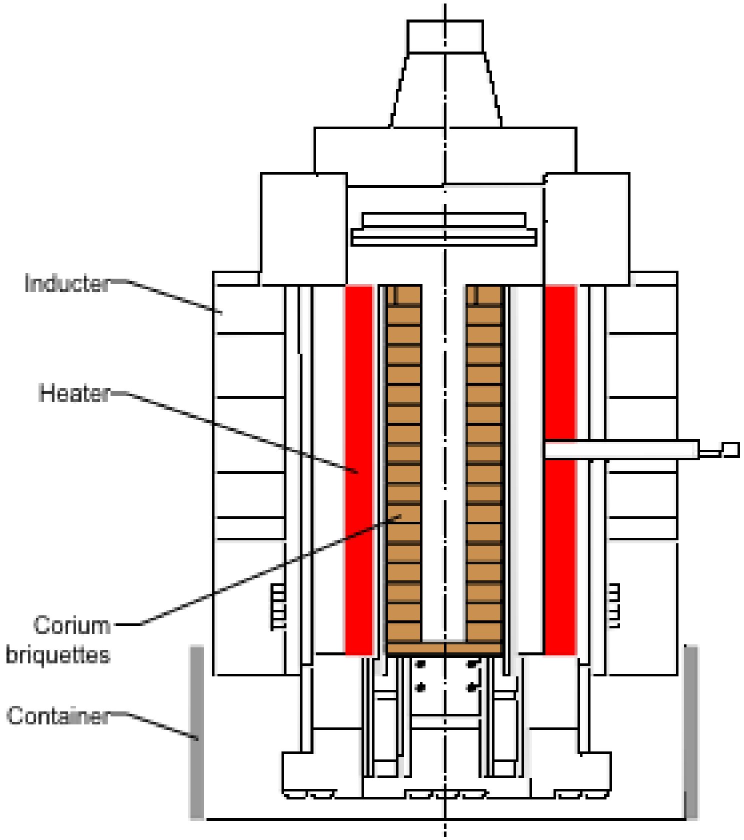
Nuclear Engineering and Design 236 (2006)

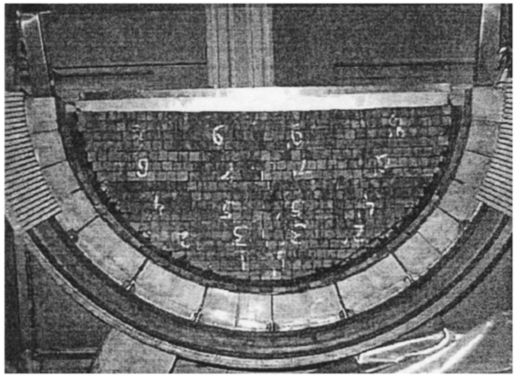
FARO



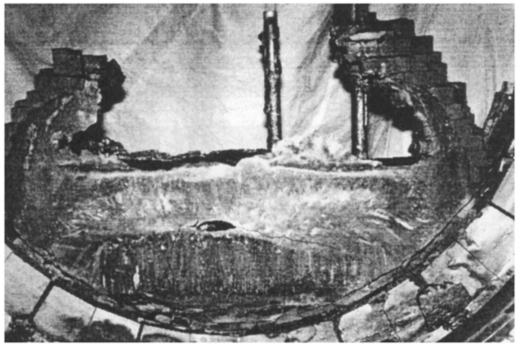
- **Furrows observed in relocated debris**
- **Intermittent contact between relocated debris and test plate**

RASPLAV provides insights about stratification in relocated molten corium materials





Before



After

Stratification dependent on presence of carbon and fraction of unoxidized zirconium
(AW-200-2 used C-22 with 81.8 wt% UO_2, 5.0 wt% ZrO_2, 13.2 wt% Zr, and 0.3 wt% C)

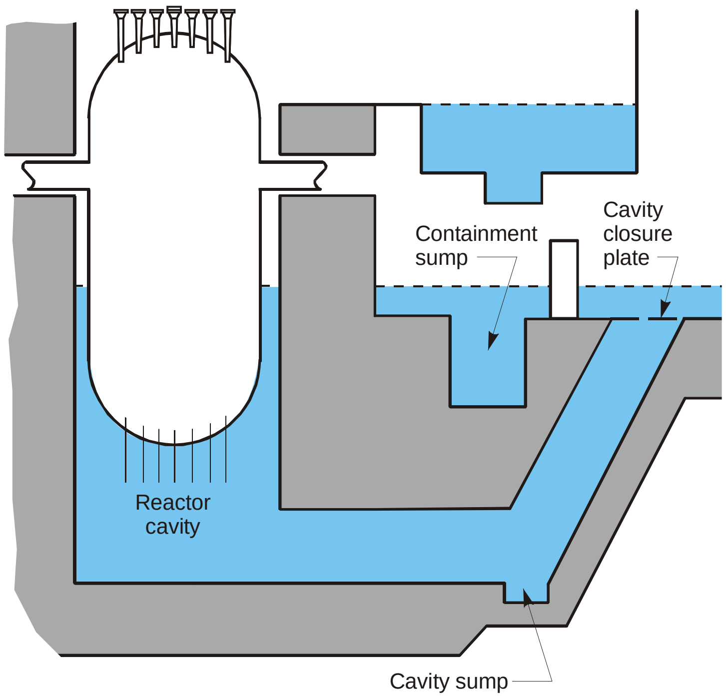
Summary

- Experimental data suggest range of debris endstates possible
  - Data insufficient to select one bounding configuration
  - Data suggest melt progression scenario dependent
  - Additional research needed to assess potential for various configurations to occur and heat transfer conditions associated with various configurations
- Experimental data provide insights related to heat transfer from various configurations
  - Gaps, cracks, and increased upper surface area enhance ceramic melt coolability

Several mechanisms available to reduce potential for vessel failure

- External Reactor Vessel Cooling (ERVC)
  - Enhanced vessel/insulation arrangement
  - Enhanced vessel coatings
- RCS depressurization
  - Intentional
  - Unintentional

Requirements for Successful External Reactor Vessel Cooling



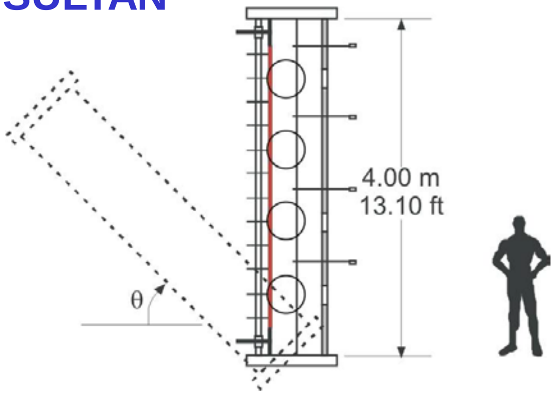
- Water must quickly cover lower vessel external surfaces
  - Flooding must occur prior to melt relocation
  - Sufficient coolant ingress and steam egress
  - Insulation must be designed to withstand forces associated with ERVC
- Heat flux to vessel must be less than heat removed from the vessel
  - Often translated to vessel heat flux must be less than Critical Heat Flux (CHF) for nucleate boiling on vessel outer surface
  - CHF dependent on angle, surface treatment, geometry(penetrations, junctions, insulation) and water height

External Reactor Vessel Cooling (ERVC) Proposed or Used for Several Plants

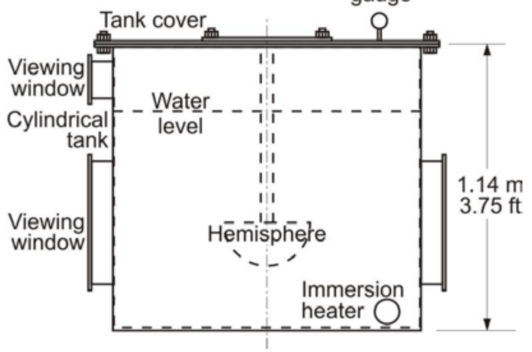
- In many Individual Plant Examinations (IPEs), cavity flooding assumed to preclude vessel failure and reduce event consequences
  - Westinghouse vessels (Zion, Byron, etc.) penetrated by instrumentation tubes that travel through reactor cavity
  - CE vessels (Palisades, etc.) without lower head instrumentation tubes
- All four generic vendor Severe Accident Management Guidelines (SAMGs) invoke ERVC, although extent of reliance varies in plant-specific SAMGs
- Finnish safety authorities approved ERVC as an Accident Management strategy for Loviisa plant (modified to enhance ERVC)
- Proposed for many advanced reactor designs, such as Westinghouse AP600, AP1000, and the Korean APR1400.

Various Approaches used to Investigate ERVC

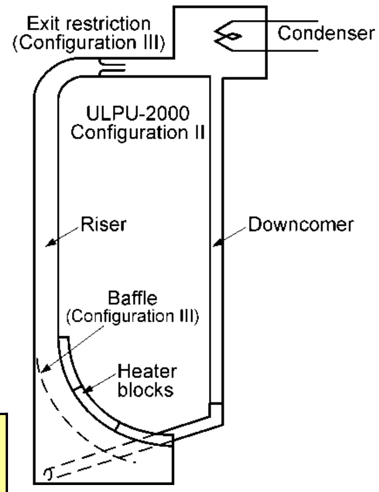
SULTAN



SBLB



ULPU

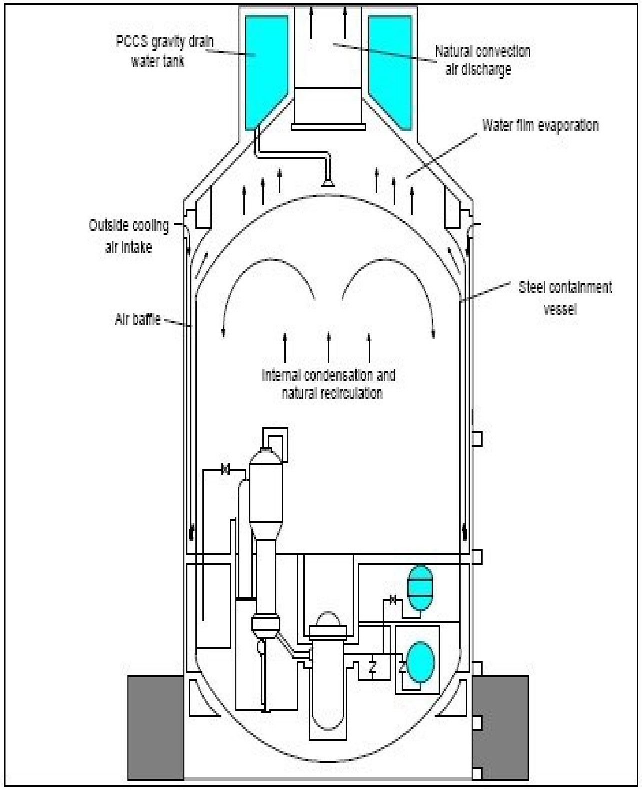


Program	Description	Subcooling (°C)	Critical Heat Flux (kW/m²)
UCSB ULPU	SS heated 2D full-scale slice [2 m [6.6 m] outer radius]	0-14 (32-57 °F)	~500 to 1500 (~1.59E5 to 4.76E5 Btu/hr-ft²)
CEA SULTAN	SS electrical heating of a flat plate [15 cm (49.2 ft) wide/ 4 m (13 ft) long]	0-50 (32-122 °F)	~500 to 1500 (~1.59E5 to 4.76E5 Btu/hr-ft²)
Penn State SBLB	Quench and SS heated hemisphere [0.31 m (1.0 ft) OD]	0-10 (32-50°F)	~400 to 2000 (~1.27E5 to 6.34E5 Btu/hr-ft²)

Drawings not to scale

Mitigating High Pressure Scenarios

- Progression of core damage under high pressure presents unique challenges
  - Steam generator tube rupture (bypa risk)
  - Hot leg/surgeline failure
  - Safety valve failure to close
  - RCP seal leakage
  - High pressure melt ejection
  - Direct containment heating
- Mechanisms to prevent by design
  - Primary depressurization system
  - Lower core power density
  - Minimize head penetrations
  - Minimize pathways to upper containment



AP1000 Passive containment cooling system

Source: ansaldonucleare.it

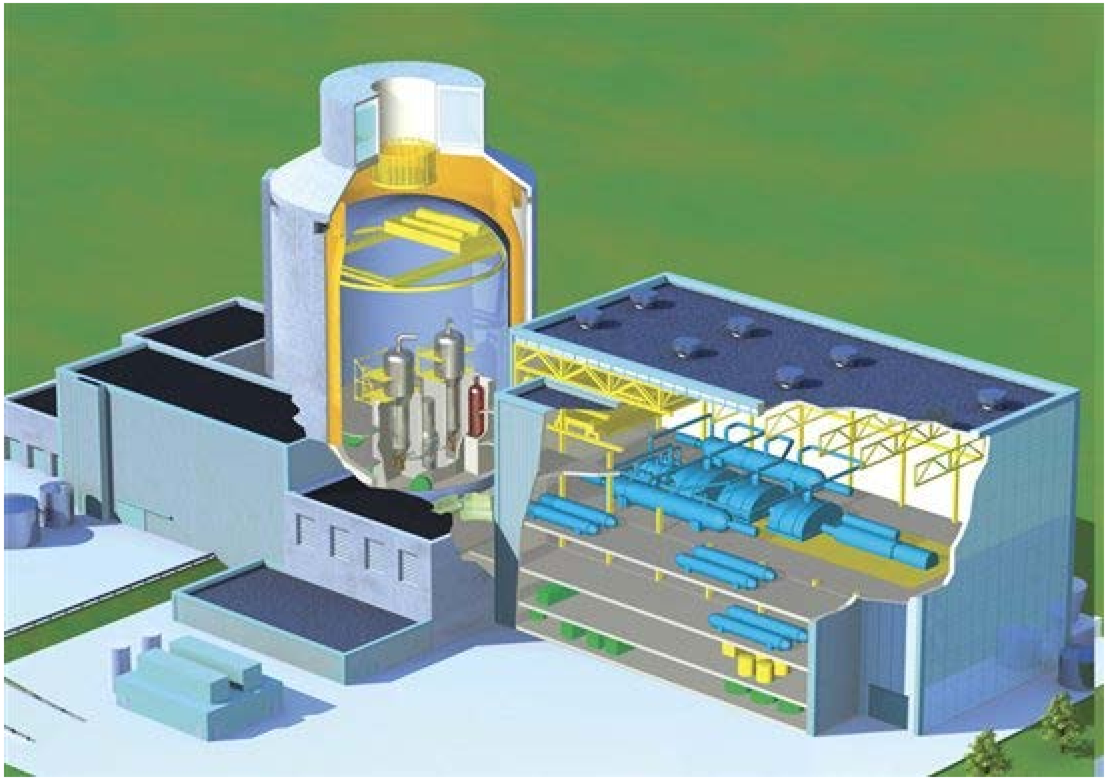
SCDAP/RELAP5 calculations suggest induced RCS piping failure prior to significant core relocation.

- Calculations performed for wide spectrum of SBLOCAs assuming unflawed steam generator tubes
- Wide spectrum of plants (Zion, Surry, Calvert Cliffs, Arkansas Nuclear One) analyzed
- Results suggest
  - natural circulation promotes hot leg or surge line failure before core relocation
  - RCS depressurizes and accumulators discharge prior to vessel failure
  - small amounts of steel and zirconium relocate
  - H_2 generation consistent with 20-60% Zr oxidation

Summary

- External Vessel Reactor Cooling (ERVC) may prevent vessel failure
  - Plant-specific evaluations needed to assure timing of flooding, sufficient water ingress, and steam egress.
  - Methods available to enhance ERVC.
- Several RCS depressurization mechanisms offer potential for accident mitigation:
  - RCP seal leakage
  - Induced RCS piping failures
  - Safety valve failing open
  - Intentional PORV depressurization.

Case Study: AP1000



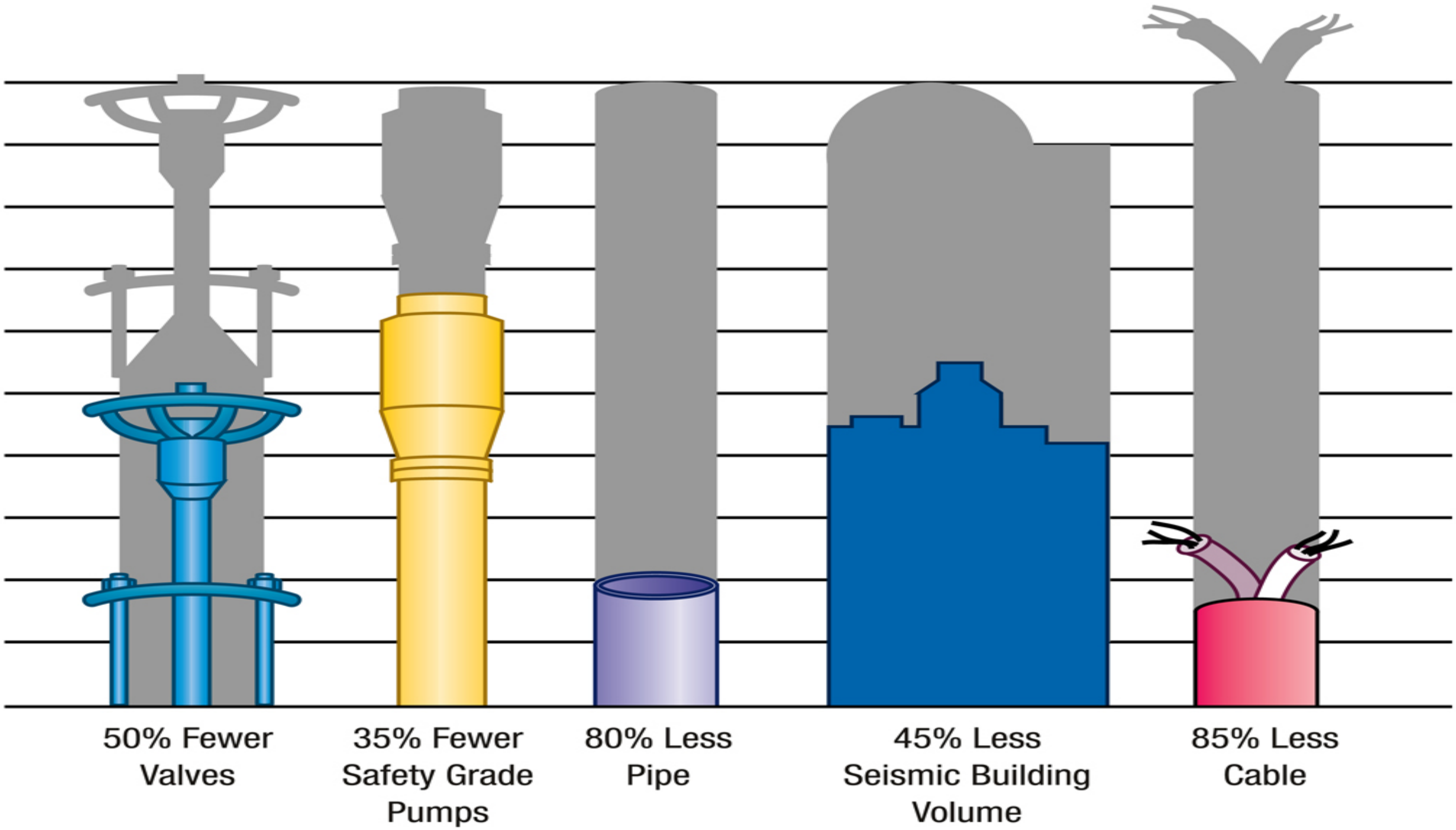
Westinghouse Advanced PWR 1000 MWe (AP1000) focused on simplicity

– Heavily reliant on passive, rather than active, safety systems
– Reduced outages and maintenance

ERVC Central to Westinghouse AP1000 Severe Accident Treatment

The AP1000™ is designed to mitigate a postulated severe accident such as core melt. In this event the AP1000 operator can flood the reactor cavity space immediately surrounding the reactor vessel with water to submerge the reactor vessel. The cooling is sufficient to prevent molten core debris in the lower head from melting the steel vessel wall and spilling into the containment. These water storage tanks hold enough water to cool the containment vessel for seventy two hours.

AP1000 Relies on Design Simplicity and Passive Cooling



Study Questions

- What key parameters may influence vessel integrity during a severe accident?
- Why is vessel failure mode and timing important in assessing the risk associated with an accident sequence?
- Name several vessel failure modes.
- Name two mechanisms for RCS depressurization.
- Describe ERVC and factors that may influence its success.
- Draw several possible configurations for relocated core materials. Show where peak heat fluxes will occur and describe why they will occur at these locations.

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External Reactor Vessel Cooling

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TMI Damage Implications for Fukushima and Status

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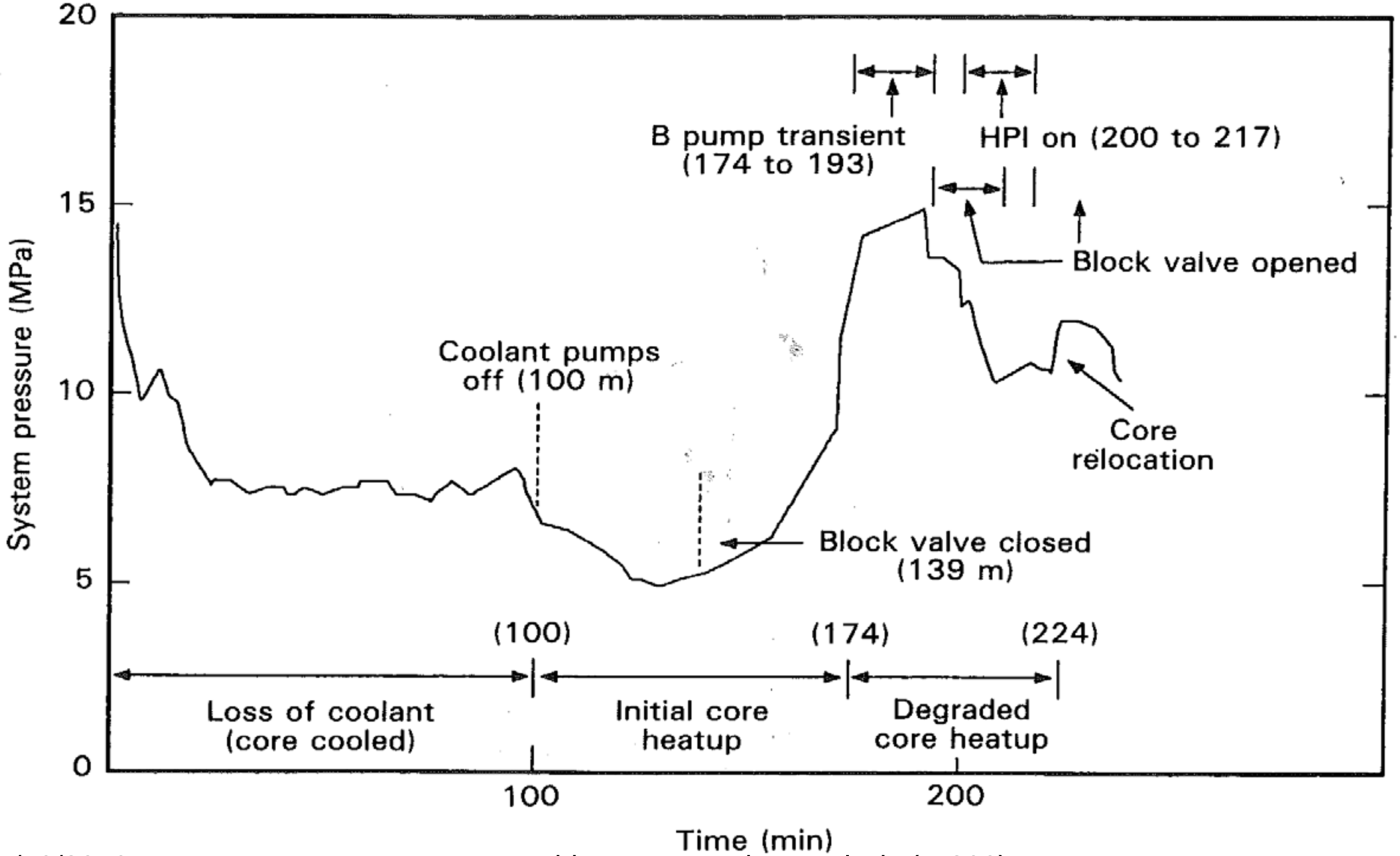
4/19/2016

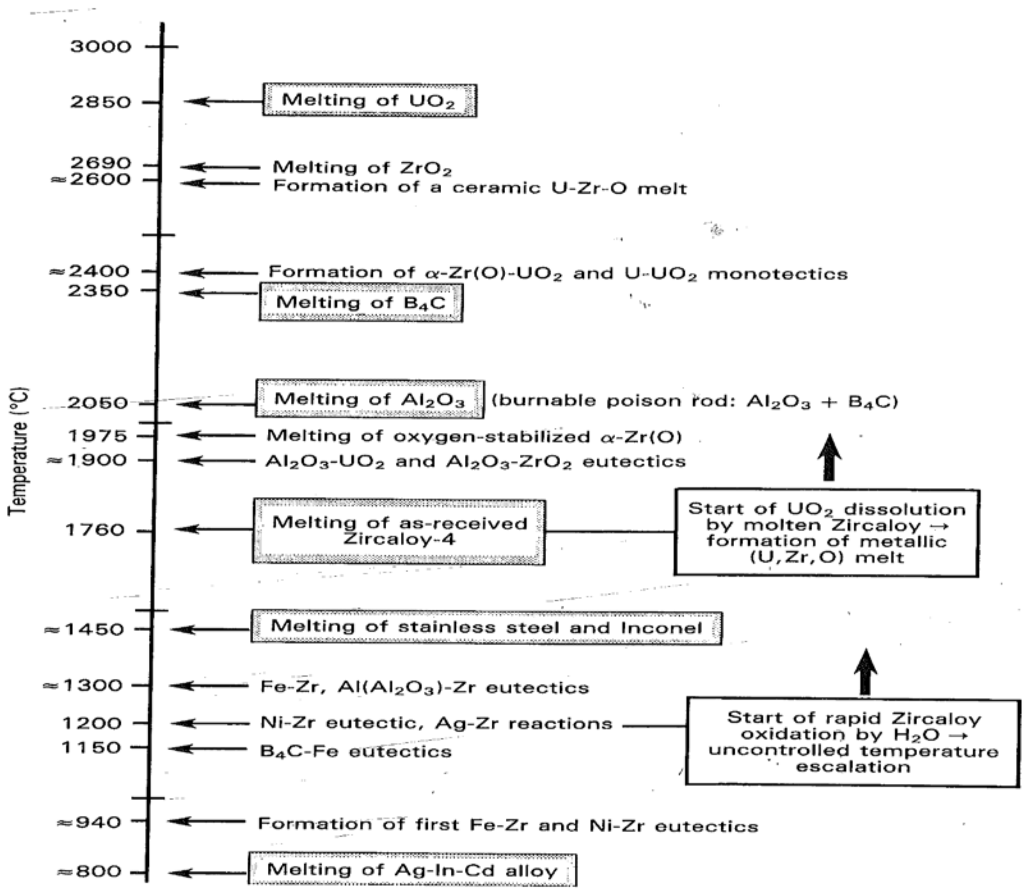
Overview

- TMI reactor accident core damage progression
- TMI reactor core damage and fuel relocation
  - Core material melt behavior
  - Relocation of fission products and core materials
- Damage to the lower head of the TMI reactor vessel
- RPV design differences between PWRs and BWRs
- Implications of the TMI accident for the Fukushima recovery
- Fukushima status
- Fukushima path forward and schedule
- Nuclear material accountability issues and TMI approach

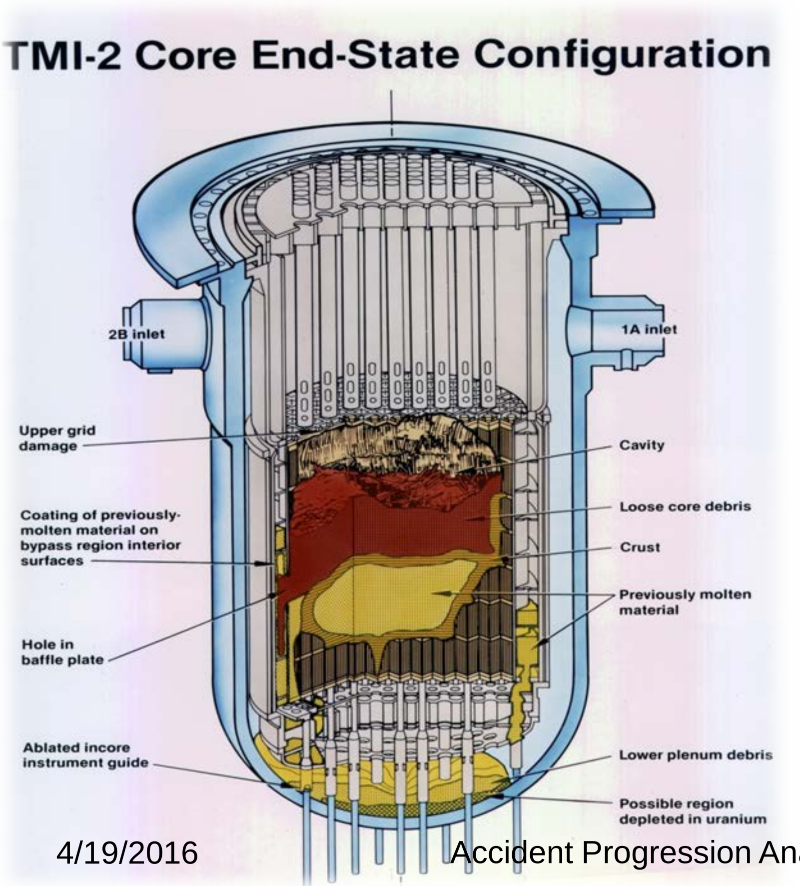
TMI Core Damage Occurred Within 224 Minutes Including Core Relocation to RPV Lower Head



Melt Behavior Defines Core Damage Progression



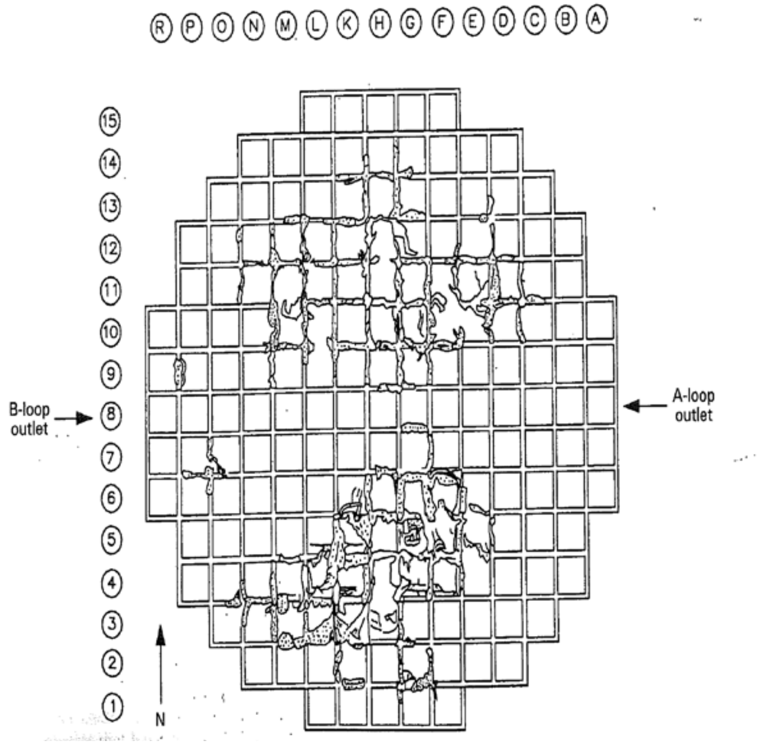
Extensive Sampling and Coring Used to define Core damage and Materials/Fission Product Behavior



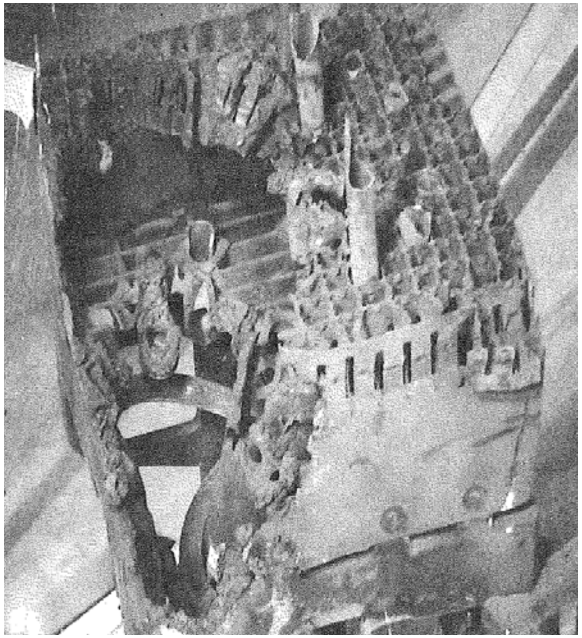
- **Core Examinations included**
  - Control rod leadscrews
  - Upper RPV core debris
  - Core bores
  - lower RPV debris
- ***Fuel - (UO_2) and Zirconium Relocation***
  - Upper core debris – 26,000 kg
  - Center core melt – 21000 kg
  - Lower crust – 5400 kg
  - Partial fuel assemblies - 40000 kg
  - Core support assembly – 8900 kg
  - RPV lower head -19000 kg

Limited Damage to Reactor Pressure Vessel Above Fuel Assemblies

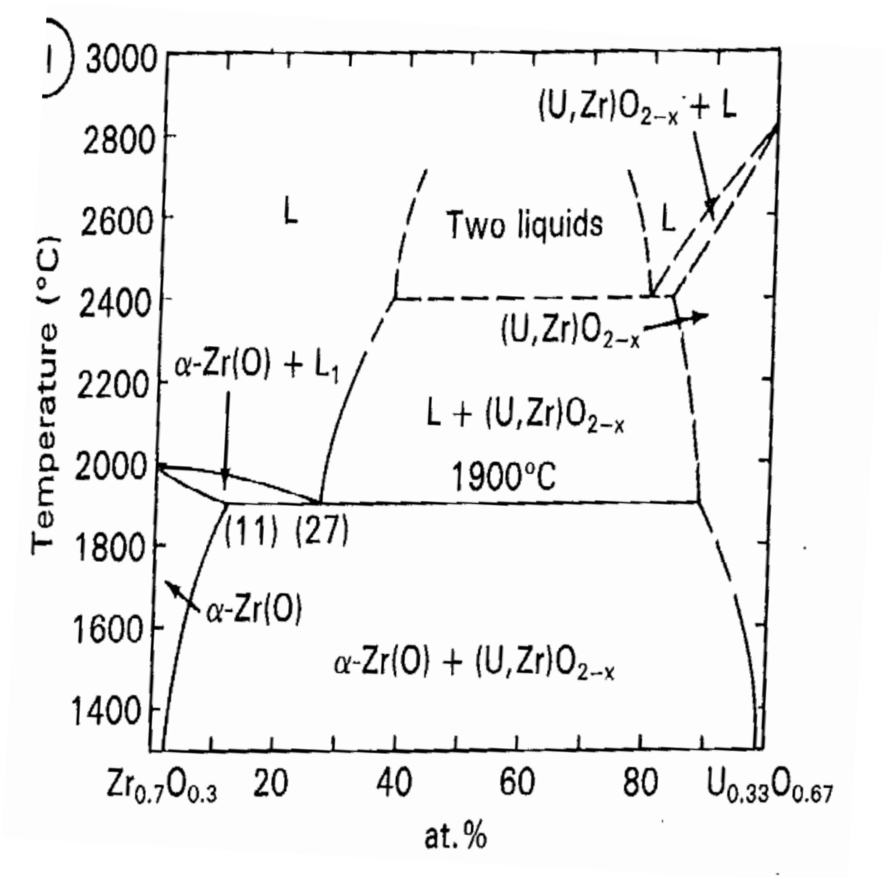
Damage to Grid Above Fuel



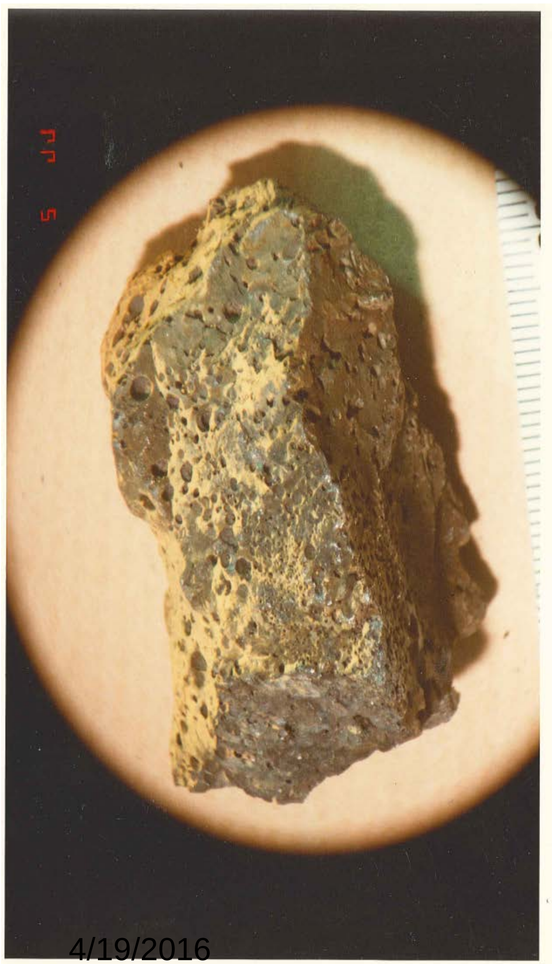
Damage to Fuel Assembly End Fitting



$(U,Zr)O_2$ Previously Melted Reactor Fuel -Highly Inert and Depleted of Volatile Fission Products

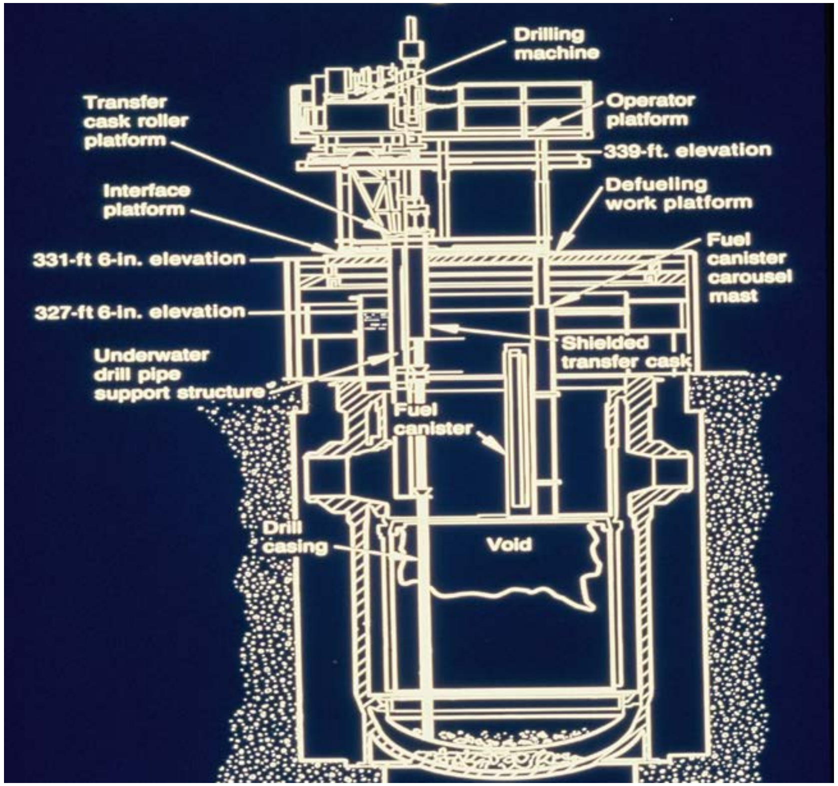


$(U,Zr)O_2$ Melted Reactor Fuel -Highly Inert and Similar Composition in Central Core and Lower Head

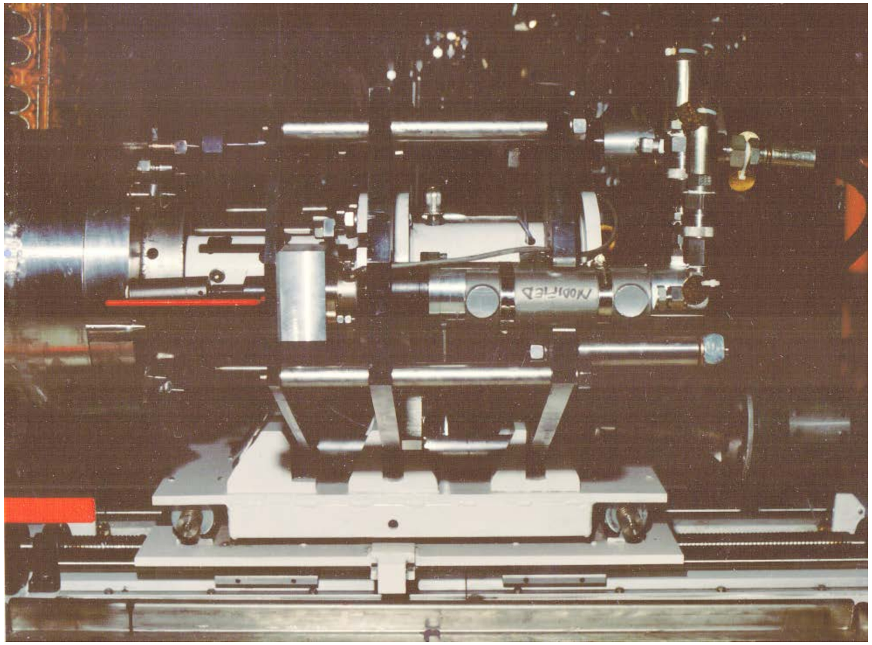


- Composition is primarily (U,ZrO_2) nominally - 69%U, 26%Zr and 4.6% O
- Density of debris ranges from 7-9 g/cm^3
- Nominally <1% of volatiles inventory (e.g., ^{134,137}Cs, radioiodine and noble gases)
- Dissolution of bulk specimens (>50 g) in some cases required multiple step processes and required days to complete dissolution

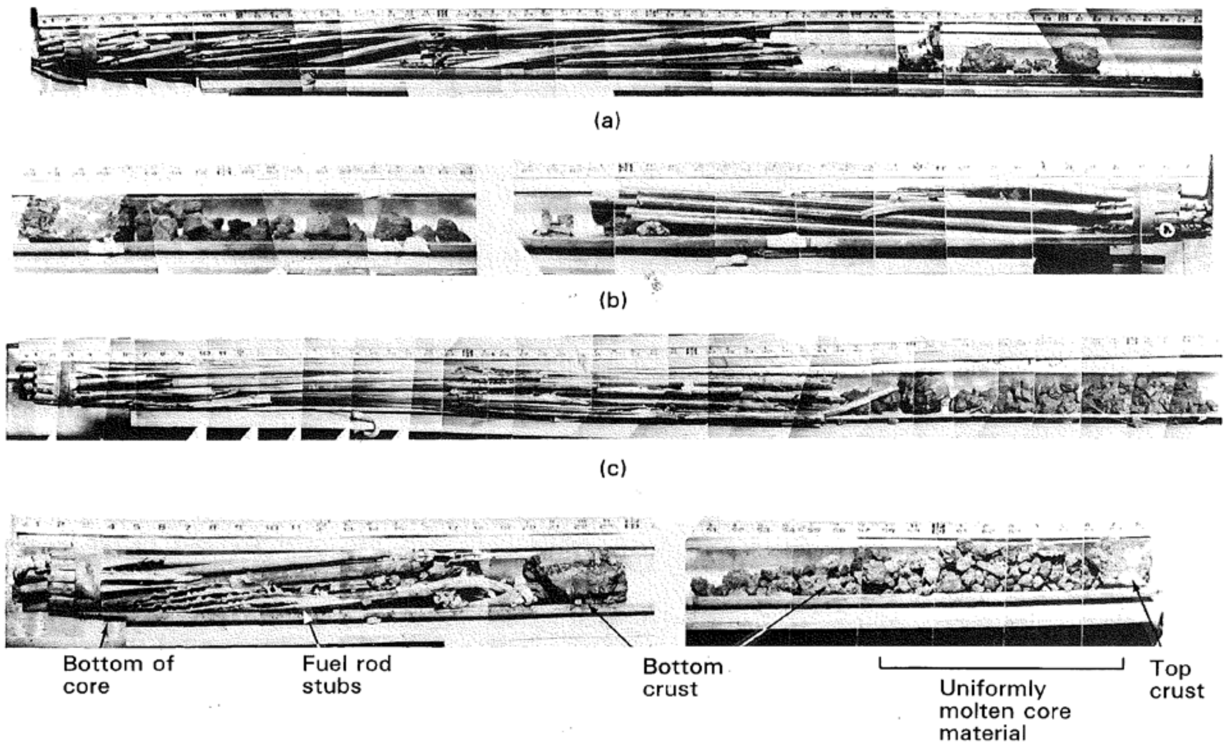
TMI Core Boring System (Modified Drilling System) to Break Up Debris



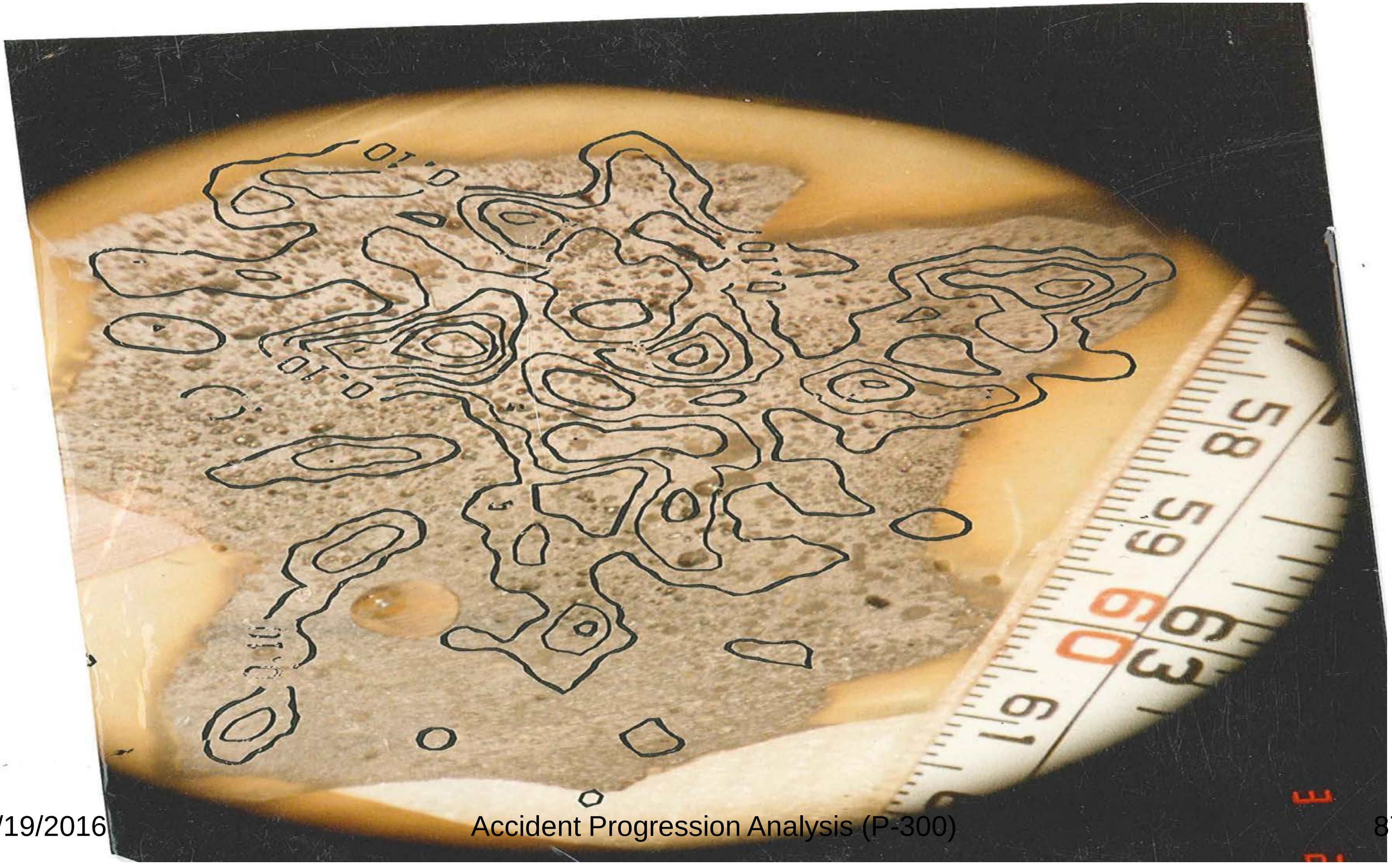
Core bore head



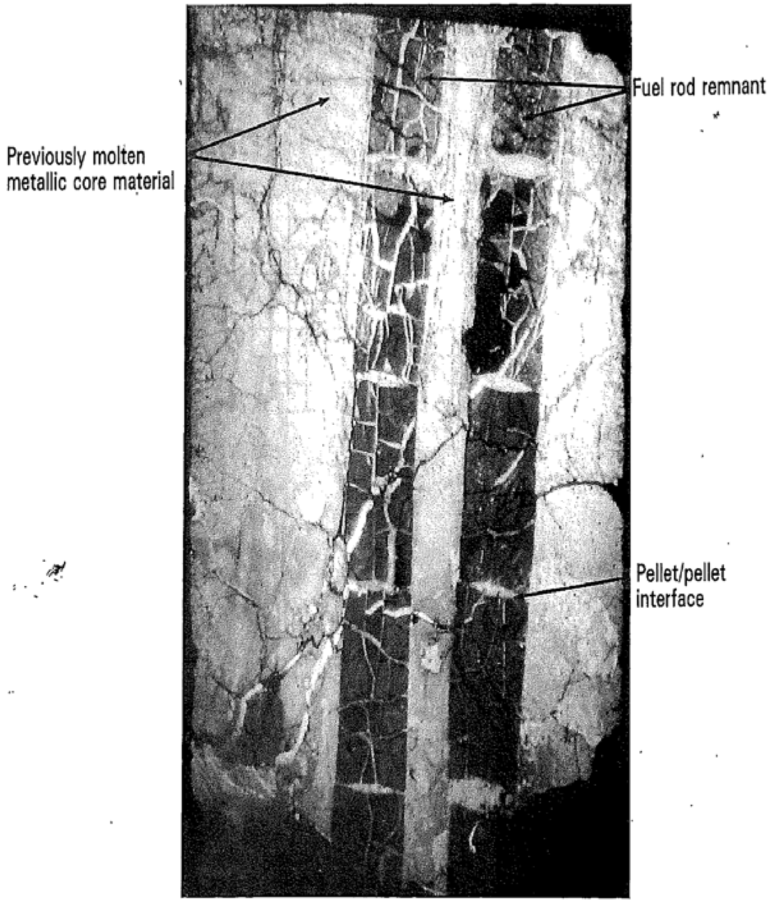
Specialized Gamma Spectrometry System used to Reconstruct Fuel and Fission Product Relocation

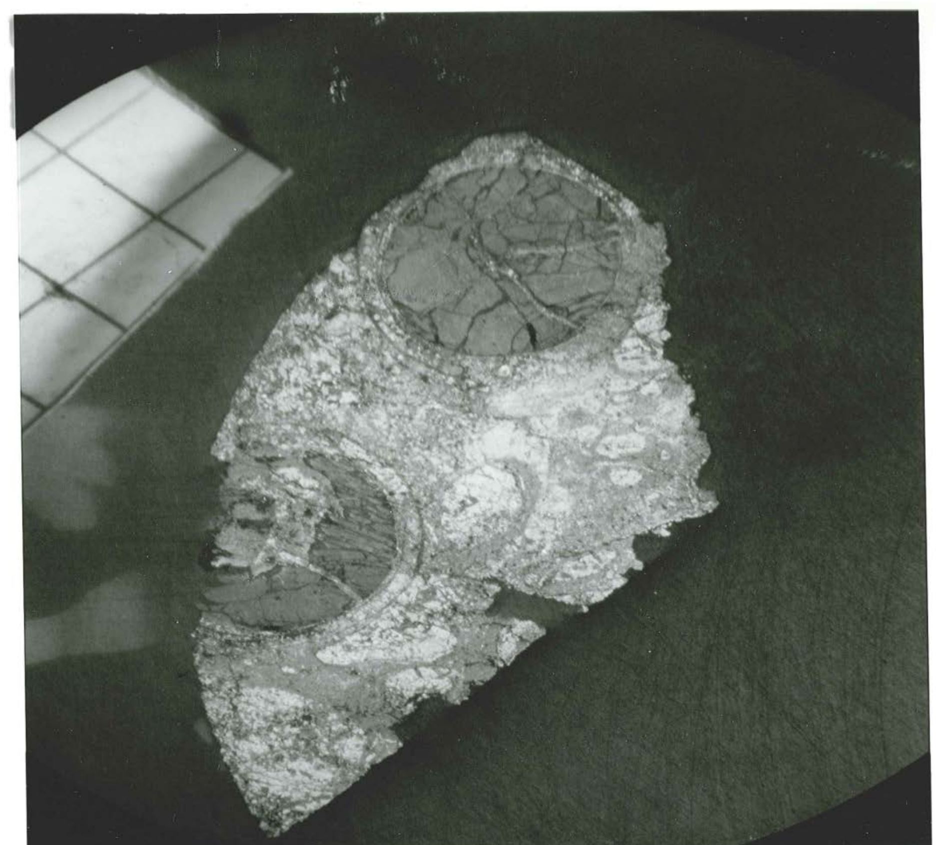


Fission Product Release (e.g., ^{137}Cs) from Prior molten Reactor Fuel Evaluated using Gamma Tomography System

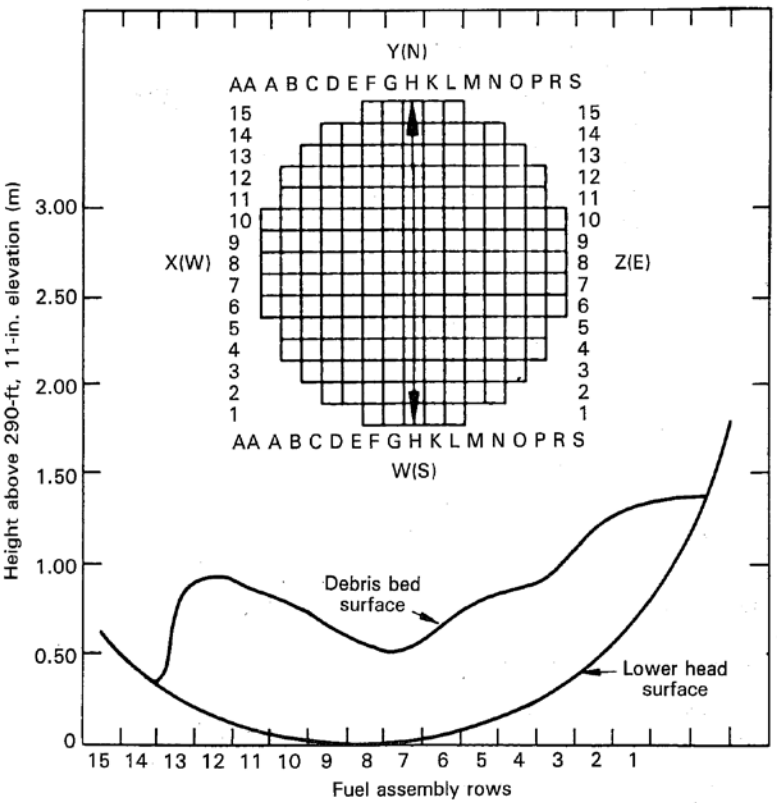


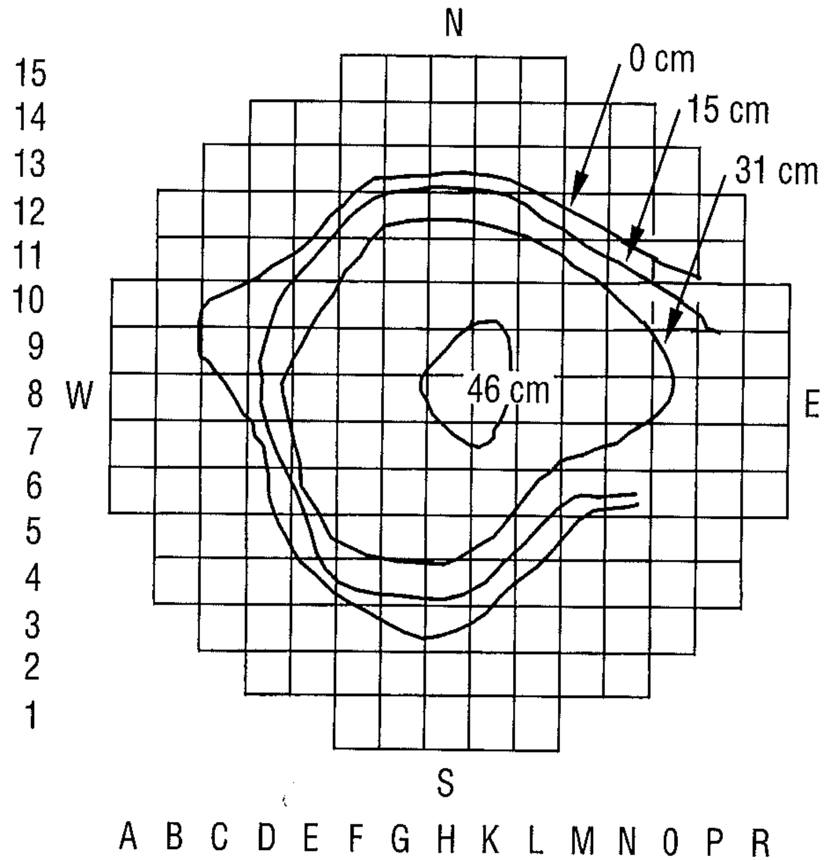
Relatively Intact Fuel In Metal Layer Below Central Core





19000 kg of Fuel Melt On TMI RPV Lower Head

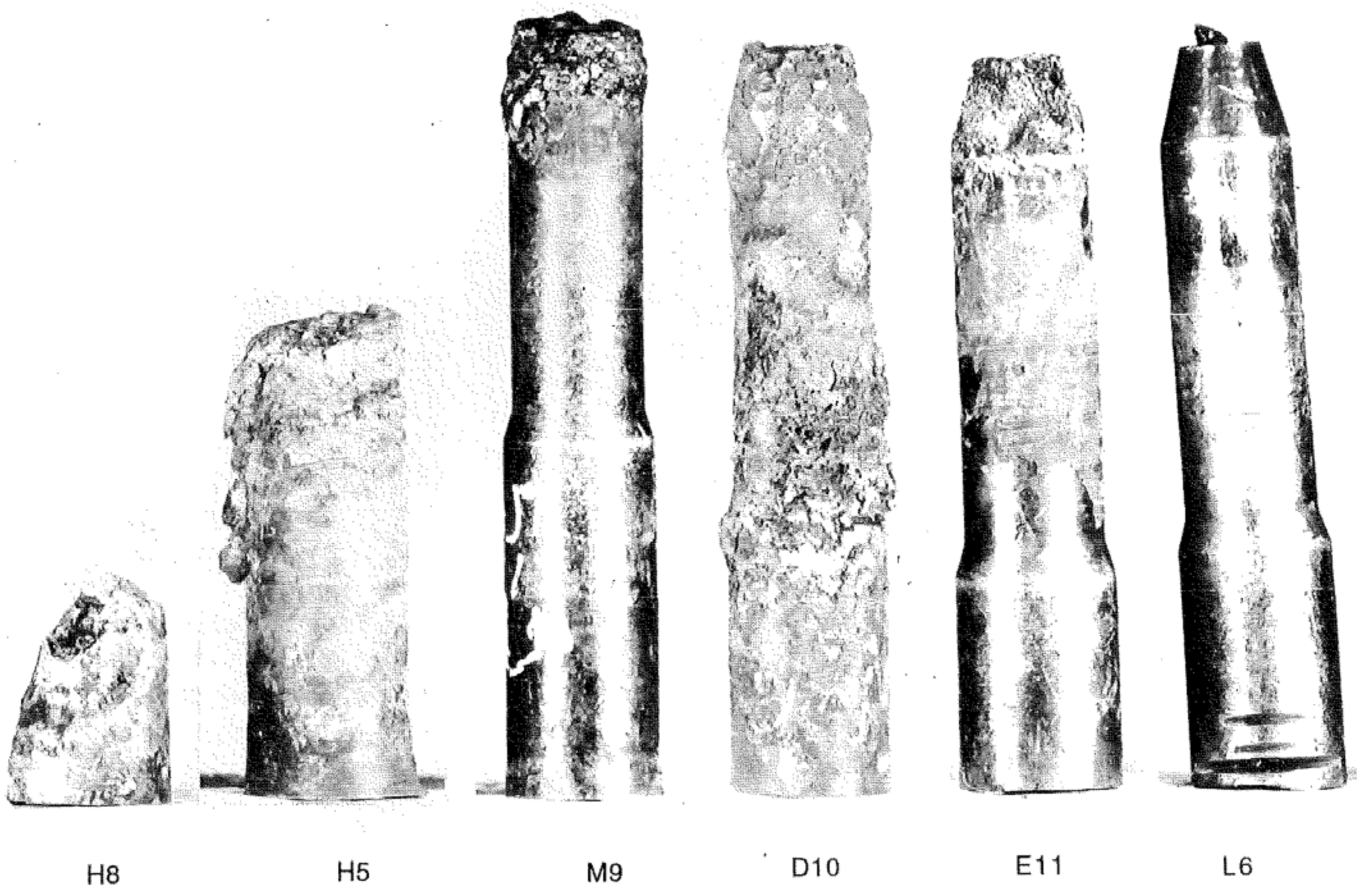




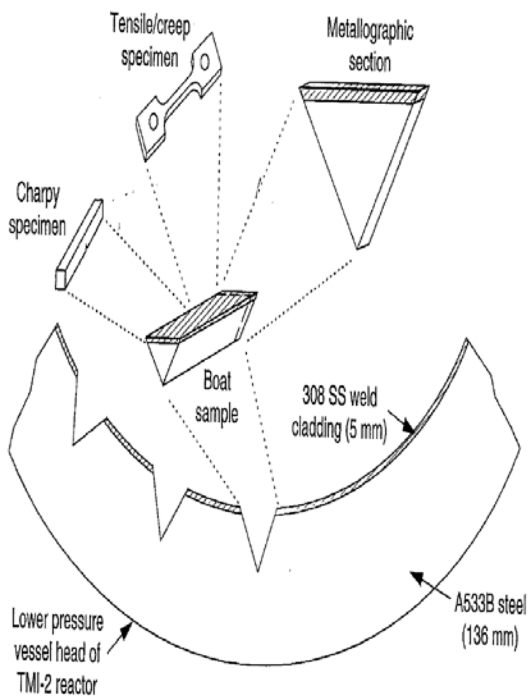
Burned off Incore Instrument Penetrations on Head Indicate Protection by Soldified Melt

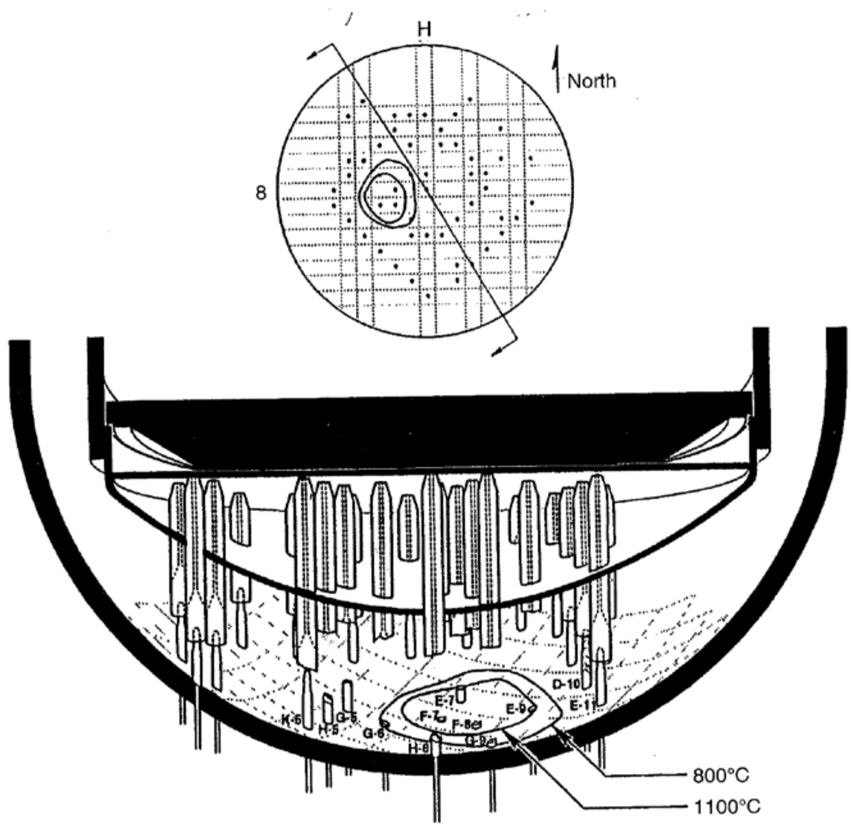
TMI -2 Incore Nozzle Protected by Solidified Fuel Melt

Varying damage to Lower Head RPV Nozzles

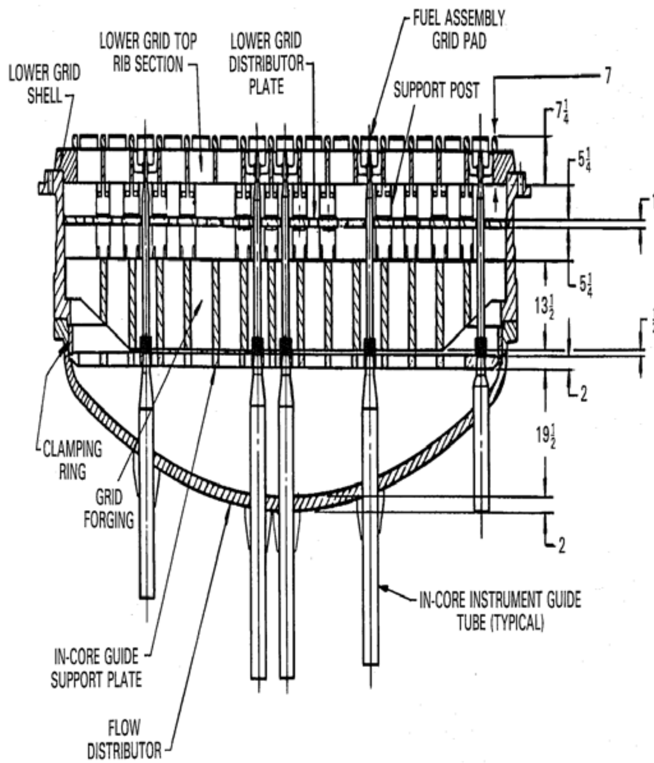


Lower Head Boat Samples Indicate Max Temperature < 1100ºC





BWR RPV Assembly More Massive Than PWR



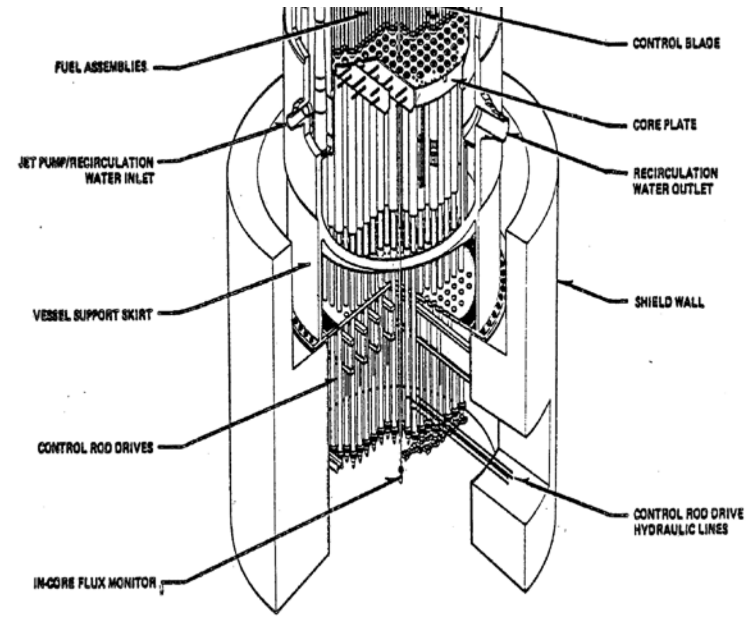


Figure 4 Reactor Assembly

PWR Lower RPV Head **BWR Lower RPV Head**

Significant Relocation of Volatile Fission Products From the Reactor Vessel to the Containment with little Release to the Environment

- Significant release of all highly volatility fission products where melting occurred - Approximately 50% of noble gases, iodines and cesium radionuclides
- Medium volatility radionuclides ^{125}Sb and ^{104}Ru accumulated in metal layer below mid core location
- Low volatility $^{\mathbf{144}}Ce$, ^{154}Eu and ^{155}Eu fully retained in fuel
- Tc-99 releases also expected due to volatility and long half-life

^{137}Cs Retained in RB water(47%) and noble gases in containment (54%)

Repository	Core Inventory (%)				
	^{129}I	^{137}Cs	^{90}Sr	^{125}Sb	^{106}Ru
RCS coolant	1.20	0.79	1.00	---[a]	---
RCS surfaces	0.91	0.36	0.02	0.46	0.4
RCDT	0.01	0.00	0.07	0.00	0.0
RB structural surfaces	0.07	0.03	0.00	0.00	0.0
RB air cooling assembly surfaces	0.23	0.01	0.00	0.00	0.0
RB basement water	14.10	40.10	1.69	0.18	0.0
RB basement sediment	7.91 to 100.00	0.78	1.40	2.91	0.0
Auxiliary building media	6.11	5.00	0.00	---	---
Total	30.54 to 100.00	47.07	4.18	3.55	0.49

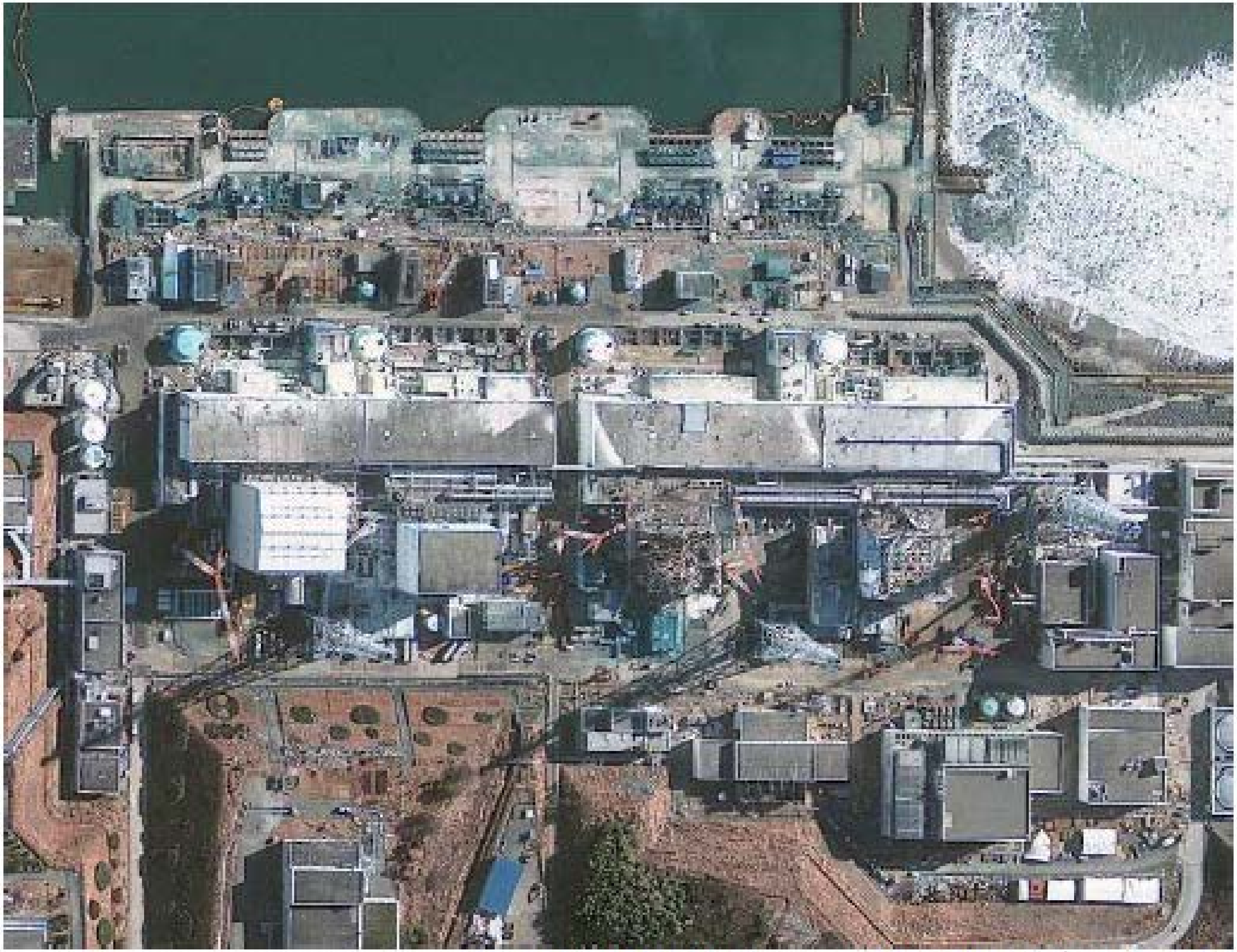
[a]Not detected.

Post Accident Fuel Distribution Outside the Reactor Core

- Auxiliary and fuel handling buildings <17 kg
- Reactor building outside the RCS - <75 kg
- RCS outside the RV - <133 kg
  - Primarily steam generator tube sheet
- Reactor vessel following defueling <900 kg
- Several techniques used for post accident defueling assessment
  - Visual examination estimate 630 kg
  - Passive neutron measurement – 1332 kg

Fukushima Dahiichi NPP after Tsunami and explosions

Recent View of Fukushima Daiichi (Units 1 to 4)



As of 1/31/2012 10:24 (C)GeoEye / 日本スペースイメージング

Unit Specifications

Unit	Output (MW)	Start of Operation	Reactor Type	Containment Model	General Contractor
1	460	3/26/1971	BWR-3	Mark I	GE
2	784	7/18/1974	BWR-4	Mark I	GE & Toshiba
3	784	3/27/1976	BWR-4	Mark I	Toshiba
4	784	10/12/1978	BWR-4	Mark I	Hitachi
5	784	4/18/1978	BWR-4	Mark I	Toshiba
6	1100	10/24/1979	BWR-5	Mark II	GE & Toshiba

JAEA Activities for Environmental Restoration

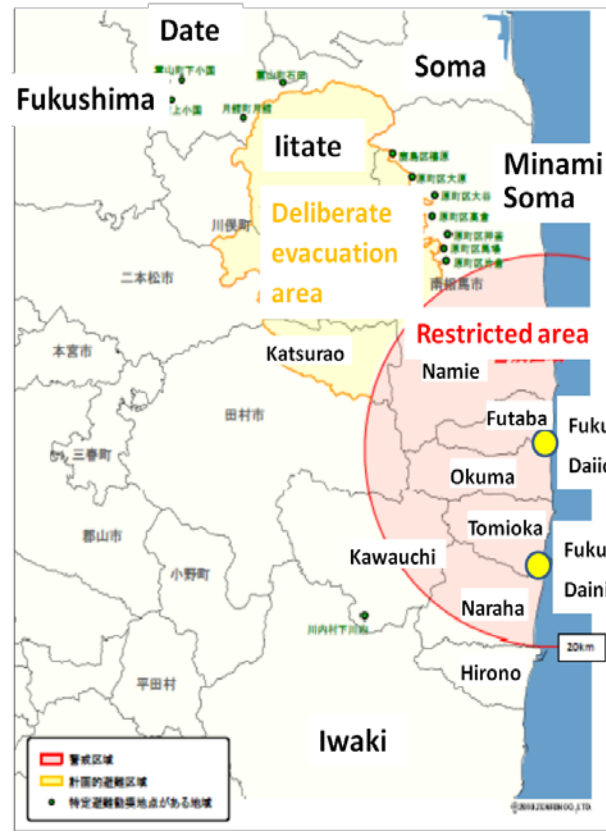
econtamination technology developme

○ Challenge for development of macromolecule cesium collection material, soil exfoliation technology with solidification agent, etc.

○ Demonstration of areal decontamination at the model areas, totaling 221 ha in size with various components and various dose rate levels from 5 to over 100mSv/y, and decontamination technologies were carried out to provide valuable data for full scale decontamination work in the future.

Results of decontamination in Otto-zawa area in Okuma-machi

	Before decontamination (μSv/h)	After decontamination (μSv/h)	Reduction rate
Forest	136.8	63.1	54%
Farmland	62.4	12.4	80%
Housing site	55.3	14.5	74%
Road	55.2	17.3	69%
Road	112.5	76.4	32%



Communications and instructions

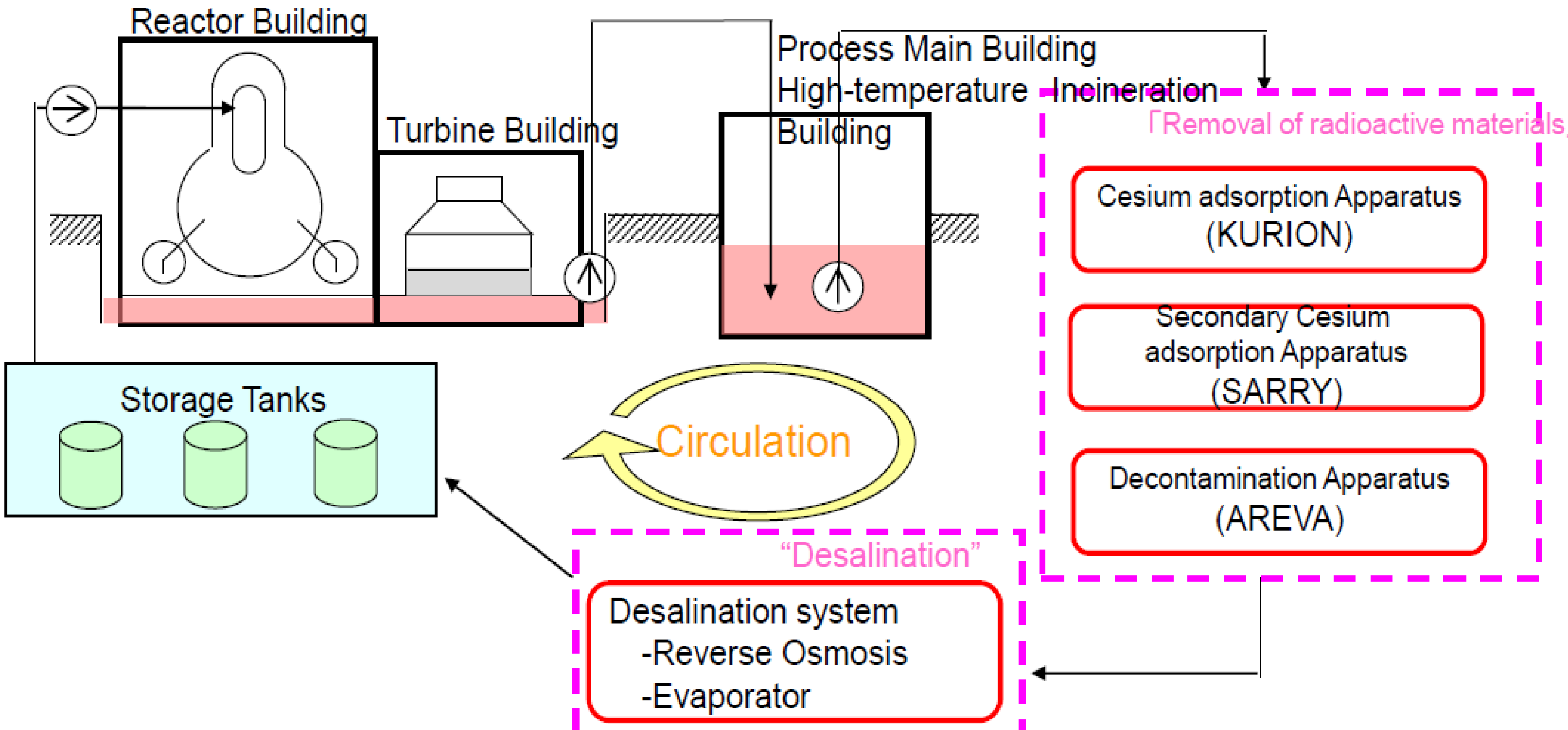
○ JAEA staff members talk face-to-face with parents and teachers, answering their questions on radiation and its health effects. 177 sessions have been held since July 2011, and a total of about 12,900 people joined.

○ WBC measurement for Fukushima pref. residents (A total of 18,600 persons have been measured and those who were estimated as more than 1 mSv (a maximum of 2.8 mSv) are 0.07% of the whole.

WBC

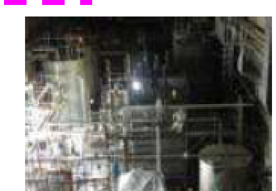
Overview of Circulating Water Cooling

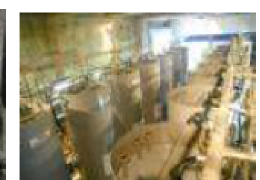
"Circulated cooling water injection" has been established to reuse the contaminated water in the buildings (accumulated water) for injection into the reactors (since 2011/6/27.)

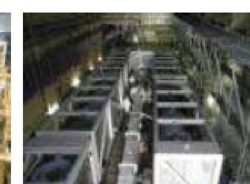



Evaporator

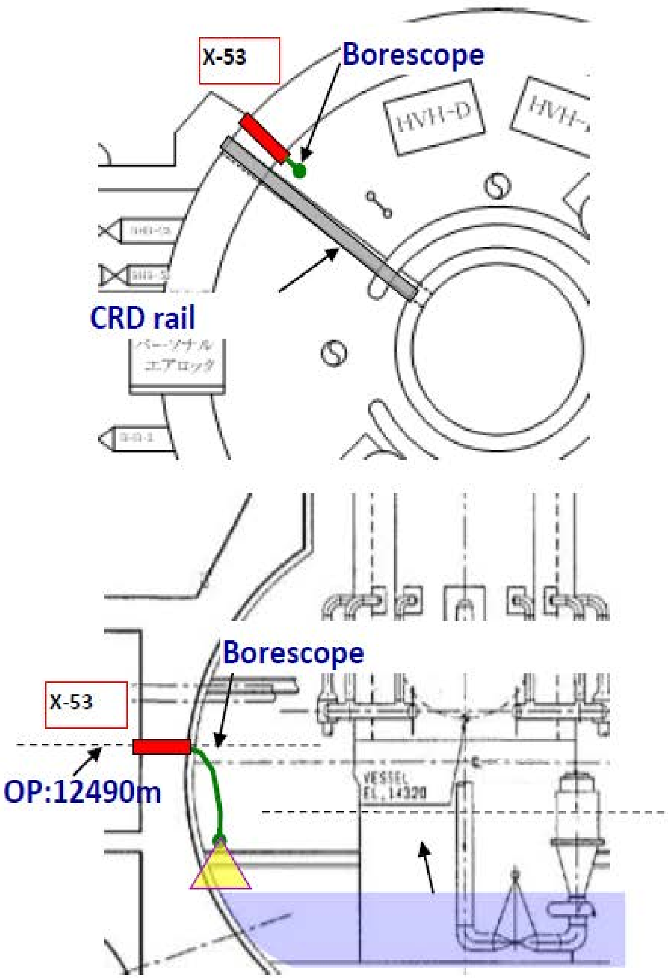

Reverse Osmosis


Decontamination


2^{nd} Cs adsorption


Cs adsorption

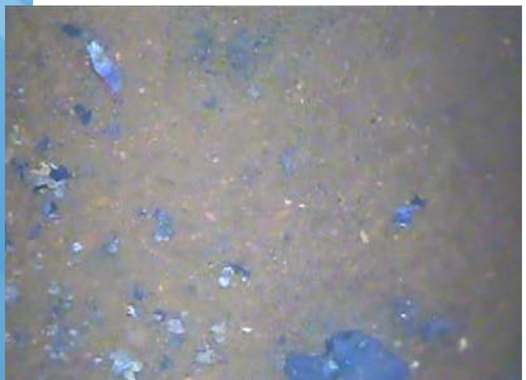
Photos inside of PCV, Unit2 on Jan.20th



: Shooting to this direction

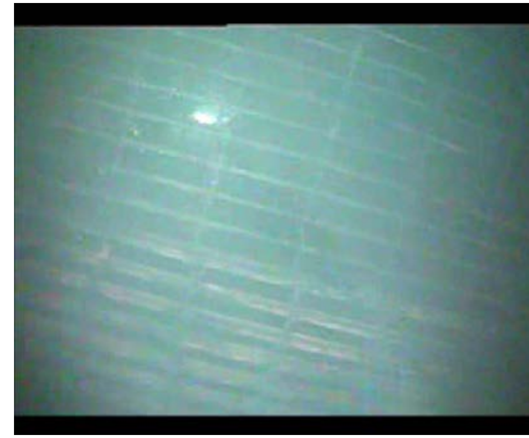
Around the grating, 1FL, Dry well

Inside inspection of damaged PCV, Unit1 on Sep.27h

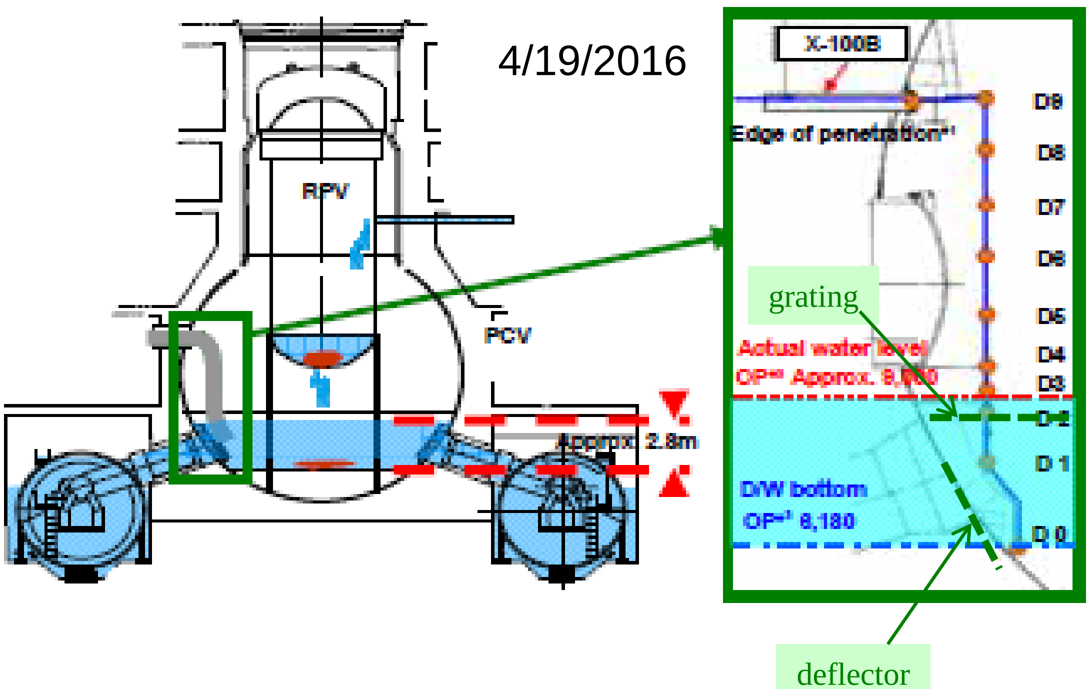


Bottom surface (dry well)

Surface on the deflector



CCD camera inspection



Measurement point	Distance from the bottom of D/W*	Radiation dose (Sv/h)
The edge of penetration	8,686	Approx. 11.1
D9	8,686	9.8
D8	Approx. 7,800	9.0
D7	Approx. 6,800	9.2
D6	Approx. 5,800	8.7
D5	Approx. 4,800	8.3
D4	Approx. 3,800	8.2
D3	Approx. 3,300	4.7
D2 (Water surface)	Approx. 2,800	0.5
D1	—	—
D0	0	—

*D/W (PCV)

Muon Model of Unit 1 Vessel and Core

Density-length image from Detector-1 based on design drawing

Vhen the density of substances existing inside is higher, the more muon are absorbed. The Black part inside the reactor show eactor core location. (Assuming fuel is not damaged)

Density-length is the multiplying structure density and length along with incidence path from the detector, which represents the extent of muon attenuation

Muon Imaging of Unit 1 Reactor – 26 days

'. Results of twenty-six day measurement with detectors 1 and 2

The results gained from the detector 1 (North West side) do not identify fuel debris inside the reactor, while those from the detector 2 (North side) appear to show something exists inside.

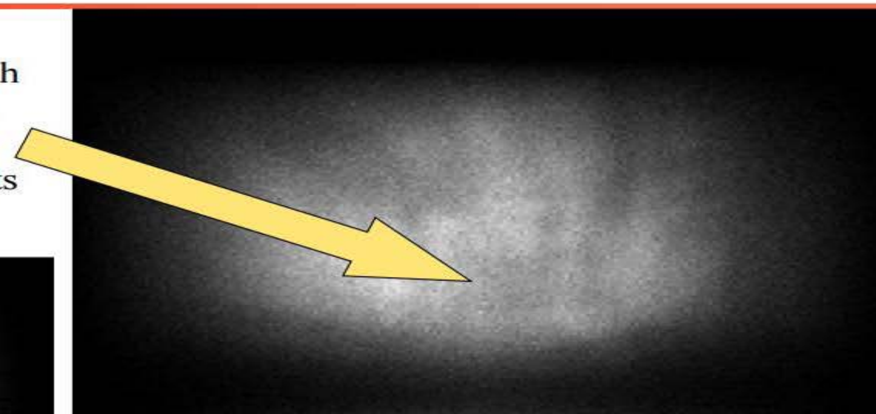


Figure 1. Measured image from the detector 2 (North side)

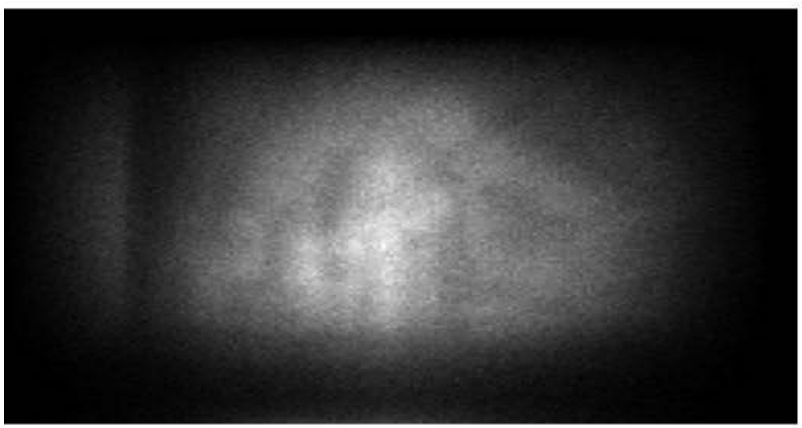
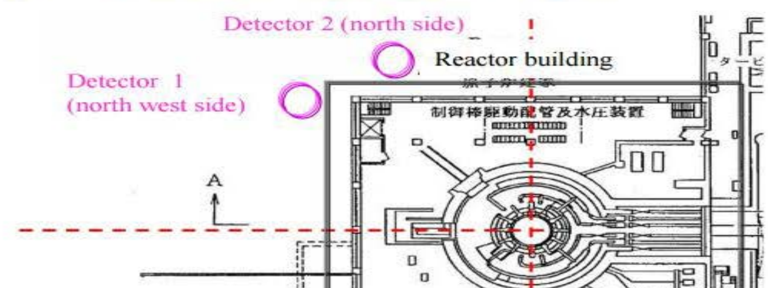


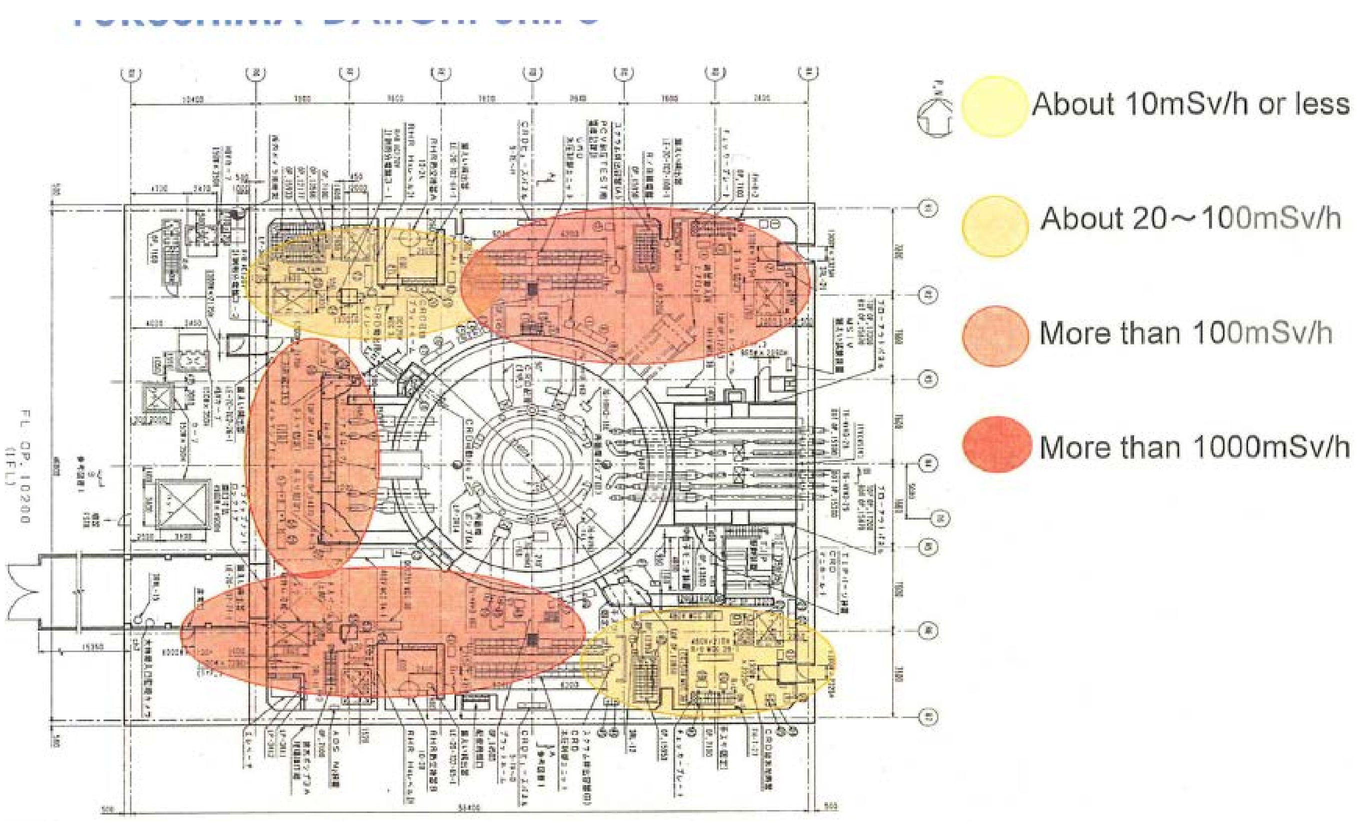
Figure 2. Measured image from the detector 1 (North west side)



東京電力

8

Defueling Floor Radiation Fields -100 R/hr Maximum (1 mSv = 100 mR)



Summary

- TMI data provides a basis for understanding the Fukushima reactor accident
- Improvements in nuclear technology provided methods for characterizing the reactor accident
- Significant retention of core debris in the control rod assemblies is likely
- Direct measurement of relocated fuel material is possible

5. Phenomena Affecting Containment Integrity

- Introduction
- Failure Analyses
- Phenomena
- Case Study and Problem
- Study Questions
- References

Objectives

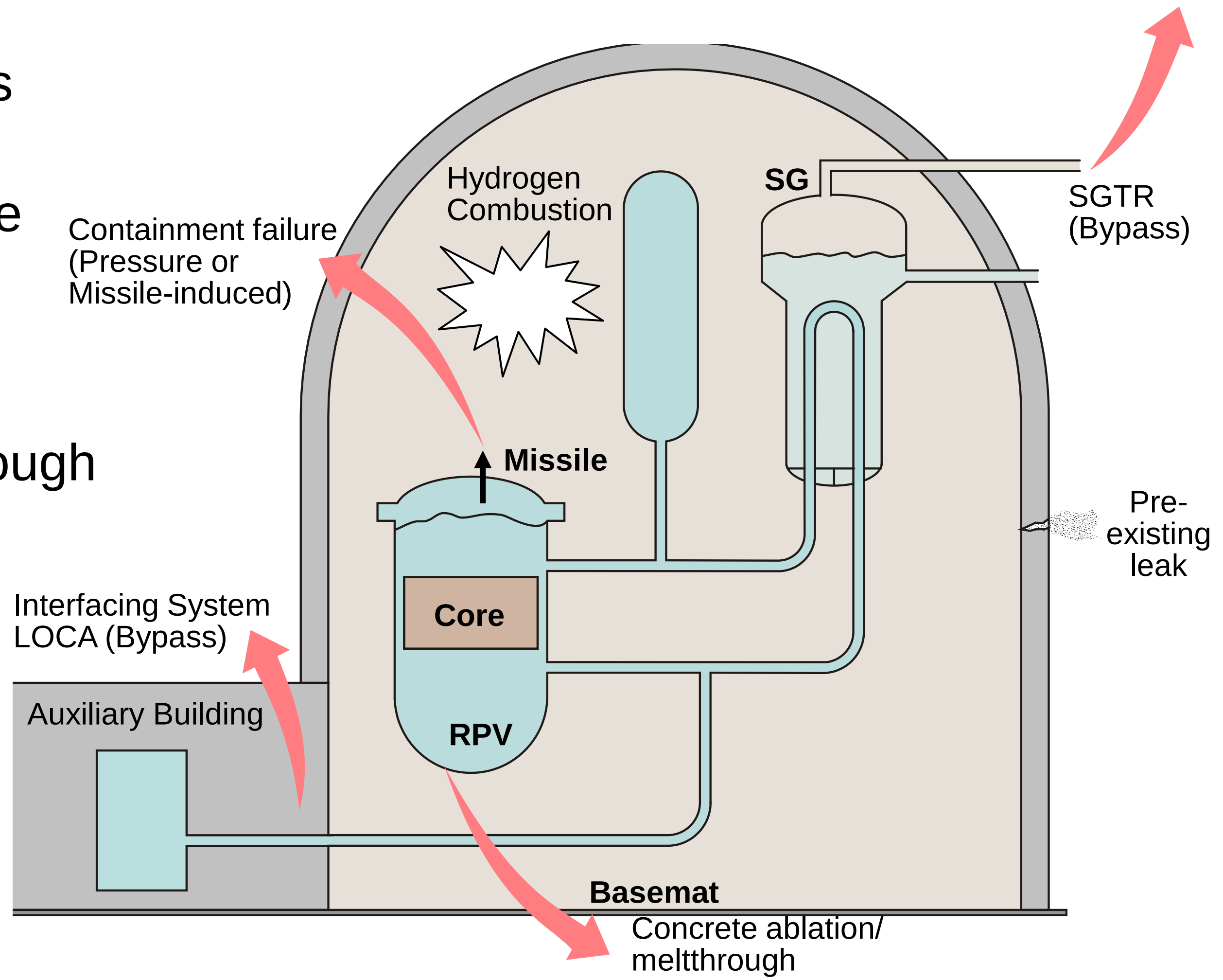
- Identify various containment failure modes and understand their likelihood for various accident scenarios.
- Identify and describe parameters affecting various challenges to containment integrity.

Ex-vessel Severe Accident Progression

- Hydrogen Combustion
- Hydrogen Recombination/Burn
- Molten Core-Concrete Interaction
- Hydrogen/CO Generation
- Melt Spreading
- Steam/Hydrogen Transport
- Long-term Containment Heat Removal

Several Challenges to Containment Integrity

- Pre-existing leaks
- Overpressure
- Dynamic pressure (shock wave)
- Internal missiles
- External missiles
- Basemat meltthrough
- Bypass
- Isolation failures



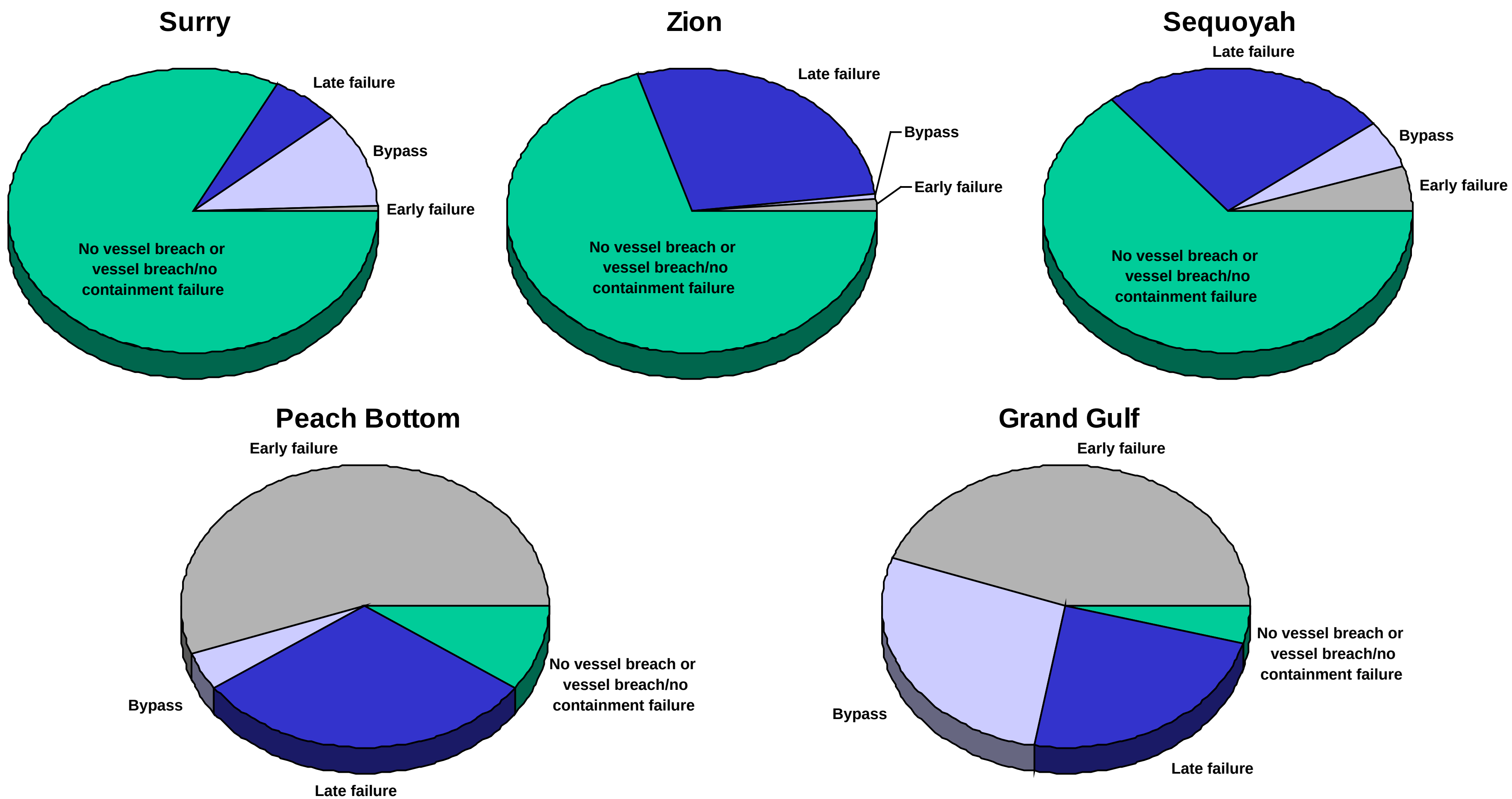
Challenges Dominate at Different Time Periods

Time Regime		Challenge
Early	Start of accident	pre-existing leak, isolation failure, bypass
	At or soon after vessel breach	RCS blowdown, hydrogen combustion, bypass, steam explosion, liner meltthrough
Late (> 2 hours after vessel breach)		containment heat removal system failure, hydrogen combustion, non-condensable gas generation, basemat meltthrough

Containment Failure addressed in NUREG-1150 Using Expert Elicitation

- What is the probability distribution function for various challenges to the containment for various events?
  - What is the pressure and temperature load distribution given that each challenge occurs?
  - What is the conditional probability of each containment failure mode for given temperature and pressure loads?

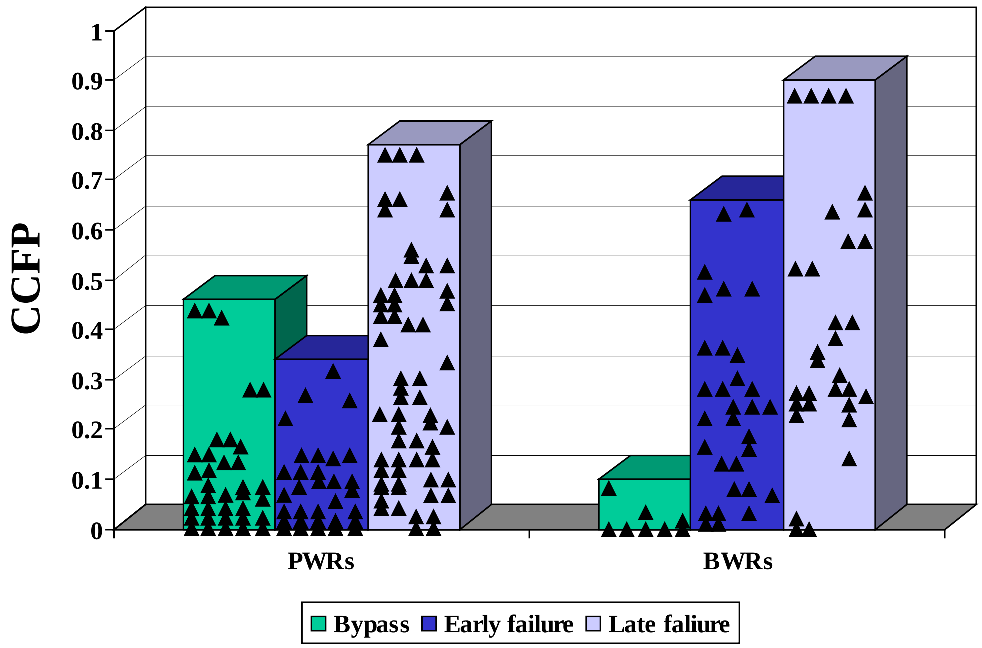
NUREG-1150 Results Indicate BWR Early Containment Failures More Likely



NUREG-1150 relative probability of containment failure modes from internal events

Individual Plant Examinations (IPEs) Suggest Late Failures Dominate

- PWR containments less likely to experience early failures than smaller BWR containments
- Bypass probabilities higher in PWRs due to higher operating pressures and use of steam generators

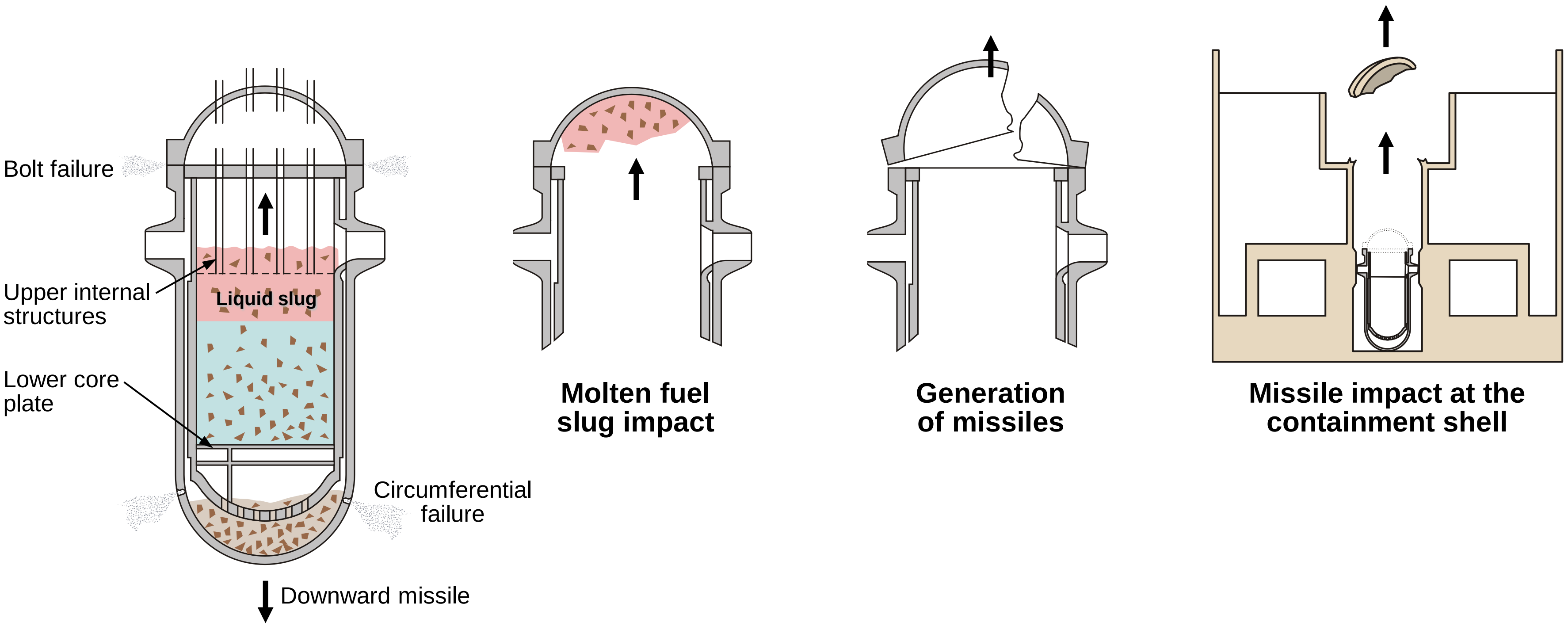


- Result variability due to differing containment features and analysis

Key Phenomena Challenging Containment Integrity

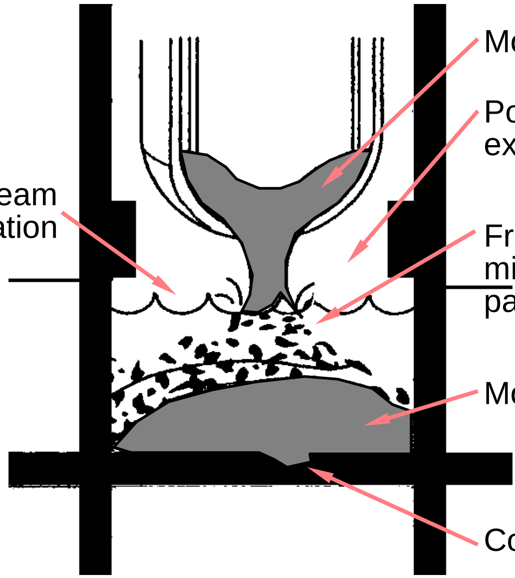
- In-vessel steam explosions
- Ex-vessel steam explosions
- Direct containment heating (DCH)
- Molten core concrete interactions (MCCI)
- Hydrogen combustion
- Meltthrough

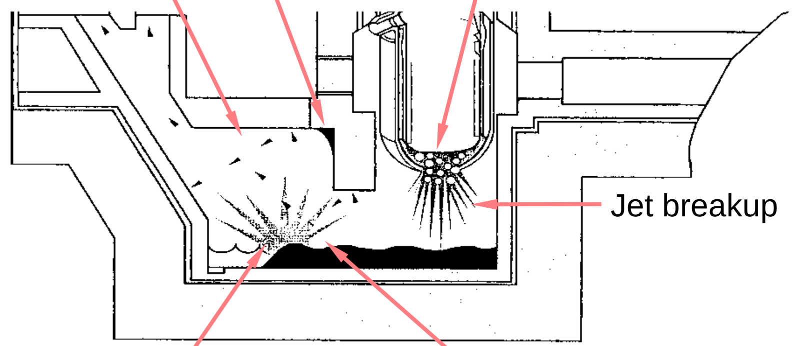
In-vessel Steam Explosion Issues



- Will in-vessel fuel/water interactions cause rapid energetic reactions?
- Are such reactions sufficient to accelerate a slug that fails vessel upper head and/or creates a missile that causes early (α) containment failure?

Ex-Vessel Steam Explosion Issues





- Is sufficient water present in the reactor cavity or pedestal region for an energetic ex-vessel fuel/water reaction?
- Are such reactions sufficient to lead to containment failure?

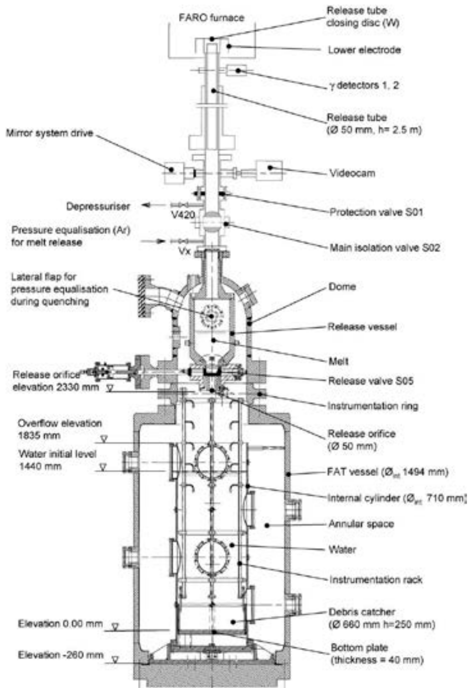
NUREG-1150 Addresses SEs using Sensitivity Studies

- Issues so controversial at time NUREG-1150 completed, expert panel refused to address.
- SNL staff internally developed distribution based on opinions expressed by SERG (NUREG-1116).
- Sensitivity studies performed assuming PDF derived by "averaging" published frequency estimates from diverse group of representative researchers.

Recent Experimental Data Provides Key Insights about Steam Explosions

Facility/ *Location*	Pheonomena Investigated	Test Section Diameter	Melt Jet Diameter (mm)	Water Depth	System Pressure	Melt Composition and Mass
FARO/ ***ISPRA***	Integral tests investigating premixing, quenching, propagation, and FCI energetics	700 mm (27.6 in.)	100 mm (4.0 in.)	0.1-5.0 m (0.3 -1.4 ft)	0.1 – 5.0 MPa (15 -730 psi)	UO_2-ZrO_2 (w/ and w/o Zr & SS) 18 - 250 kg (40 - 550 lb)
KROTOS/ ***ISPRA***	Smaller scale tests investigating premixing, quenching, propagation, and FCI energetics	95-200 mm (3.7-7.9 in.)	30-50 mm (1.2-2.0 in)	1.0 m (3.3 ft)	0.1 - 1.0 MPa (15-150 psi)	UO_2-ZrO_2 Al_2O_3 1.4 - 6.0 kg (3.1 - 13 lb)
TROI/ ***KAERI***	Integral tests investigating premixing, quenching, propagation, and FCI energetics	600 mm (24.0 in.)	~38 to 50 mm (~1.5 – 2.0 in.)	0.67 m (2.2 ft)	0.1 to 2.0 MPa (15.0 -290 psi)	ZrO_2 and UO_2-ZrO_2 5 to 14 kg (11- 30 lb)

Experimental Data Provides Key Insights about Steam Explosions (continued)



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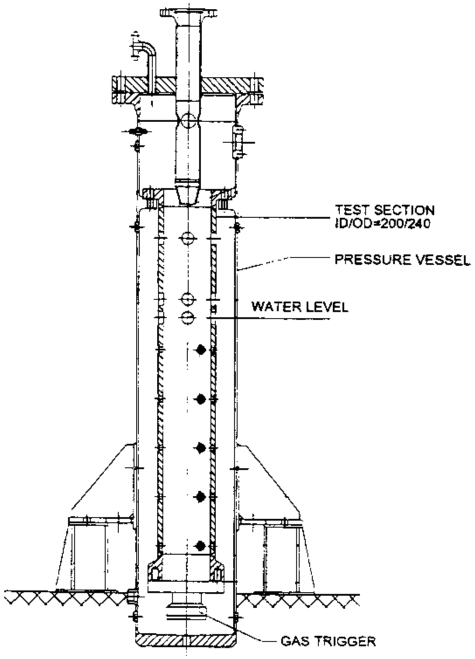
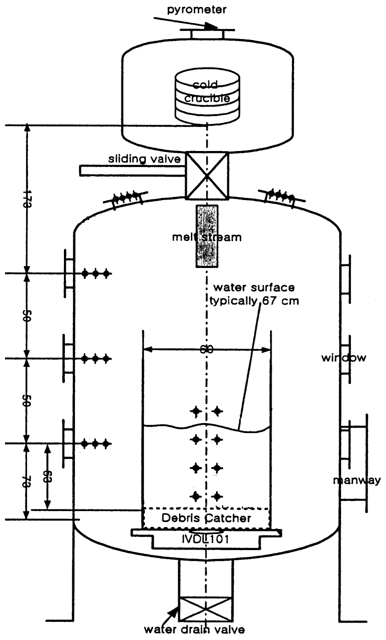
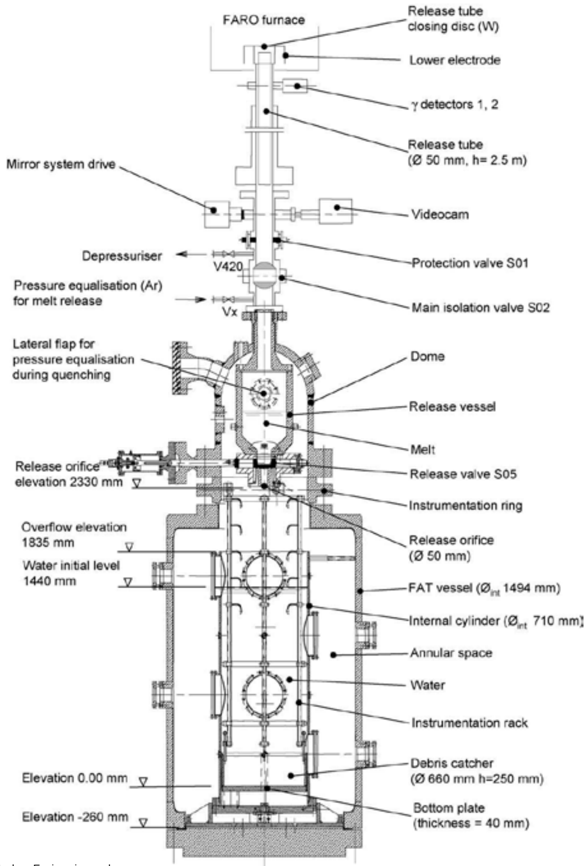




FARO ***KRYTOS*** ***TROI***

Prototypic Large-scale FARO Data Suggest Steam Explosions Less Likely



Nuclear Engineering and Design 236 (2006)

- In tests with UO_2, ZrO_2, and Zr, complete fragmentation occurred
- In tests with UO_2 and ZrO_2, relocated materials consisted of a "cake" with an overlying layer of fragmented debris
- Mean particle size of fragmented debris ranged from 3.4 to 4.8 mm (0.13 to 0.19 in.)
- No energetic steam explosions observed in tests simulating in-vessel conditions.

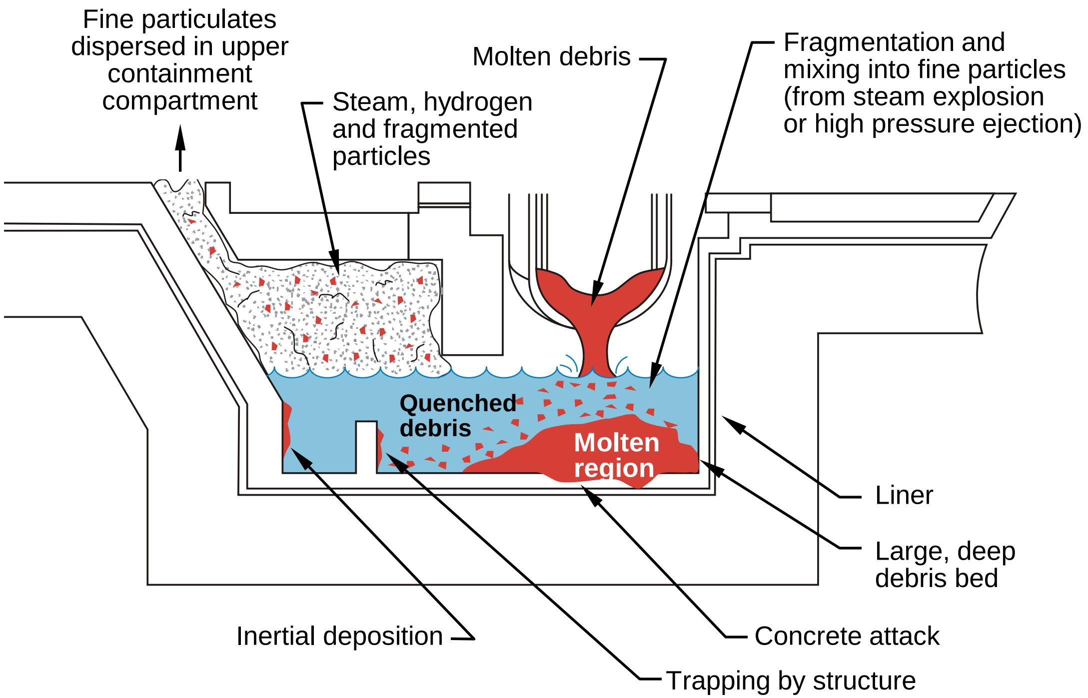
Key Parameters for Evaluating Ex-Vessel Steam Explosion Potential

- Sequence
  - Melt composition (amount of unoxidized metals)
  - Melt mass and energy
  - Melt pour area, rate, and geometry
  - Water availability
- Containment design
  - Cavity or pedestal geometry
  - Potential for shock wave transmission
  - Water availability

Recent Findings Suggest Lower Probability for Steam Explosions

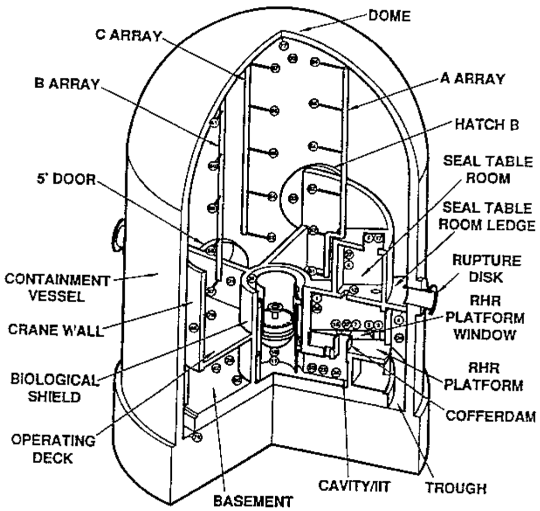
- Experimental results indicate:
  - At low pressure (0.1 MPa/15 psia), limited fuel mass participates
  - At higher pressures (>1 MPa/150 psia), difficult to trigger
  - Debris composition affects ability to trigger spontaneous SE
- All eleven SERG-2 experts estimated low probabilities for α-mode failure
  - Low conversion energy
  - Lower explosivity of corium
  - Intervening structures
- Nine of eleven SERG-2 experts declared issue of α-mode failure induced by steam explosion resolved from risk perspective
- OECD-sponsored SERENA program designed to compare various SE models with data from FARO, KROTOS, and TROI.

Direct Containment Heating (DCH) Issues

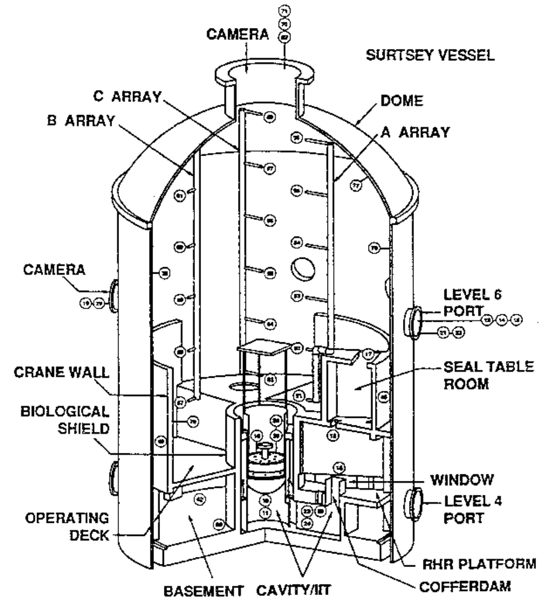


- Is sufficient melt entrained as vessel depressurizes?
- Does sufficient heat transfer, oxidation, and/or hydrogen combustion occur to threaten containment integrity?

Unique Experimental Facilities Provide Insights About Potential for DCH



CTTF



SURTSEY

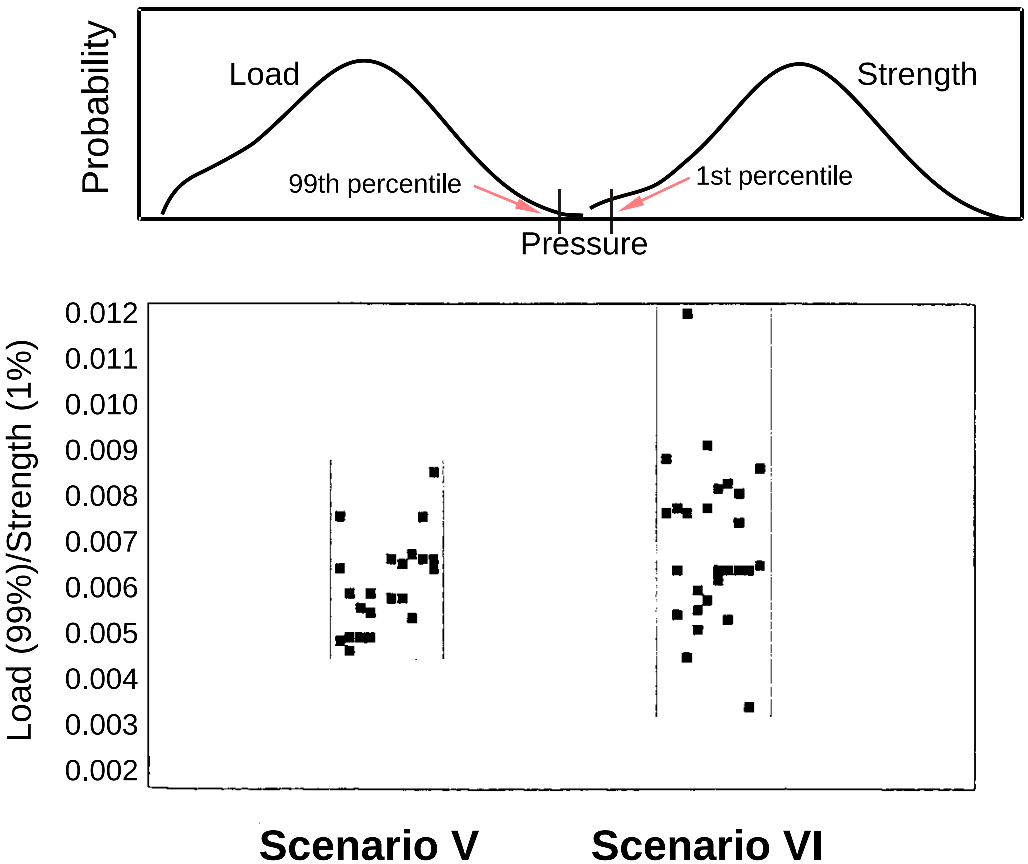
Facility capabilities allowed measurement of :

- Pressure load
- Hydrogen distribution and combustion
- Containment compartment geometry effect
- Post-test debris distribution
- Effectiveness of safety equipment

Key Parameters for Evaluating DCH Potential

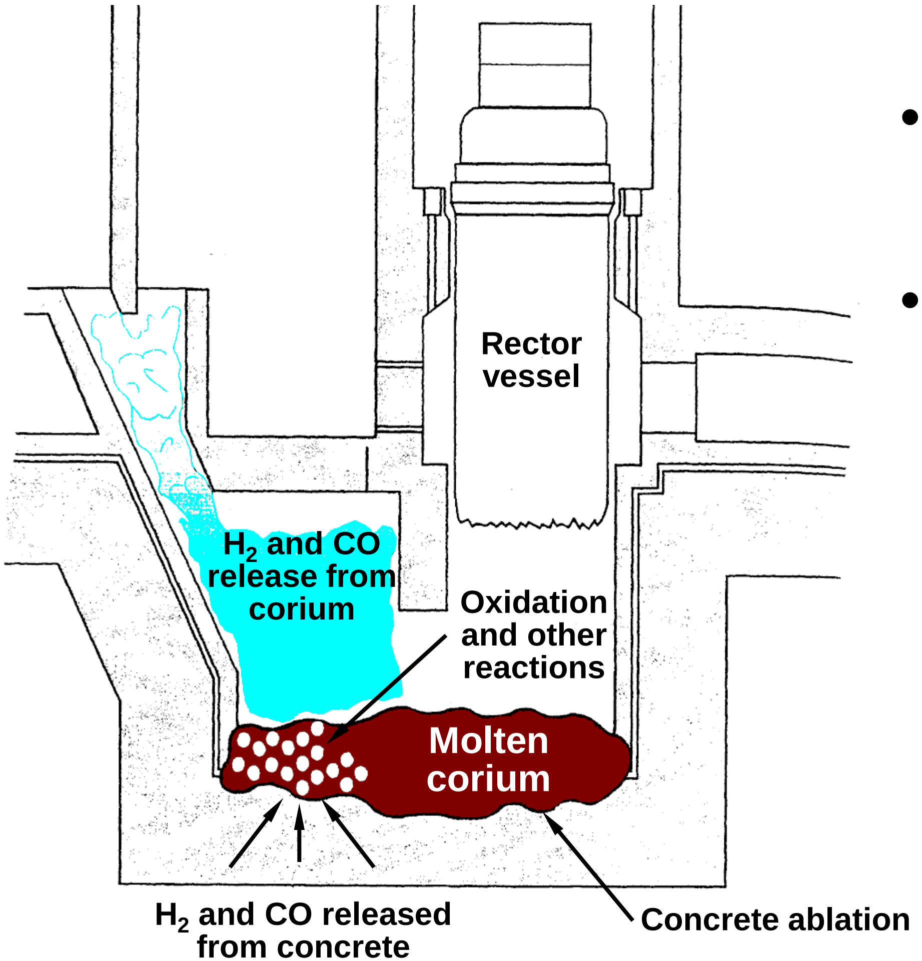
- Sequence
  - Melt composition (amount of unoxidized metals)
  - Melt mass
  - Vessel pressure and failure area
  - Water availability (via containment sprays, etc.)
- Containment design
  - Subcompartment configuration
  - Cavity flow paths
  - Water availability (flooded height)

Recent results suggest very low potential for DCH in large dry or subatmospheric containments.



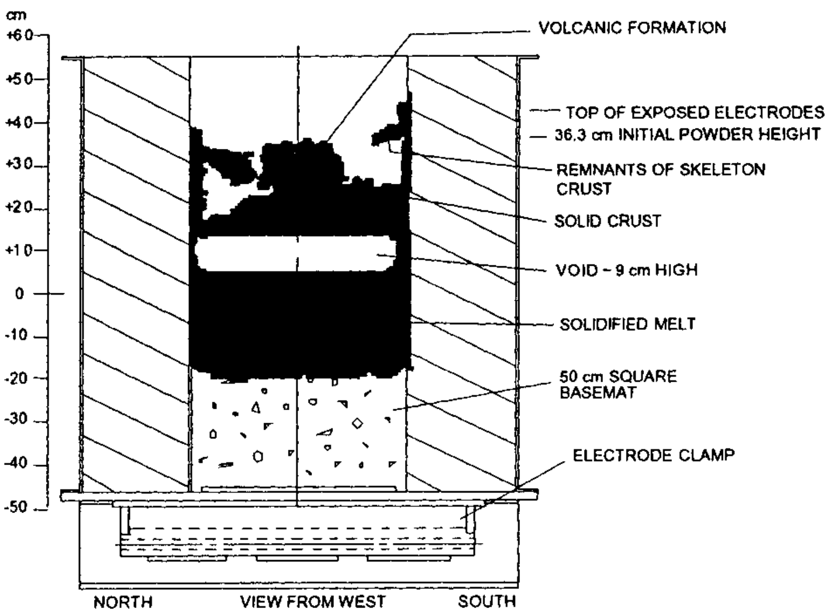
- Compartmentalization (CCFP < 0.01 for most W plants)
- Higher potential for induced RCS depressurization (lower likelihood for HPME)
- Realistic initial melt conditions based on SCDAP/RELAP5 calculations (smaller melt mass, less unoxidized metallics)

Molten Core Concrete Interaction (MCCI) Issues



- Is corium released from the vessel coolable?
- If not, does MCCI lead to:
  - combustible and/or noncondensible gas release?
  - radioactive and/or nonradioactive aerosol release?
  - basemat melt-through/failure

MACE Tests Provide Key MCCI Insights



- Large scale, prototypic tests:
  - 100 to 2000 kg (220 to 4400 lbs) prototypic corium
  - 30 cm x 30 cm to 120 cm x 120 cm (1 ft x 1 ft to 4 ft x 4 ft) concrete basemat area
  - UO_2, ZrO_2, and Zr corium materials heated up to 2350 K (3770 ° F)
  - Electrodes to simulate decay heat
  - Water added after corium melts

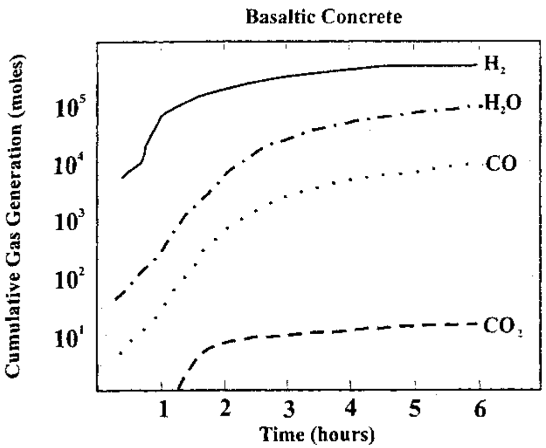
- Observed:
  - High initial heat transfer from corium
  - Significantly lower heat removal after crust forms on upper surface
  - Voiding in corium region beneath crust
  - Pool swelling followed by eruptions enhances heat removal.
- CEA-sponsored VULCANO underway (with ~30-50 kg prototypic materials)

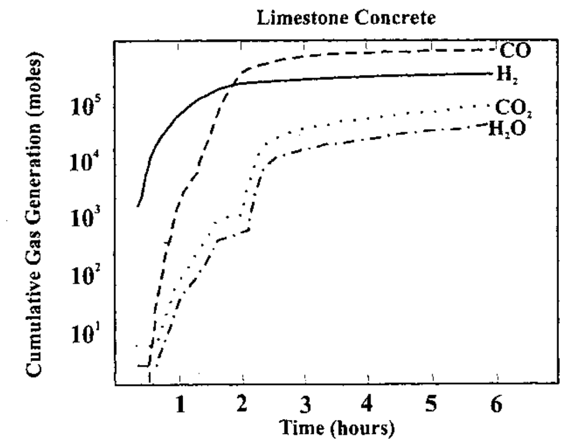
Several Factors Influence MCCI

- Containment design dependent
  - Type of concrete (limestone quickly ablates isotropically and generates more gases than basalt-based/silica-rich concrete)
  - Basemat thickness
  - Cavity size and geometry
- Sequence dependent
  - Melt mass released
  - Melt composition
  - Melt configuration (coolability)
  - Presence of water

Concrete Composition Affects Gas Generation

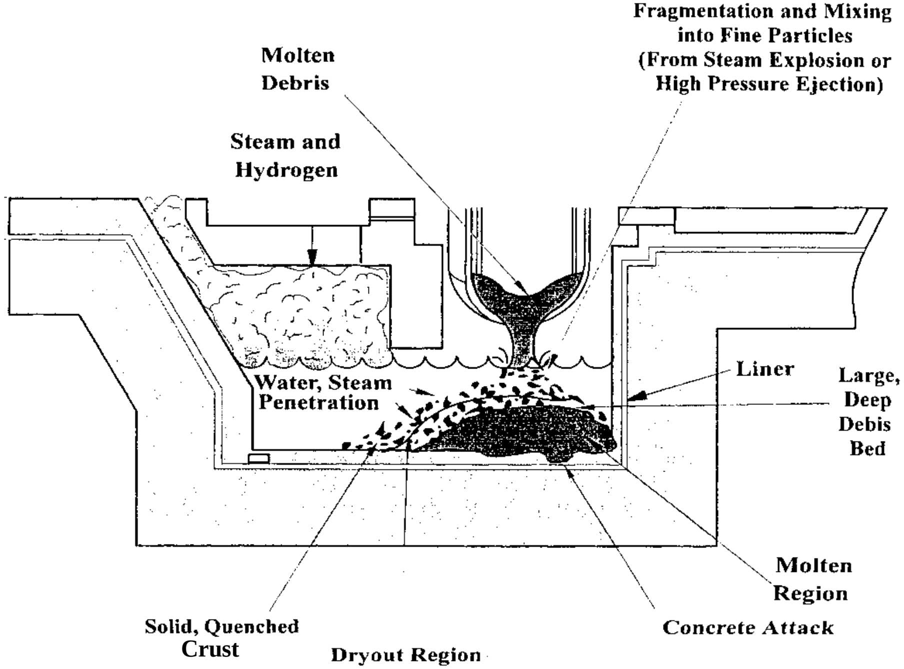
Typical chemical composition (wt.%)			
Oxide	**Basaltic Concrete**	**Limestone Concrete**	**Limestone/Common Sand Concrete**
SiO_2	54.73	3.60	35.70
CaO	8.80	45.40	31.20
Al_2O_3	8.30	1.60	3.60
MgO	6.20	5.67	0.48
Fe_2O_3	6.25	1.20	1.44
K_2O	5.38	0.68	1.22
TiO_2	1.05	0.12	0.18
Na_2O	1.80	0.08	0.82
MnO	-	0.01	0.03
Cr_2O_3	-	0.004	0.014
H_2O	5.00	4.10	4.80
CO_2	1.50	35.70	22.00





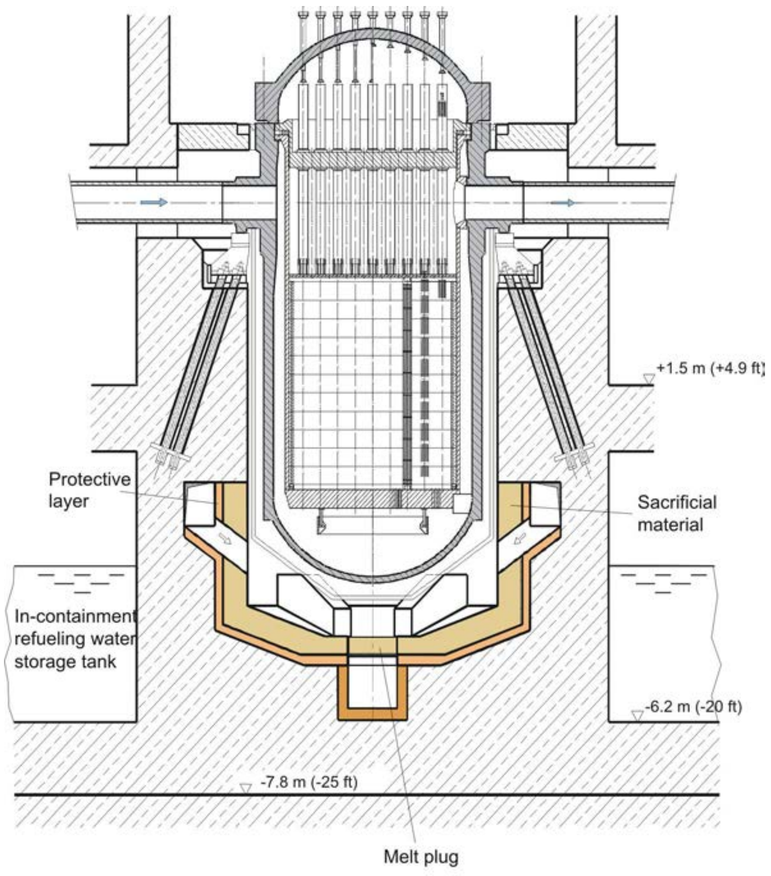
Limestone concrete ablates more rapidly and generates more gases

Presence of Water Does Not Guarantee Coolability

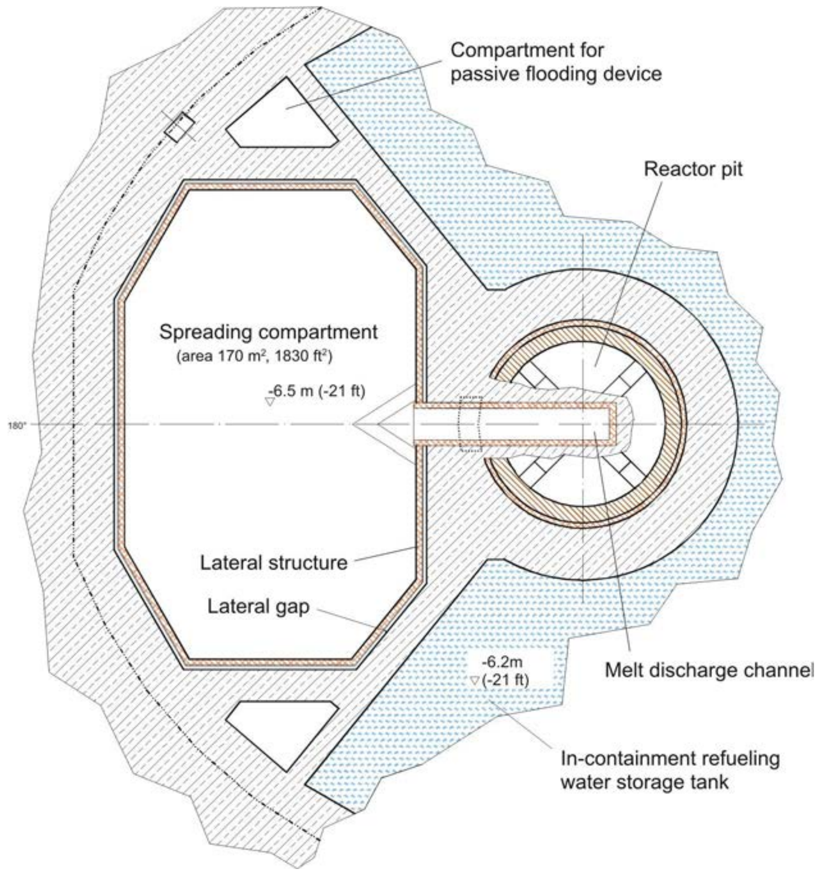


Water can cool released gases and retain some released fission products

EPR Relies on Large Spreading Area to Guarantee Coolability



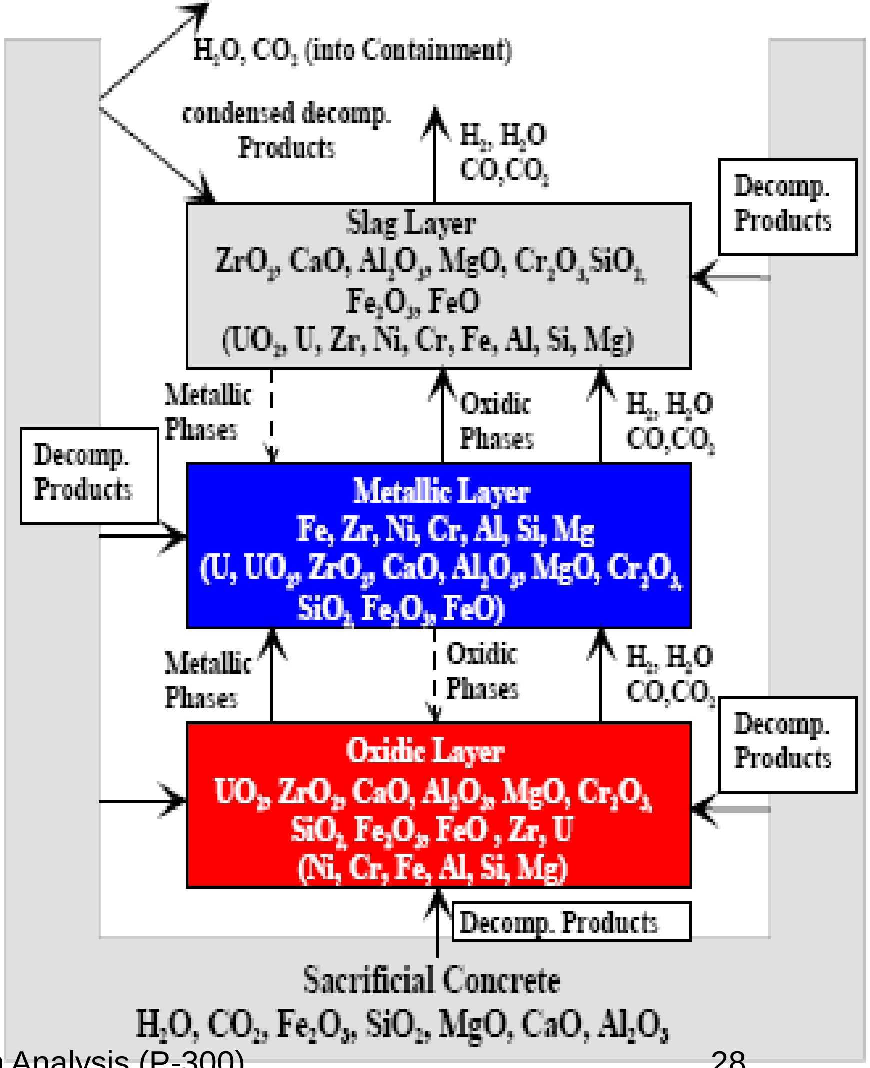
Reactor Cavity



Spreading Area

Incorporating MCCI Benefits in Event Mitigation Strategy – EPR Example

- **Refractory layer ensures melt discharge from cavity only occurs at the gate**
- **Admixture of concrete constituents during MCCI conditions melt to facilitate spreading**
- **Heavy and light oxides fully miscible – oxide layer eventually rises above metallic layer**
- **Metals react with H_2O and CO_2 with H_2 and CO as products**



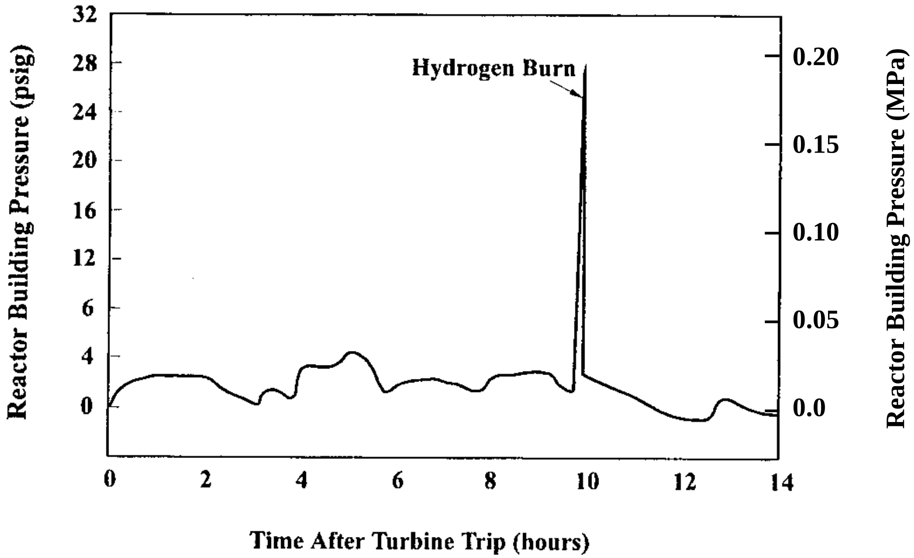
Hydrogen Combustion Issues

$$2\,H_2 + O_2 \rightarrow 2\,H_2O + 57.8 \text{ kcal /mole } H_2 \text{ consumed}$$

$$(229 \text{ Btu /mole } H_2 \text{ consumed})$$

- Under what conditions will hydrogen combustion occur?
- Are pressure loads associated with hydrogen combustion sufficient to threaten containment integrity?

Hydrogen ignition significantly increased TMI-2 containment pressure

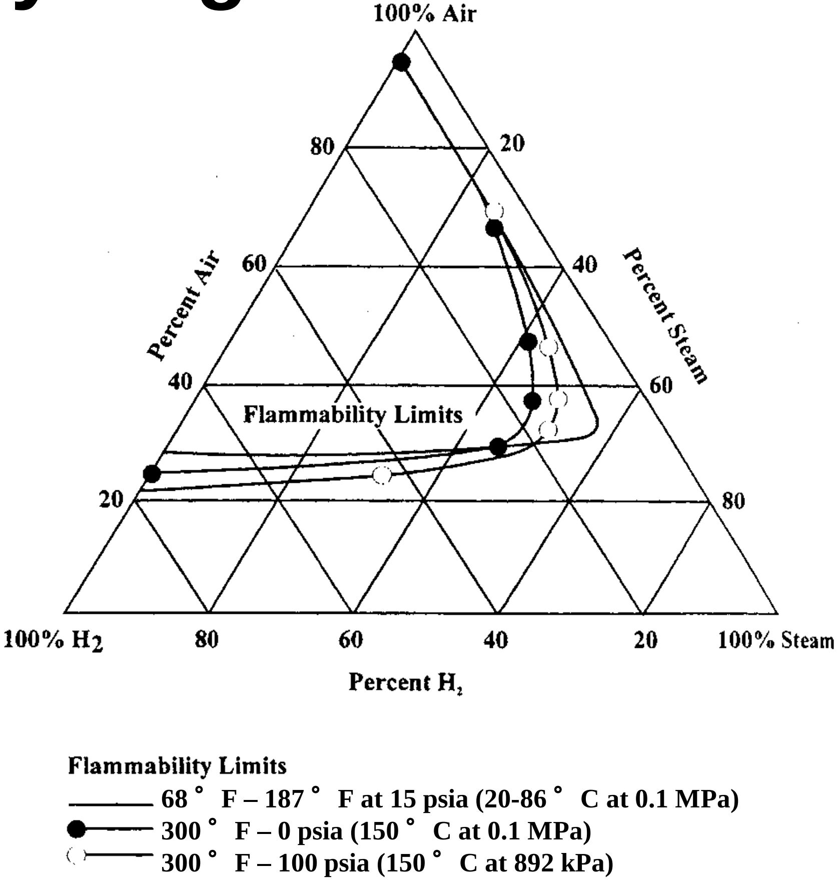


- During core heatup, between 270 to 370 kg (600 to 820 lb_m) hydrogen released through PORVs (~40% of zirconium oxidized)
- Pressure rise corresponds to complete combustion of approximately 8% hydrogen atmosphere
- Concerns exist about the integrity of containments with smaller net free volumes or smaller design pressures exposed to similar threats

Two Types of Combustion

- Deflagration waves
  - requires low energy ignition source
  - requires $[H_2] > 4$ vol% and $[H_2O] < 60$ vol%.
  - travel subsonically (< 35 m/s or < 120 ft/s)
  - heat unburned gases to temperatures high enough for chemical reactions to occur
  - produce quasi-static containment loads
- Detonation waves
  - requires high energy ignition source
  - requires $[H_2] > 18$ vol%
  - travel supersonically (at least 2200 m/s or 7200 ft/s)
  - heat unburned gases by compression
  - produce dynamic or impulsive containment loads in addition to static loads (can generate missiles and challenge containment steel shell).

Shapiro and Moffette Diagram Depicts Hydrogen: Air: Steam Flammability Limits



Limits vary with:

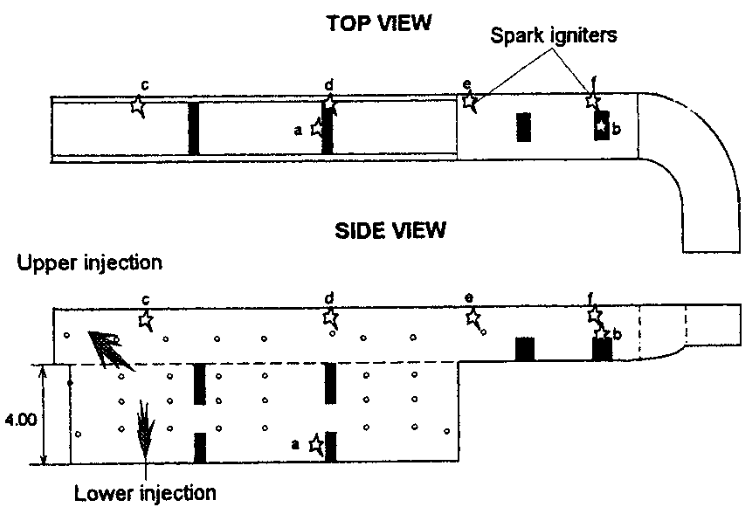
- pressure
- temperature
- presence of steam or other diluents.

Mixture non-flammable if:

- $[H_2]$ < 4 vol%,
- $[O_2]$ < 5 vol%, or
- $[H_2O]$ > 60 vol%

RUT Experimental Data Provides Insights about Hydrogen Ignition

- Series of tests with dynamic hydrogen injection and spark ignition
  - Up to 480 m^3 (17,000 ft^3)
  - 0.6-1.0 kg/s (1.3 – 2.0 lb/s) and 0.1-0.2 kg/s (0.2-0.4 lb/s) H_2 injection
  - Ignition made by electric spark operating at 0.1 and 1 Hz.



- Ignition observed to depend most on:
  - Distance between injection and ignition point
  - Mean H_2 concentration

- Results used to optimize number and location of igniters and develop H_2 combustion criteria
  - σ criterion to estimate risk of flame acceleration
  - 7 λ criterion to assess non-occurrence of DDT

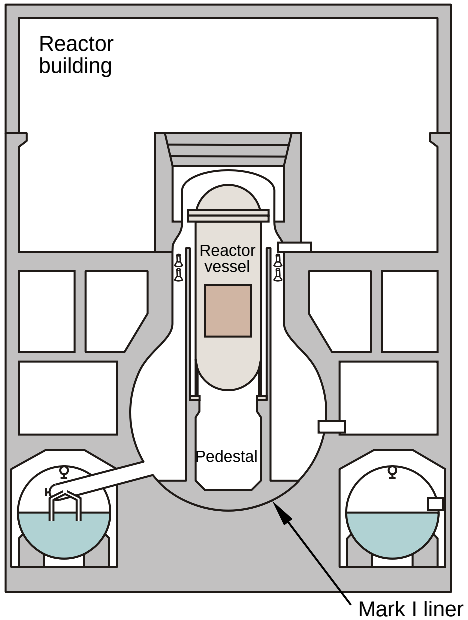
Localized Effects May Be Important

- Higher concentrations of hydrogen
  - near release points
  - under ceilings or dome due to density stratification,
  - near steam removal locations, such as ice condensers, suppression pools, and fan coolers
  - within smaller volume compartments
- Equipment susceptible to high pressure or temperature
- Ignition sources
  - structures / regions at higher temperature
  - electrical equipment sparks

10CFR50.44 Hydrogen Control Requirements Instituted after TMI-2

- All BWR Mark I and Mark II containments must be inerted during normal operation
- Deliberate ignition required in BWR Mark III and PWR ice condenser containments

BWR Mark I Liner or Shell Meltthrough Issues



- Is sufficient melt released?
- Does melt contact carbon steel Mark I liner/shell?
- Is heat load from melt sufficient to fail Mark I liner/shell?

Several Factors Influence Melt-Through

- Design dependent
  - Pedestal door, drywell floor, sump, and downcomer entrance size and geometry
- Sequence dependent
  - Melt mass released
  - Melt composition
  - Melt superheat
  - RCS pressure
  - Presence of water

Mark I Liner Failure Studies Led to Several Actions to Reduce Contribution Potential for Liner Meltthrough

- Mark I Liner failure studies grouped cases by key parameters affecting liner failure
  - Pressure
  - Drywell Flooding
  - Vessel Failure Mode
- Studies recommended several actions to
  - Improve success for vessel depressurization
    - Revised procedures
  - Improve success for drywell flooding
    - Availability of alternate water sources to drywell spray header
    - Revised criteria for initiation of containment sprays
    - Improved diesel pump and spray nozzle designs

Case Study: DCH in Westinghouse Plants with Large Dry Containments or Subatmospheric Containments

DCH Resolution Methodology

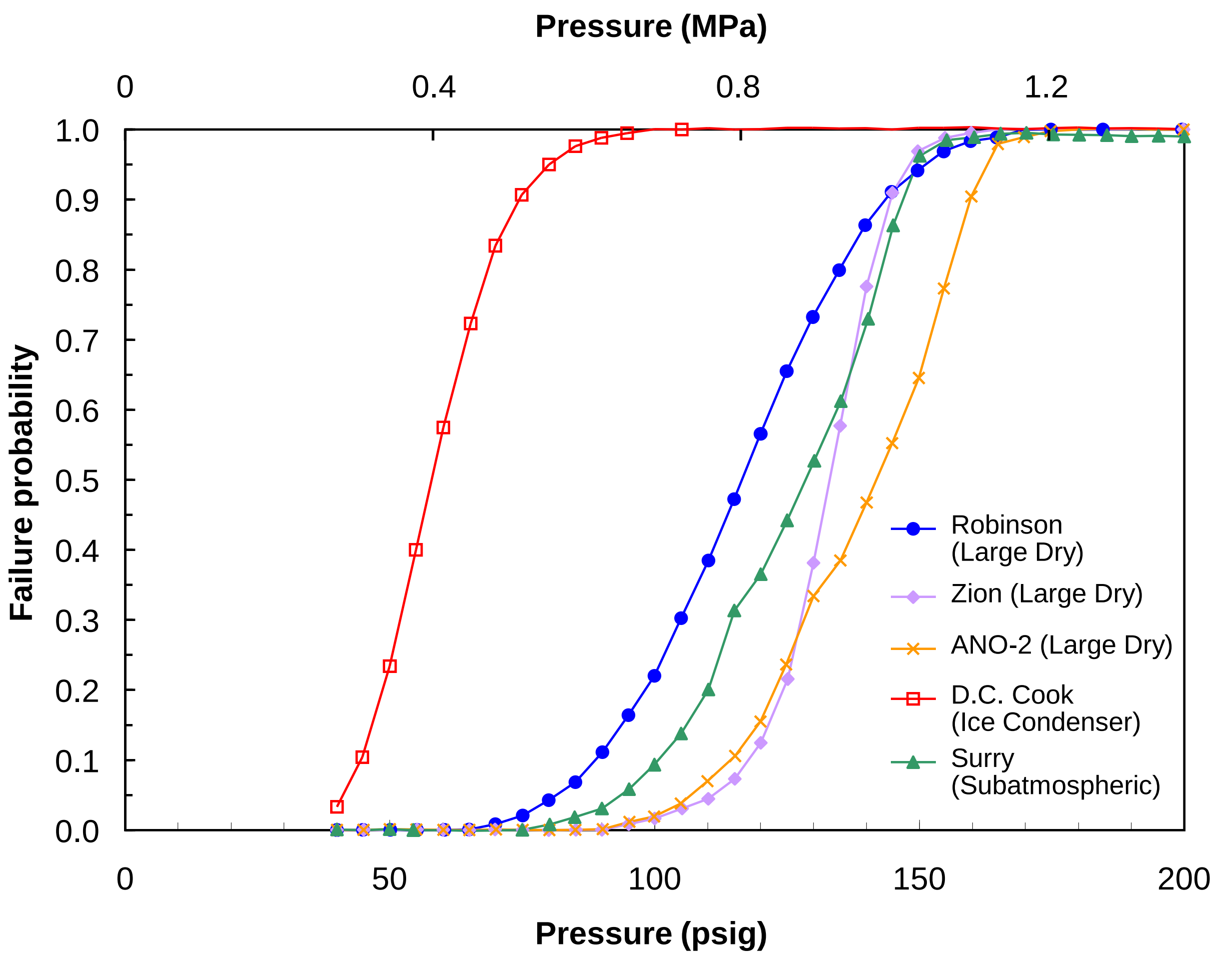
Resolution Criterion:

For events with core damage, threat of early containment failure due to DCH $\leq$ 0.1

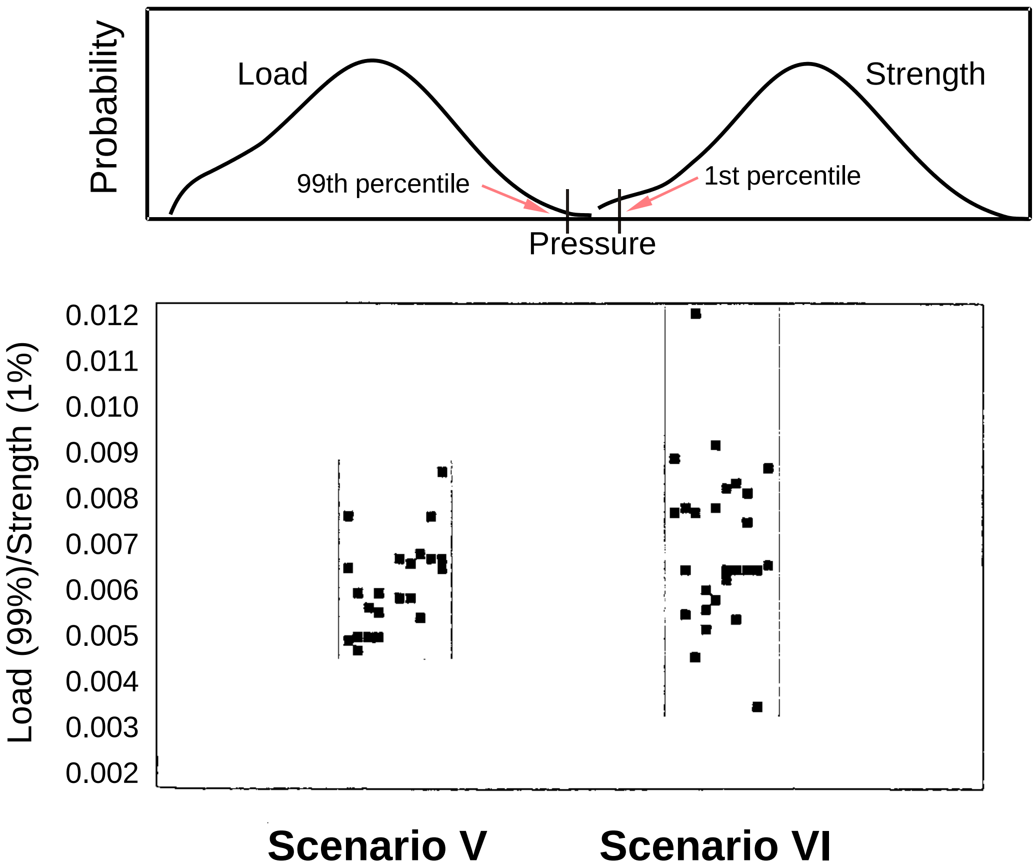
Procedure:

- Analyze several splinter scenarios to envelop conditions for release (melt mass, composition, vessel pressure, etc.)
- Predict containment pressurization pdf.
- Estimate CCFP using plant specific containment fragility curve (from IPEs).
- If CCFP > 0.01 (screening criterion), perform more detailed evaluation, considering probabilities of HPME and/or more refined containment load/strength analysis.

DCH resolution study assumed IPE containment fragility curves

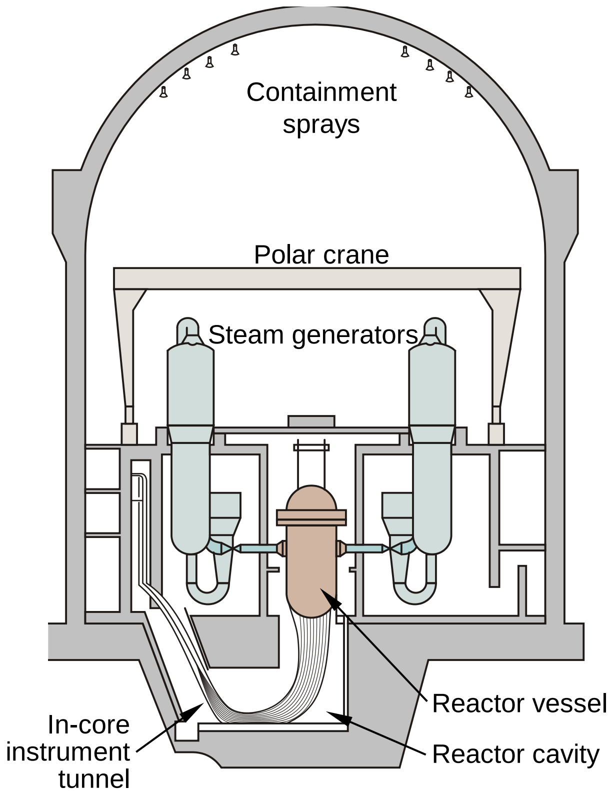


Mean CCFP < 0.01 for all Westinghouse Large Dry and Subatmospheric Containments

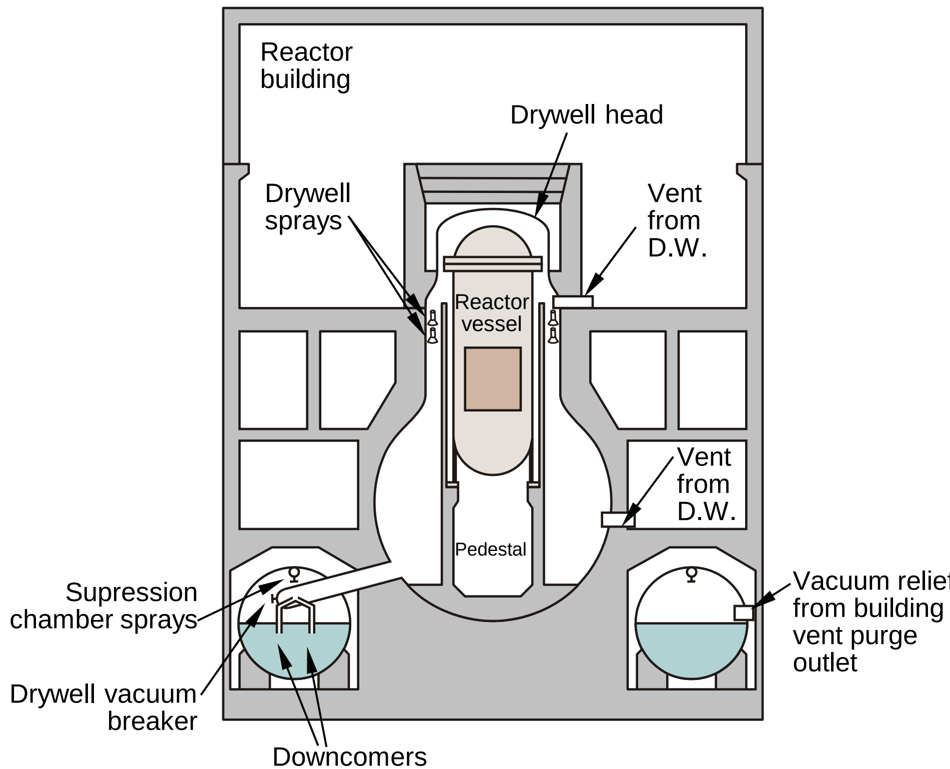


- No intersections of load distributions with fragility distributions for most plants (CCFP ~ 0).
- Finite, but negligible, intersection predicted for H.B. Robinson (broad containment fragility distribution and dome transport characteristics).

Problem: How would DCH analysis change if a Mark I containment were considered?

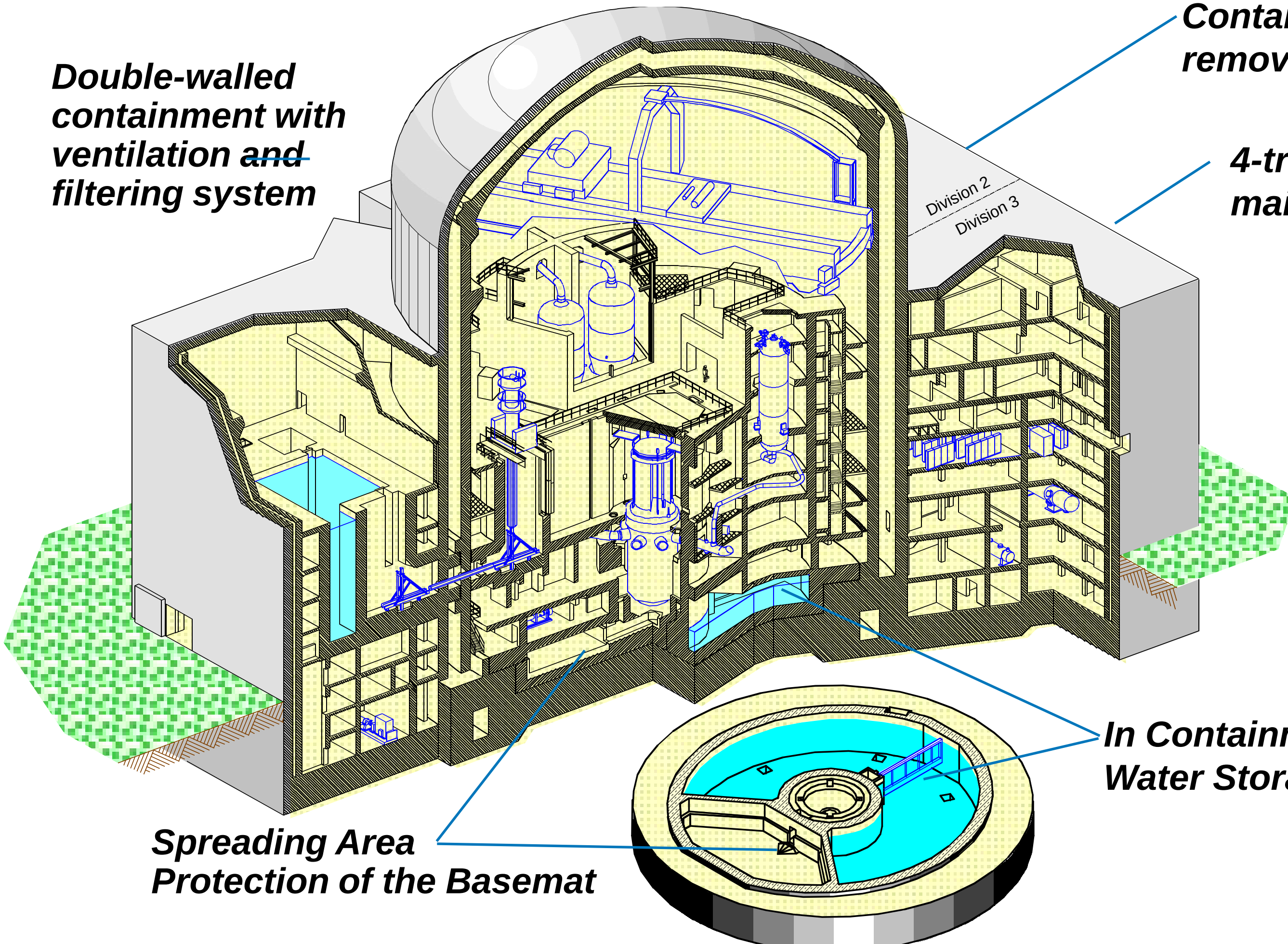


Large dry



Mark I

Problem: How would EPR containment integrity evaluations differ?



Study Questions

- Why is containment failure timing important in assessing the risk associated with an accident sequence?
- State the time period when the following challenges to containment integrity dominate.
  - Steam explosions
  - Direct containment heating
  - Molten core concrete interactions
  - Hydrogen combustion
  - Meltthrough/impingement
- What are key sequence and containment design parameters for evaluating the above challenges to containment integrity?

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Accident Progression Analysis (P-300)

6. CET Development

April 2016

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Idaho National Laboratory

Session Objectives

- To Understand the basic steps and information needs in the CET development process

Level of Detail in CET Varies

- CET models can be very simple or very complex
  - WASH-1400, many IPE’s only consist of 6 to 12 top events in event tree
  - NUREG-1150 APET’s comprised 75 to 125 top events
    - Not displayed graphically

CET Details Determined by Purpose of Level-2 Analysis

- Is objective of Level-2 analysis to support Level-3 (i.e., generate source terms for health consequences)?
- Is objective of Level-2 limited to a containment analysis?
- Is objective to calculate LERF (i.e., Reg Guide 1.174)?
- Each of the above will yield different looking CET, Compare:
  - NUREG-1150 APETs,
  - IPE CETs,
  - LERF CETs (NUREG/CR-6595)

CET Covers Multiple Phases

- Either explicitly or implicitly CET needs to:
  - Delineate boundary conditions (i.e., details of level-1 CD sequence, containment isolation, etc.)
  - Update/establish status of containment systems (e.g., Recovery of AC power)
  - Model progression of accident with respect to actual core damage and RPV/RCS failure
  - Model resulting loads on containment structure
  - Assign probability of release/source-term to each accident sequence

Level-2 Analysis Typically Represented as an Event Tree

- Event trees appropriate modeling choice for chronological progression of a sequence of event
- Ideally, Level-2 analysis would be incorporated into expanded level-1 models (i.e., single integrated ET)
  - Direct linking would better accommodate dependencies and obviate much manual manipulation of intermediate results
- Single integrated model, often not practical
  - Level-2 analyst usually different from Level-1 analyst
  - Modeling and bookkeeping requirements very extensive
  - Large, integrated models more difficult to quantify and review

Potential CET Top Event Sources

- NUREG-1560 (IPE Insights Report) provides a good overview on likely containment failure mechanisms for all containment types
- Specific IPEs could be utilized
- NUREG/CR-6595 outlines relatively simple CETs for use in estimating a screening LERF
- Standardized Plant Analysis Risk (SPAR) model program developed CETs for several PWR plants

Containment Failure Categories

- Bypass Events
  - Vessel failure not required for release
    - Event V or Interfacing System LOCA (ISLOCA)
    - SGTRs
  - Largely determined by level-1 CD sequence information
- Early Failures
  - Early - usually in relation to the timing of vessel failure (i.e., before, during or shortly after vessel failure)
  - Typically within a few hours of the start of core damage
- Late Failures
  - Several hours after vessel failure

Containment Failure

- If containment is not bypassed, need to assess the likelihood and mode of containment failure
- Containment failure mechanisms are scenario dependent
  - Mode of RPV failure has major impact on magnitude of containment challenges
    - e.g., Does RPV fail while RCS is at high pressure or low pressure?

Analyze Containment Loads

- Many challenges need to be considered
  - Internal pressure rises (usually considered “static”)
  - High temperatures
  - Thermo-mechanical erosion of concrete structures (molten core concrete interaction)
  - localized dynamic loads (e.g. shock waves and internally generated missiles)
- Analyses often distinguish between catastrophic failures and leaks
- Location of failure is also important
  - e.g., wetwell versus drywell

Loads Can be Characterized at Different Levels of Detail

- A series of specific "small" estimates can be made, or a single estimate of the total pressure
  - What is the pressure?
  - Add the pressure from a number of contributors
    - Initial pressure
    - Pressure from DCH
    - Pressure from steam explosion
    - Pressure from hydrogen combustion
    - etc.
- Both approaches have been used

Estimate Challenges to Containment Integrity (for example)

- Hydrogen generation and combustion
- Fuel-coolant interactions (steam explosions)
- Melt/debris ejection following RV failure (DCH)
- Debris bed coolability and core-concrete interaction
- Shell melt-through failure in Mark-I containments
- Long-term overpressure
- Basemat melt-through
- Each phenomena depends on accident progression characteristics and containment design

Early Containment Failures

- Early containment failure mechanisms include:
  - direct contact of the core debris with steel containments
  - rapid pressure and temperature loads
  - hydrogen combustion
  - missiles generated by fuel-coolant interactions (sometimes referred to as steam explosions or alpha-mode failures)
  - containment isolation failures
  - sometimes include containment venting (depending on when vents are opened)

Late Containment Failures

- Late containment failures include:
  – gradual pressure and temperature increases
  – hydrogen combustion
  – basemat melt-through by core debris
  – sometimes include containment venting (depending on when vents are opened)

CET Endstate Defines Source Term

- Primary purpose of CET
  - Frequency and characteristics of source term
    - Possibly as simple as large and early (LERF)
    - Possibly very complex
      - Amount of radioactive material released
      - Start and end time of release
      - Energy of release
      - Location (elevation) of release

CET End-State Descriptions Vary

- For example, common output forms include:
  - Large Early Release Frequency (LERF)
    - Large early containment failure plus bypass
  - Containment Failure (CF) Mode Descriptions
    - Accident Progression Bins
    - Often segregated into:
      - Early CF, Late CF and Containment Bypass
  - Source Term Descriptions
    - For input to a Level-3 (Consequence) analysis

CET Provides Needed Source Term Information

- Specific information needed determined by the source term analysis method
- Example: SEQSOR (Sequoyah NUREG-1150)
  - Simple, fast-running parametric code that extrapolates and interpolates results from more detailed mechanistic codes and expert judgement
  - Early and late radioactive release fractions calculated for nine isotope classes (comprising 60 radionuclides)
  - Information needed by SEQSOR organized into a 14-character Accident Progression Bin (APB) vector

SEQSOR Input (APB Vector)

1 Time of containment failure
2 Period in which sprays operate
3 Occurrence of CCI
4 RCS press before VB
5 Mode of VB
6 SGTR
7 Amount of core available for CCI
8 Fraction of Zr oxidized in vessel
9 Fraction of core in HPME
10 Size or type of containment failure
11 # of large holes in RCS after VB
12 Early ice condenser function
13 Late ice condenser function
14 Status of air return fans

Example: SEQSOR Characteristic 1 - Containment Failure Time

A	*V-Dry*	*Event V, releases not scrubbed by fire suppression sprays*
B	*V-Wet*	*Event V, releases scrubbed by fire suppression sprays*
C	*CF-E*	*Containment failure during core degradation*
D	*CF-VB*	*Containment failure at vessel breach*
E	*CF-L*	*Late containment failure (during initial CCI, nominally a few hours after VB)*
F	*CF-VL*	*Very late containment failure (from 12 to 24 hours after VB*
G	*NoCF*	*No containment failure*

Parametric Source Term Code

- XSOR codes written specifically for NUREG-1150 plants
- Parametric Source Term (PST) code developed in 1996 under Accident Sequence Precursor (ASP) program
  - PST developed to provide source terms for all U.S. PWRs
  - Estimates source terms for 9 release classes comprising approximately 60 isotopes

PST Input Uses 10-Character Vector

1	Containment Failure Mode
2	Status of Containment Heat Removal Systems
3	Occurrence of Core Concrete Interactions
4	RCS Pressure at Vessel Breach
5	Mode of Vessel Breach
6	Occurrence of SGTR
7	Presence of Water in Reactor Cavity
8	Amount of Oxidation in Vessel
9	Containment Failure Size
10	Core Damage Time

Example: PST Characteristic 1 - Containment Failure Mode

ID	*Definition*
A	*Containment bypass*
B	*Containment not isolated*
C	*Early containment failure (near time of vessel breach)*
D	*Late containment failure*
E	*No containment failure*

Most Level-2 Analyses Involve a Mix of Supporting Information

- Plant-specific code calculation
  - MAAP, MELCOR, SCDAP/RELAP5
- Analyses from other prior PRAs or severe accident studies
  - NUREG-1150, IPEs
- Engineering analyses of specific issues
  - Threat from hydrogen combustion
- Experimental data
  - Debris coolability

Accident Progression Phase	Containment Failure Mode	Phenomena or Mechanism	Lg Dry	Ice Cond	Mark-I	Mark-II	Mark-III
Bypass	ISLOCA		Yes	Yes	Yes	Yes	Yes
	SGTR		Yes	Yes	No	No	No
	Induced SGTR		Yes	Yes	No	No	No
	Induced Isol Cond tube failure		No	No	BWR/2&e3	No	No
CF before VB	Isolation Failure (includes pre-existing leak)		Yes	Yes	Yes	Yes	Yes
	Venting		No	No	Yes	Yes	Yes
	Over Pressure	Steam	Yes	Yes	Yes	Yes	Yes
		H2 combustion	Yes	Yes (SBO)	inerted	inerted	Yes (SBO)
CF at VB							
LP-RCS	IVSE (FCI)		Yes	Yes	Yes	Yes	Yes
	EVSE (FCI)		Yes	Yes	Yes	Yes	Yes
	H2 combustion		Yes	Yes	inerted	inerted	Yes
	Liner (Shell) Melt-Thru		No	No	Yes	No	No
HP-RCS	IVSE (FCI)		Yes	Yes	Yes	Yes	Yes
	HPME (RPV blowdown)	DCH	Yes	Yes	Yes	Yes	Yes
		Steam	Yes	Yes	Yes	Yes	Yes
		H2 combustion	Yes	Yes (SBO)	inerted	inerted	Yes (SBO)
		Direct Impingement	Yes	Yes	No	Yes	Yes
CF after VB	Venting		No	No	Yes	Yes	Yes
	Over Pressure (CCI)	Steam	Yes	Yes	Yes	Yes	Yes
		Non-Cond.	Yes	Yes	Yes	Yes	Yes
		H2 combustion	Yes	Yes	Yes	Yes	Yes
	Basemat melt-thru		Yes	Yes	Yes	Yes	Yes

Dry Cavity	Some steam produced, but core concrete interaction (CCI) can produce H2 and non-condensible gas	
Wet Cavity	coolable geometry	Large amount of steam but no CCI
	non-coolable	Steam plus H2 and non-cond. gas (from CCI)
Ice Condenser and Mark III	H2 combustion	possible only if igniters have failed (i.e., SBO)
Direct Impingement	Depends on geometry of reactor cavity	
	[i.e., does a direct path (instrument tunnel) exist for molten core to contact containment wall?]	
	Also, only important for steel shell containments	
Over Pressure	Steam - requires failure of containment heat removal (CHR)	
IVSE	In-Vessel Steam Explosion (also see alpha-mode, below)	
EVSE	Ex-Vessel Steam Explosion	
FCI	Fuel-Coolant Interaction	Such interactions can lead to steam explosions (encompasses both IVSE and EVSE)
alpha-mode	Scenario where-by an IVSE breaks the vessel head free with such force that its impact on containment results in	
	containment failure, currently judged a very low probability event	
BWR/2&e3	Only BWR /2 and early /3 designs include isolation condensers	

SPAR Level-2 Models (SAPHIRE ver. 7)

April 2016

www.inl.gov

Idaho National Laboratory

Work Began July 2008

- July 2008: Work began to develop integrated Level-1/Level-2 SPAR model using SAPHIRE ver. 7
- SOW specified three models
  - Surry
  - Peach Bottom
  - Sequoyah

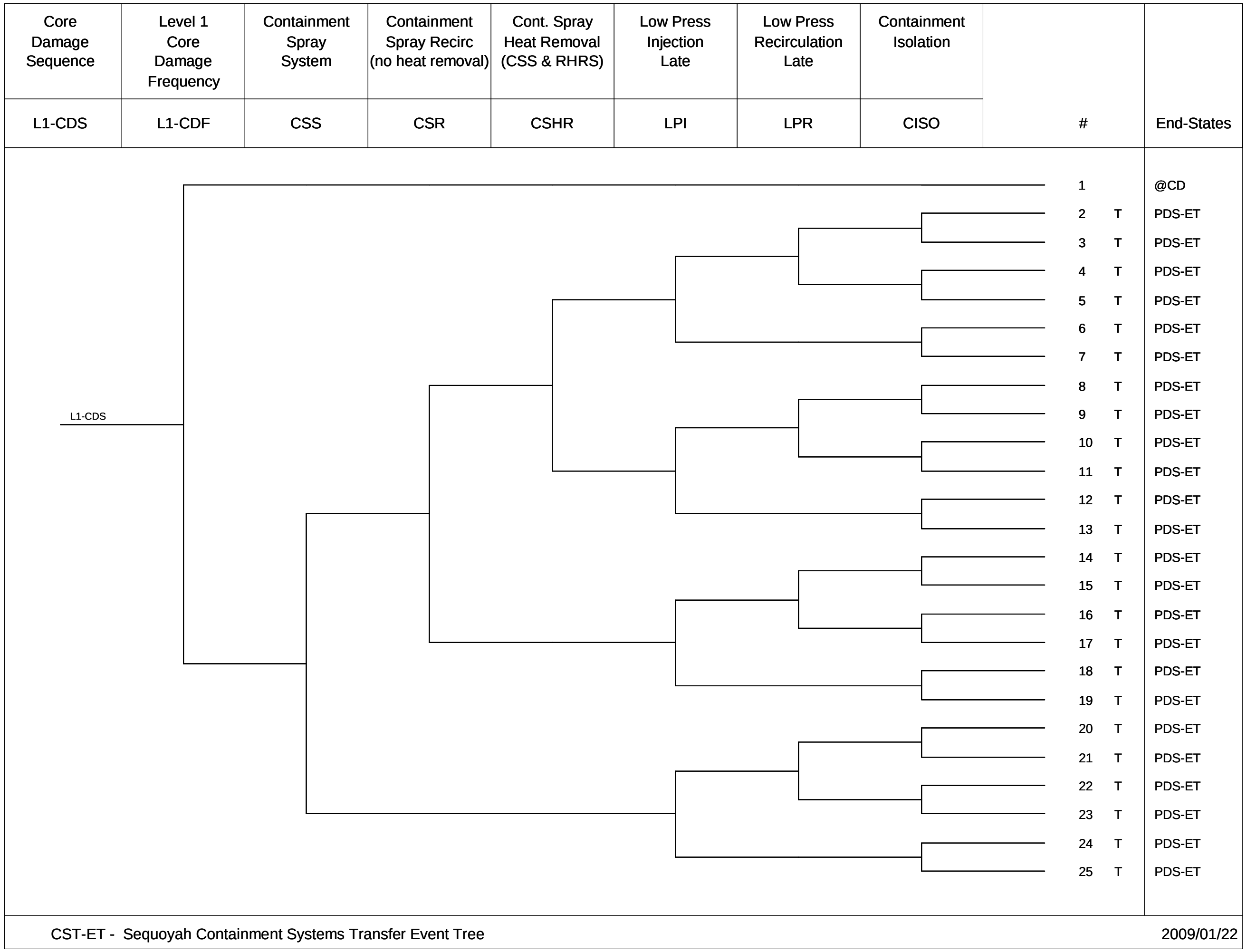
Level-2 Modeling Relies on Series of Event Trees Linked Together

- Level-1 core damage sequences extended using Containment Systems Transfer Event Tree (CST-ET)
  - Simple transfer from Level-1 ET (sometimes called Bridge Event Tree)
- CST-ET then transfers to Plant Damage State Event Tree (PDS-ET)
  - Binning of CD sequences to PDSs provides detailed characteristics on each CD sequence
  - PDS becomes “Initiating Event” for level-2 portion of analysis
  - Only PDS identifier and associated frequency are transferred to level-2

CST-ET Simple Transfer from CD ET

- Objective is to capture dependencies between level-1 systems analysis and level-2 systems analysis
- Also referred to a Bridge Tree
- Level-1 SPAR models commonly use event tree transfers – this is just one more
- However, top event substitutions via logic rules need to be coordinated between level-1 event trees and CST-ET

CST-ET for Sequoyah



CST-ET End States Transfer to PDS-ET

- PDS-ET assigns each CD-containment systems sequence to one of the Plant Damage States
- PDS-ET logic rules search on combined level-1/CST-ET sequence logic
- Complete sequence logic carried through to PDS binning process
- Process relies on two types of SAPHIRE rules
  - Logic rules for development of sequence logic
  - Partitioning rules for generating PDS vector information

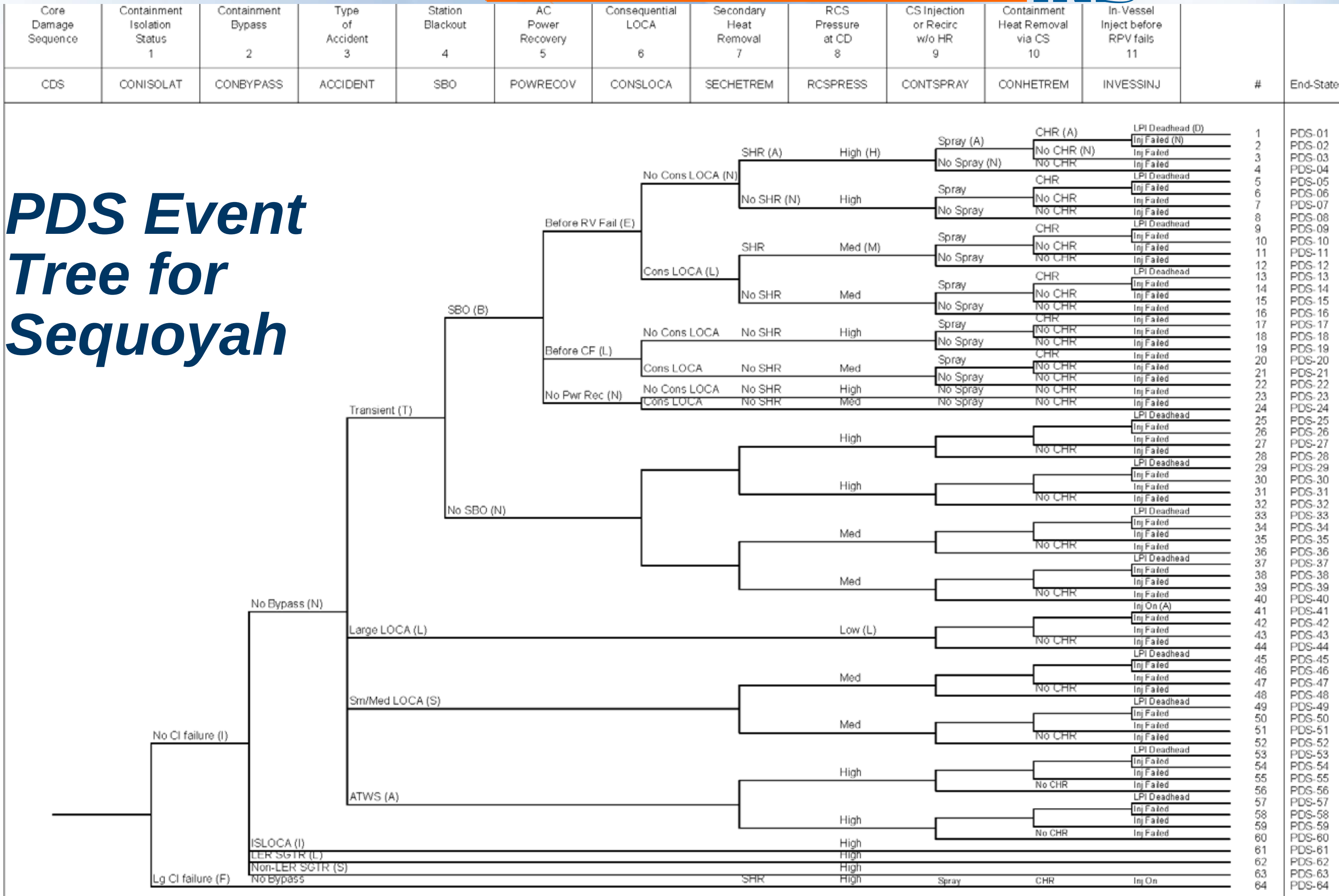
PDS Identifier Uses Vector Format

- SAPHIRE capable of producing end-state information in two ways
  - End state identified on event tree
    - PDS-#
  - End state generated via “Partition” rules
    - Partition rules used to produce PDS vector
- PDS captures 11 characteristics of CD sequence
  - Each position of PDS vector associated with one of the 11 characteristics (top events on PDS-ET)
    - PDS-ABCDEFGHIJK

PDS Characteristics for Sequoyah

1. Containment isolation Status
2. Containment bypass status
3. Type of CD accident
4. Occurrence of SBO
5. Status of AC power recovery
6. Occurrence of severe accident induced LOCA
7. Status of secondary side heat removal
8. RCS pressure at CD
9. Status of containment spray (CS)
10. Operation of CS for containment heat removal
11. Status of In-vessel injection before RPV fails

PDS Event Tree for Sequoyah



PDS-ET - Sequoyah Plant Damage State Event Tree

2008/12/08

PDS-ET Logic Rules Example

```
| 2 - CONBYPASS
| Containment Bypass
| Branch[0] = No Bypass
| Branch[1] = ISLOCA
| Branch[2] = Large Early Release SGTR
| Branch[3] = No-LER SGTR

| if ISLOCA initiating event
 if init(IE-ISL-HPI) + init(IE-ISL-LPI) + init(IE-ISL-RHR) then
 /CONBYPASS = skip(CONBYPASS);
 CONBYPASS[1] = DE-ISLOCA;      | set DE to 1.0
 CONBYPASS[2] = skip(CONBYPASS);
 CONBYPASS[3] = skip(CONBYPASS);

| SGTR but no LER
 elsif init(IE-SGTR) * /FW * /SGI * (/SSC-SGTR + /SSC1)
    then
 /CONBYPASS = skip(CONBYPASS);
 CONBYPASS[1] =  skip(CONBYPASS);
 CONBYPASS[2] = skip(CONBYPASS);
 CONBYPASS[3] = DE-NLR-SGTR;   | set DE to 1.0

| SGTR with LER
 elsif init(IE-SGTR) then
 /CONBYPASS = skip(CONBYPASS);
 CONBYPASS[1] = skip(CONBYPASS);
 CONBYPASS[2] = DE-LER-SGTR;   | set DE to
   1.0
 CONBYPASS[3] = skip(CONBYPASS);

| Default to No-Bypass
 else
 /CONBYPASS = DE-N-NOBYPASS;
  | complimented, so set to zero
 CONBYPASS[1] = skip(CONBYPASS);
 CONBYPASS[2] = skip(CONBYPASS);
 CONBYPASS[3] = skip(CONBYPASS);
 endif
```

PDS Serves as an Intermediate Calculation Point

- PDS-ET Logic Directs Sequence Freq to Appropriate End-State
- Process referred to as Binning
  - PDS-ET end states only identified with a number (e.g., PDS-23, PDS-41)
- PDS will be the start of the severe accident analysis
  - i.e., will be the “initiating” event for the containment analysis
- Containment Event Tree (CET) is the “real” level-2 PRA
  - (NUREG-1150 used the name APET – Accident Progression Event Tree, a much more detailed CET)

PDS Vector Generated Via Partitioning

- SAPHIRE term used to describe process of allocating sequence cut sets using rules
  - Partitioning can be done on the sequence cut-sets or on sequence logic (as was done for SPAR)
  - SAPHIRE allows partitioning rules to construct the PDS vector
  - Partitioning generates an alternate version of the event tree end-state
    - E.g., PDS-35 ≡ PDS-INTNZLAMANZ

PDS Vector Partitioning Example

- 2
- CONBYPASS - Containment Bypass
- Branch[0] = No Bypass (N)
- Branch[1] = ISLOCA (I)
- Branch[2] = Large Early Release SGTR (L)
- Branch[3] = SGTR but not a Large Early Release (S)
- Define Partition Macros (PM) for top event parameters

PM-NO-BYPASS = SYSTEM(/DE-N-NOBYPASS);

PM-ISLOCA = SYSTEM(DE-ISLOCA);

PM-LER-SGTR = SYSTEM(DE-LER-SGTR);

PM-SGTR = SYSTEM(DE-NLR-SGTR);

```
If PM-NO-BYPASS then
  GlobalPartition = "PDS-?N";

  Elsif PM-ISLOCA then
  GlobalPartition = "PDS-?I";

  Elsif PM-LER-SGTR then
  GlobalPartition = "PDS-?L";

  Elsif PM-SGTR then
  GlobalPartition = "PDS-?S";

  Else
  GlobalPartition = "PDS-?Z";

endif
```

PDS Results for Sequoyah

PDS #	PDS Vector	Frequency
SBO		
PDS-08	PDS-INTBENNHNNN	3.80E-06
PDS-16	PDS-INTBELNMNNN	1.48E-06
PDS-19	PDS-INTBLNNHNNN	1.09E-06
PDS-22	PDS-INTBLLNMNNN	3.57E-07
PDS-23	PDS-INTBNNNHNNN	9.69E-07
PDS-24	PDS-INTBNLNMNNN	4.10E-07
SBO Subtotal		8.10E-06
Trans		
PDS-25	PDS-INTNZNAHAAD	2.25E-10
PDS-26	PDS-INTNZNAHAAN	4.19E-07
PDS-27	PDS-INTNZNAHANZ	1.40E-07
PDS-28	PDS-INTNZNAHNNN	3.68E-11
PDS-29	PDS-INTNZNNHAAD	1.84E-07
PDS-30	PDS-INTNZNNHAAN	9.97E-09
PDS-31	PDS-INTNZNNHANZ	9.73E-08
PDS-32	PDS-INTNZNNHNNN	0.00E+00
PDS-33	PDS-INTNZLAMAAD	2.76E-08
PDS-34	PDS-INTNZLAMAAN	1.01E-06
PDS-35	PDS-INTNZLAMANZ	2.98E-05
PDS-36	PDS-INTNZLAMNNN	7.27E-09
PDS-37	PDS-INTNZLNMAAD	0.00E+00
PDS-38	PDS-INTNZLNMAAN	3.06E-11
PDS-39	PDS-INTNZLNMANZ	1.04E-09
Transient Subtotal		3.17E-05

LLOCA		
PDS-41	PDS-INLNZZZLAAA	1.00E-07
PDS-42	PDS-INLNZZZLAAN	1.01E-08
LLOCA Subtotal		1.10E-07
S/M LOCA		
PDS-45	PDS-INSNZZAMAAD	7.38E-06
PDS-46	PDS-INSNZZAMAAN	1.62E-06
PDS-47	PDS-INSNZZAMANZ	2.93E-08
PDS-48	PDS-INSNZZAMNNN	1.20E-09
PDS-49	PDS-INSNZZNMAAD	0.00E+00
PDS-50	PDS-INSNZZNMAAN	0.00E+00
PDS-51	PDS-INSNZZNMANZ	0.00E+00
S/M LOCA Subtotal		9.02E-06
ATWS		
PDS-53	PDS-INANZZAHAAD	1.27E-07
PDS-54	PDS-INANZZAHAAN	1.39E-10
PDS-55	PDS-INANZZAHANZ	0.00E+00
PDS-56	PDS-INANZZAHNNN	0.00E+00
PDS-57	PDS-INANZZNHAAD	3.09E-07
PDS-58	PDS-INANZZNHAAN	0.00E+00
PDS-60	PDS-INANZZNHNNN	8.02E-11
ATWS Subtotal		4.37E-07
ISLOCA		
PDS-61	PDS-IIZNZZZHZNZ	5.70E-07
LER SGTR		
PDS-62	PDS-ILZNZZZHZNZ	8.95E-08
nLER SGTR		
PDS-63	PDS-ISZNZZZHZNZ	4.39E-08
CI Failure		
PDS-64	PDS-FZZNZZZHZNZ	2.74E-07
Total		5.04E-05

Each PDS Vector Becomes an IE

- SAPHIRE converts each PDS vector into an Initiating Event
  - SAPHIRE automatically generates a “dummy” event tree with PDS name
  - This is directed by the user in the Partitioning Rules
- PDS vector ET then transfers to CET for actual severe accident (i.e., level-2) analysis
  - Note: PDSs are just core damage sequences with additional descriptive information on details of the accident

CET Models Plant Response to CDS

- Containment Event Tree models the response of the Reactor Pressure Vessel (RPV) and containment to the Core Damage Sequence (CDS)
  - Mode and severity of RPV failure affects challenge to containment structure
- CET logic rules query status of plant systems and then assign appropriate probabilities to various phenomena in severe accident progression
  - PDS vector contains information on status of plant systems

CET Top Events

CONBYPAS – Status of Containment Bypass

RCSFAIL – Mode of Induced RCS failure

SGTRPATH – Path of release from SGTR

INVCOOL – Status of core debris cooling in-vessel

CF-EARLY – Mode of Early Cont. Failure

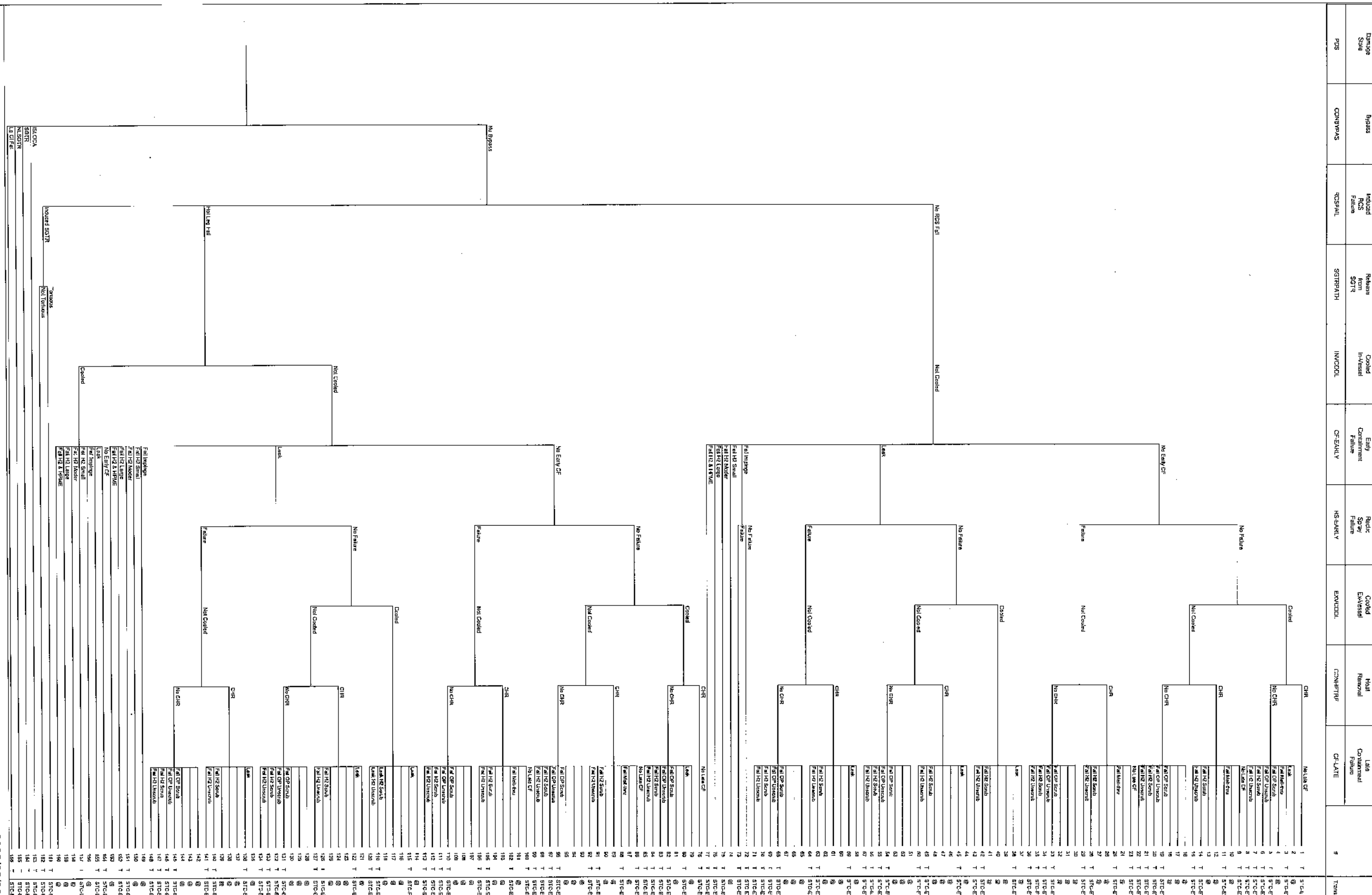
RS-EARLY – Early status of Recirc. Spray

EXVCOOL – Status of core debris cooling ex-vessel

CONHETRE – Status of Cont. Heat Removal

CF-LATE – Mode of Late Cont. Failure

CET - Sequoyah Containment Event Tree (Level-2 model)



2009/02/09

CET Logic Based on PDS Vector

```
| Using positions 1 and 2 in PDS vector
| CONBYPAS - Containment Bypass
| Branch[0] = No Bypass (N)
| Branch[1] = ISLOCA (I)
| Branch[2] = Large Early Release SGTR (L)
| Branch[3] = SGTR but not a Large Early Release (S)
| Branch[4] = Large Containment Isolation Failure

if  "PDS-F*" then
  /CONBYPAS = SKIP(CONBYPAS);
  CONBYPAS[1] = SKIP(CONBYPAS);
  CONBYPAS[2] = SKIP(CONBYPAS);
  CONBYPAS[3] = SKIP(CONBYPAS);
  CONBYPAS[4] = SYS-TRUE;

 elsif "PDS-?I*" then
  /CONBYPAS = SKIP(CONBYPAS);
  CONBYPAS[1] = SYS-TRUE;
  CONBYPAS[2] = SKIP(CONBYPAS);
  CONBYPAS[3] = SKIP(CONBYPAS);
  CONBYPAS[4] = SKIP(CONBYPAS);

elsif  "PDS-?L*" then
  /CONBYPAS = SKIP(CONBYPAS);
  CONBYPAS[1] = SKIP(CONBYPAS);
  CONBYPAS[2] = SYS-TRUE;
  CONBYPAS[3] = SKIP(CONBYPAS);
  CONBYPAS[4] = SKIP(CONBYPAS);

elsif  "PDS-?S*" then
  /CONBYPAS = SKIP(CONBYPAS);
  CONBYPAS[1] = SKIP(CONBYPAS);
  CONBYPAS[2] = SKIP(CONBYPAS);
  CONBYPAS[3] = SYS-TRUE;
  CONBYPAS[4] = SKIP(CONBYPAS);

else  | default to No Bypass
  /CONBYPAS = SYS-FALSE;
        | complimented, so becomes a TRUE
  CONBYPAS[1] = SKIP(CONBYPAS);
  CONBYPAS[2] = SKIP(CONBYPAS);
  CONBYPAS[3] = SKIP(CONBYPAS);
  CONBYPAS[4] = SKIP(CONBYPAS);

endif
```

CET End States Transfer to STC-ET

- Source Term Category Event Tree (STC-ET) sorts the CET sequences into release categories
  - Direct event tree transfer
- Logic rules in STC-ET used to query CET top event logic

STC-ET Assigns Release Category to Each CET Sequence

STC-ET Collects Sequence Frequencies

Release Category	Description	Frequency
REL-LER	Large Early	4.76E-06
REL-MER	Medium Early	3.03E-06
REL-SER	Small Early	0E+00
REL-LLR	Large Late	1.14E-05
REL-MLR	Medium Late	1.46E-05
REL-SLR	Small Late	6.40E-06
REL-LK	Leak	1.27E-07
REL-NO	No Release	1.00E-05
Total		5.03E-05

7. Severe Accident Simulation Codes

- Introduction
- Codes – SCDAP/RELAP5, MELCOR, MAAP MAAP
- Case Studies
- Methods
- Study Questions
- References

Objectives

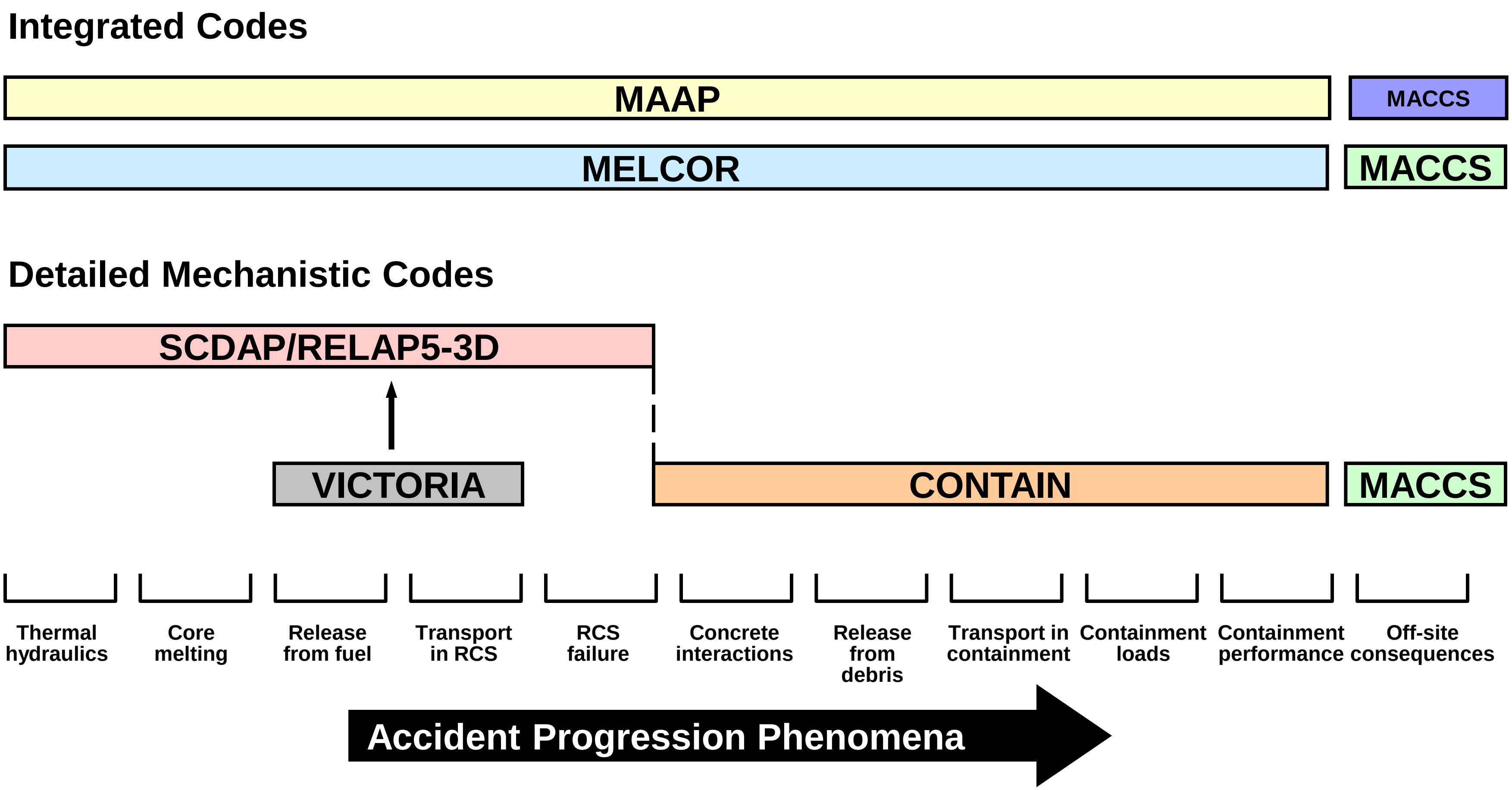
- Identify various methods used in the US for modeling severe accident progression.
- Understand what phenomena are modeled by each method.
- Understand differences in modeling approaches that may impact code predictions.

Code Design Philosophies Differ

Method	Developer/Sponsor	Design Philosophy
SCDAP/RELAP5 ***SCDAP/RELAP5-3D©***	ISL/NRC/United States INL/DOE	Detailed mechanistic models Limited to RCS Limited user parameters
MELCOR	SNL/NRC/ United States	Simplified or mechanistic models (depending on phenomena) Integrated RCS and containment analysis Extensive user parameters
MAAP	FAI / EPRI/ United States	Simplified, parametric models Integral RCS and containment analysis Extensive user parameters Separate versions for each reactor type (BWR, PWR, etc.)
ICARE/ASTEC	IPSN /CEA/France	Detailed models Limited to RCS Limited user parameters
ATHLET-CD	GRS/Germany	Detailed models Limited to RCS Limited user parameters
IMPACT SAMPSON	NUPEC / METI/Japan	Detailed models Integral RCS and containment analysis

Presentation focuses on US severe accident analysis codes.

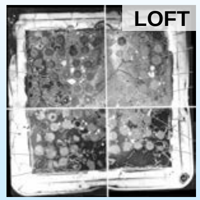
Approximate Accident Phenomena Covered by U.S. Severe Accident Computer Codes

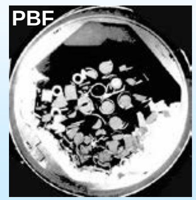


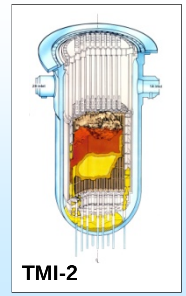
SCDAP/RELAP5-3D© Embodies Understanding of Severe Accident Processes

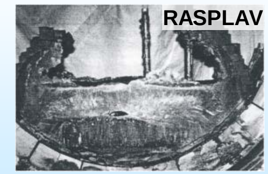
Model Development and Assessment Based on Data from:

- **DF/XR**
- **PHEBUS**









Experiments and Analyses

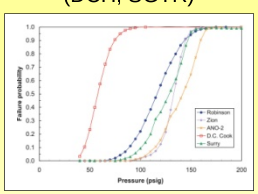
Model Development and Assessment

SCDAP/RELAP5-3D©

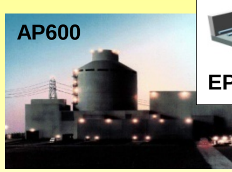
Applications

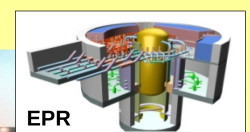
LWR

Severe Accident Resolution
(DCH, SGTR)

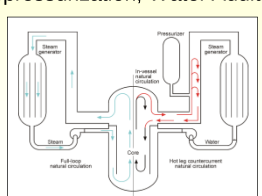


ALWR Evaluations
(AP600, APR 1400,EPR, SBWR)



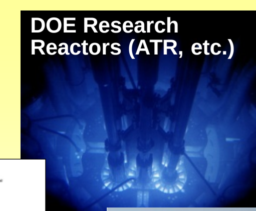


Severe Accident Mitigation Strategies
(Depressurization, Water Addition)

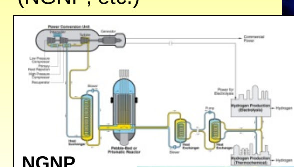


Existing LWRs

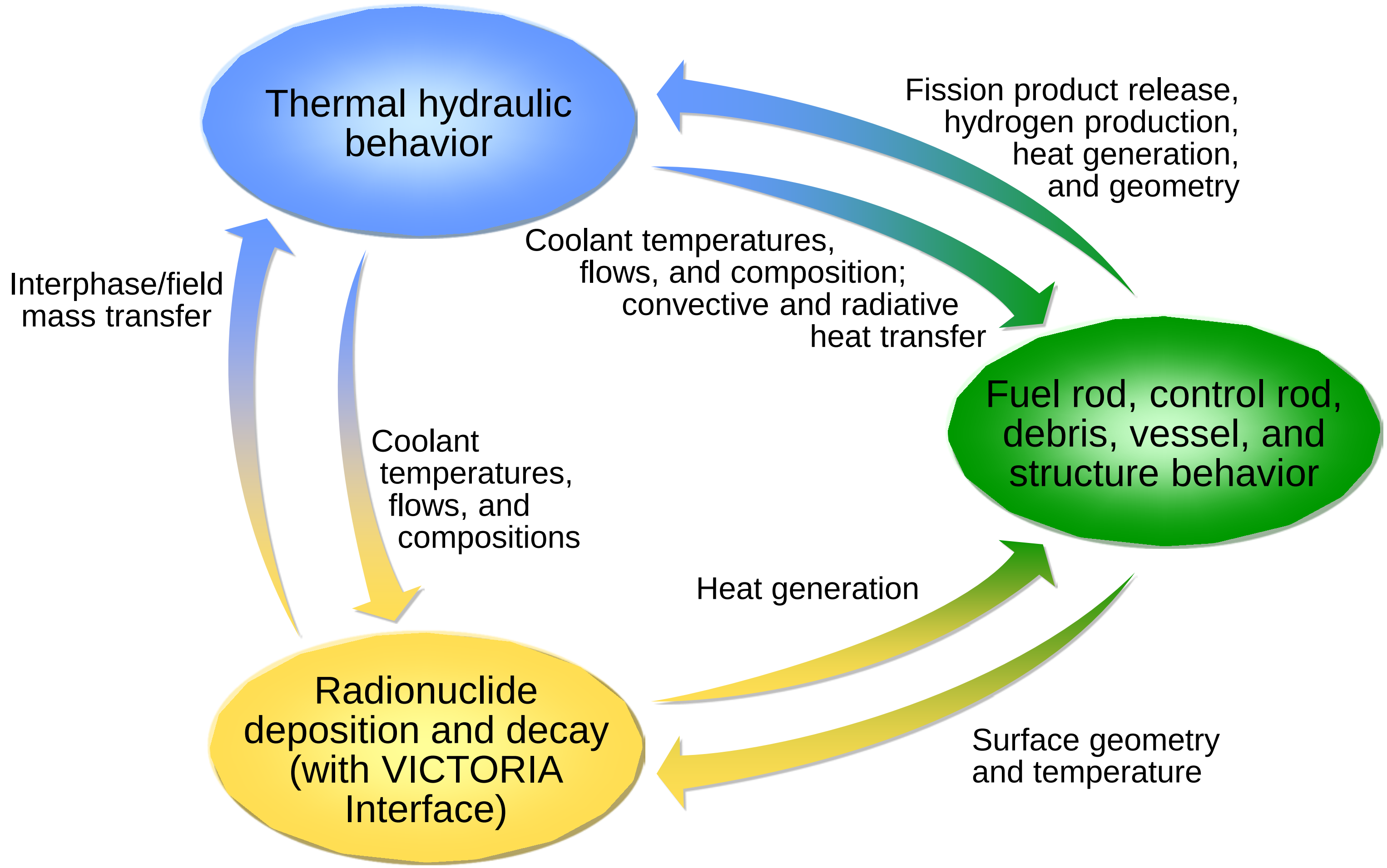
Non-LWR



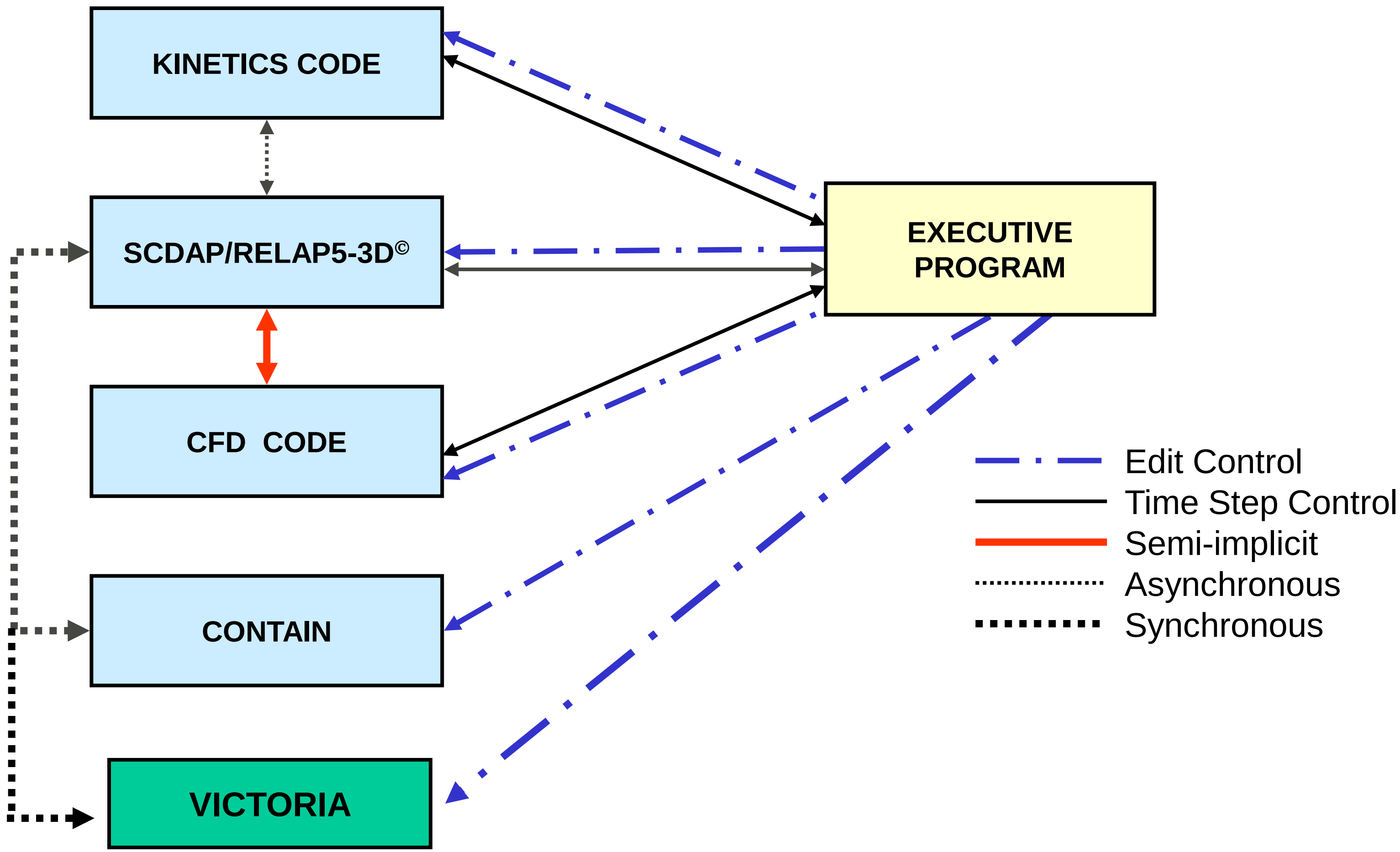
GENIV Reactors
(NGNP, etc.)



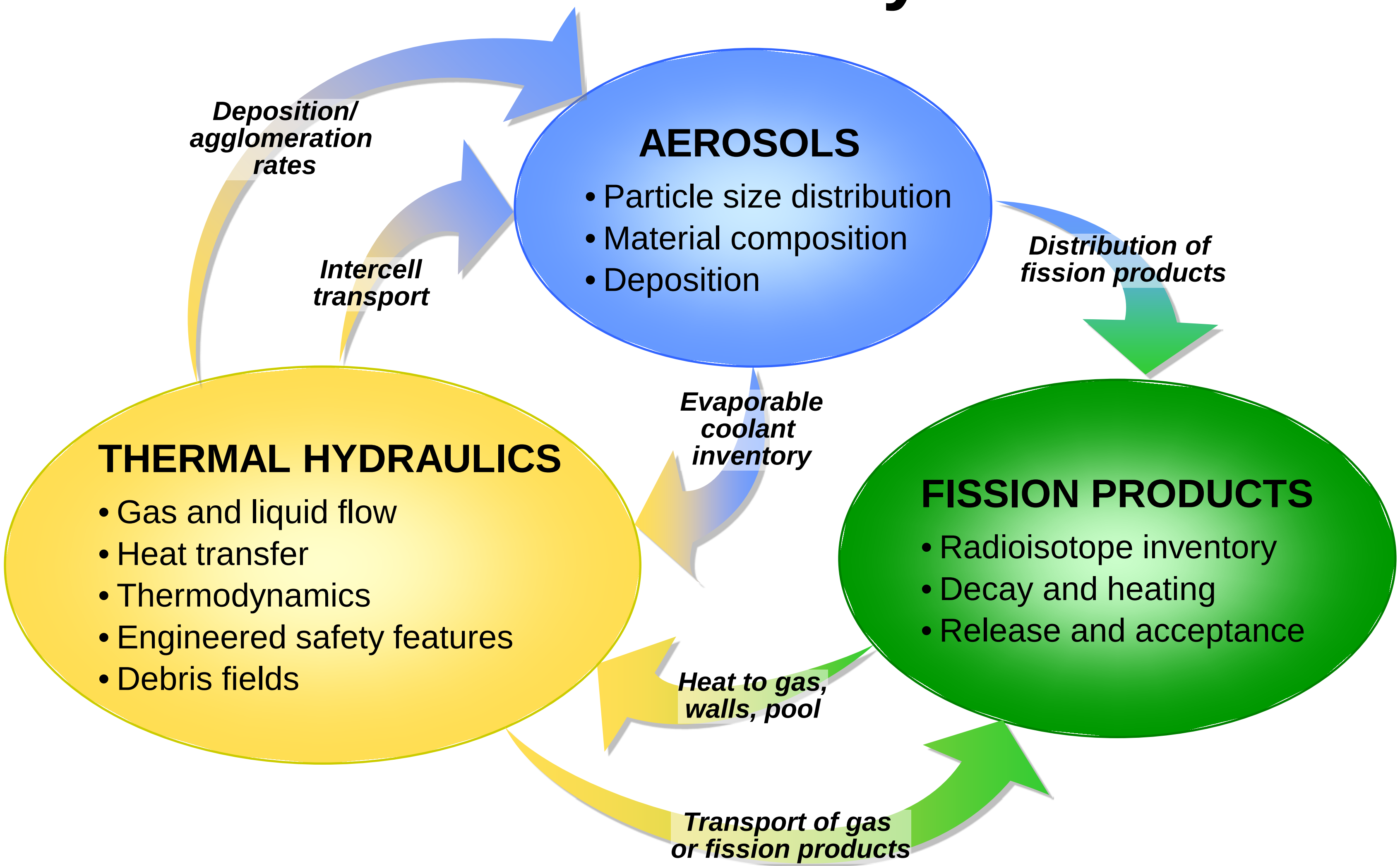
SCDAP/RELAP5-3D© Provides Mechanistic Severe Accident Modeling Tool



PVM linkage provides options not available with other analysis tools



CONTAIN provides mechanistic containment analyses tool.



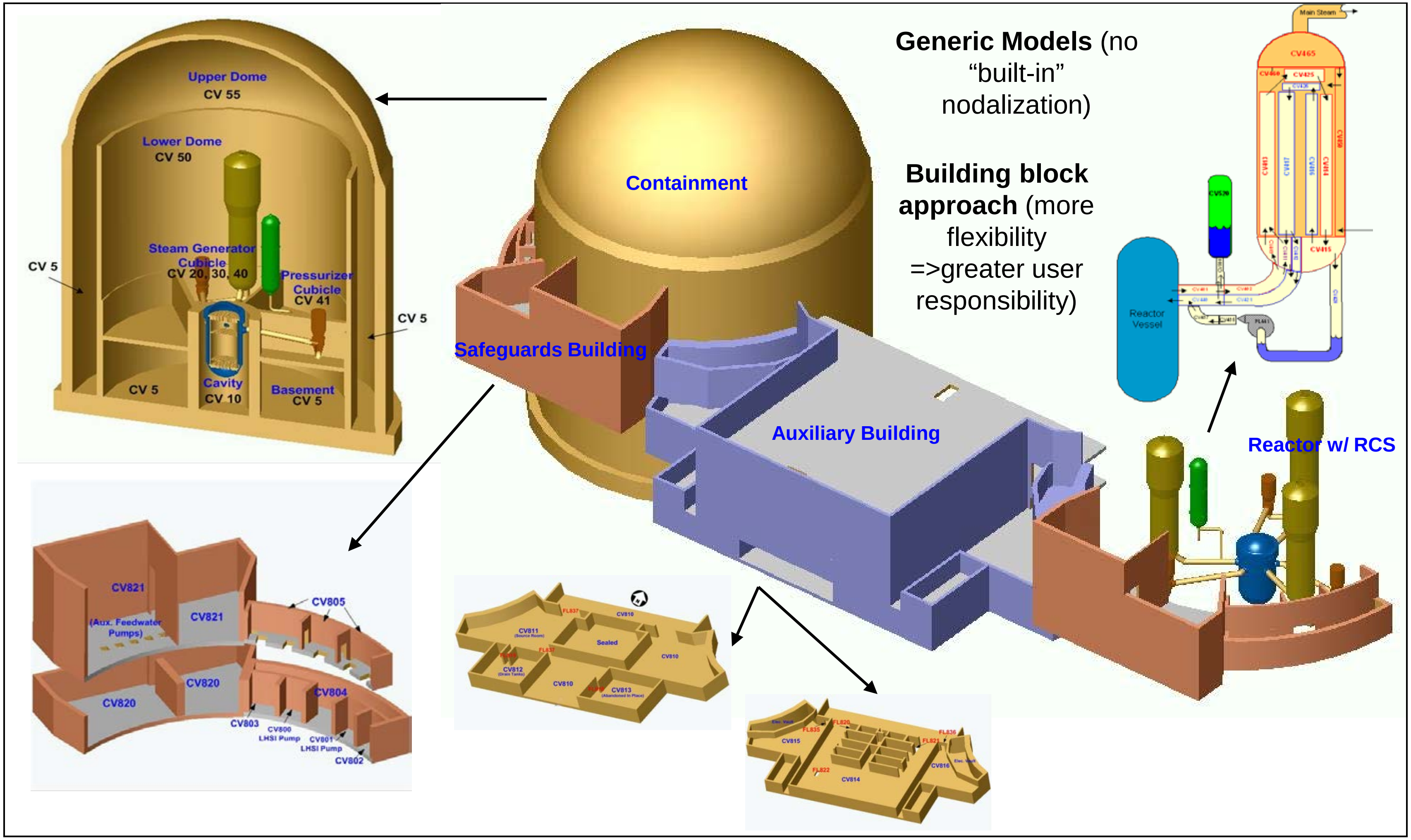
MELCOR Code Physics Description

- MELCOR is a fully integrated, engineering-level computer code that models the progression of severe accidents in light water reactor nuclear power plants
- MELCOR is being developed at Sandia National Laboratories for the U.S. Nuclear Regulatory Commission as a second-generation plant risk assessment tool.
- A broad spectrum of severe accident phenomena in both boiling and pressurized water reactors is treated in MELCOR in a unified framework
- Reactor plant systems and their response to off-normal or accident conditions include:
  - ✓ Thermal-hydraulic response of the primary reactor coolant system, the reactor cavity, the containment, and the confinement buildings,
  - ✓ Core uncovering (loss of coolant), fuel heat-up, cladding oxidation, fuel degradation (loss of rod geometry), and core material melting and relocation,
  - ✓ Heat-up of reactor vessel lower head from relocated fuel materials and the thermal and mechanical loading and failure of the vessel lower head, and transfer of core materials to the reactor vessel cavity,
  - ✓ Core-concrete attack and ensuing aerosol generation,
  - ✓ In-vessel and ex-vessel hydrogen production, transport, and combustion,
  - ✓ Fission product release (aerosol and vapor), transport, and deposition,
  - ✓ Behavior of radioactive aerosols in the reactor containment building, including scrubbing in water pools, and aerosol mechanics in the containment atmosphere such as particle agglomeration and gravitational settling, and,
  - ✓ Impact of engineered safety features on thermal-hydraulic and radionuclide behavior

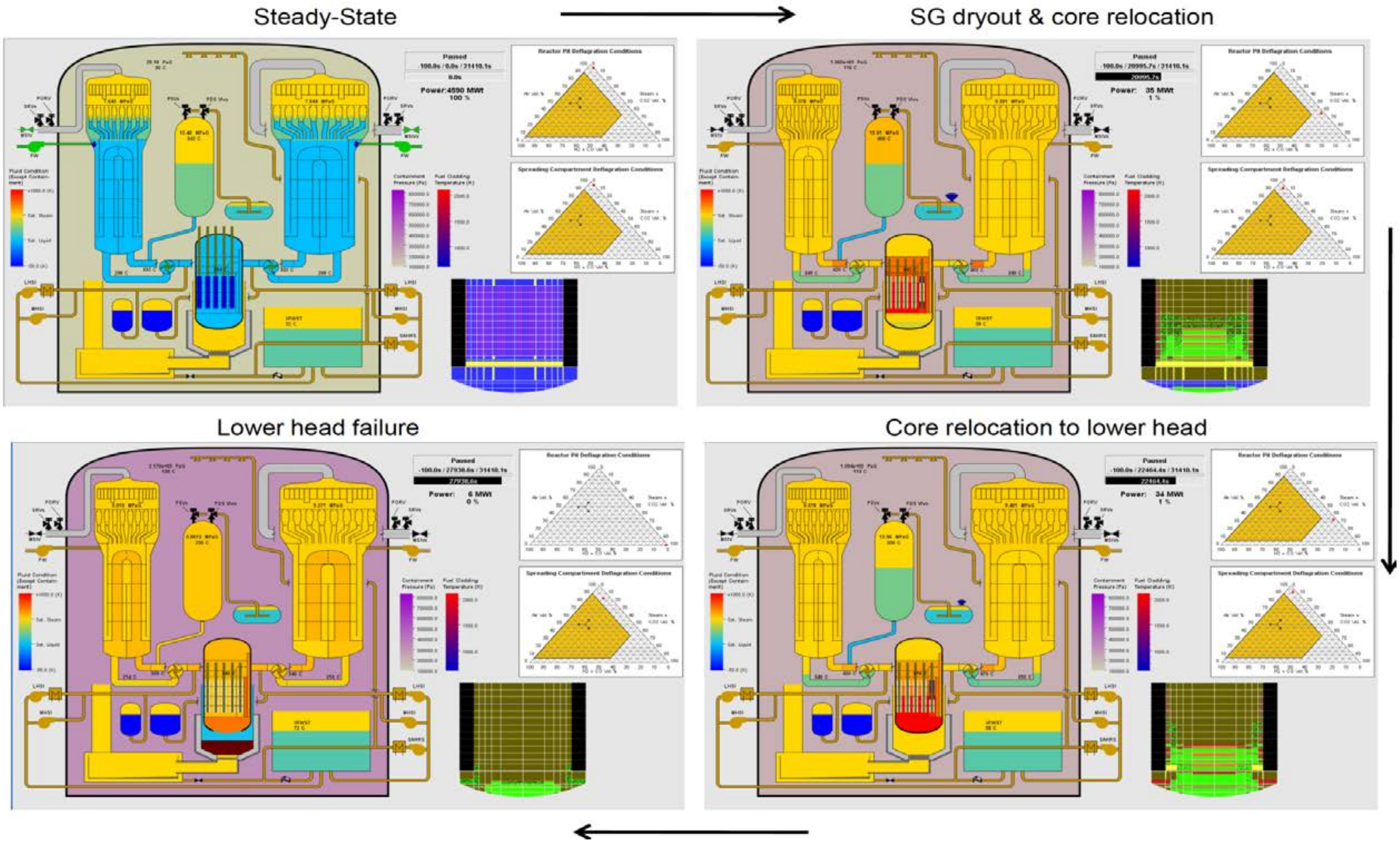
MELCOR Modeling Approach

Generic Models (no "built-in" nodalization)

Building block approach (more flexibility =>greater user responsibility)



MELCOR User Interface



MELCOR Models Fission Facilities

- A six equation non-equilibrium fluid flow model for fluid flow in a facility by using control volumes, flow paths, and heat structures

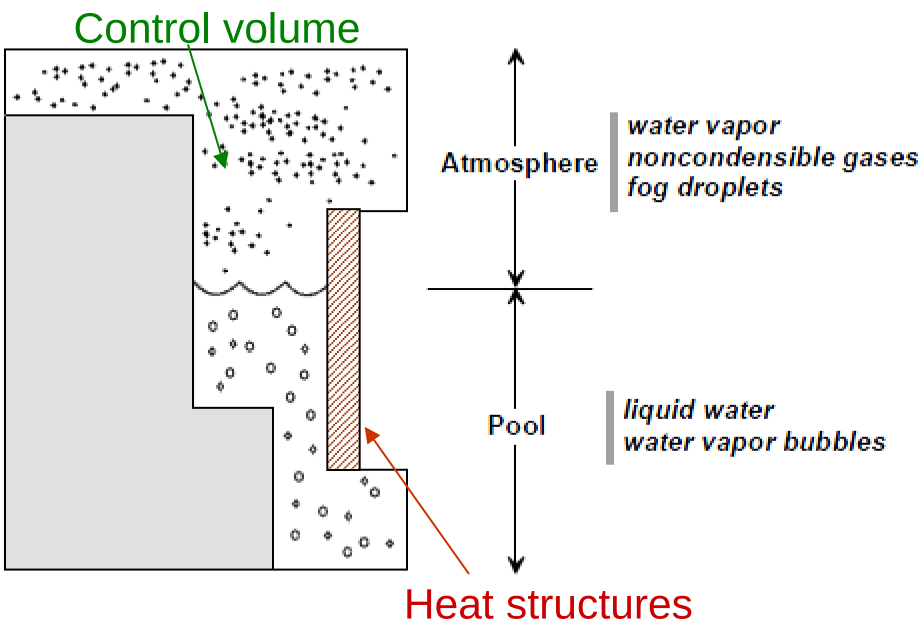
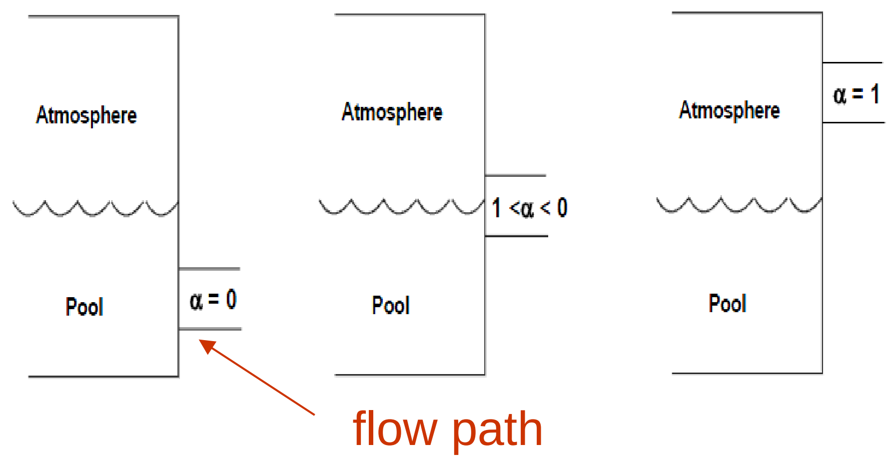


Figure 2.3 Control volume contents and pool surface

- Multiple flow paths can connect any two control volumes. Height of flow path determines time dependent phase of flow entering or leaving the flow path

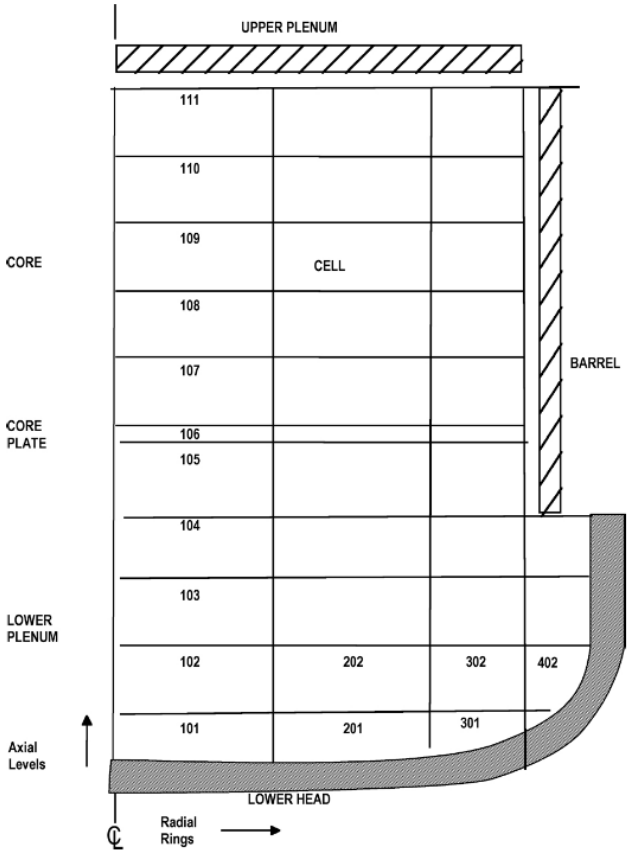


- Heat structures for walls, piping, vessels, etc. with pool and atmosphere natural, force convective heat transfer (pool includes boiling heat transfer)

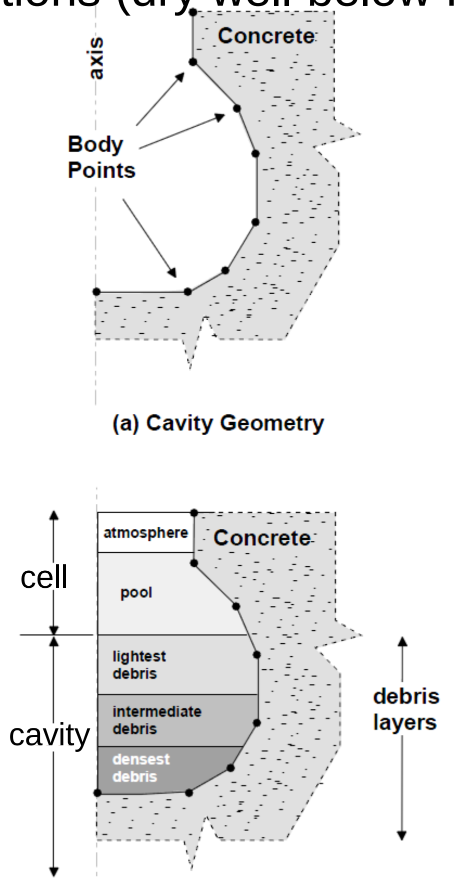
- Aerosols and fission products are transported both in the vapor and liquid phases.

MELCOR Models Fission Facilities (cont.)

A core model for fuel/cladding response



A cavity model for debris concrete reactions (dry well below RPV)



(a) Cavity Geometry

(b) Cavity Contents and Boundary Conditions

MELCOR Models Fission Releases and Transport

- Radionuclide releases can occur from the core fuel, from the fuel-cladding gap, and from material in the cavity
- Three options are currently available for the release of radionuclides from the core fuel component; the CORSOR, CORSOR-M or CORSOR-Booth
  - Cesium release fraction, f, at time t is calculated from an approximate solution of Fick's law assuming spherical fuel grains
- Release of the radionuclides in the fuel-cladding gap (initial inventory plus masses from fuel release) occurs on cladding failure. Cladding failure is assumed to occur if either a temperature criterion is exceeded or if the intact cladding geometry has been lost due to candling or oxidation
  - For release of radionuclides from the cavity due to core-concrete interactions, the VANESA model has been implemented in MELCOR coupled to the CORCON model
- The condensation and evaporation of fission product vapors to and from heat structures, pool surfaces, and aerosols is evaluated by the same equations as in the TRAP-MELT2 code

MELCOR Models Fission Releases and Transport (cont.)

- The MELCOR calculation of changes in aerosol distribution and location within a plant considers the following general processes:
  - Aerosol phenomenological sources from other packages, such as release from fuel rods or during core-concrete interactions, and/or arbitrary user-specified sources;
  - Condensation and evaporation of water and fission products to and from aerosol particles;
  - Particle agglomeration (or coagulation), whereby two particles collide and form one larger particle;
  - Particle deposition onto surfaces or settling through flow paths into lower control volumes;
  - Advection of aerosols between control volumes by bulk fluid flows
  - Removal of aerosol particles by Engineered Safety Features (ESFs), such as filter trapping, pool scrubbing, and spray washout

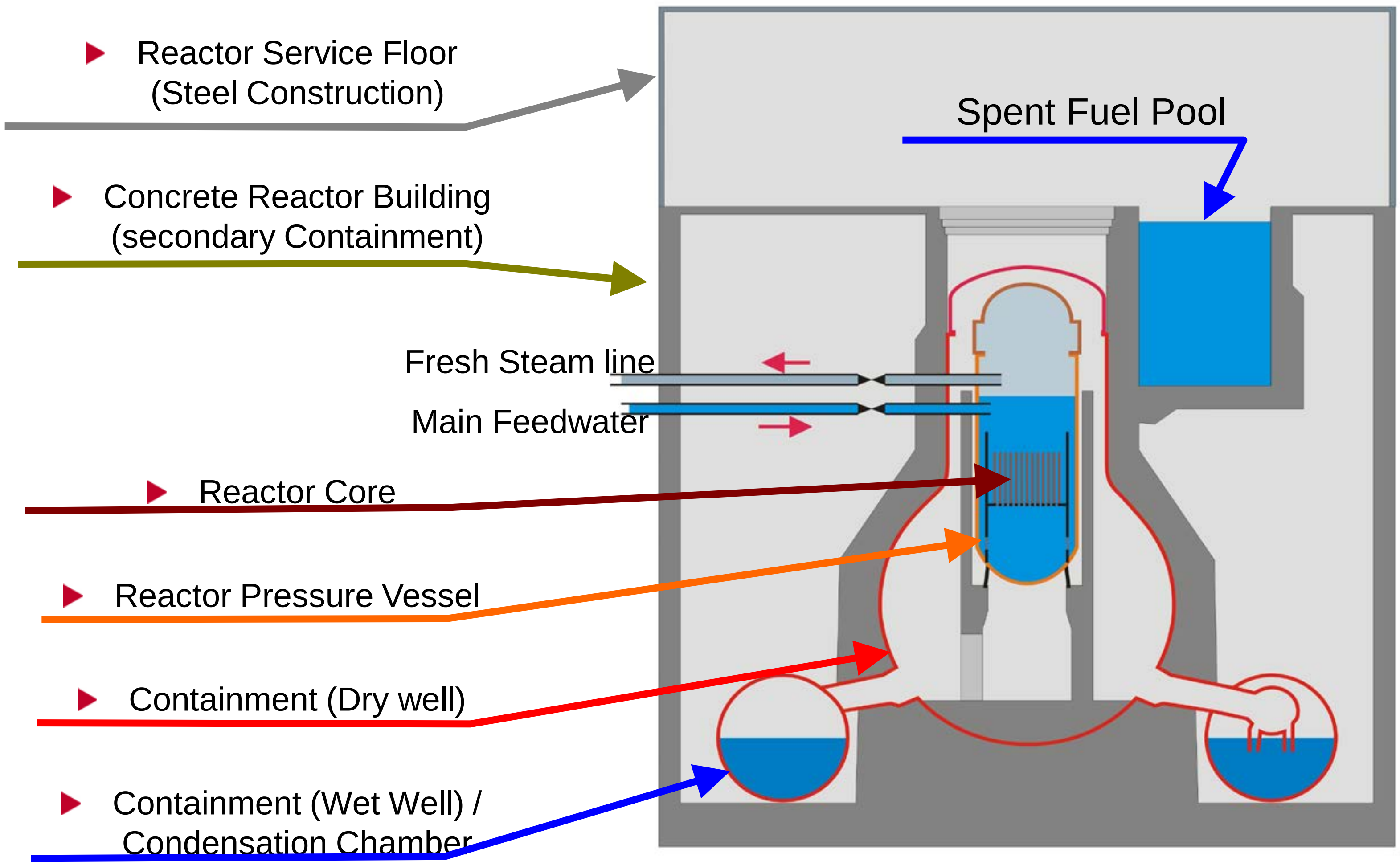
MELCOR Models Fission Releases and Transport (cont.)

- Fission Product Chemistry effects can be simulated in MELCOR through the use of class reactions and class transfers.
  - The class reaction process uses a first-order reaction equation with forward and reverse paths.
  - The class transfer process, which can change the material class or location of a radionuclide mass, can be used to simulate fast chemical reactions.
  - With these two processes, phenomena including adsorption, chemisorption, water chemistry, and chemical reactions can be simulated

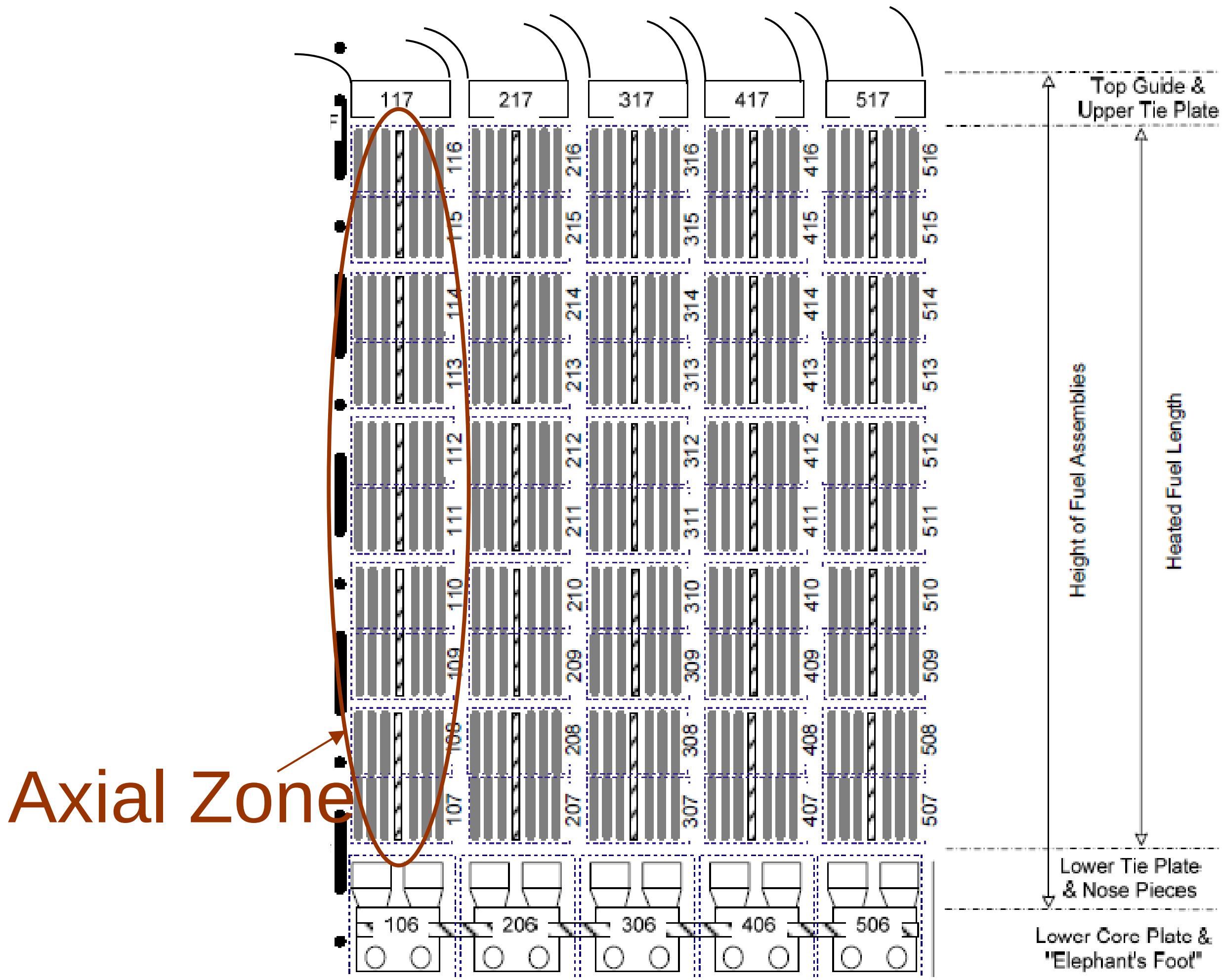
This reaction can be simulated by the RN package by the following sequential class reactions and transfers:

$CsI(g) \rightarrow CsI(ad)$	rate constant for adsorption is supplied through input
$CsI(ad) \rightarrow CsOH(ad) + HI(s)$	instantaneous and complete transfer between classes when water is present. **Note that the water mass is not included in the model; water mass is not explicitly conserved.**
$CsOH(g) \rightarrow CsOH(ad)$	rate constant for adsorption supplied or condensation limited
$CsOH(ad) \rightarrow CsOH(g)$	reaction with zero rate constant below T_1 positive value or instantaneous above T_1
$HI(s) \leftrightarrow HI(g)$	controlled by condensation/evaporation

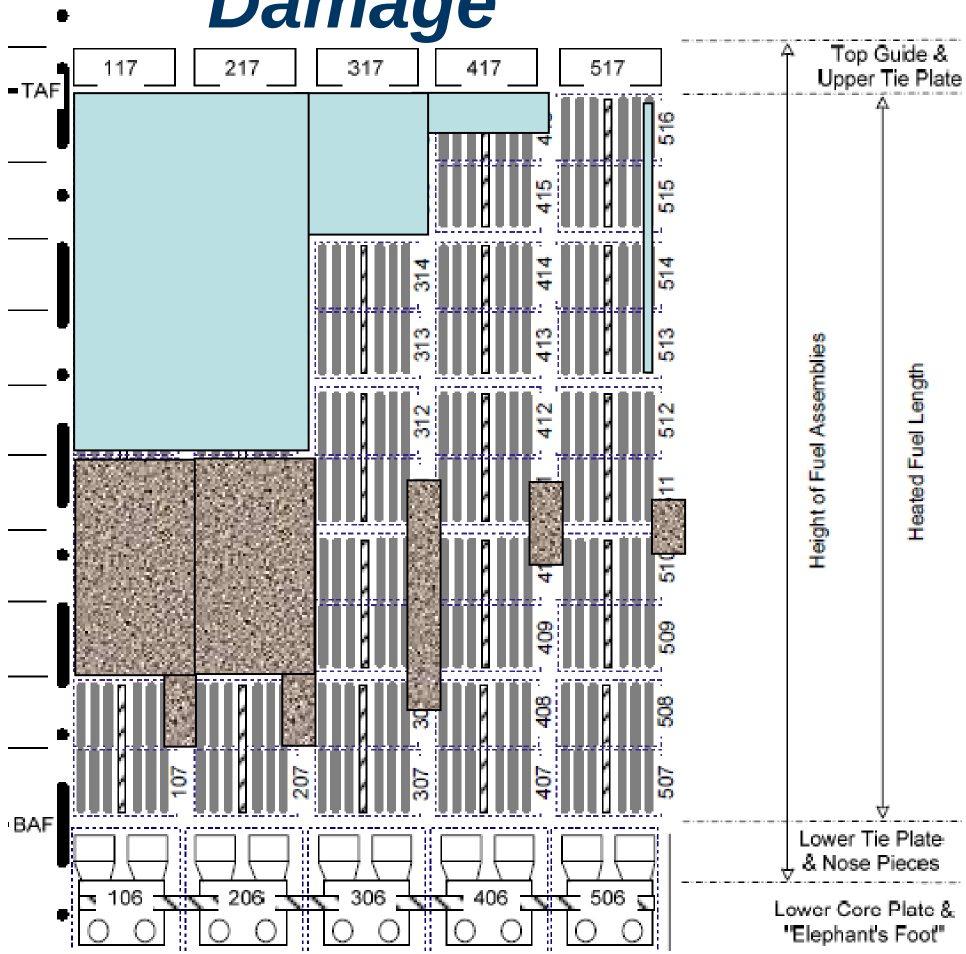
A MELCOR Model of a BWR that includes Reactor Building, Plus All Emergency Cooling Systems was used for analyzing Fukushima Unit 1



MELCOR Core Zones Modeled

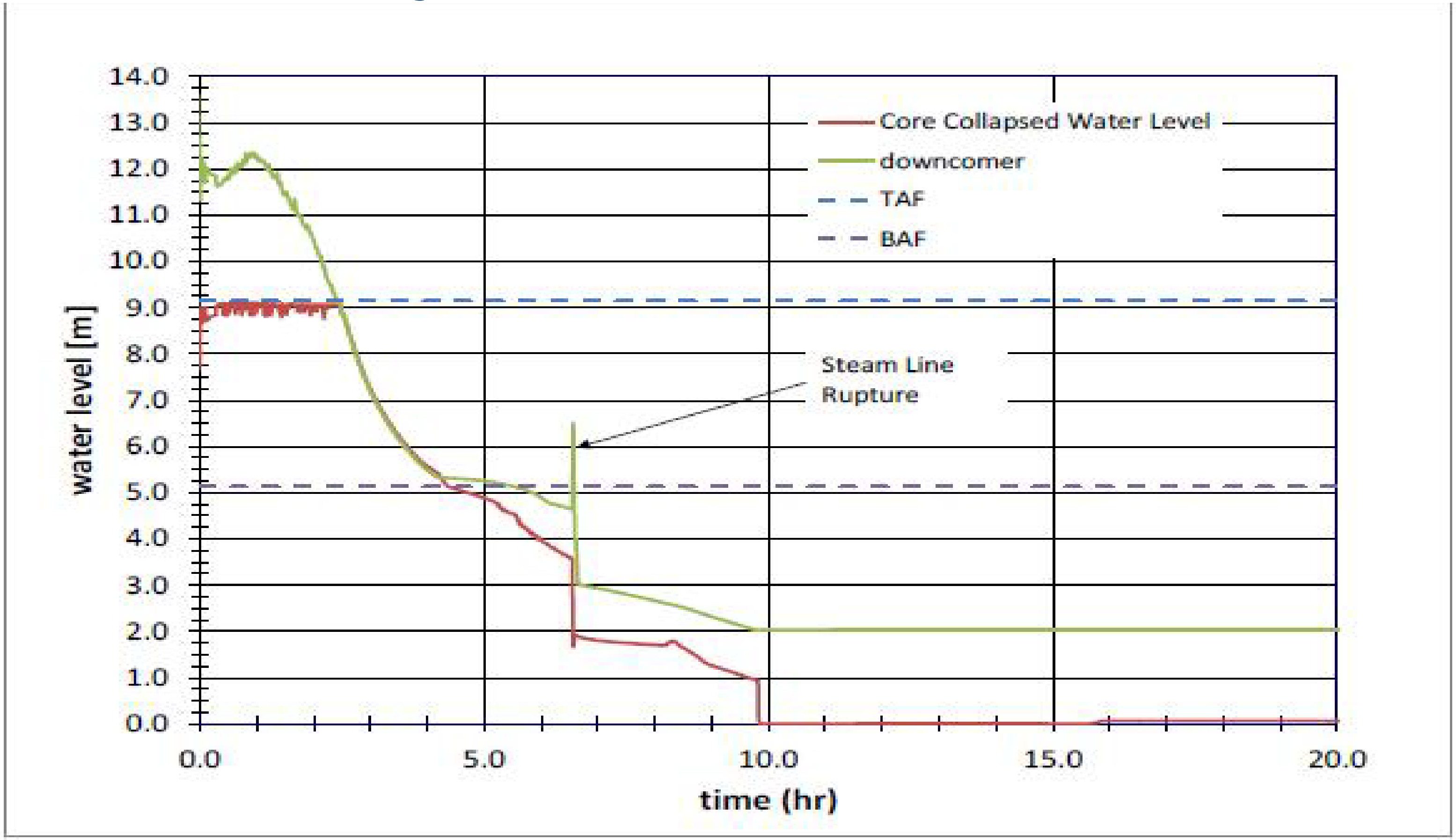


Fukushima Unit 1 Schematic of Predicted Core Damage

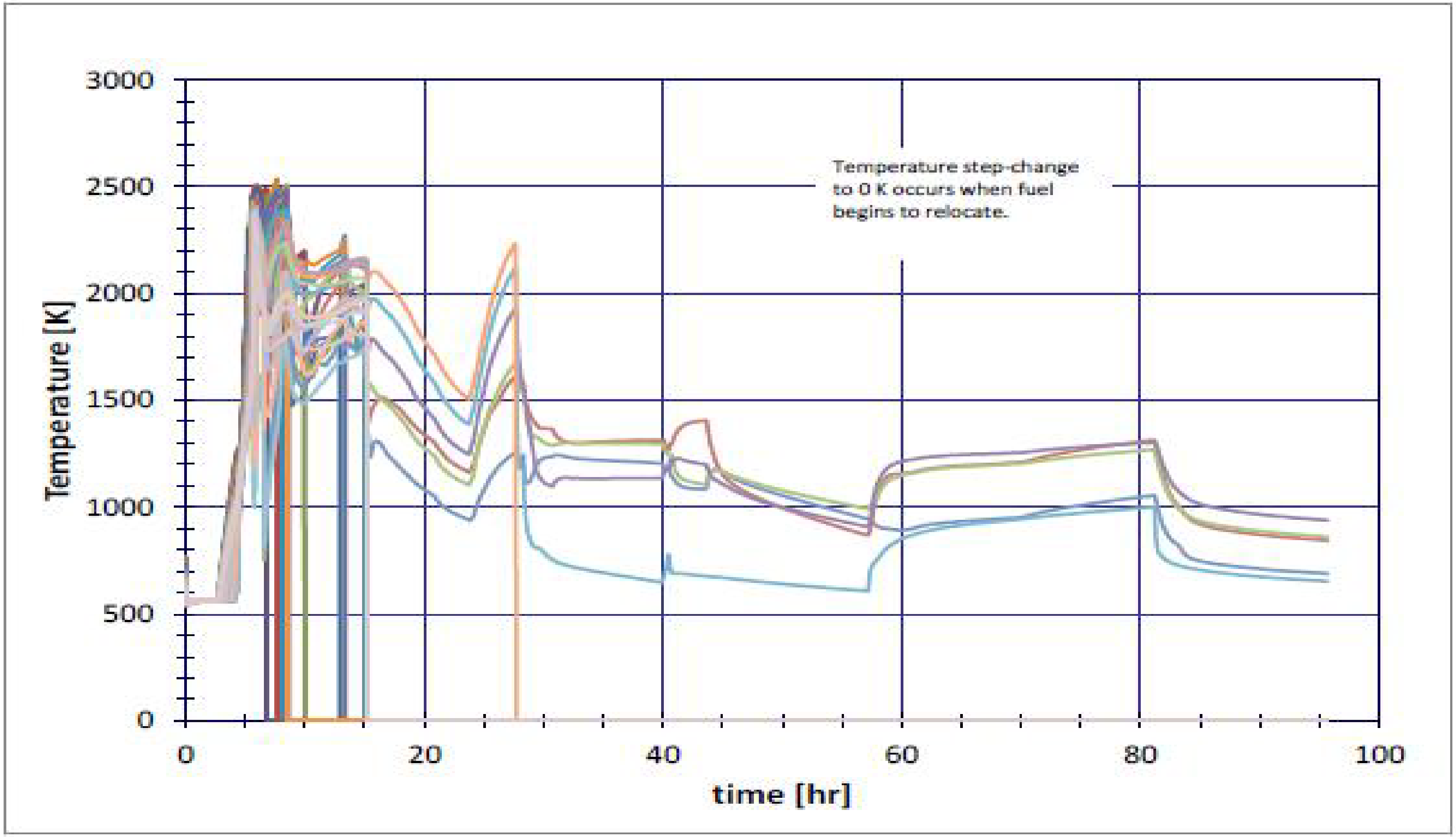


Time = 45 hrs

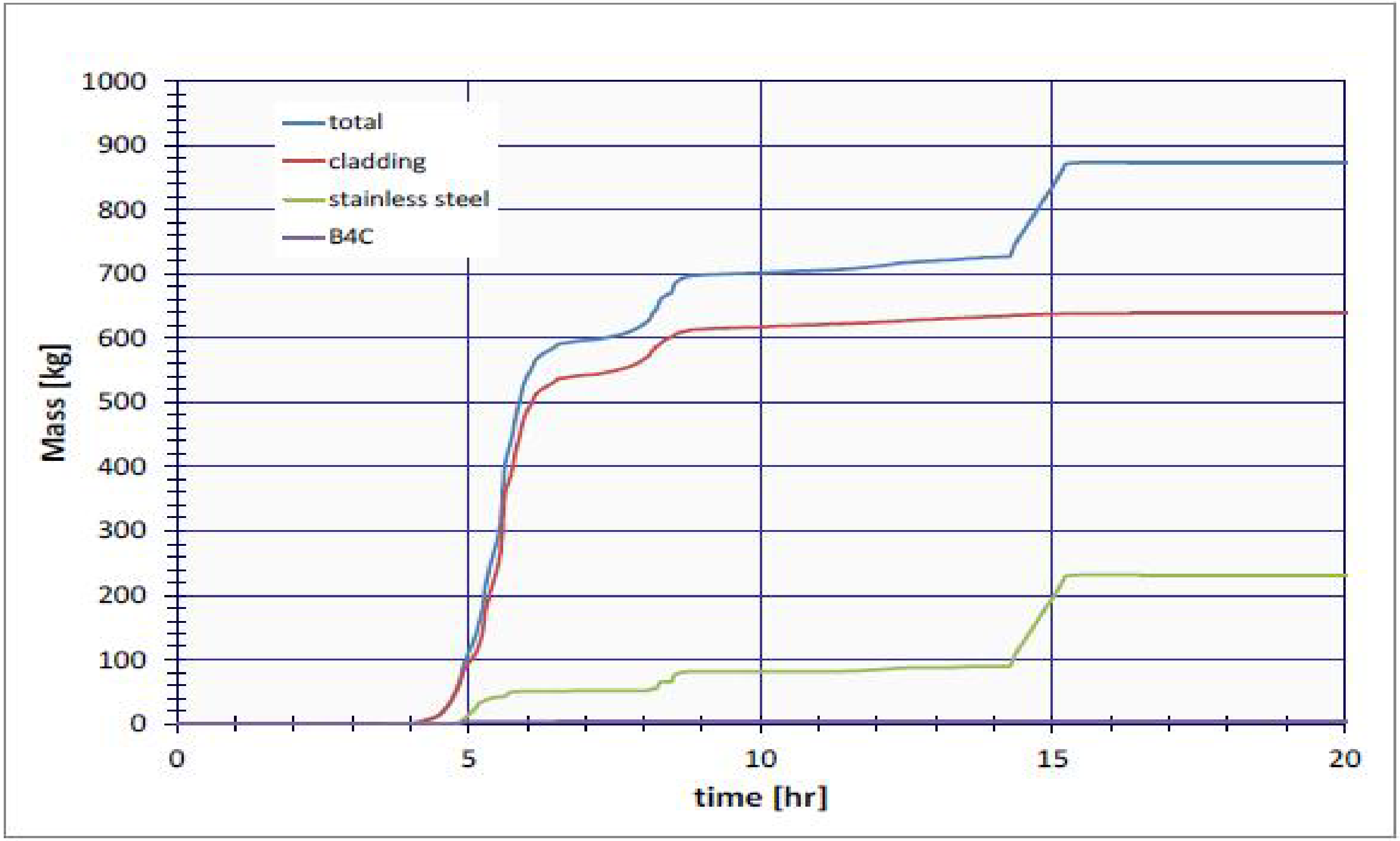
Fukushima Unit 1 MELCOR Calculated/Defined RPV Water Injection Rate-SAND2012-6173



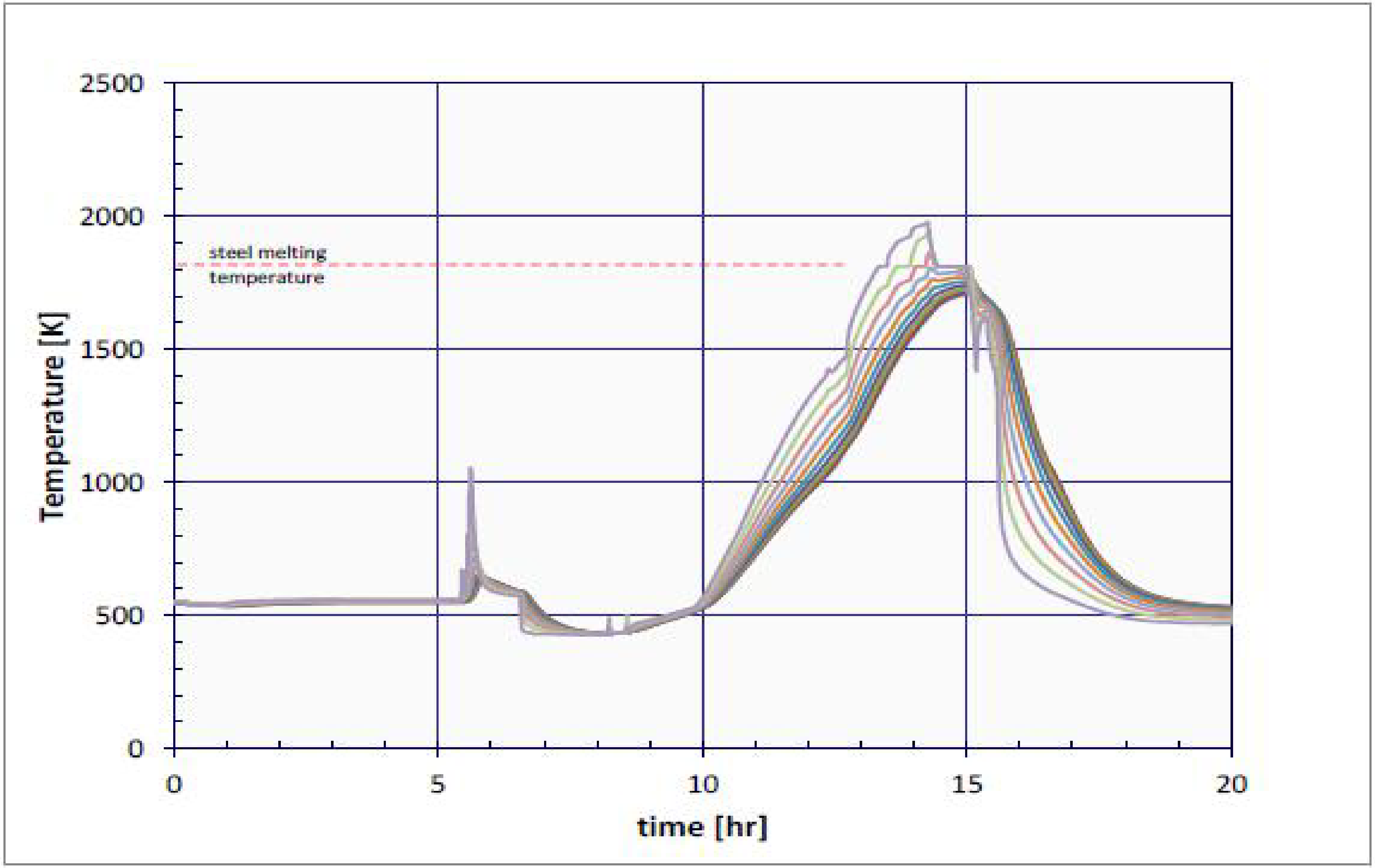
Fukushima Unit 1 Fuel Temperatures – SAND2012-6173



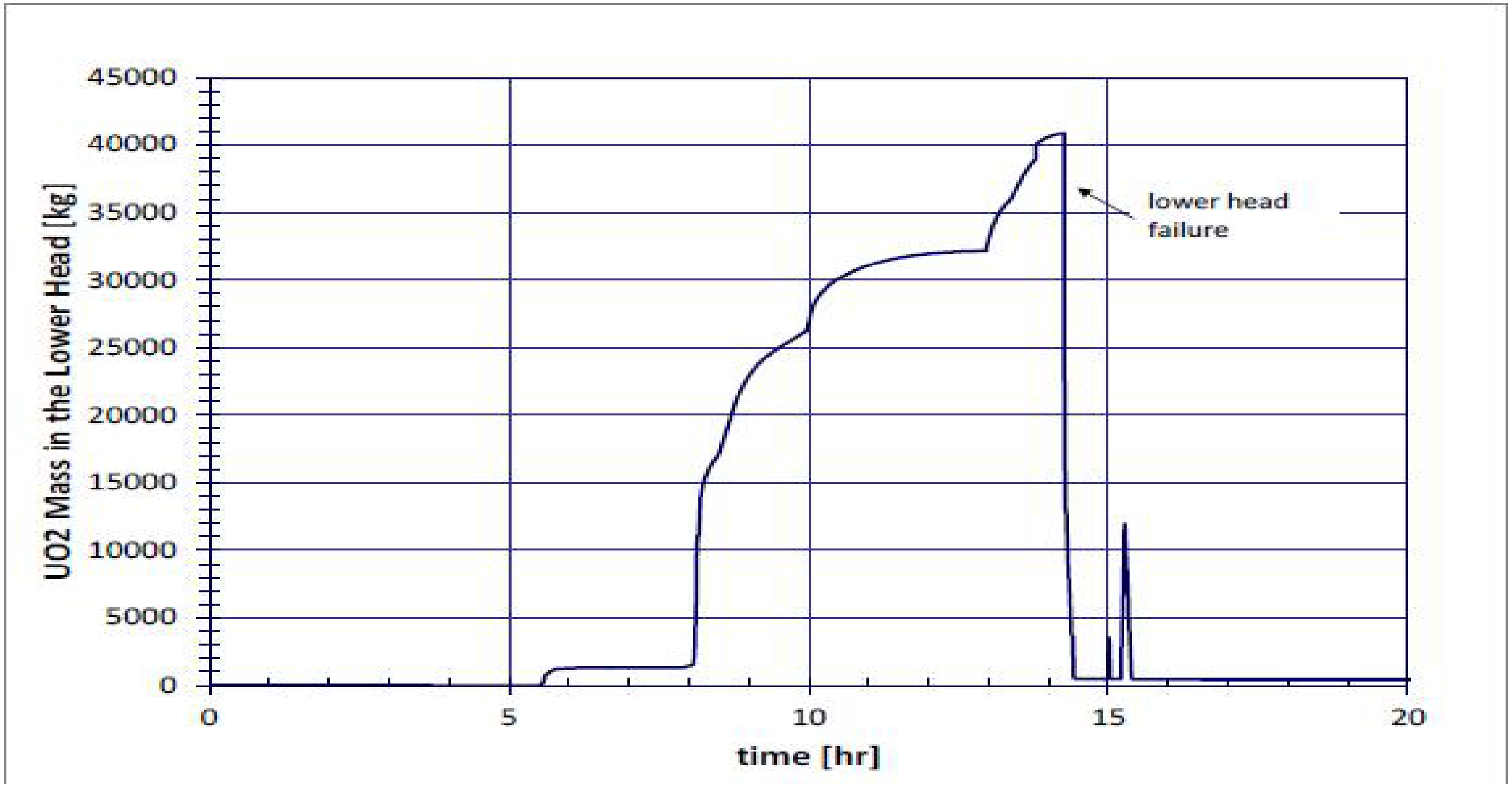
Fukushima Unit 1 Hydrogen Generation from Cladding Stainless Steel and B4C –SAND2012-6173



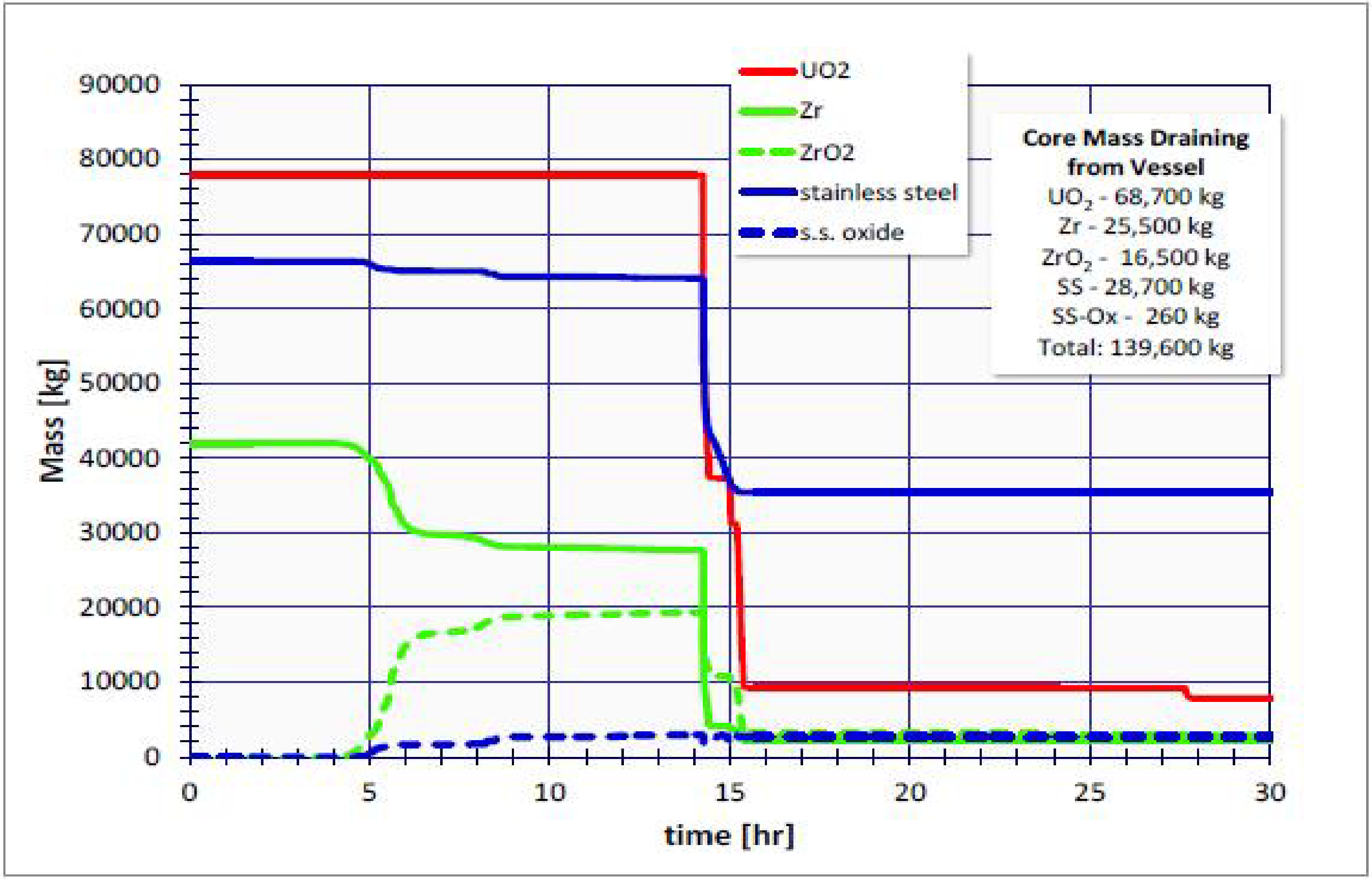
Fukushima Unit 1 Lower Head Fuel Temperatures –SAND2012-6173



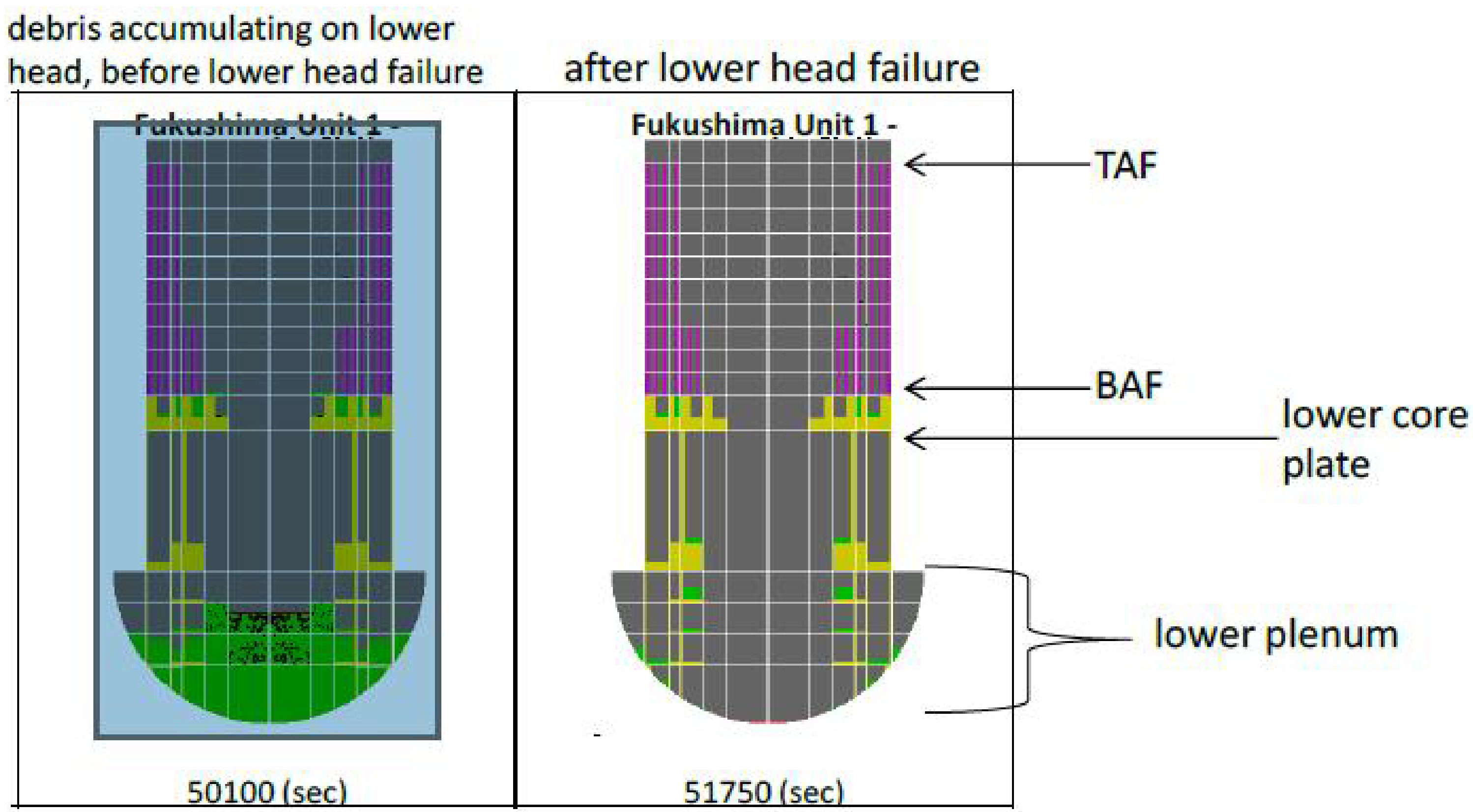
Fukushima Unit 1 Accumulation of Fuel in Lower Plenum 139,000 kg on concrete – SAND2012-6173



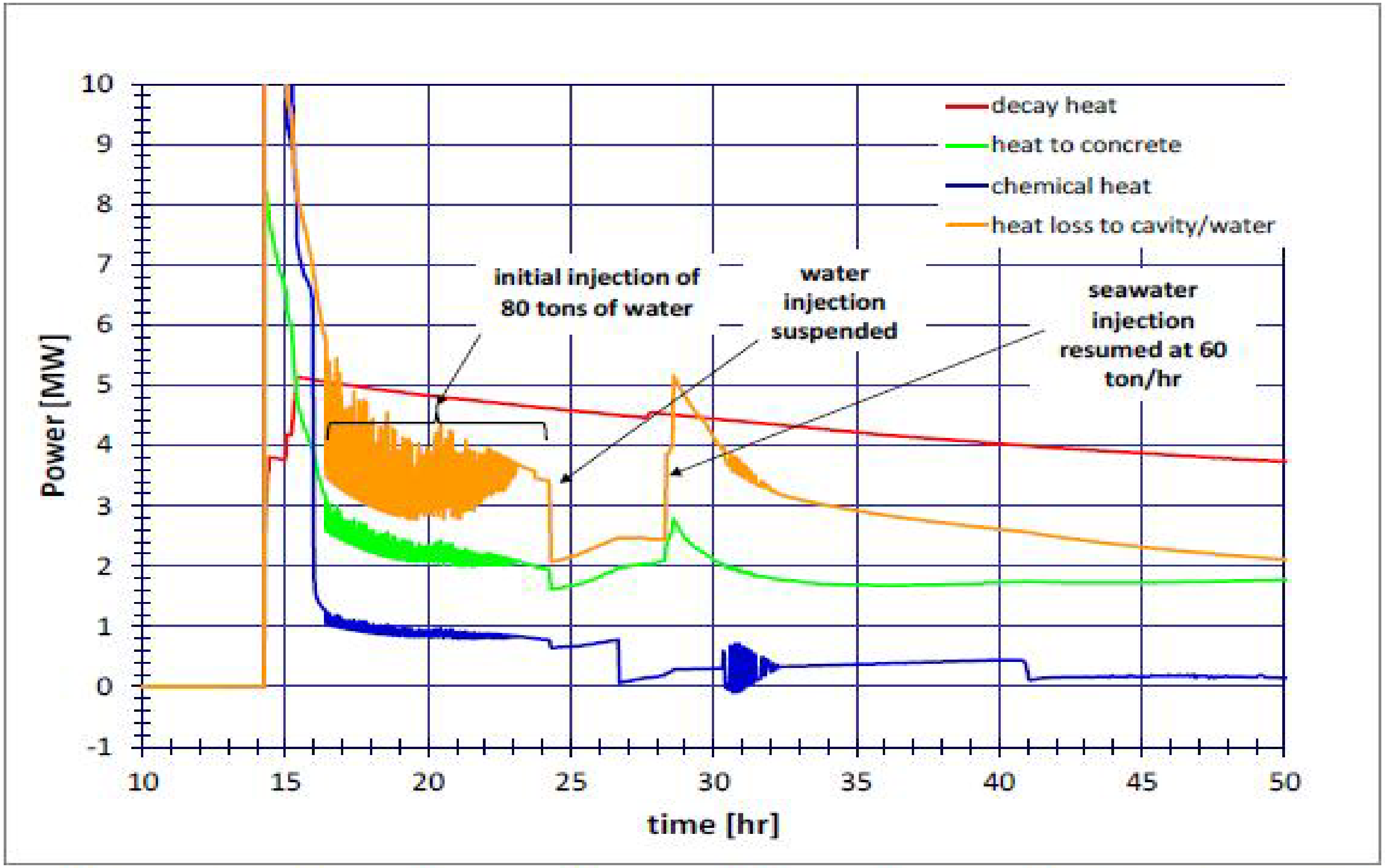
Fukushima Unit 1 Fuel Relocation –SAND2012-6173



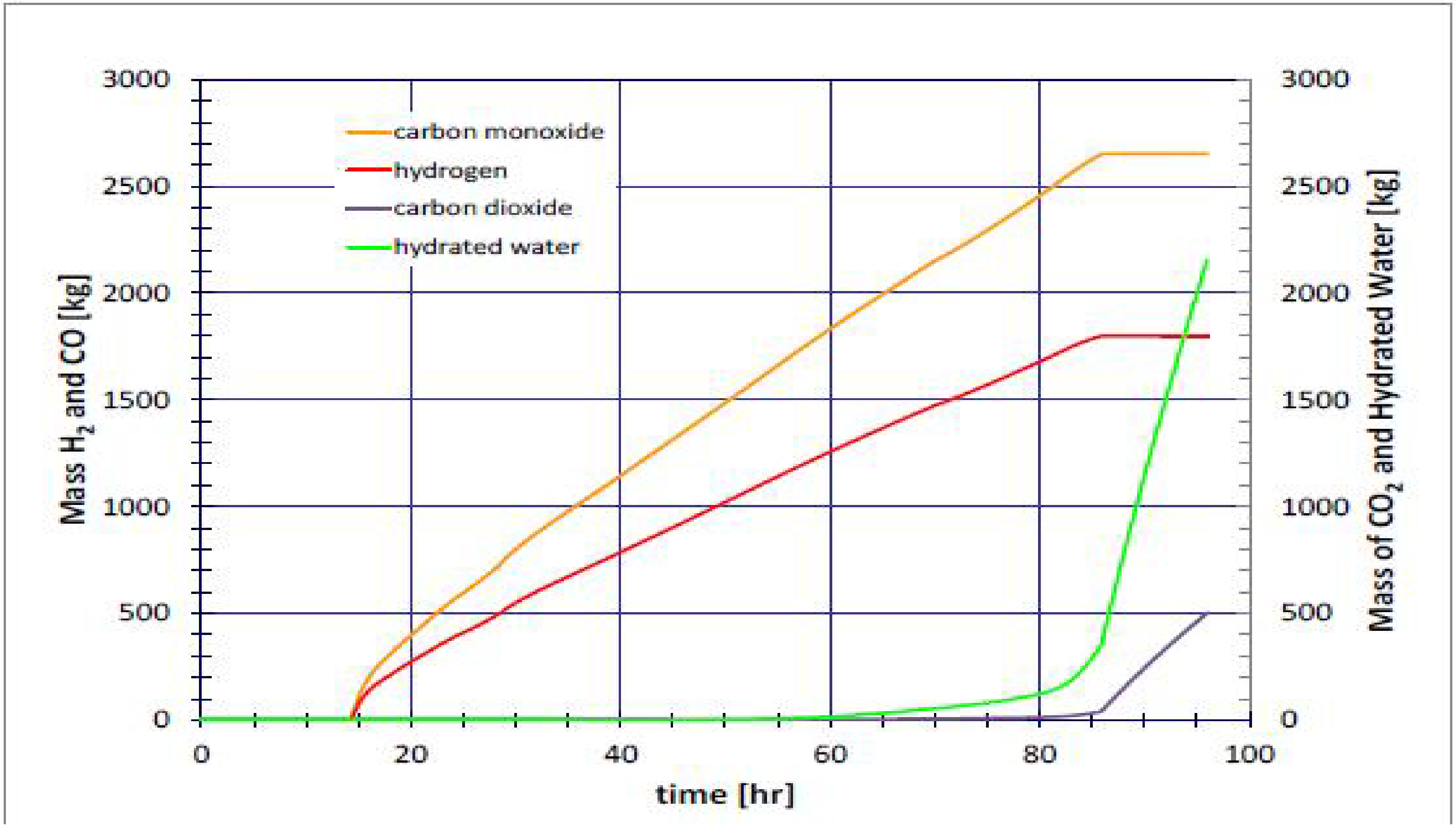
Fukushima Unit 1 Core Condition –SAND2012-6173



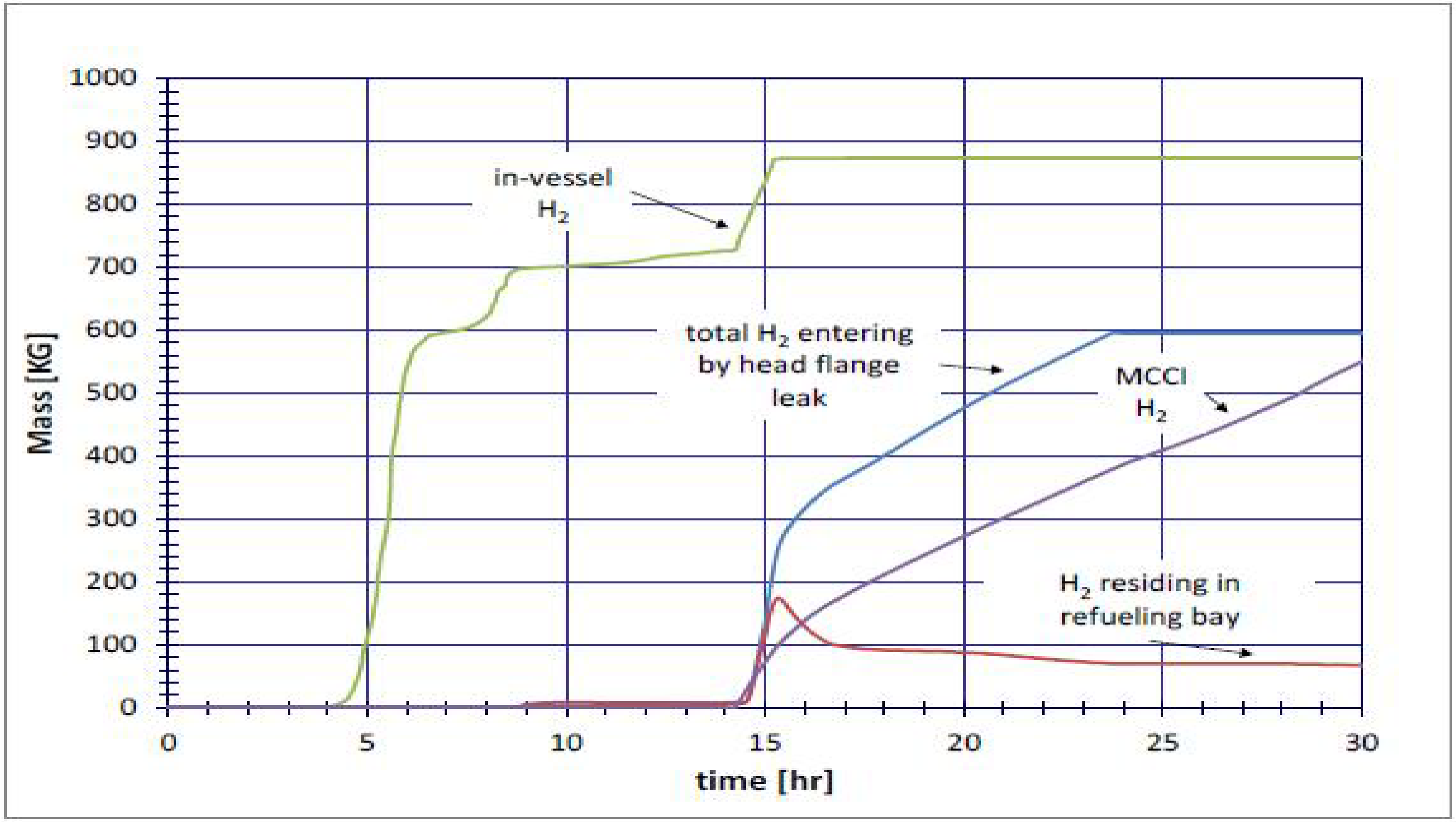
Fukushima Unit 1 MCCI Interaction–SAND2012-6173



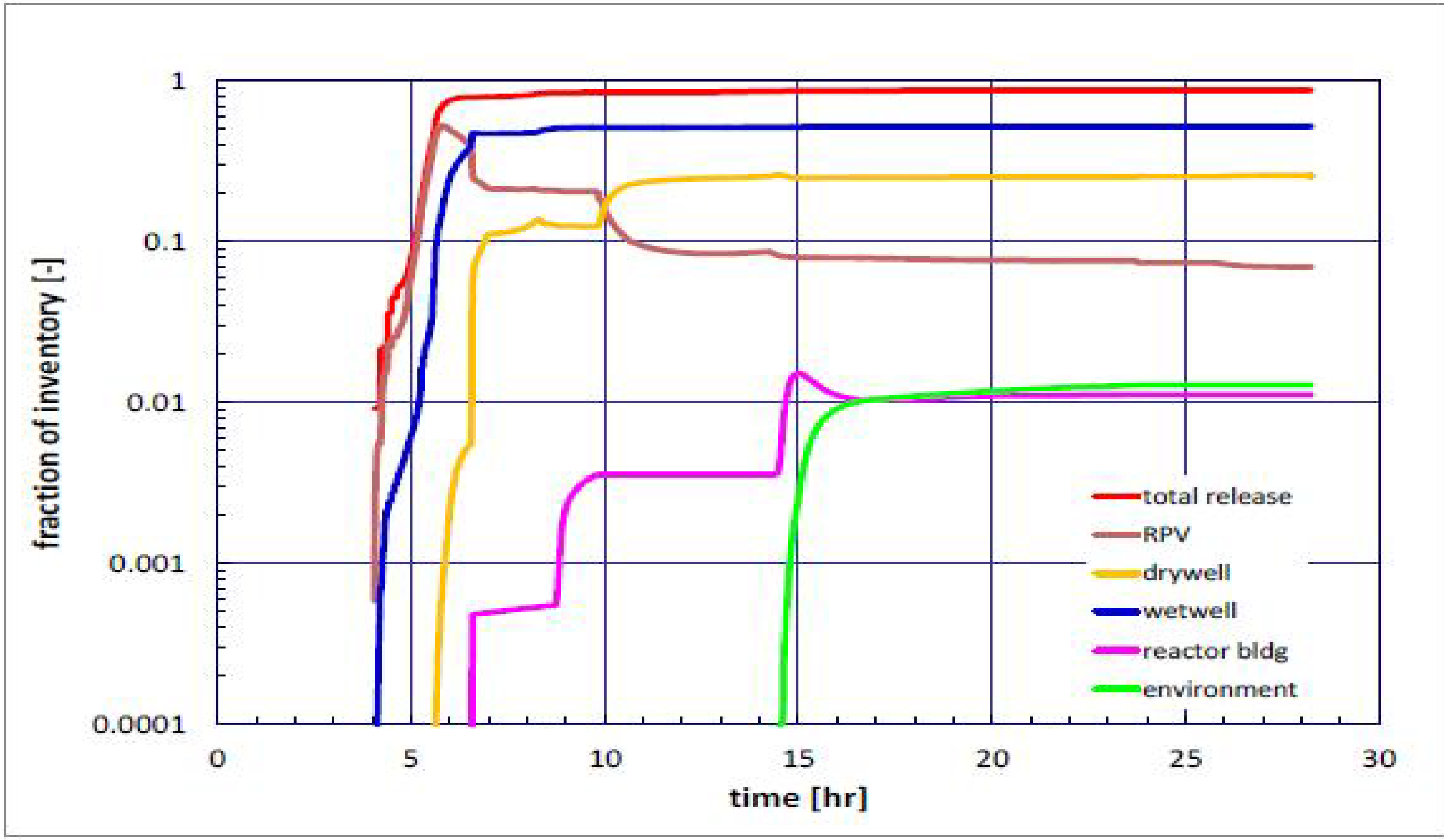
Fukushima Unit 1 MCCI Products–SAND2012-6173



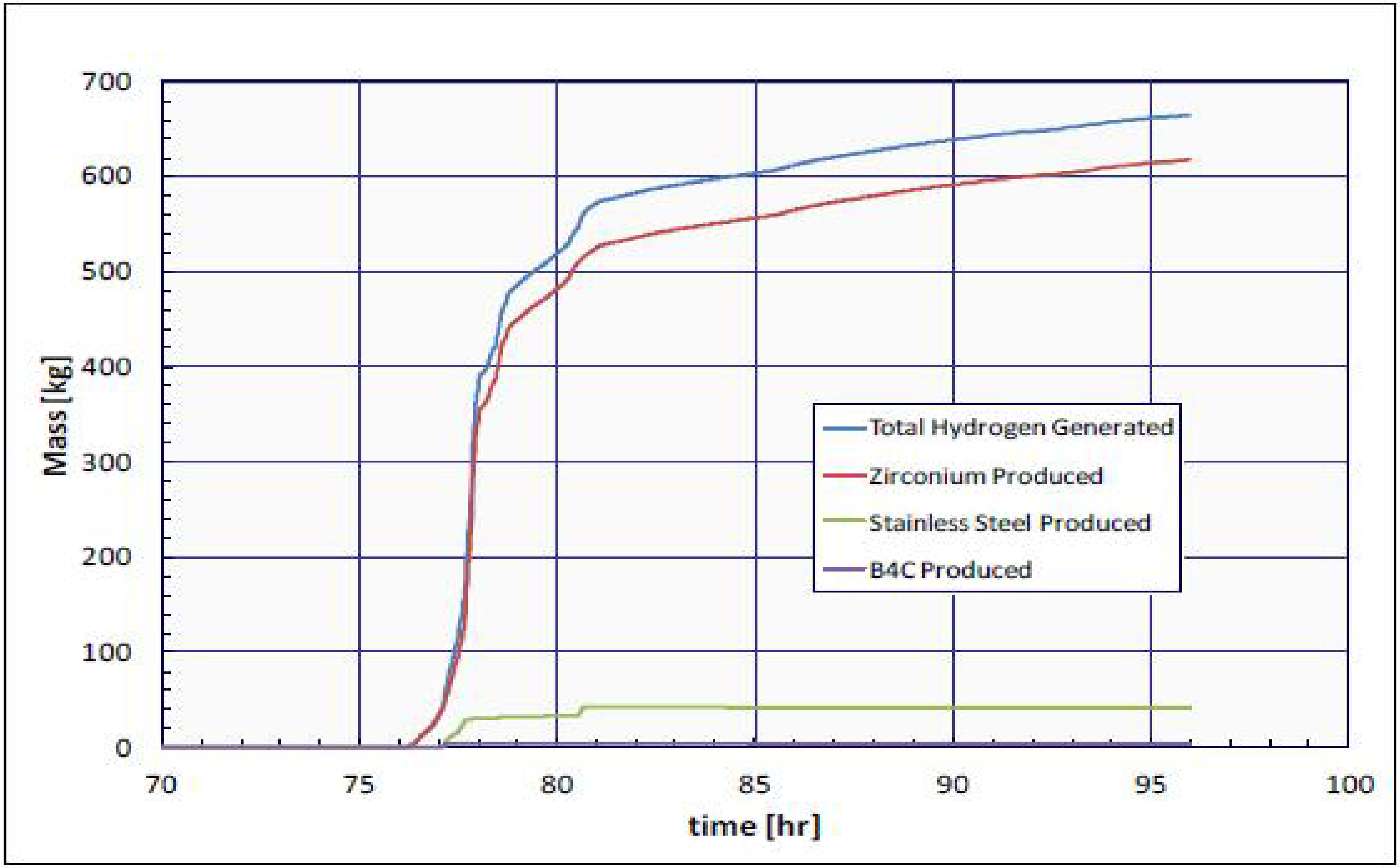
Fukushima Unit 1 MCCI and other hydrogen production –SAND2012-6173



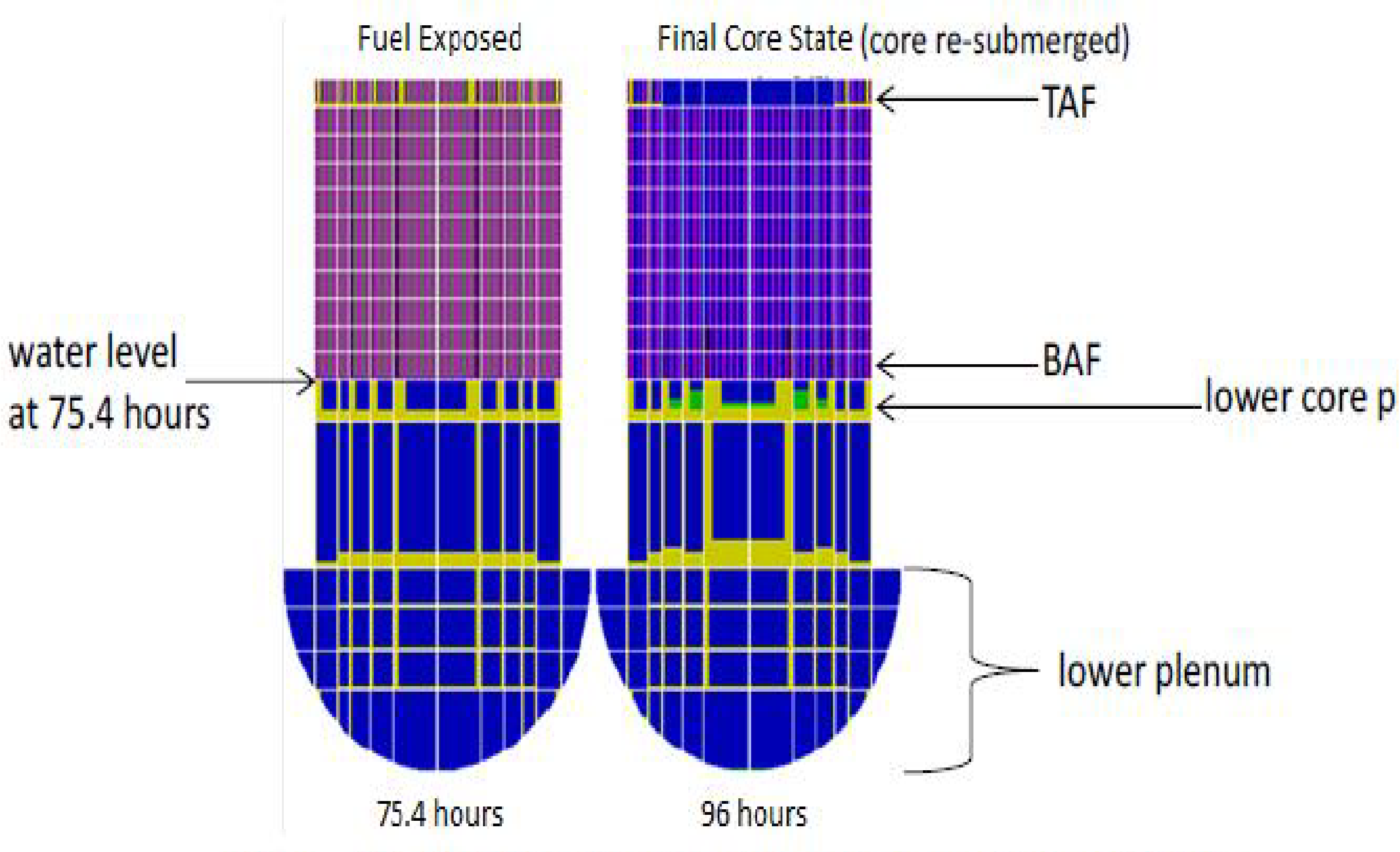
Fukushima Unit 1 CsI Most Retained in Suppression pool with at 1-2% Release– SAND2012-6173



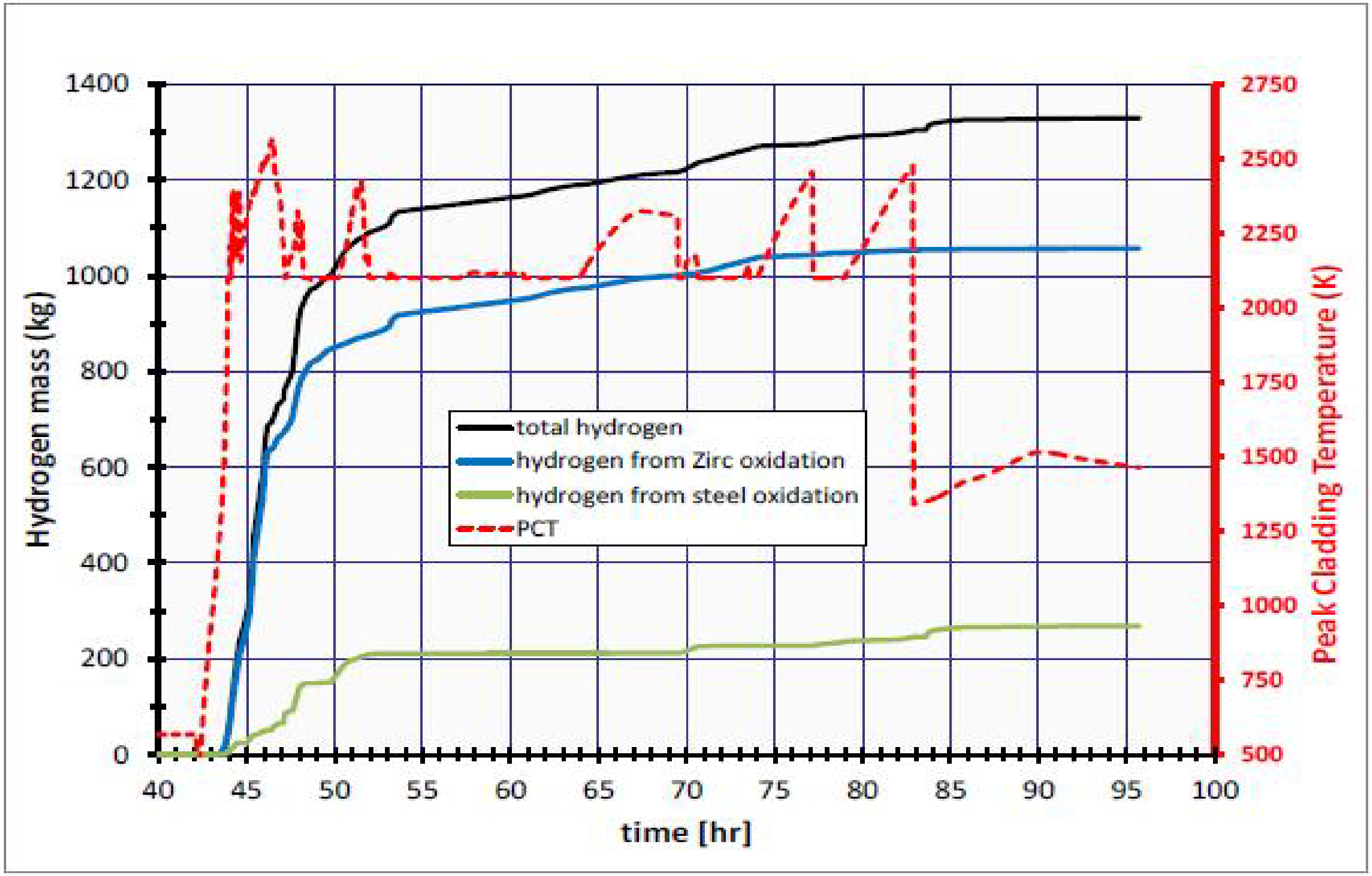
Fukushima Unit 2 Significant Fuel Melt – SAND2012-6173



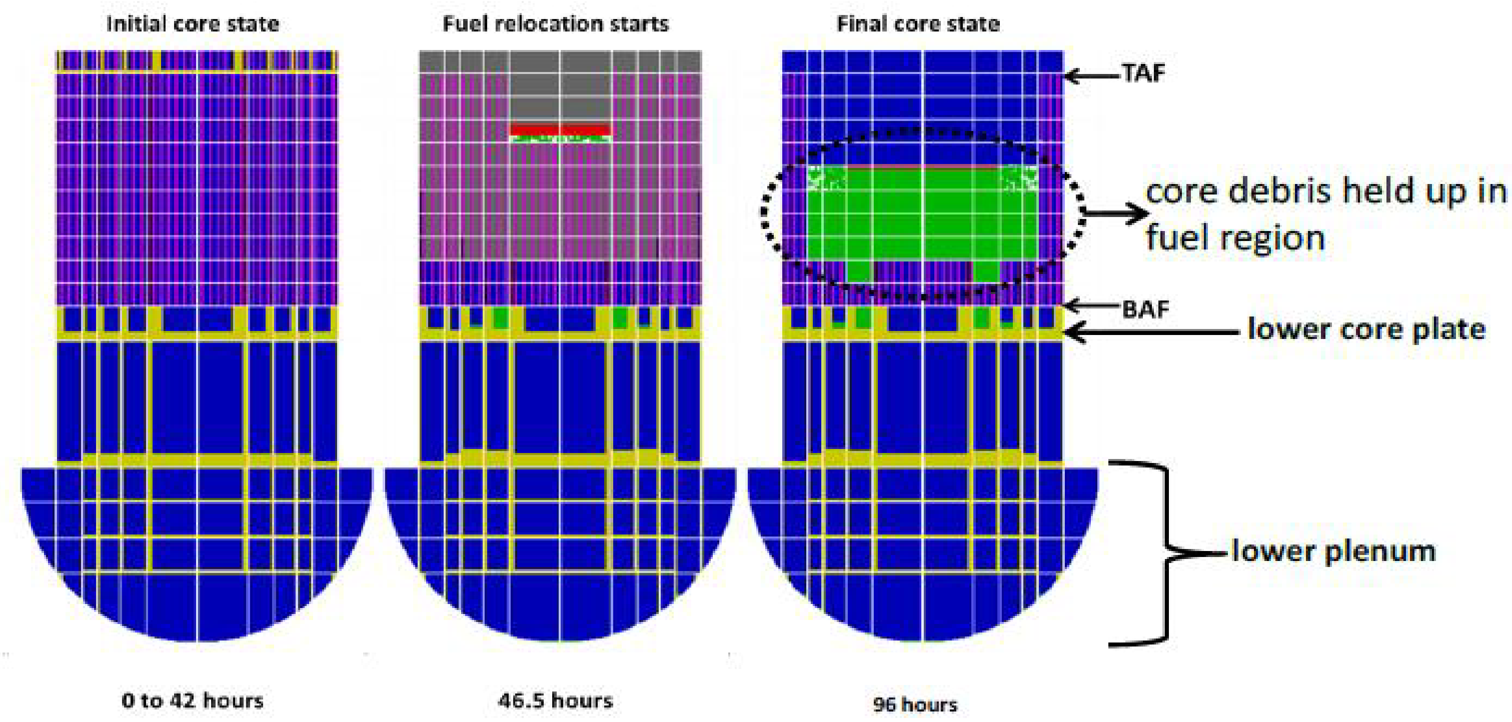
Fukushima Unit 2 Significant Fuel Damage and Fission Product Release –SAND2012-6173



Fukushima Unit 3 hydrogen production little damage to fuel –SAND2012-6173



Fukushima Unit 3 Fuel Retained in the RPV retained damage 58% of Noble Gases Released –SAND2012-6173



MELCOR Role Evolving

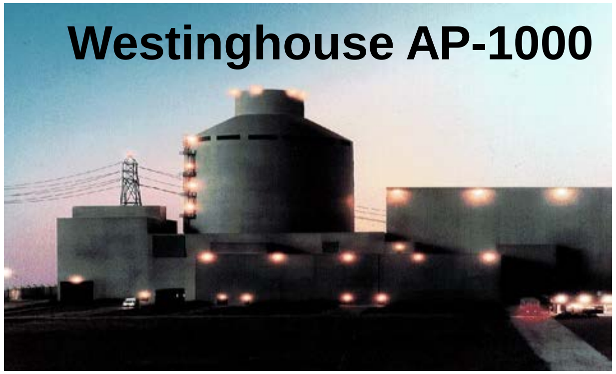
- Original role for PRAs required simpler, fast-running code
  - Uncertainties assessed through sensitivity studies
  - Substantial user flexibility allowed for parametric studies
- Recent role uses more detailed models
  - NRC consolidating to one code
  - Assessments against more detailed codes used to determine required model complexity
  - More mechanistic models implemented as necessary
- Recent role using more flexible modeling geometry
  - More generic modeling without “built-in” nodalization
  - Control volume approach used to define plant system
- Application NOT limited to LWR reactor accident analysis

MELCOR Modeling Improvements Assessed with Mechanistic Codes

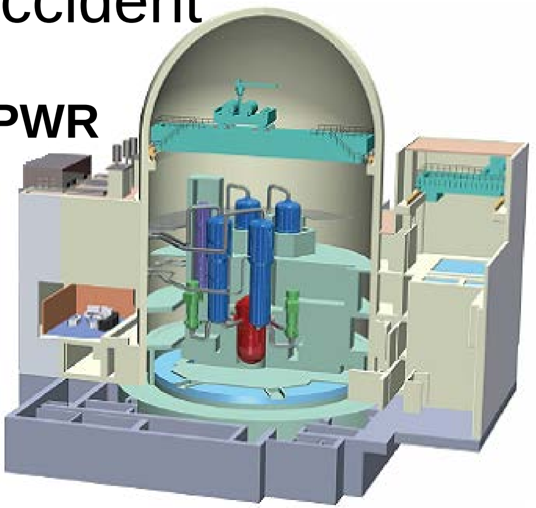
- CONTAIN for containment modeling (completed)
- SCDAP/RELAP5 for core and in-vessel degradation modeling (underway)
  - RCS natural circulation
  - TMI-like core melt progression
  - plant sequence comparisons
- VICTORIA for fission product chemistry and transport models (planned)
  - fission product speciation
  - fission product deposition

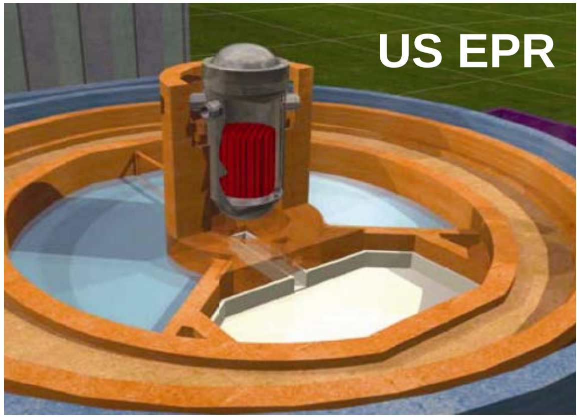
Audit Tool in New Plant Design Certification

- Severe accident response and source term
- Containment response to design basis accident

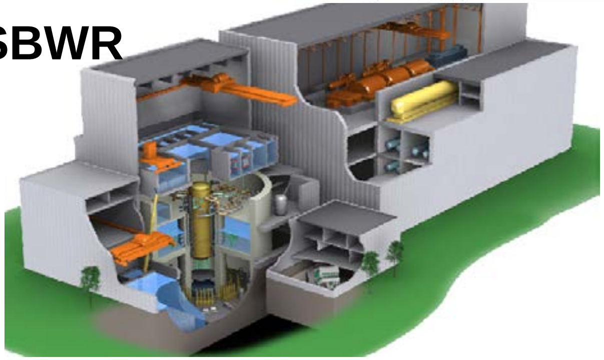


US APWR





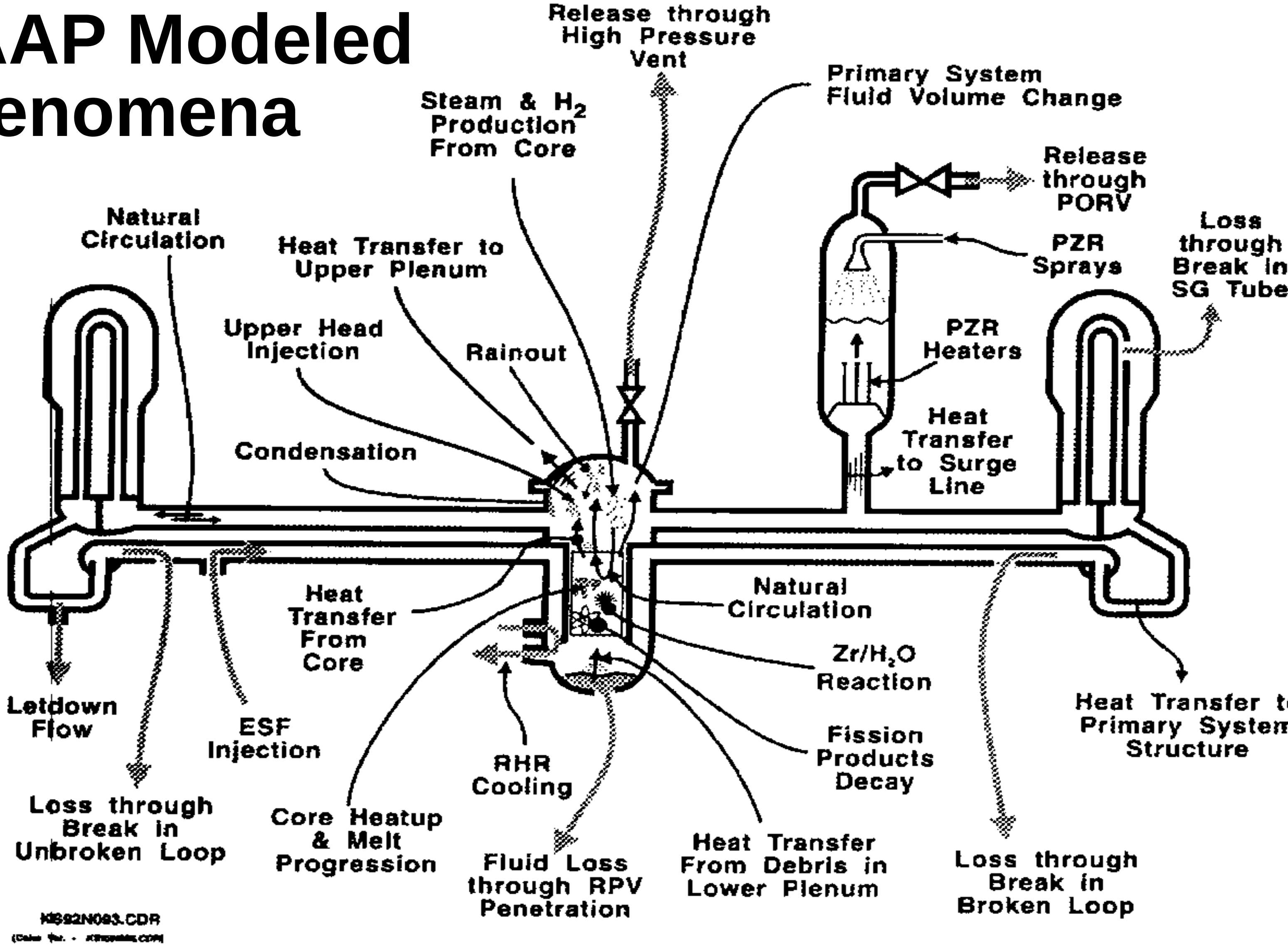
ESBWR



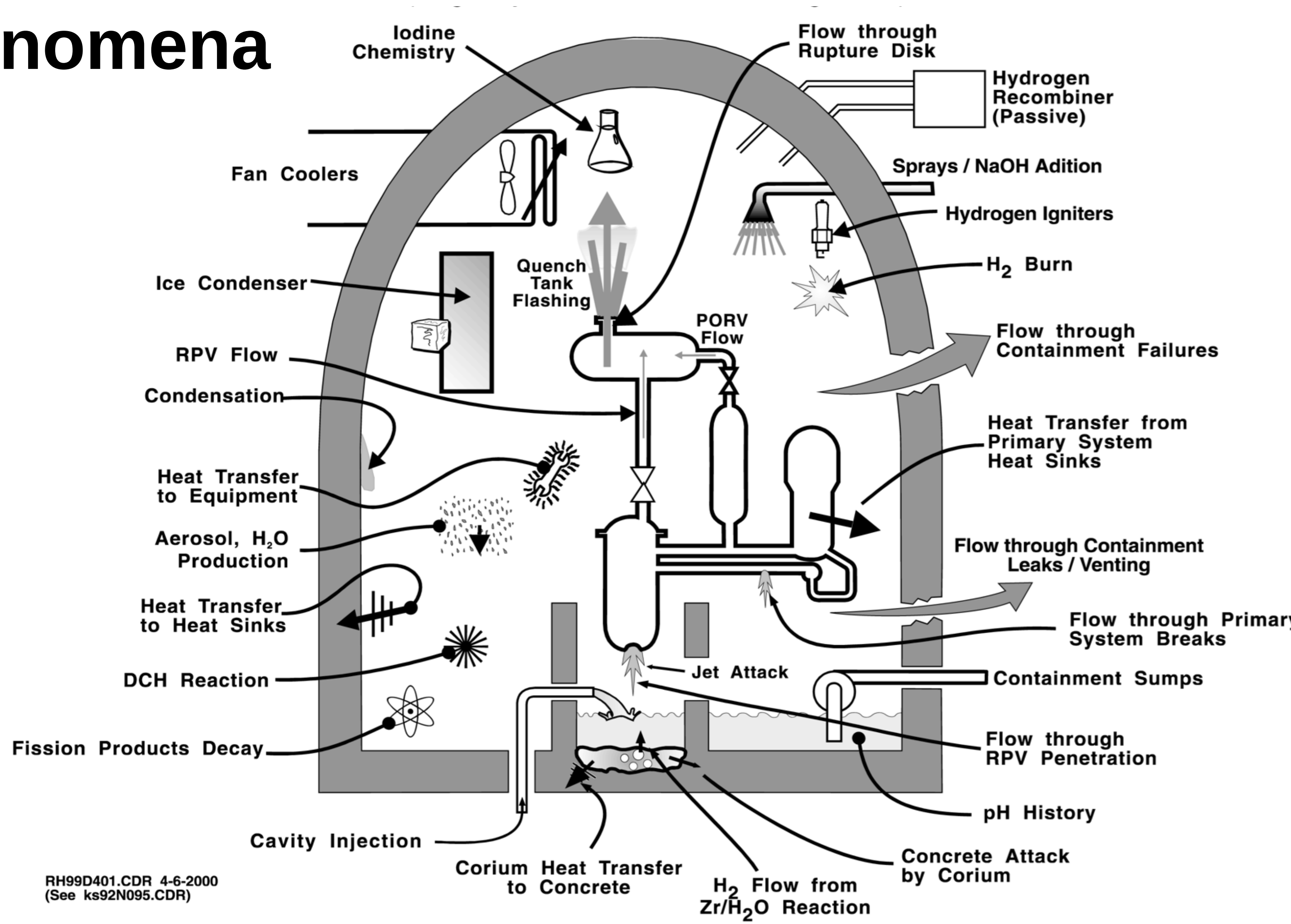
MAAP Designed for Full-Plant Calculations

- Developed & used by industry for PRA and phenomenological studies
- Integrated RCS and containment analysis
- Control system/trip logic functions
- Lumped parameter models provide fast, global approximations
- Design specific versions (e.g., BWR, PWR) with relatively fixed thermal-hydraulic system representations
- Provides for free-form containment modeling
- Model validation against experimental data requires ***special*** models or versions.

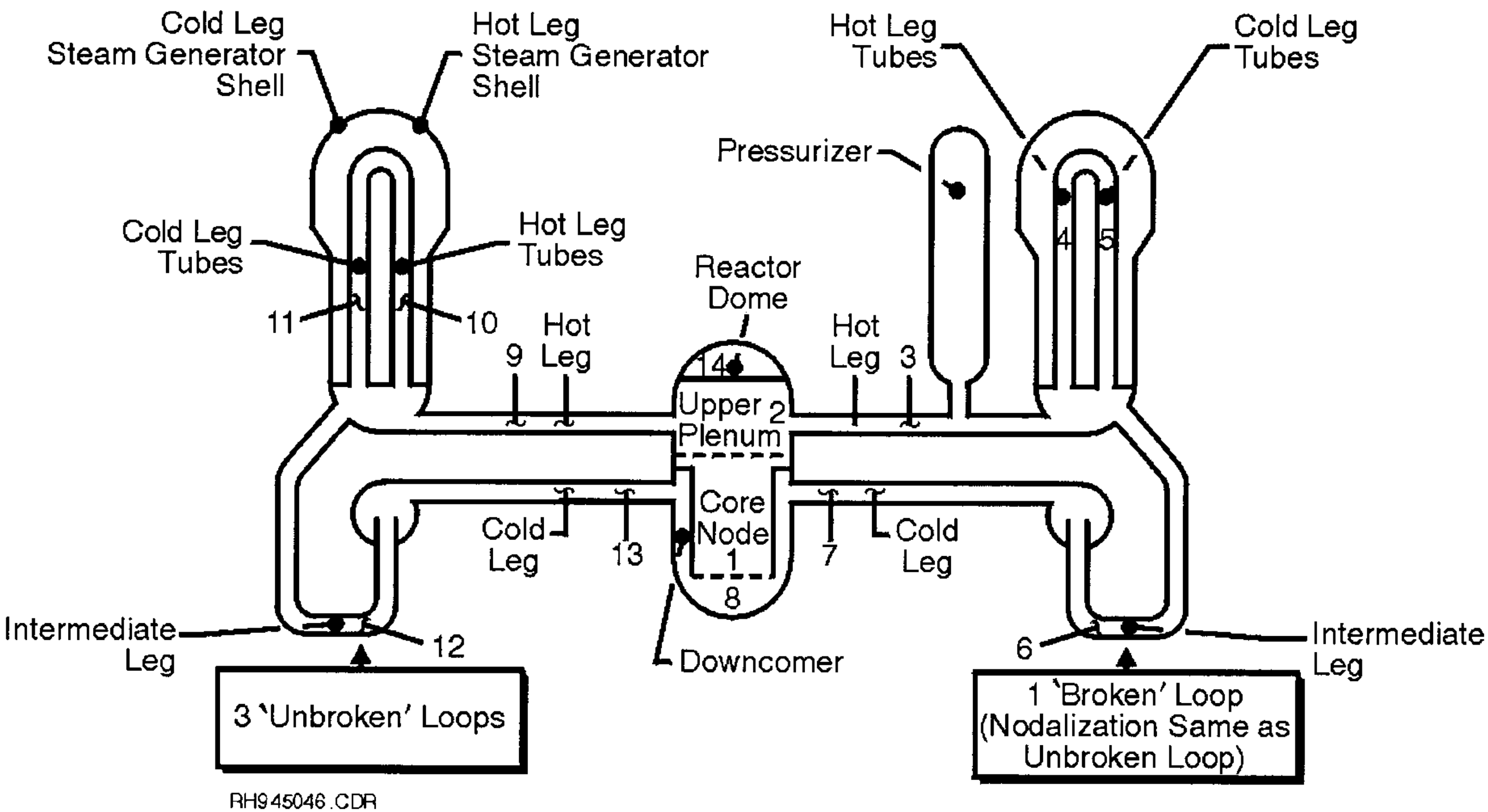
MAAP Modeled Phenomena



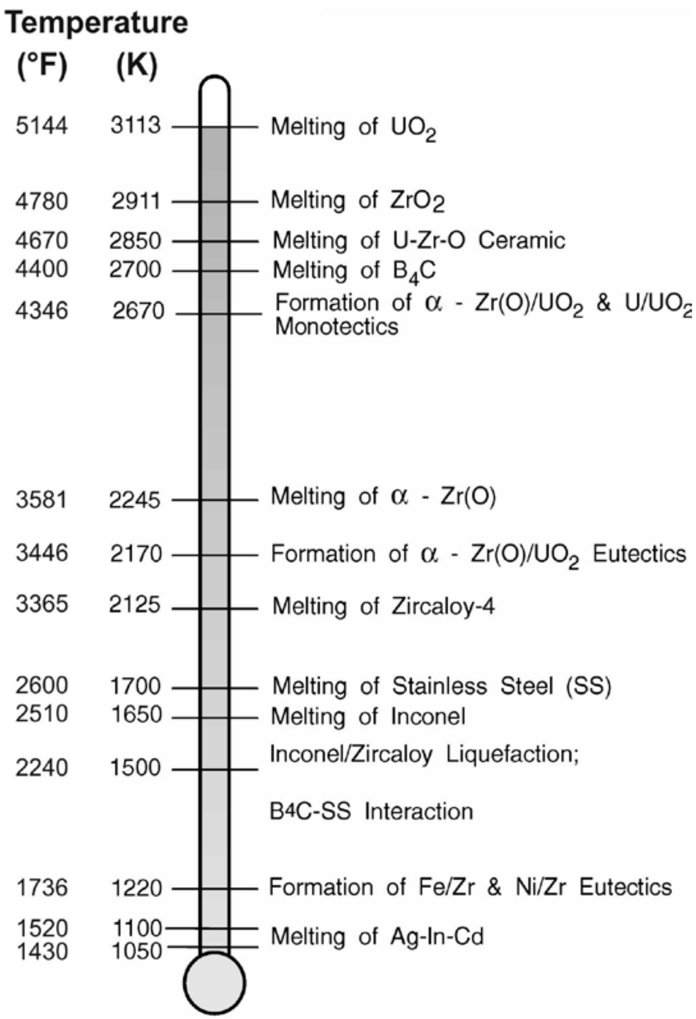
MAAP Modeled Phenomena

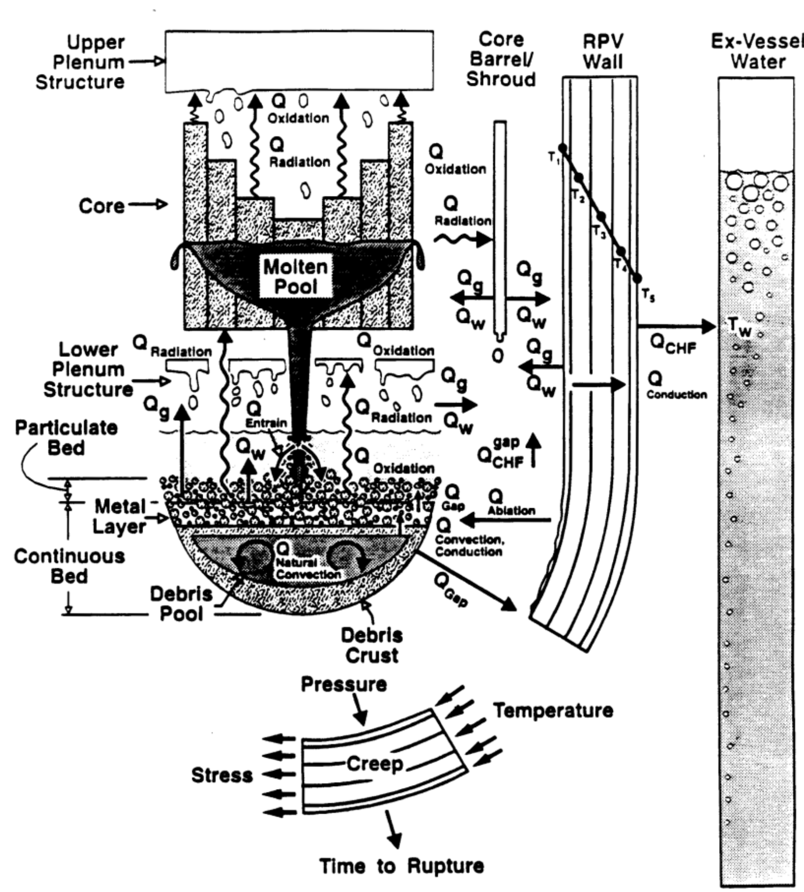


Representative MAAP PWR Analysis Considers Gas Nodes, Heat Structures, and Water Nodes

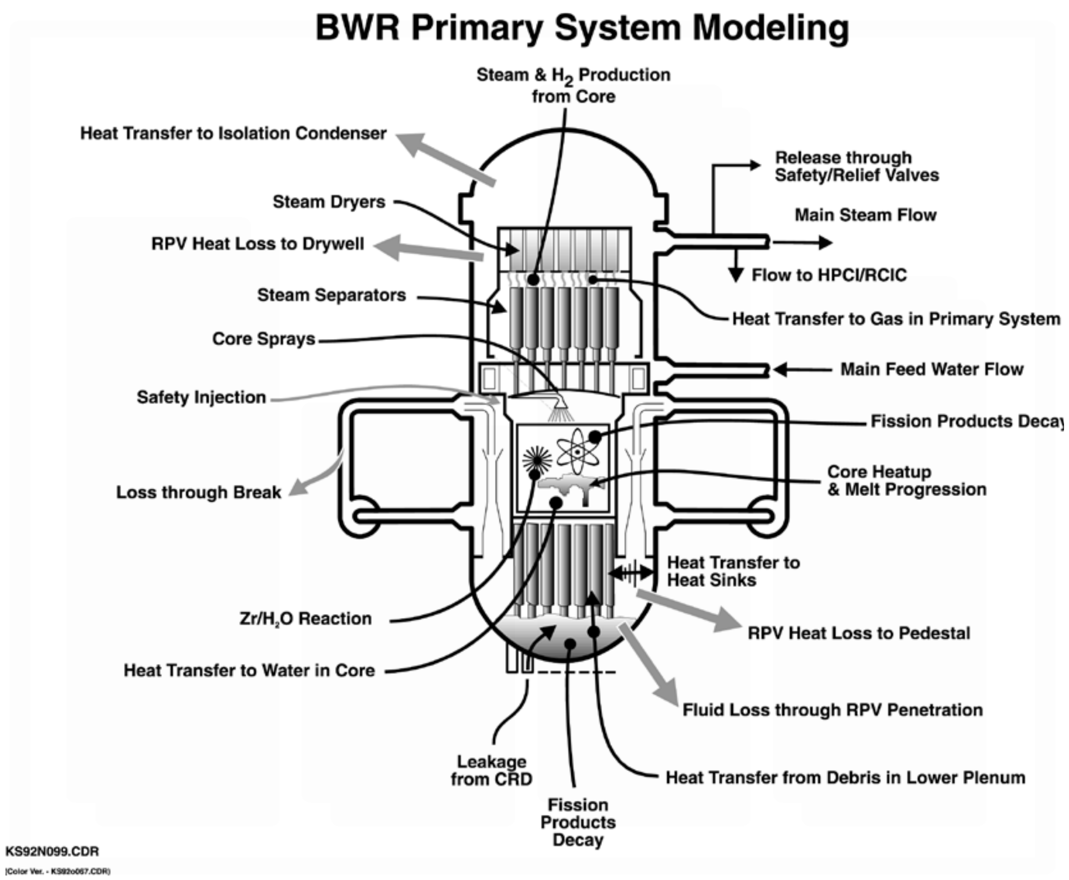


MAAP4 Melt Progression Phenomena

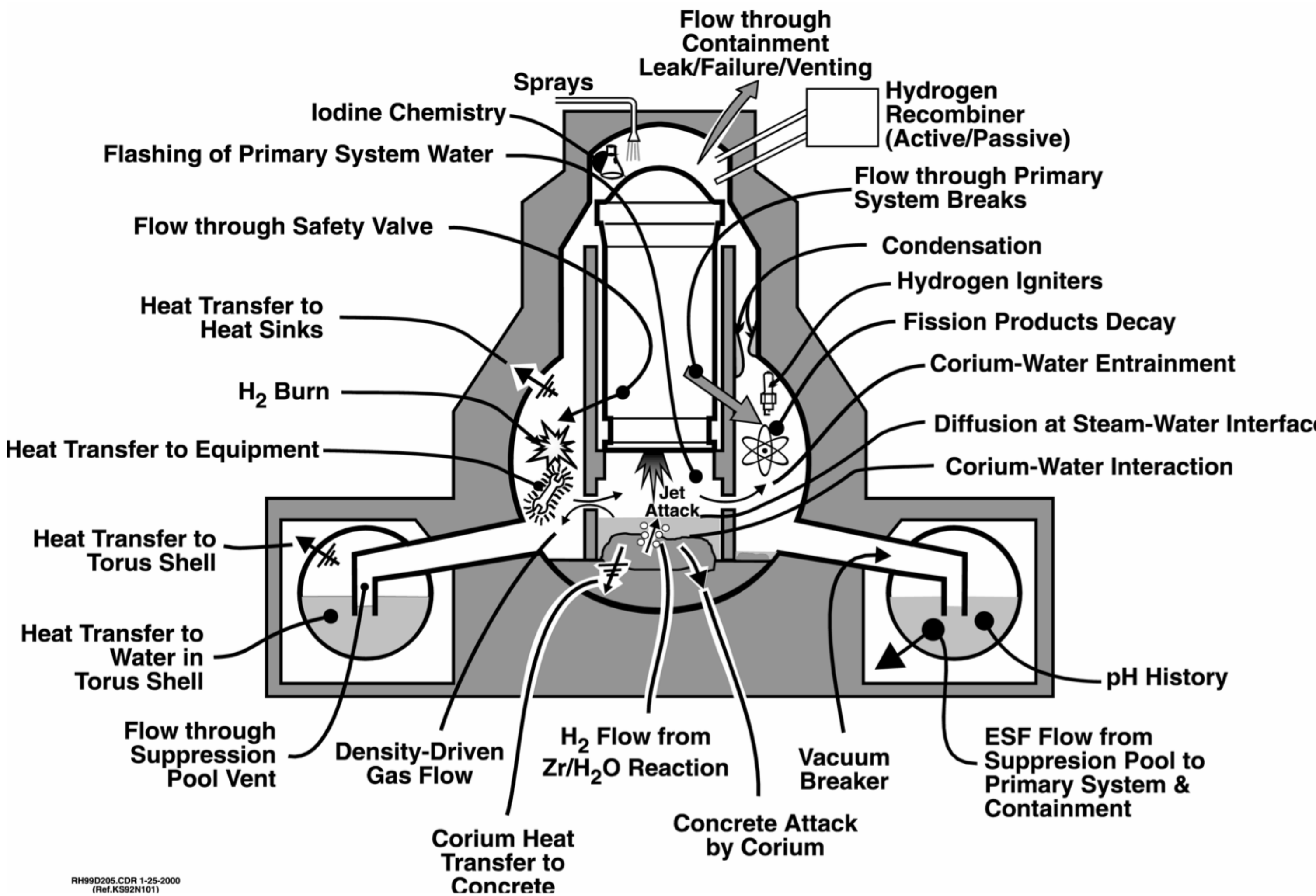




MAAP Considers Unique BWR RCS Phenomena



MAAP Modeled BWR Containment Phenomena



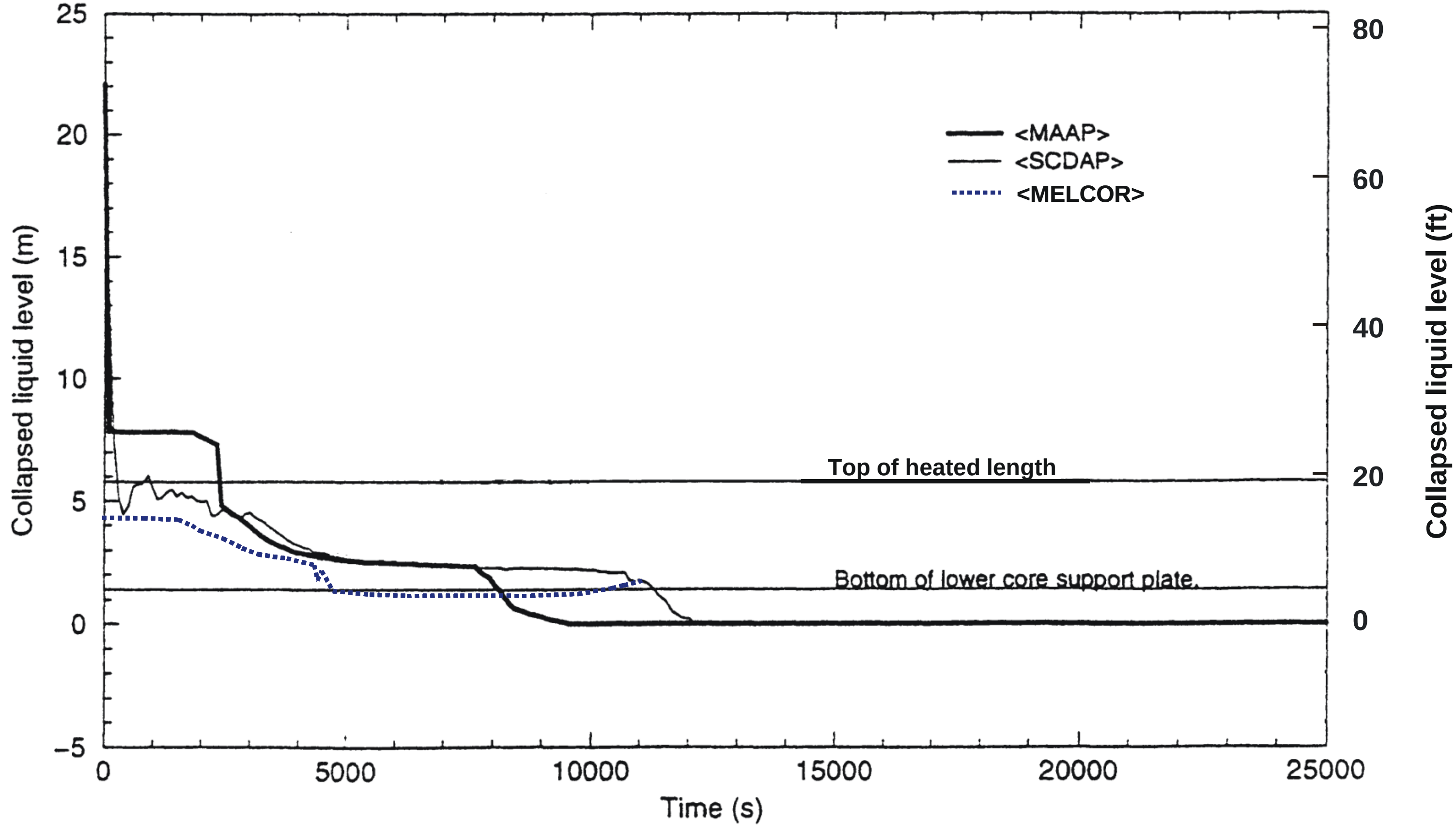
Case Study 2: Comparison of Code Results for AP600 Analysis

Code Models and Assumptions Impact 3BE AP600 Analysis Results

- 3BE transient initiated by large break at location that precludes reactor vessel reflood.
- Key assumptions affecting results:

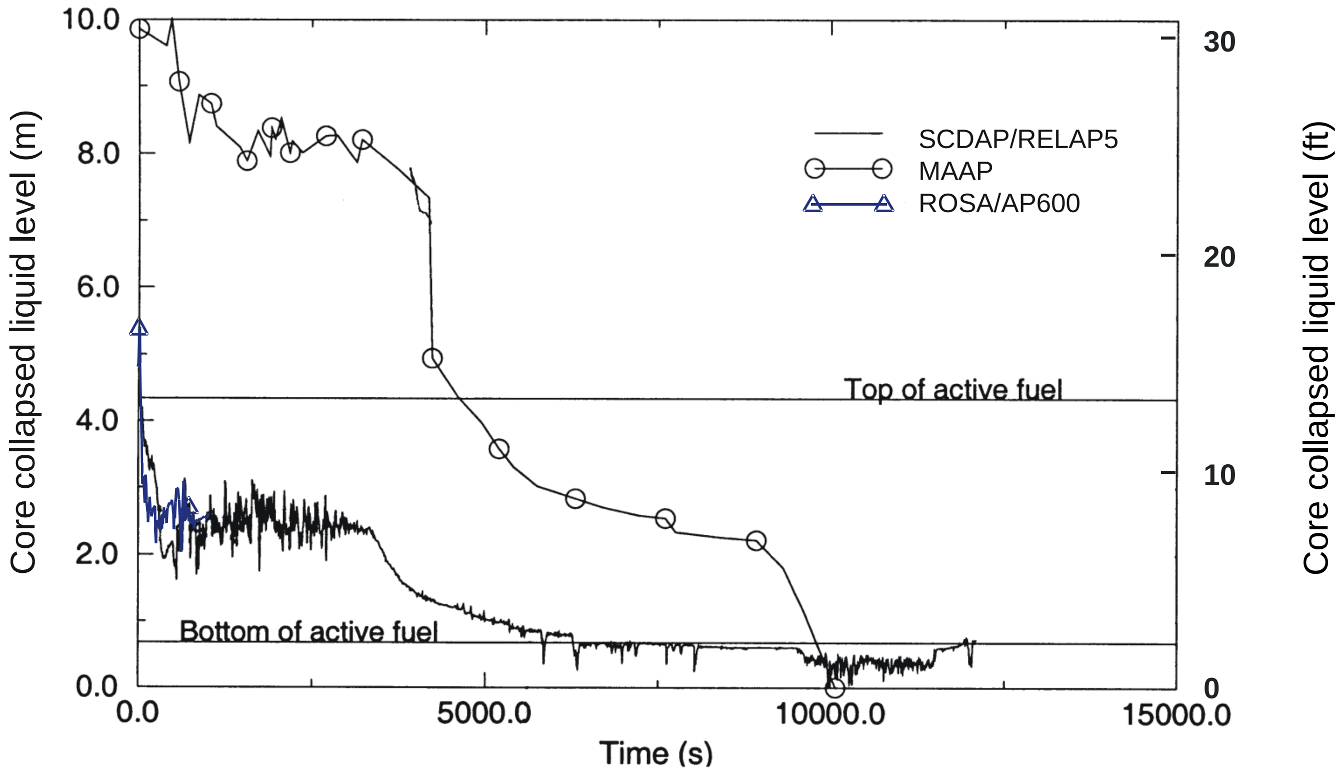
Phenomenon	SCDAP/RELAP5-3D	MAAP	MELCOR
RCS Depressurization Model	Ransom/Trapp critical flow model (results consistent with ROSA/AP600 data)	Single phase critical flow model (unexplained mass retained in RCS)	Two-phase critical flow model (with user supplied discharge coefficients)
Fuel melting	At 2870 K / 4710 ºF due to eutectic formation	At 3100 K / 5120 ºF (UO_2 melting temperature)	At user-specified temperature.
Hydrogen generation	Throughout core degradation	Until first relocation	Until cladding failure temperature.
Relocation to vessel	If crust cannot support molten material	When melting temperature is predicted	When fuel melting occurs, material relocates to core plate and is retained until core plate reaches user-specified temperature.
Debris-to-vessel heat transfer	No enhanced debris cooling (model developed and data now available)	Enhanced cooling from water in user-specified gaps with user-specified heat transfer	No enhanced debris cooling (model developed, and data now available)

Code Models and Assumptions Impact 3BE AP600 Analysis Results (continued)



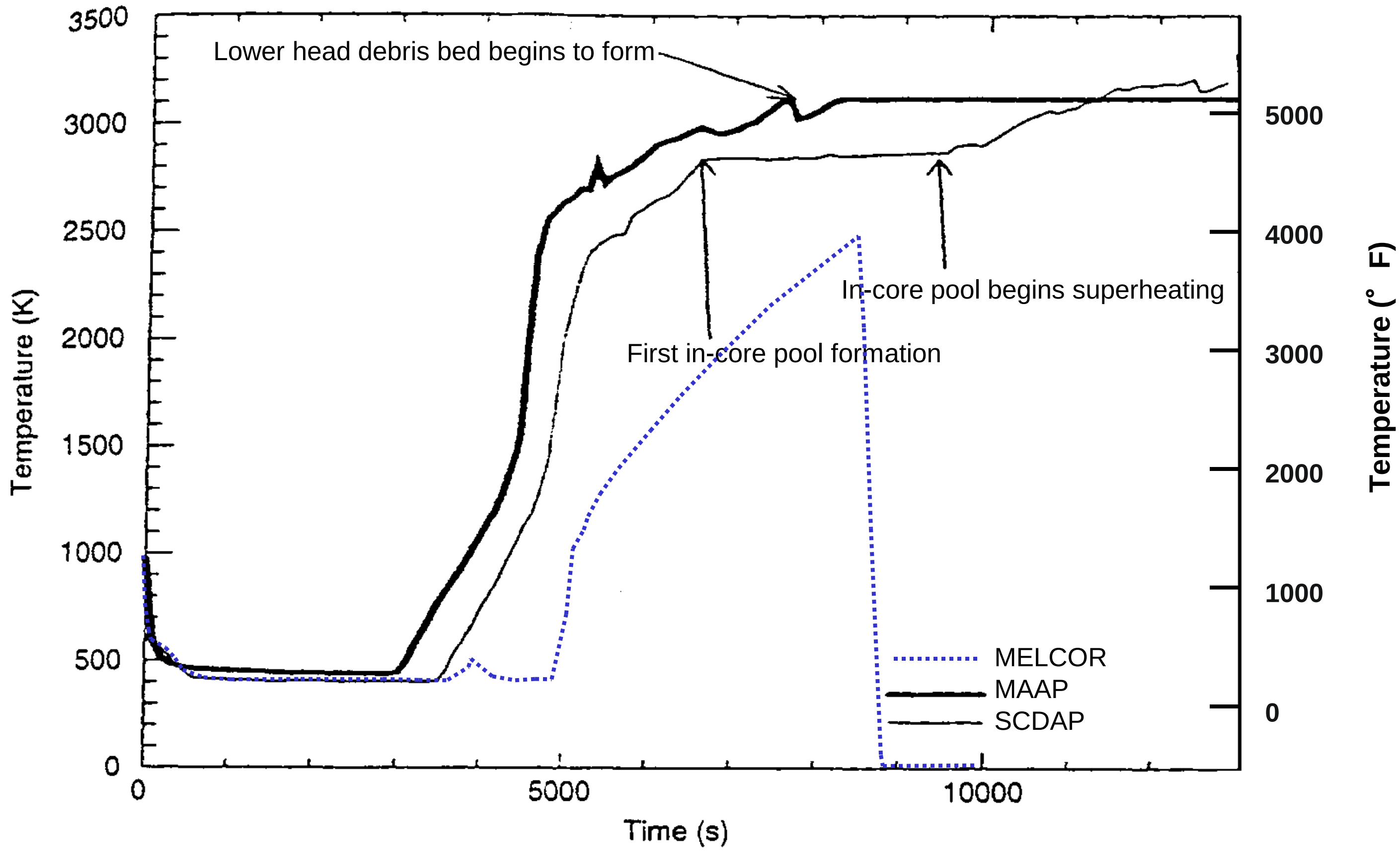
Unexplained additional coolant retained in RCS for MAAP calculation

Code Models and Assumptions Impact 3BE AP600 Analysis Results (continued)



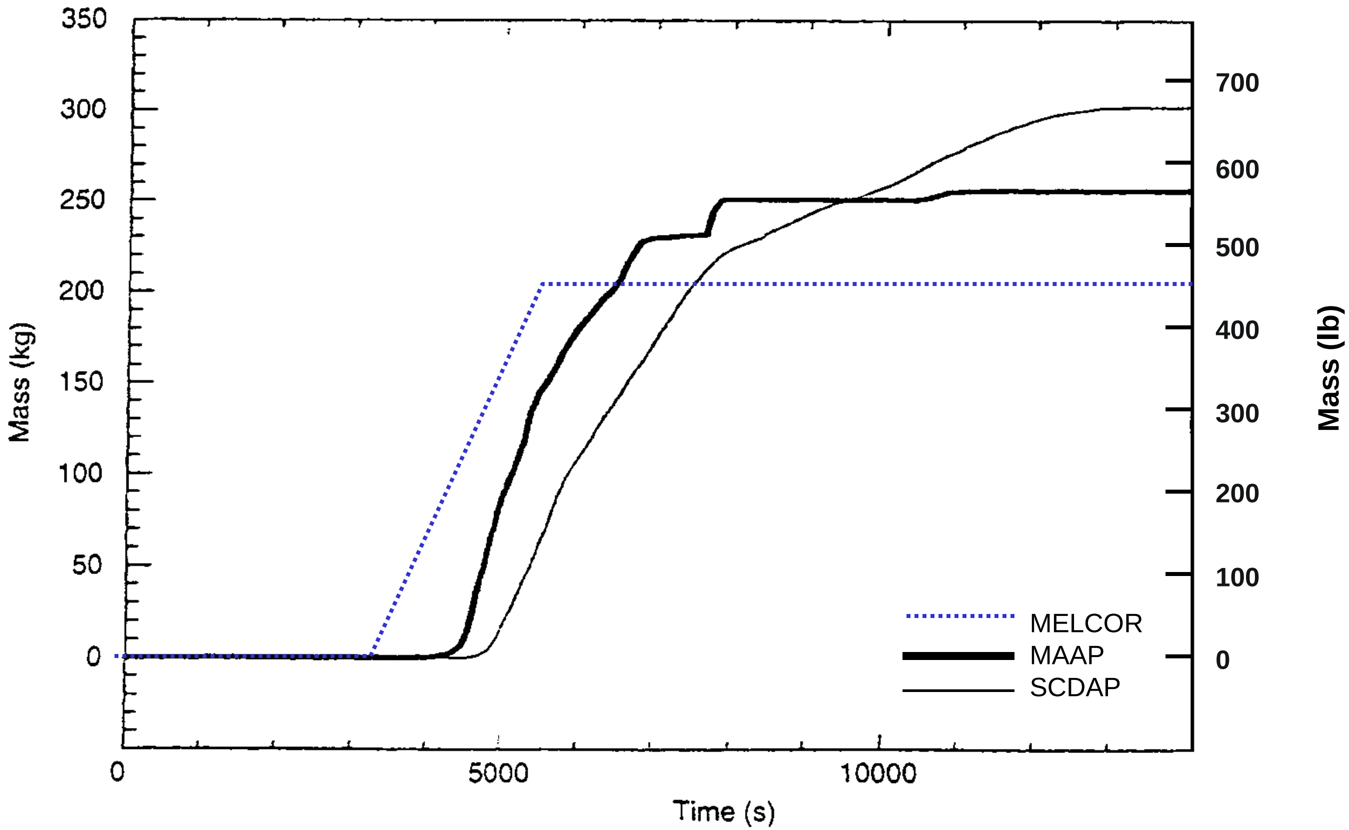
SCDAP/RELAP5-3D core uncovery consistent with ROSA/AP600 data.

Code Models and Assumptions Impact 3BE AP600 Analysis Results (continued)



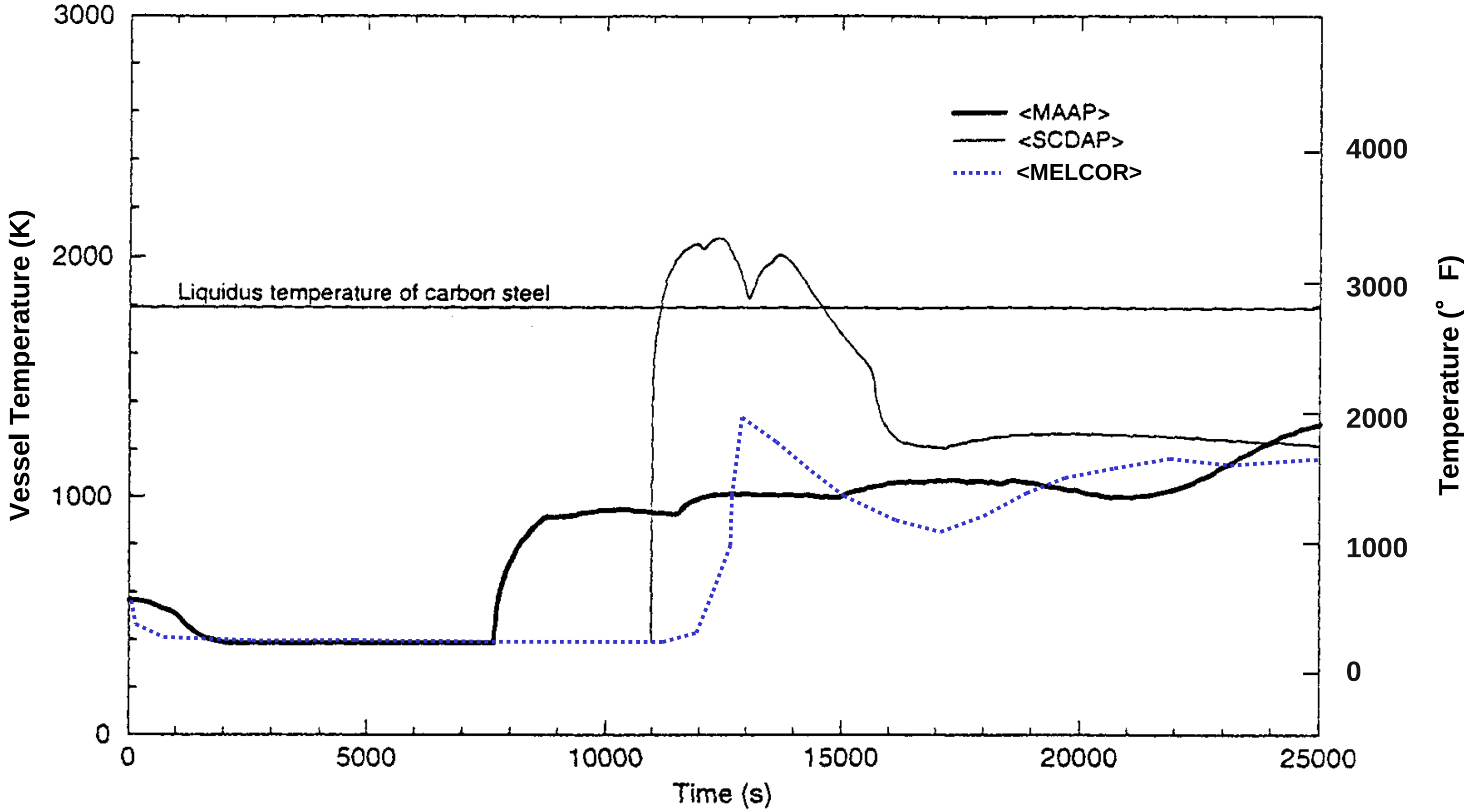
MELCOR shows delayed core heatup despite early core uncovery.

Code Models and Assumptions Impact 3BE AP600 Analysis Results (continued)



MAAP and MELCOR predict much lower total hydrogen generation.

Code Models and Assumptions Impact 3BE AP600 Analysis Results (continued)



MELCOR and MAAP predict lower debris heat load on vessel wall

Summary and Discussion

- Selection of mature US severe accident analysis codes available.
  - Codes differ in modeling approaches
  - Codes have undergone fairly extensive code-to-data comparisons.
  - Insights from code calculations have played a key role in resolving accident management issues
- Analysis reviews must consider impact of code modeling assumptions and approaches

Regulatory Considerations (SECY-93-087)

- Hydrogen Control
  - 10CFR50.44, "Combustible Gas Control for Nuclear Reactors"
  - Capability to ensure a mixed atmosphere
  - Maintain atmospheric concentration of hydrogen below 10% by volume
  - Maintain containment integrity in the event of a deflagration

- Core Debris Coolability
  - Provisions to spread and quench molten core debris
  - Ensure that the environmental conditions (pressure and temperature) resulting from core-concrete interactions do not exceed established criteria

- Containment Performance
  - Maintain role as a leak-tight barrier for 24 hours following core damage
  - Post-24 hours, provide a barrier against uncontrolled fission product release
  - Consideration of in-vessel and ex-vessel steam explosion

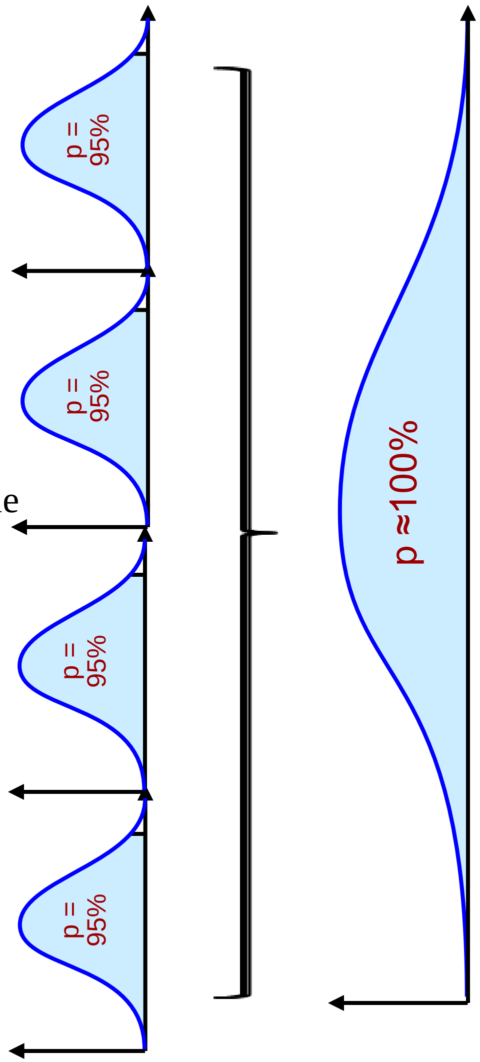
- High Pressure Melt Ejection
  - Reliable depressurization system
  - Features to decrease ejected debris in the upper containment

Methods

Uncertainty Convolution: Deterministic vs. Probabilistic

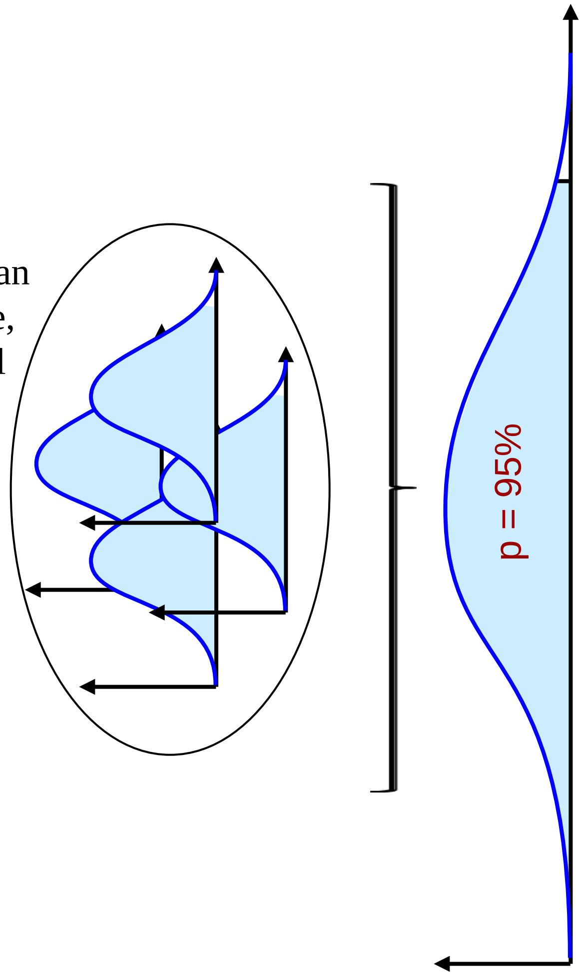
Deterministic Treatment

Key parameters are conservatively bounded, effectively “stacked” upon other conservatisms in a single calculation



Probabilistic Treatment

Key parameters are sampled over an uncertainty range, requiring several calculations



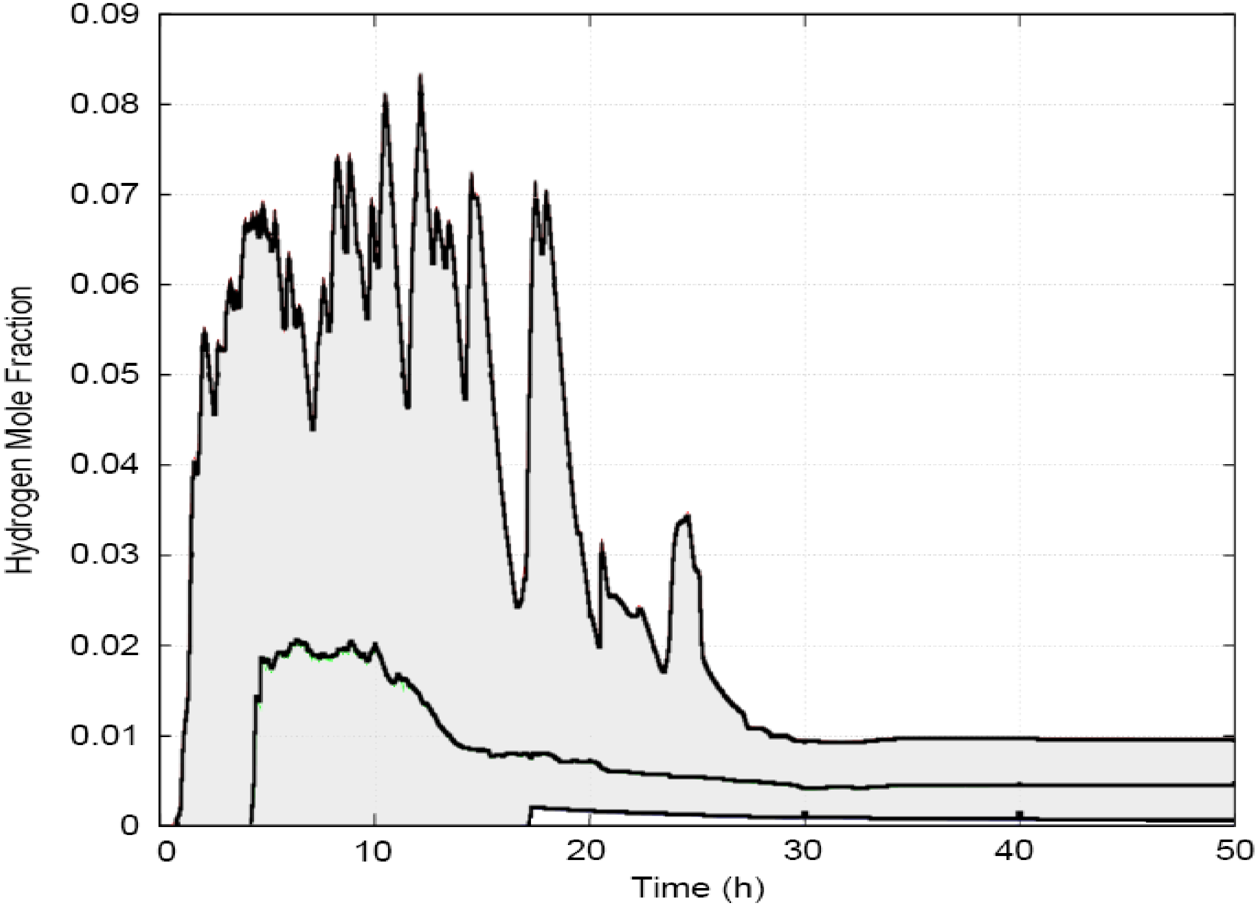
Severe Accident Phenomena (EPR)

- A selection of MAAP4 model parameters
- Perform numerous simulations from random sampling of model parameters
- Statistics can reveal limiting condition, important phenomena

Description	Low Value	High Value
HYDROGEN UNCERTAINTY PARAMETERS		
Zr-H2O Oxidation Multiplier	1.5	2.0
fraction of Zr oxidized to keep cladding intact	0.0	NA
Cladding Melt Breakout Temperature	2500 K	3000 K
Fuel Rod Collapse Temperature (i.e. L-M coef.)	46	54
enable/disable the U-Zr-O eutectic model	NA	1
Fuel Melt Temperature	2500 K	2800 K
Control Rod Melt Temperature	1500 K	2500 K
Melt relocation HTC	0.0	0.15
Particulate debris size in lower plenum	0.01	0.1
Porosity of fuel debris beds	0.26	0.53
CORE DEBRIS COOLABILITY UNCERTAINTY PARAMETERS		
Total Power (decay power)	100%	106%
Initial radius of the local vessel failure	0.005 m	0.25 m
Lower head damage fraction for failure	0	1
corium friction coefficient	0.001	0.1
Flat Plate CHF Kutateladze #	0.2	0.3
Steaming rate (kg/s)	13	17
Emissivities	0.7	1.0

CGCS Analysis: Tolerance Limit of H2 Concentration

- Licensing limit is 10%



Study Questions

- Name U.S.-developed codes used in severe accident analysis
- What phenomena are considered in each severe accident analysis code?
- Discuss differences in code modeling approaches that may impact code predictions
- List some key questions to ask when reviewing an analysis

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Tier 2 NRC Recommendations

- Spent fuel pool makeup capability *(Recommendation 7.2, 7.3, 7.4, and 7.5)*
- Emergency preparedness regulatory actions *(Recommendation 9.3)*
- Other External Hazards Reevaluation (tornados, hurricanes, drought, etc.)*(additional Issue)*

Tier 3 NRC Recommendations

Potential enhancements to the capability to prevent or mitigate seismically-induce fires and floods (long-term evaluation) *(Recommendation 3)*
•Reliable hardened vents for other containment designs
•(long-term evaluation) *(Recommendation 5.2)*
•Hydrogen control and mitigation inside containment or in other buildings
•(longterm evaluation) *(Recommendation 6)*
•Emergency preparedness enhancements for prolonged station blackout and multiunit events
•(dependent on availability of critical skill sets) *(Recommendation 9.1/9.2)*

Tier 3 NRC Recommendations

•Emergency Response Data System capability (related to long-term evaluation Recommendation 10) *(Recommendation 9.3)*
•Additional emergency preparedness topics for prolonged station blackout and multiunit events (long-term evaluation) *(Recommendation 10)*
•Emergency preparedness topics for decision-making, radiation monitoring, and public education (longterm evaluation) *(Recommendation 11)*
•Reactor Oversight Process modifications to reflect the recommended defense-indepth framework (dependent on Recommendation 1) *(Recommendation 12.1)*
•Staff training on severe accidents and resident inspector training on severe accident management guidelines (dependent on Recommendation 8) *(Recommendation 12.2)*
•Basis of emergency planning zone size *(additional issue)*
•Prestaging of potassium iodide beyond 10 miles *(additional issue)*
•Ten-year confirmation of seismic and flooding hazards (dependent on Recommendation 2.1) *(Recommendation 2.2)*
•Transfer of spent fuel to dry cask storage *(additional issue)*

8. Radionuclide Release and Transport

- Introduction
- Characterization
- Phenomena
- Quantification
- Study Questions
- References

Objectives

- Identify and understand factors affecting radionuclide release and transport during a severe accident.
- Identify and describe differences between various methods and approaches used to estimate severe accident releases.

Inventory Characterized in Terms of Decay Rates

One curie (Ci) of material undergoes radioactive decay at 3.7×10^{10} dps

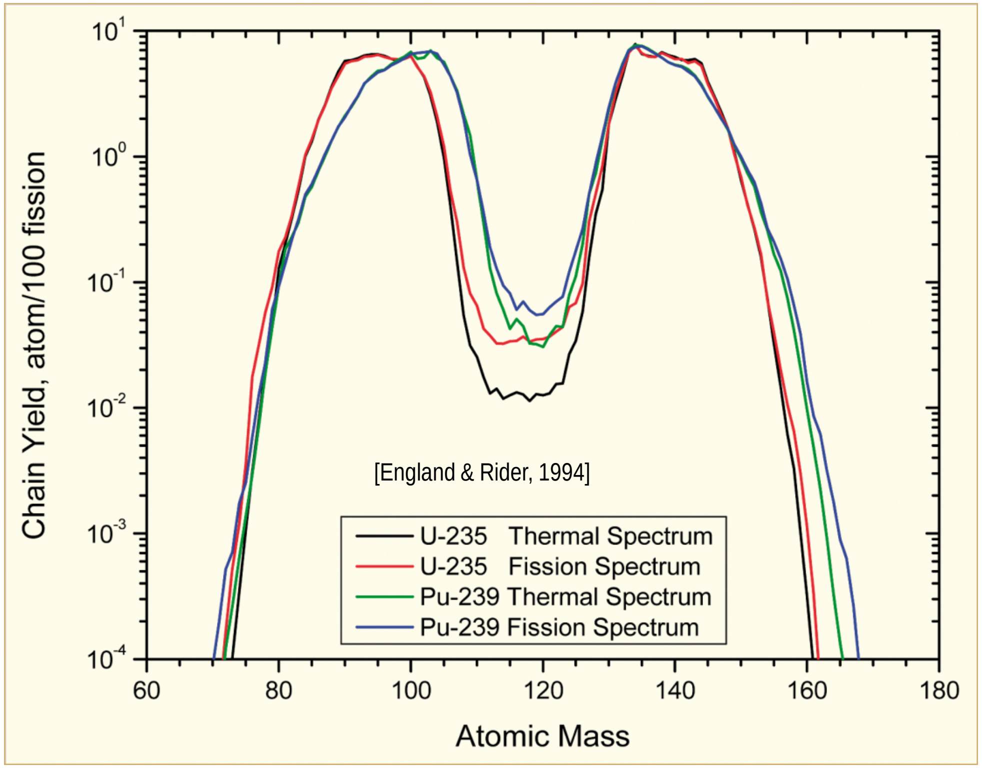
– 1 Becquerel (Bq) = 1 dps, or

– 1 Ci = 3.7×10^{10} Bq

Categories of Fission Product Inventory

- Volatile
  - Gases and evaporated elements (e.g., I, Cs, and Br)
  - Transport dominated by diffusion
- Semi-volatile
  - Liquids and aerosols, elements susceptible to evaporation
  - Rates influences by chemistry and temperature
  - Transport dominated by evaporation-driven mass transfer
- Non-volatile
  - Solids and aerosols
  - May become volatile only at very high temperatures
- Non-radioactive
  - Solids, liquids, or gases
- Inert vs. chemically reactive

Fission Product Yields Vary Based on Source and Burnup



(Rudling and Peterson -2012)

- Wide range of elements produced by fission
  - Probabilistic process with “light” and “heavy” distributions
  - Yields vary significantly by atomic mass and slightly by the fissile nuclide and neutron energy
- Cumulative production rate is ~0.1% per GWd/MTU

Most Volatile Radionuclides Reside in Reactor Core

Location	Inventory, Ci	
	Noble Gases (Xe, Kr)	Iodine (I)
Core	4.0E+8 (1.48E+19 Bq)	7.5E+8 (2.775E+19 Bq)
Gap between UO_2 fuel and Zr cladding	3.0E+7 (1.11E+18 Bq)	1.4E+7 (6.29E+17 Bq)
Spent fuel storage pool	1.0E+6 (3.7E+16 Bq)	5.0E+5 (5.18E+15 Bq)
Primary coolant[3]	1.0E+4 (3.7E+14 Bq)	6.0E+2 (2.22E+13 Bq)

[3]*Nominal value, varies depending on fuel leakage.*

Average Annual Plant Release Considerably Lower than Accident Releases

	Noble Gases, Ci	**Iodine, Ci**
Average annual reactor release (1975-1979)	1.00 (3.7E+10 Bq)	0.13 (4.81E+9 Bq)
TMI-2 accident (March 1979)	2.50E+6 (9.25E+16 Bq)	15 (5.55E+11 Bq)
Chernobyl accident (April 1986)	1.90E+8 (7.03E+18 Bq)	4.5E+7 (1.665E+18 Bq)

Radionuclide Inventory Time-Dependent

$$dA_i(t)/dt = -\Lambda_i(t)A_i(t) + Q_n(t)$$

where

$\Lambda_i(t)$ - fractional loss rate due to deposition, decay, leakage, sprays, etc.

$A_i(t)$ - activity of species, i,

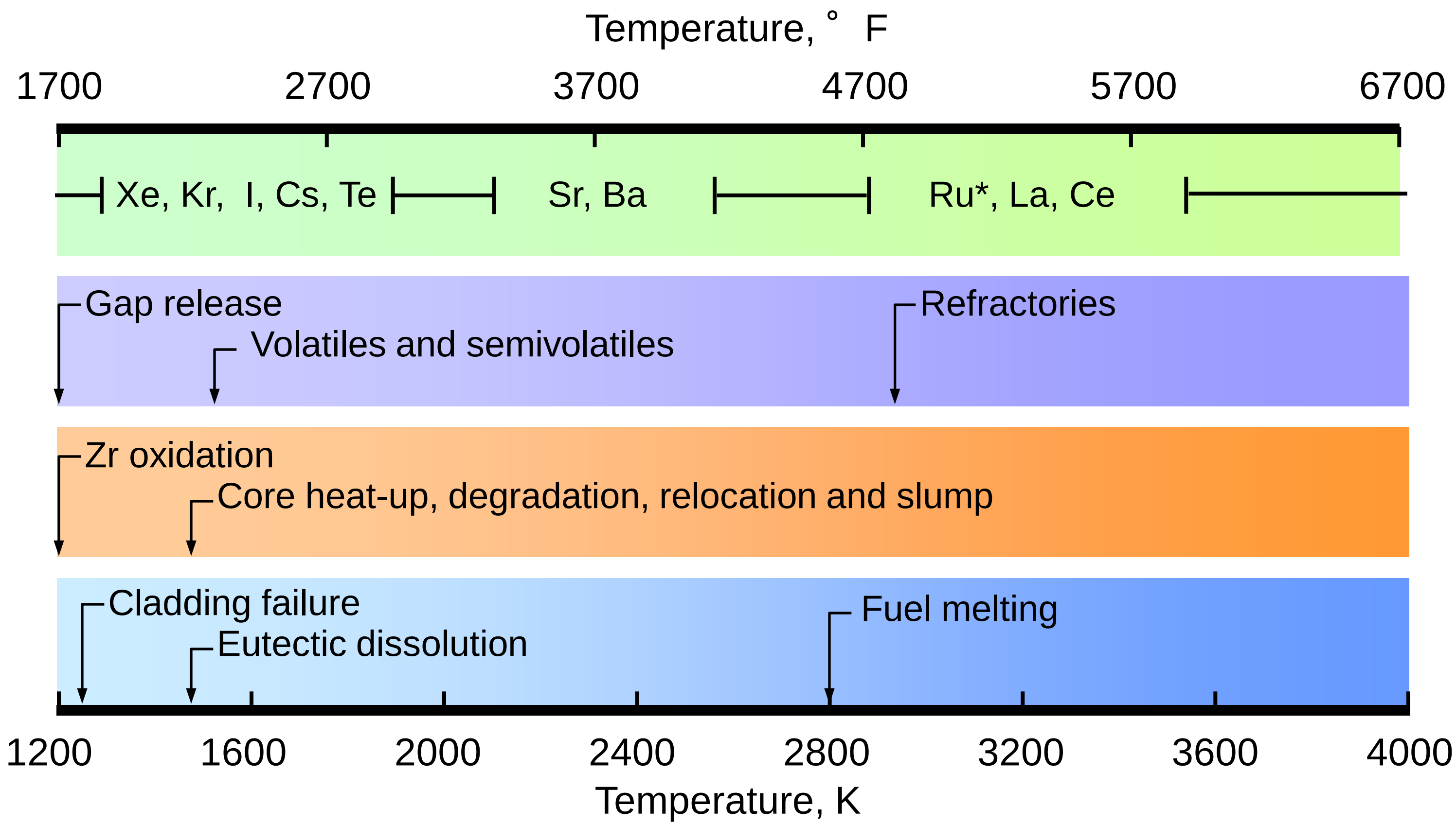
$Q_n(t)$ - activity source rate due to fuel release, MCCI, contribution entering from another volume, etc.

Radionuclide Inventory Grouped by Chemical Properties and Volatility

Group Number[1]	Release Class	Volatility	Isotopes	Group Total (Ci)[2]
1	Noble Gases	Inert	Kr-85, Kr85m, Kr-87, Kr-88, Xe-133, Xe-135	3.84 E+08 (1.4208E+19 Bq)
2	Halogens	Volatile	I-131, I-132, I-133, I-134, I-135	7.71E+08 (2.8527E+19 Bq)
3	Alkali Metals		Cs-134, Cs-136, Cs-137, Rb-86	2.18E+07 (8.066E+17 Bq)
4	Tellurium		Sb-127, Sb-129, Te-127, Te-127m, Te-129, Te-129m, Te-131m, Te-132	2.13E+08 (7.881E+18 Bq)
5	Strontium	Non-volatile	Sr-89, Sr-90, Sr-91, Sr-92	3.57E+08 (1.3209E+19 Bq)
6	Noble Metals		Co-58, Co-60, Mo-99, Rh-105, Ru-103, Ru-103, Ru-105, Tc-99m	5.94E+08 (2.1978E+19 Bq)
7	Lanthanides		Am-241, Cm-242, Cm-244, La-140, La-141, La-142, Nb-95, Nd-147, Pr-143, Y-90, Y-91, Y-92, Y-93, Zr-95, Zr-97	1.54E+09 (5.698E+19 Bq)
8	Corium (Cerium)		Ce-141, Ce-143, Ce-144, Np-239, Pu-238, Pu-239, Pu-240, Pu-241	2.15E+09 (7.955E+19 Bq)
9	Barium		Ba-139, Ba-140,	3.38E+08 (1.2506E+19 Bq)

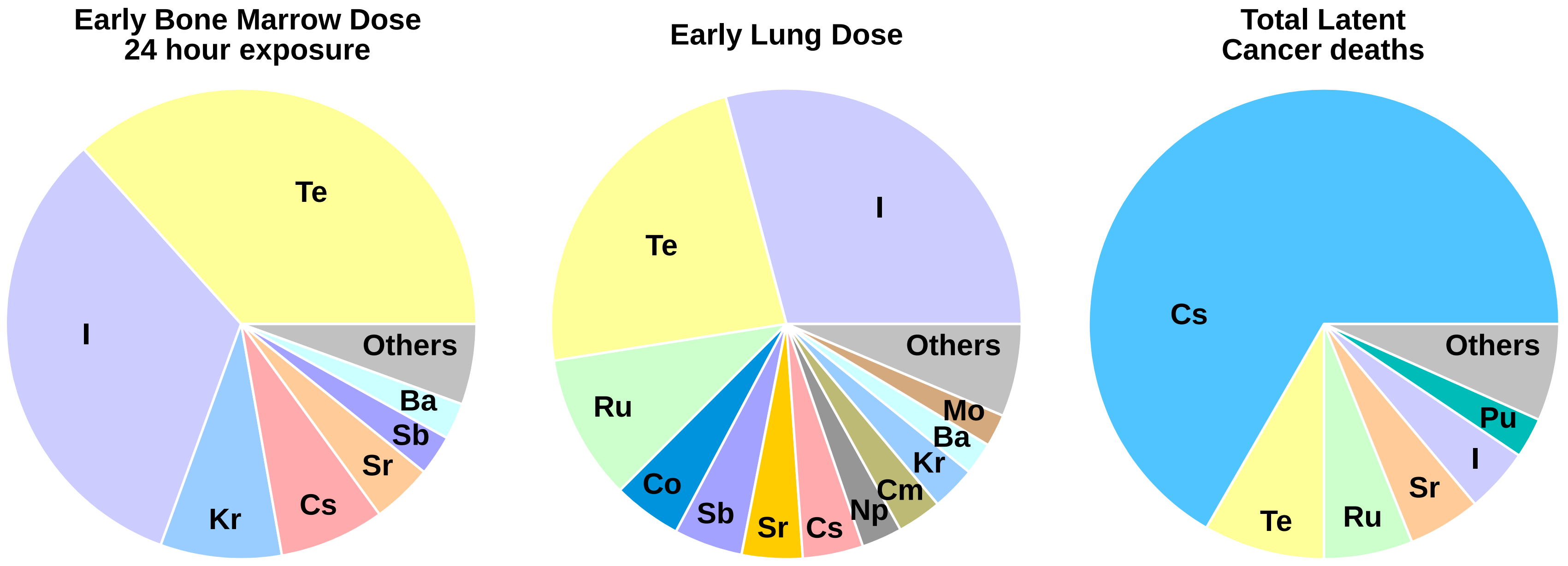
[1] Group definitions vary in different approaches.
[2] For representative large (3300 MWt) LWR 30 minutes after shutdown.

Group Release Tied to Fuel Temperature



*In highly oxidizing environment, Ru is volatile

Radiological Impact of Isotopes Differ- Overall Exposure of 600 Rem or 6 Sv Considered Potentially Fatal

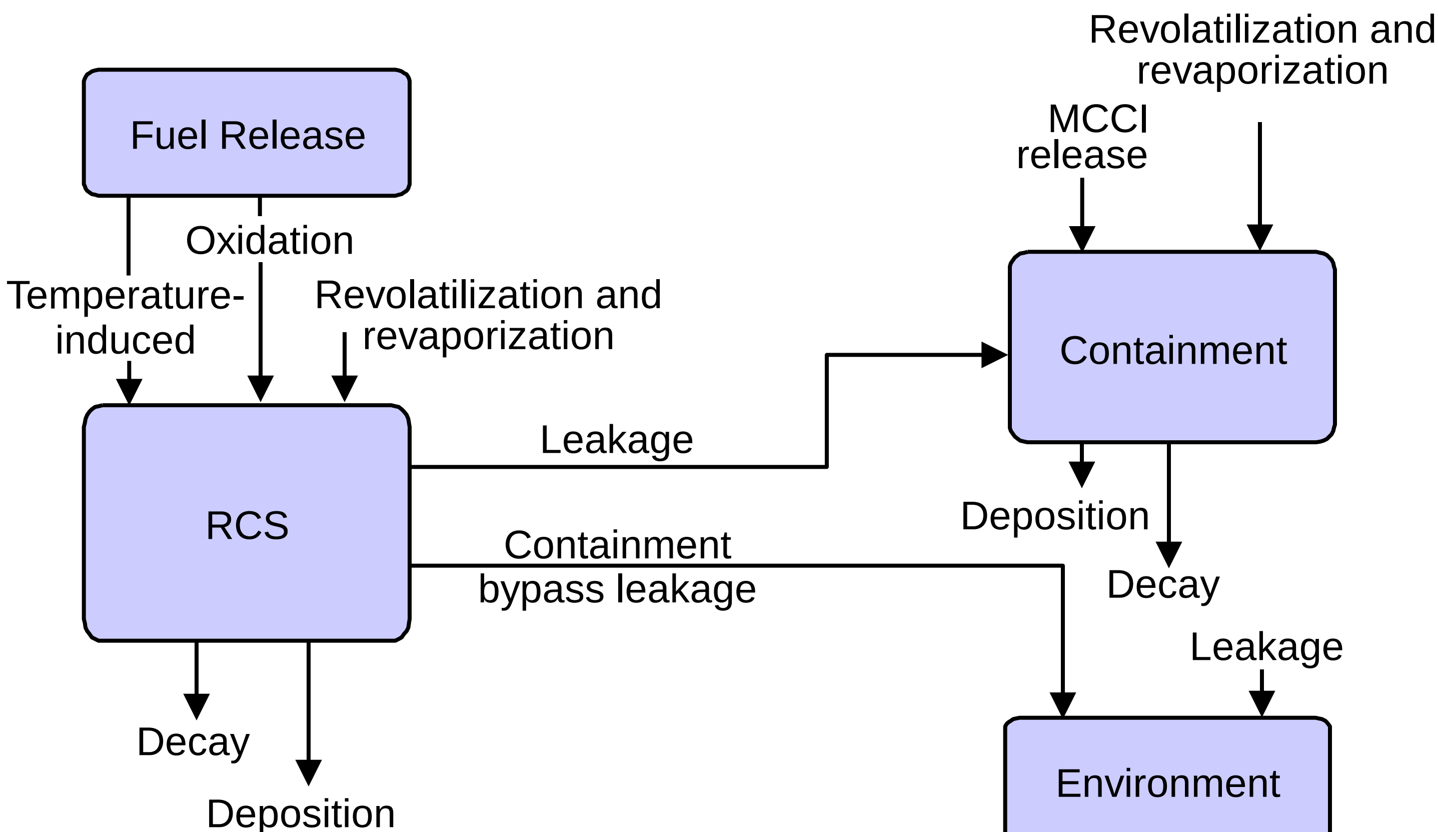


Assumes unit release of each element.

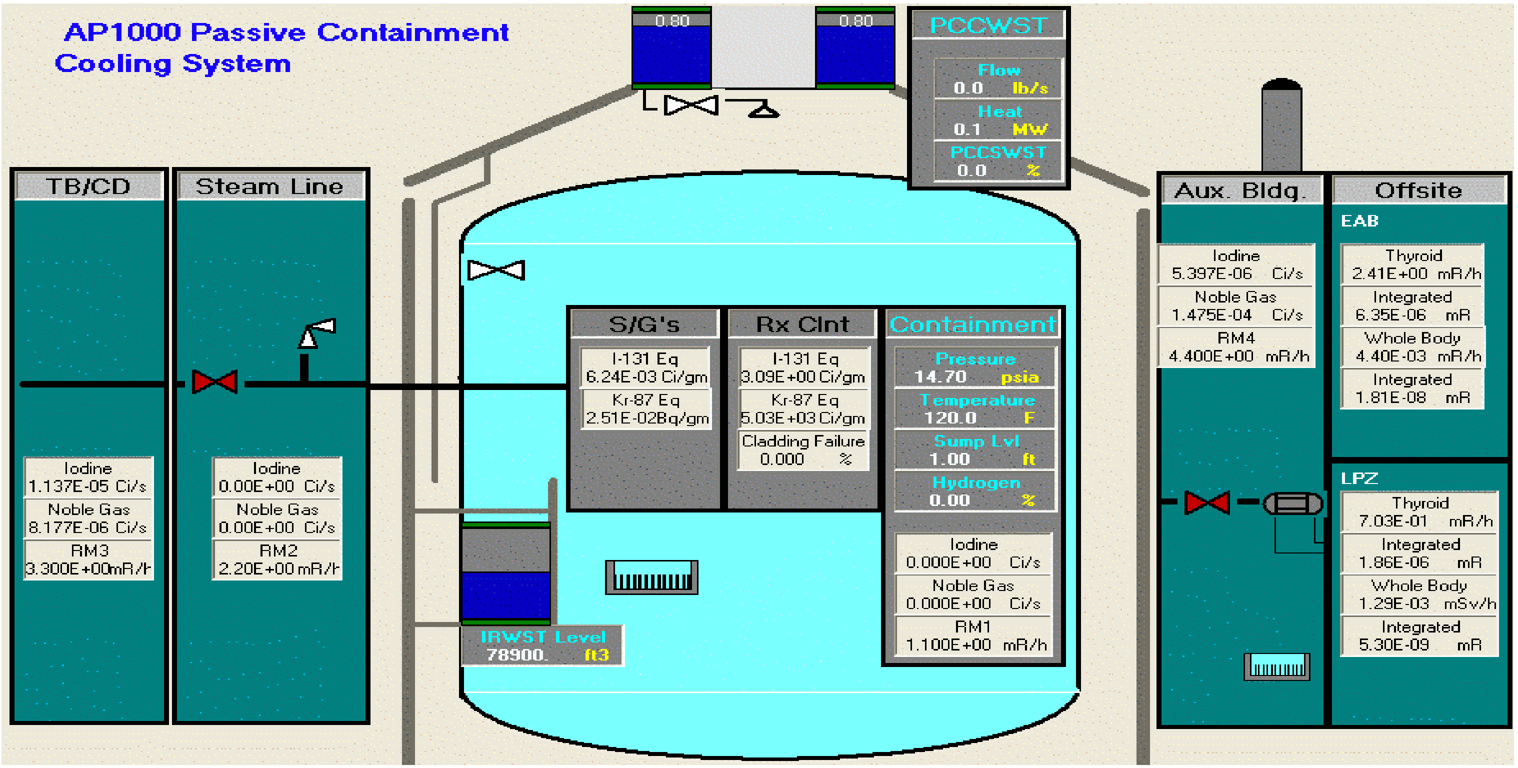
Representative Isotope Used to Characterize Group Decay

Group Number	Release Class	Representative Isotope	Half-life (days)	Daughter
1	Noble Gases	Kr-88	1.18E-01	Br-88
2	Halogens	I-131	8.04E+00	Te-131
3	Alkali Metals	Cs-134	7.53E+02	
4	Tellurium	Te-132	3.21E+00	Sb-132
5	Strontium	Sr-90	1.06E+04	Rb-90
6	Noble Metals	Co-60	1.93E+03	Fe-60
7	Lanthanides	Am-241	1.58E+05	Pu-241
8	Corium (Cerium)	Ce-143	1.38E+00	Pr-143
9	Barium	Ba-140	1.28E+01	Cs-140

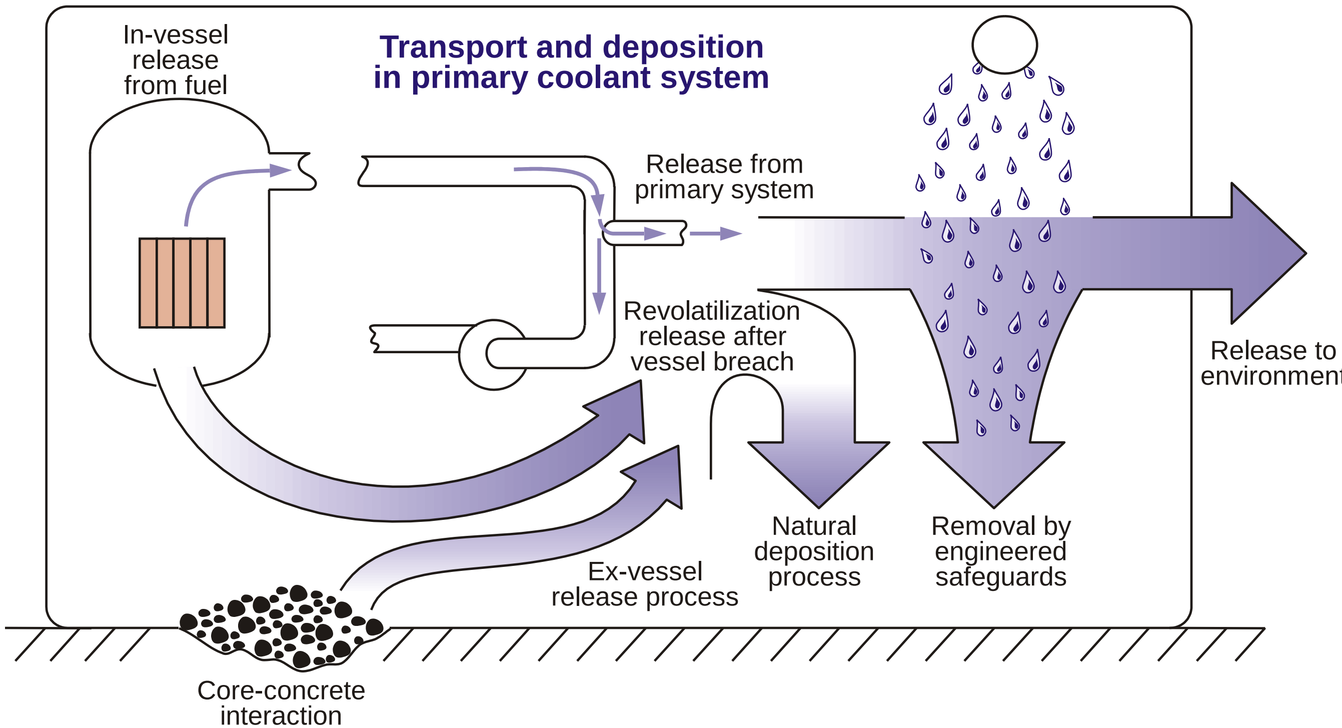
Sources and Losses Present in each Location along Release Path



AP1000 Radionuclide Containment



Several Factors Affect Release and Transport



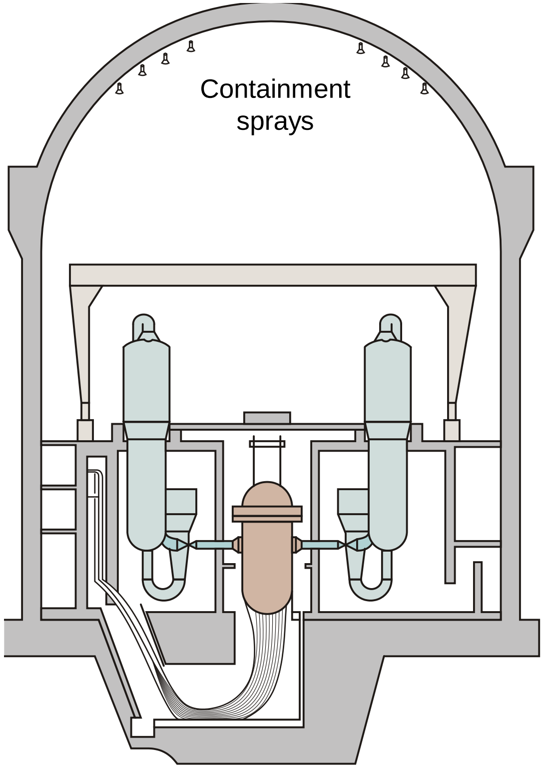
- **Sequence dependent**
  - Timing
  - Duration
  - Energy
  - Pressure
  - Chemical form
  - Physical form
  - Coolant chemistry
- **Plant design dependent**
  - Pathway (barriers, configuration, surface area, etc.)
  - Safety systems

Plant Features Significantly Reduce Release

Design Feature	Decontamination Factor[1]
Containment Sprays	100 to 1000
Ice Condensers	1 to 20 with ice present
Suppression pools	1 to 4000
Overlying water layers	1 to 4000

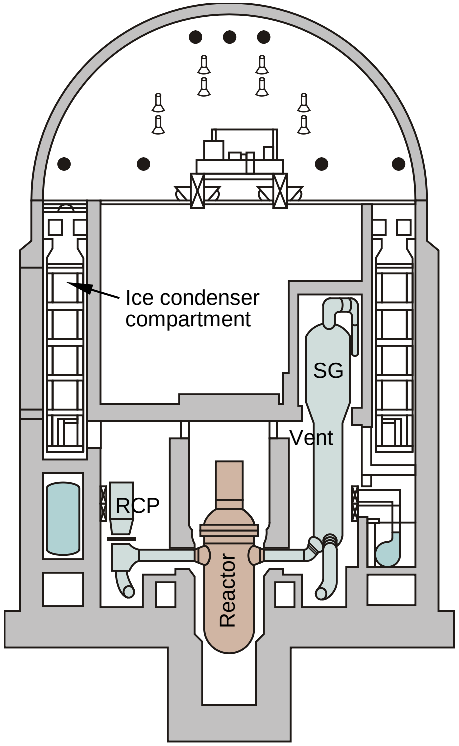
[1]Ratio of inlet to outlet concentrations.

Containment Sprays Rapidly Reduce Release



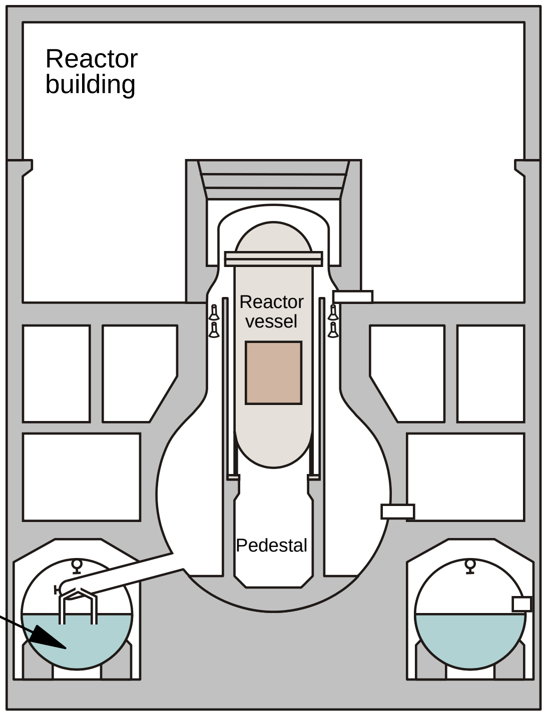
- Sprays reduce airborne concentration of aerosols and vapors in containment.
- Sprays may reduce airborne concentrations by order of magnitude in 15-20 minutes.

Ice Condensers Significantly Reduce Radioactive Release



- Retain radioactive aerosols and vapors.
- Typical decontamination factors of 1 to 20 with a median of 3.
- Decontamination factor sensitive to steam and hydrogen fraction of gas that flows through them.

BWR Suppression Pools Offer Significant Reduction



- Suppression pool water retains soluble vapors and aerosols.
- RSS (WASH-1400) assumed DF of 100 for subcooled pools and 1.0 for saturated pools.
- NUREG-1150 assumed DF between 1 and 4000 with a median value of 80.
- Suppression pool scrubbing primary reason that likelihood of early BWR fatalities is much lower in NUREG-1150.
- If suppression pool pH not maintained by chemical additives, lower pH may occur that promotes I_2 formation and vaporization (if heated) at later time periods.

Several Methods Available for Estimating Severe Accident Release

- Detailed methods
  - MELCOR
  - SCDAP/RELAP5/VICTORIA/CONTAIN
  - MAAP
- Less-detailed methods
  - TID
  - XSOR
  - Parametric Source Term (PST)
  - Alternate Approach (Revised Source Term, RST, or Alternate Source Term from NUREG-1465)

Source Terms Initially Based on TID-14844

- Based on a postulated core melt accident and 1962 understanding of fission product behavior.
- As codified in Reg. Guides 1.3 and 1.4, assumed source term consists of an instantaneous release of:
  - 100% of core inventory of noble gases
  - 50% of core inventory of iodine
    - half assumed to subsequently deposit on containment surfaces
    - 91% elemental, 5 % particulate, and 4% organic
- Assumed source term affected the site selection process and the design of engineered safety features, such as containment isolation valves, containment sprays, and filtration systems.

NUREG-1150 Release and Transport Estimated with XSOR Codes

- Developed for five NUREG-1150 plants
- Doesn't consider knowledge gained from severe accident research since1990.
- XSOR method decomposed source term into release fractions for various time periods and release barriers and quantified release fractions using expert opinion
  - Approach is time-consuming.
  - Approach isn’t reproducible.

NUREG-1465 Proposes More Realistic Source Term

- Developed more realistic source term for regulating future LWRs and for evaluating proposed changes to existing plants
  - Considers chemical and physical form
  - Provides safety and cost benefits
- Releases based on severe accident research and range of PWR and BWR STCP, MAAP, and MELCOR calculations
  - Comparisons with MELCOR comparisons suggest considerable margin between RST and best-estimate MELCOR predictions.
- Proposes time-dependent releases grouped into five phases:
  - DBA source term considers coolant, gap, and early-in-vessel releases
  - Severe accident source term considers coolant, gap, early in-vessel, ex-vessel, and late ex-vessel releases
- Implementation requires revised Part 20 dose methodology (TEDE criterion) and evaluate dose for accident's "worst two hour interval."
- Codified in Regulatory Guide 1.183

NUREG-1465 provides Time-dependent Releases

	PWR LOCA Release (fraction of core inventory)			
	Gap and Coolant	**Early In-vessel**	**Ex-Vessel**	**Late Ex-vessel**
Duration, hours	0.5	1.3	2.0	10.0
Noble gases	0.05	0.95	0	0
Halogens[1]	0.05	0.35	0.25	0.01
Alkali metals	0.05	0.25	0.35	0.01
Tellurium group	0	0.05	0.25	0.005
Barium, strontium	0	0.02	0.1	0
Noble Metals	0	0.0025	0.0025	0
Lanthanides	0	0.0002	0.005	0
Cerium group	0	0.0005	0.005	0

[1] If coolant pH greater than or equal to 7, then 95% particulate, ~5% elemental and ~0.15% organic.

Pilot plant applications demonstrate that RST reduces regulatory requirements and enhances safety

- Time-dependent source term allows:
  - delayed automatic isolation function for containment isolation valves
  - increased allowable containment and/or penetration leakage rates
- Realistic iodine chemical species allows:
  - relaxation of charcoal filtration system requirements
  - relaxation of control room habitability requirements
  - requirements for post-accident pH control of iodine particulates dissolved in water (to prevent elemental iodine formation).

SOARCA

- NRC-sponsored State of the Art Reactor Consequences Analysis
  - Realistic estimates of the potential public health effects from a severe accident
    - Health effects from previous accidents often overstated in early phases
    - Propensity to apply excessive conservatism in analyses
  - Apply understanding developed from relatively recent research programs to better assess reactor accident consequences
    - Better source term estimates
    - Credit accident management
    - Credit plant features
    - Better software and computer systems

Study Questions

- What contributes to and reduces radioactivity release during a severe accident?
- What characteristics are important in assessing radionuclide transport?
- Name several factors (and plant features) affecting radioactivity release and transport.
- Name several methods available for estimating severe accident releases.
- Define and describe differences between the RST and the TID source term.

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Accident Progression Analysis (P-300)

9. PRA Integration and Quantification

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Idaho National Laboratory

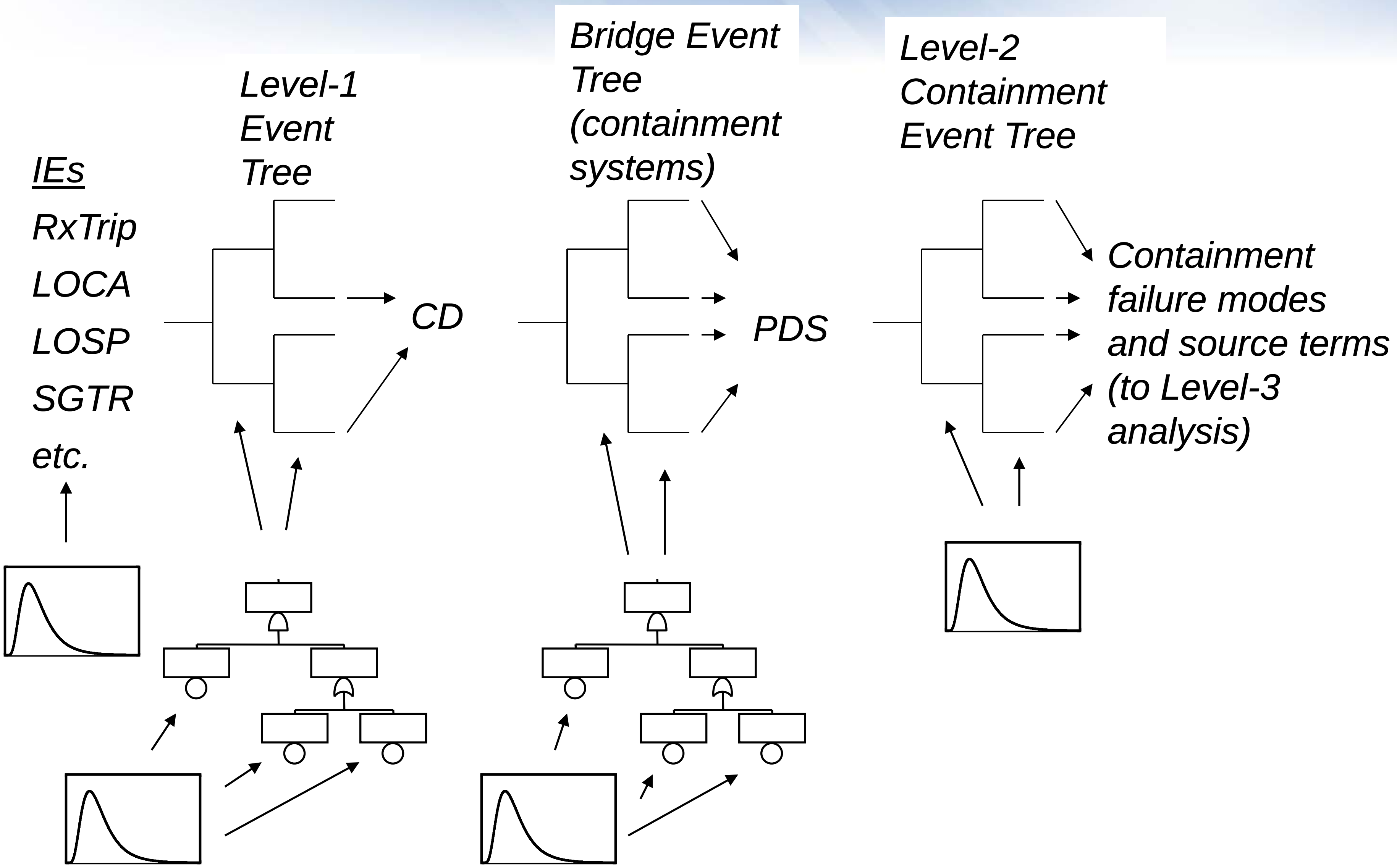
Session Objectives

- To understand the details of how the different phases of a PRA are linked to each other
  - Level-1 output = Core Damage
    - Segregation of CD sequences into Plant Damage States
  - PDSs used as input (initiator) to Level-2
  - Propagation of uncertainties

Outline

- Integration of Level-1 and Level-2
- Uncertainty
- Level-2 Results

Level-1/2 PRA Integration



Level-1/Level-2 Analysis Approach

- Assignment of core damage (CD) sequences into appropriate plant damage state (PDS) bins
- Assessment of challenges associated with each PDS bin (typically using computer codes)
- Characterization of the containment's capacity to withstand the identified challenges (i.e., fragility)
- Combining the uncertainties associated with the previous two analyses to estimate probability of containment failure (for a given PDS)
- Combining the uncertainties associated with CD frequency with those associated with conditional containment failure probabilities to estimate containment failure frequency

Level-1 CD Sequences Mapped Into PDSs

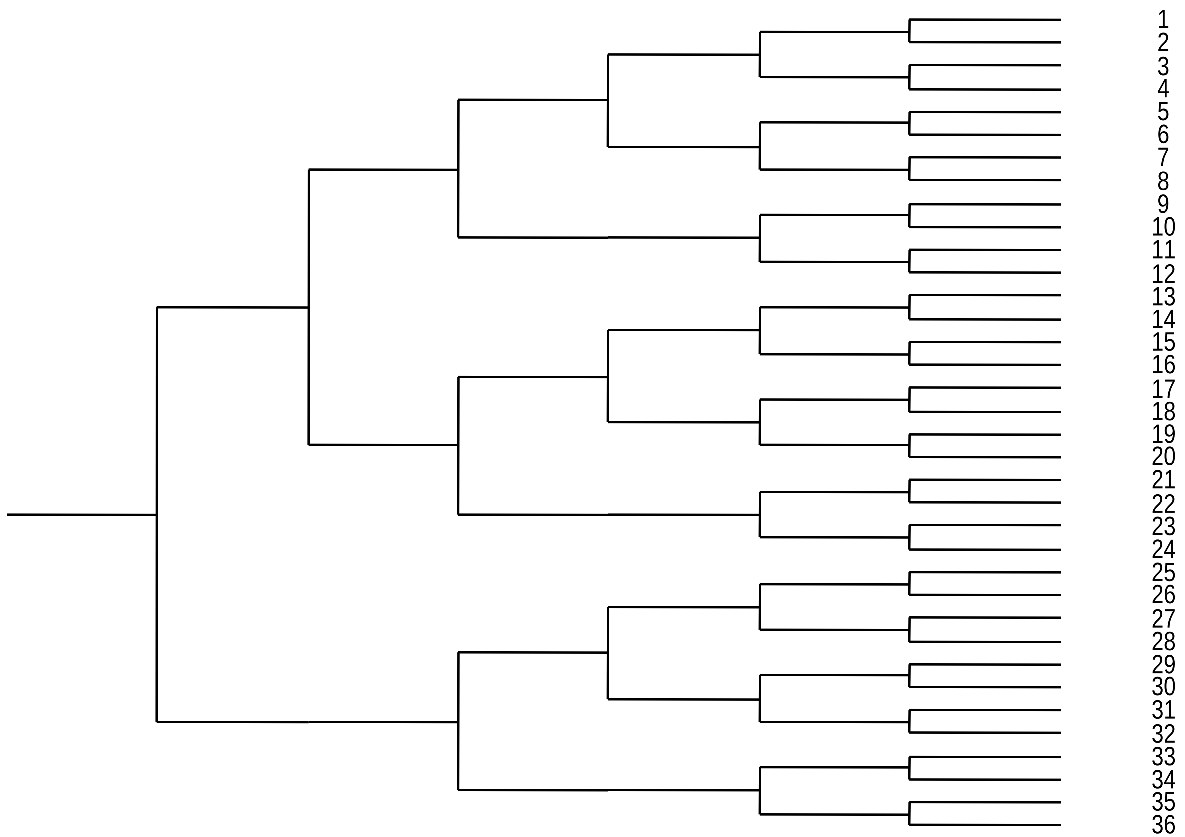
- Core Damage vs. no CD, does not provide enough information for Level-2 analysis
  - CD sequences extended to include systems and events that mitigate consequences of core damage
    - Containment spray and cooling systems
    - Need to ensure dependencies accounted for
      - SBO failing ECCS would also fail containment systems
- PDS are more detailed description of core damage sequence

Bridge Event Tree Maps CD Into PDS

- Sometimes called “binning” of CD sequences
- Bridge Tree typically straightforward extension/expansion of Level-1 event trees
  - Extends consideration beyond core damage
  - Determines status of containment systems
- Every core damage sequence propagated through bridge tree

Example Bridge (or Binning) Event Tree

Core Damage	Emergency Coolant Injection	Emergency Coolant Recirculation	Containment Spray Injection	Containment Spray Recirculation	Cooling to RCP Seals	Auxiliary Feedwater to SG		
CD	ECI	ECR	CSI	CSR	RCP-SL	AFW	#	PDS



BET-01 - Bridge Event Tree

2004/04/29 Page 1

Each CD Cut Set Unique

- Each cut set represents a unique set of events (e.g., component failures, human actions) that is expected to lead to CD (e.g., UTAF)
- Individual cut sets generated from the same CD sequence can produce different impacts on containment response
  - e.g., LOCA & ECCS failure: ECCS can fail from different causes
    - ECCS components can fail (implying containment systems are nominally operable)
    - Loss of all ac power can fail ECCS (implying containment systems are NOT operable)

Each CD Cut Set Assigned to PDS

- To accommodate different impacts on Level-2 analysis, each CD cut set explicitly mapped into a PDS (sometimes referred to as binning)
- Two approaches to binning Level-1 cut sets into PDSs
  - Two step process (often performed using “If-Then” rules)

    1 - assign PDS vector identifier to each CS

    2 - map CS into PDS based on best match of vector
  - One step process (often manually performed)
    - Directly bin each CS into a PDS (this process does not necessarily need the vector framework)

Simple Binning Example

- PWR core damage sequence
  - Small LOCA with failure of ECCS (ignore other issues for sake of simplicity)
    - Cut set #1: Small LOCA with ECCS pump fails
    - Cut set #2: Small LOCA with loss of all AC power

$$S_2D = IE\text{-}S_2 * ECCS\text{-}Pump\text{-}F + IE\text{-}S_2 * LOSP * EAC\text{-}F.$$

Simple PDS Scheme for PWR (Status of …)

1	***RCS integrity at start of CD***	***I – Intact*** ***S – Small hole***
2	***ECCS***	***A – Available*** ***U – Unavailable***
3	***CHR***	***A – Available*** ***U – Unavailable***
4	***AC Power***	***A – Available*** ***U – Unavailable***
5	***RWST***	***A – Available for injection*** ***I – Injected into containment*** ***U – Unavailable for injection***
6	***Heat Removal from S/G***	***A – Available*** ***U – Unavailable***
7	***RCP seal cooling***	***A – Available*** ***U – Unavailable***
8	***Containment Fan Coolers***	***A – Available*** ***U – Unavailable***

Different PDS Vectors for CS#1 and CS#2

	1 RCS	2 ECCS	3 CHR	4 AC	5 RWST	6 S/G	7 RCP seals	8 Fans
CS#1	S	U	A	A	A	A	U	A
CS#2	S	U	U	U	A	U*	U	U

- *Frequency from cut sets #1 and #2, even though from the same core damage accident sequence, would likely be mapped into different Plant Damage States*
- *Mapping of core damage sequences into PDS not necessarily a one-to-one process*

Each CS-Vector Then Matched to Most Appropriate PDS-Vector

- Seldom is “fit” perfect
  – Only a limited number of PDS (~10-20)
- List of available PDSs dictated by available T/H resources
  – Typically, each PDS has been analyzed using severe accident code (e.g., CONTAIN, MELCOR, MAAP)
  – Code results needed to realistically model the accident progression of each PDS
  – Strive for complete coverage of the spectrum of core damage sequences with significant contributions to total core damage frequency
    - However might include low frequency sequences that result in high consequences (containment bypass)

Each PDS Frequency Calculated (Analogous to a CDF Calculation)

- Uncertainty analysis (i.e., Monte Carlo or Latin Hypercube) generates probability histogram for each PDS
- Each PDS then used as input to (i.e., serves as the initiating event) the CET
  - CET can be manually tailored for each PDS
    - Each PDS associated with a unique CET
      - Note that vector framework NOT necessary
  - Single "general-purpose" CET can be modified during processing
    - Incorporates various "If-Then" logic rules
      - Vector framework not absolutely necessary but very useful

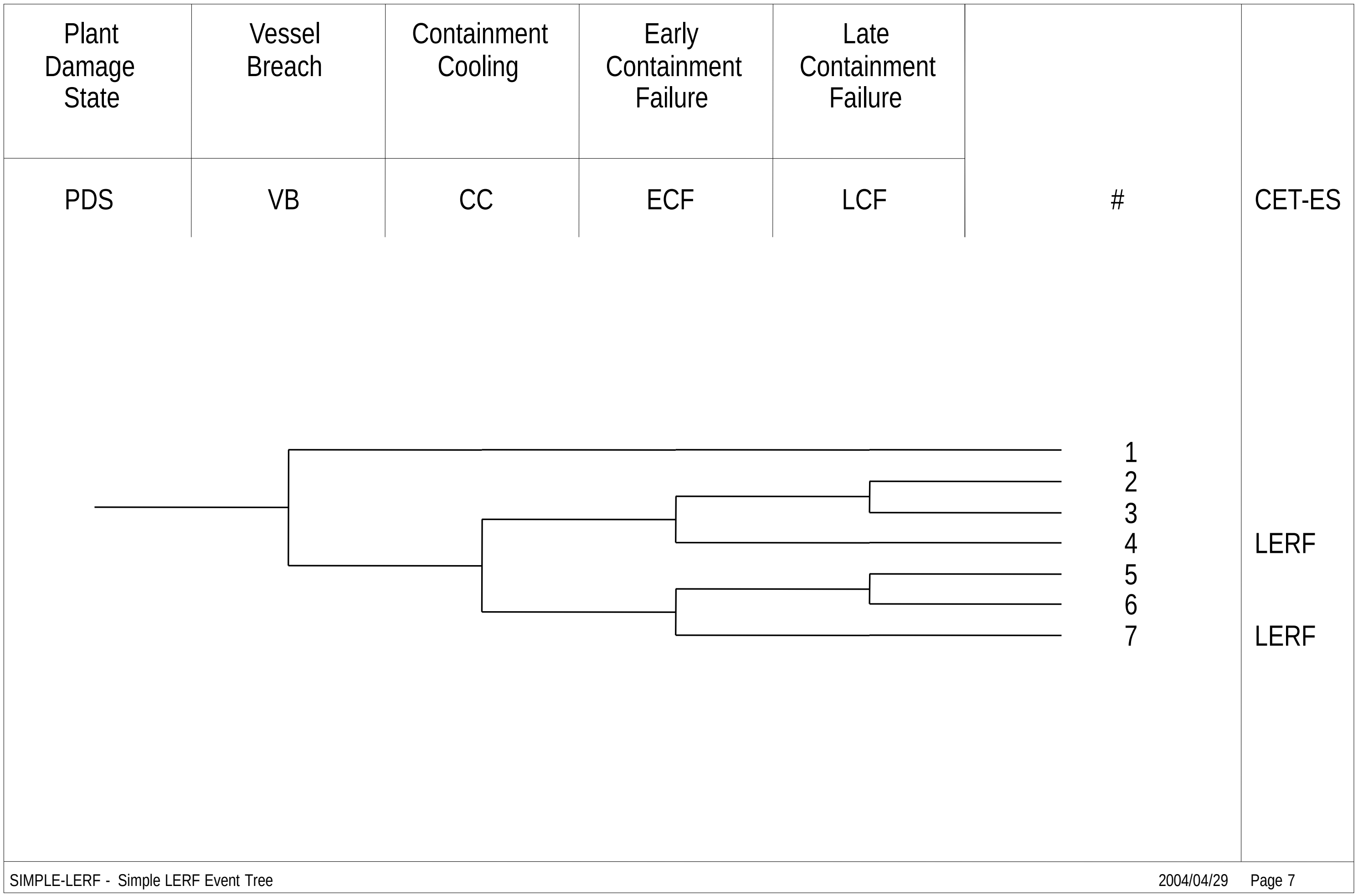
PDSs Are Level-2 “Initiating Events”

- Each PDS (or PDS group) used as Level-2 IE
  - Represent unique characteristics of core damage event
    - Influence containment challenges
    - Affect potential source term
- PDS contains relevant information needed to assess containment performance

Accident Progression Quantified Different Ways

- Depends on level of detail in CET and in "initiating event" (i.e., plant damage state vector)
- Typically use conditional split-fractions/distributions for CET branch points
  - Effectively "If-Then" statements
- Sometimes branch probability is a weighted average of different accident sequences
  - Accounts for dependencies
  - Requires detailed analysis of Level-1 sequences
    - E.g., what portion of ECCS failures are caused by SBO (implies H2 igniters won't work)

Simple LERF Quantification Example



Split Fractions for LERF-ET

- S2D1 – Small LOCA with early failure of all injection
  - Freq(S2D1) = 1E-4/yr
  - Pr(VB|S2D1) = 0.5
  - Pr(CC|S2D1) = 0.2
  - Pr(ECF|VB,CC) = 0.5
  - Pr(ECF|VB,/CC) = 0.1

Quantify LERF

What is conditional probability of LER given S2D1?

CET Output Organized

- If analysis is limited to Level-2, output usually formatted for ease of presenting results on containment failure
- If supporting Level-3, then need detailed source term information
- Output also needs to adequately represent uncertainty in the analysis

Uncertainty

- Uncertainty important in all PRA
  - Level-2 results reflect uncertainty in Level-1 results and CET uncertainties
  - Uncertainty expressed as a probability density function on the containment failure frequency (or source term release frequency)
    - “Probability of Frequency” characterization
    - Implies Bayesian techniques and interpretation

There are Different Interpretations of Probability

- Classical
  - Requires a statistical basis
  - Generates confidence intervals only (not probability distributions)
- Bayesian
  - Implies a degree of belief
    - able to accommodate sparse data and engineering judgement
  - Needed to produce and propagate probability distributions in a PRA (i.e., all PRAs employ Bayesian techniques and interpretations)

Uncertainty Often Classified by Type

- Aleatory - Stochastic, random or tolerance uncertainty
  - A product of the assumed model
    - i.e., a binomial or Poisson process
  - Can also include variability in boundary conditions
- Epistemic - State of knowledge, subjective or confidence uncertainty
  - A produced by a lack of data
    - Similar to a classical statistical confidence
    - Bayesian interpretation is the degree of belief

Aleatory Uncertainties

- Measure of randomness in process
  - e.g., coin flip - sometimes heads, sometimes tails
    - Note that this “randomness” could also be interpreted as variability in the boundary conditions of each coin flip
- Distribution is result of assumptions about the process (i.e., variability accommodated using the random process premise)
  - Additional data does not necessarily reduce aleatory uncertainty
- Distribution is a function of parameter values (i.e., λ’s), which are usually uncertain

Epistemic Uncertainties

- Uncertainty in model parameters (i.e., uncertainty in our estimate of λ)
- Distribution reflects data, relevant model predictions, engineering judgment
- As more data is accumulated, the uncertainty narrows
- Typically generated using Bayesian methods (covered in Probability and Statistics for PRA course)
  - e.g., Bayesian update process

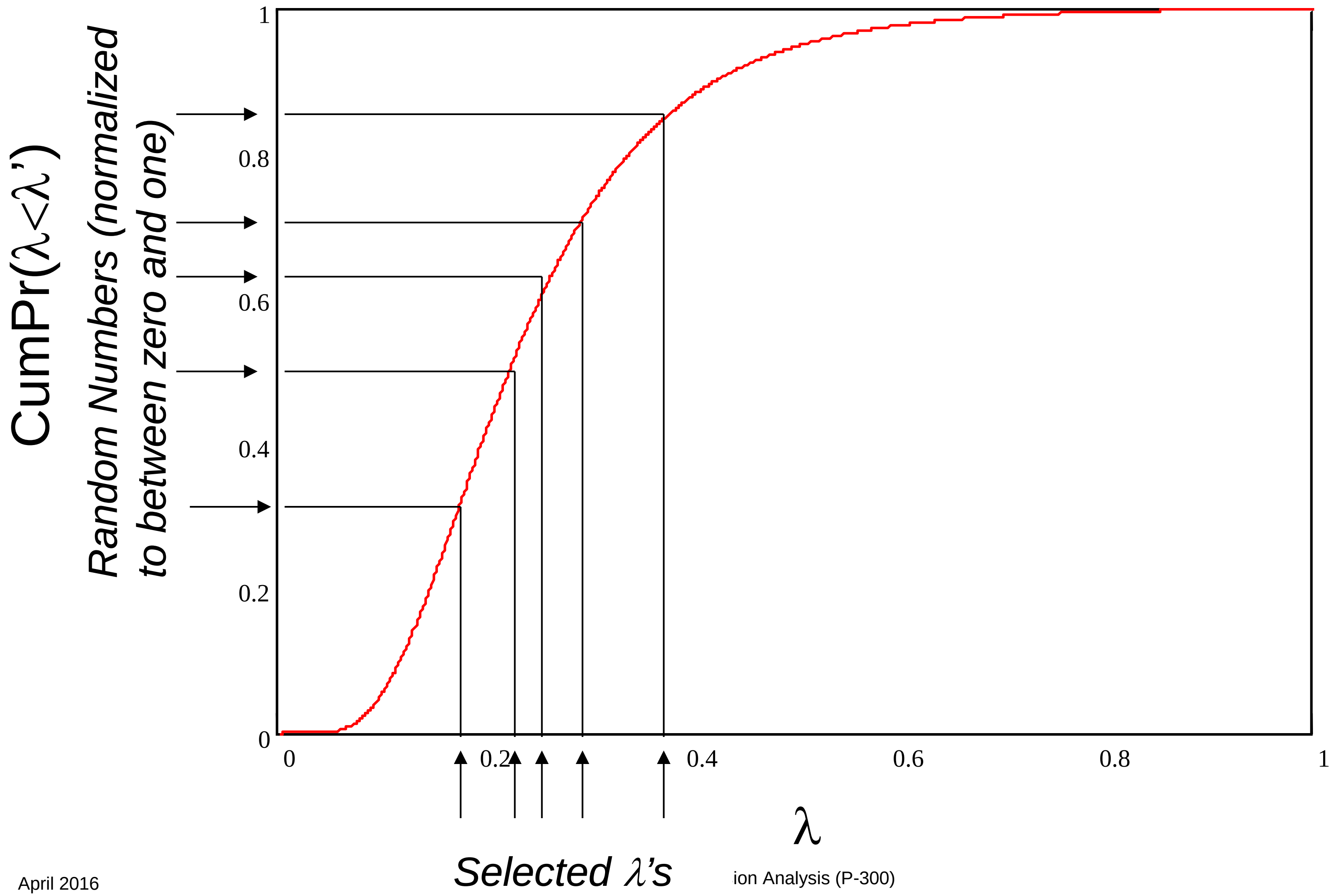
Uncertainty Needs to be Propagated Through Entire PRA

- Beginning with uncertainty on Level-1 initiating event frequencies
- Uncertainty in different input parameters represented in different ways
  - lognormal, beta, gamma, uniform distributions
- Different types and sources of uncertainty need to be accounted for in the PRA results
  - Be it core damage frequency, containment failure frequency or health risk

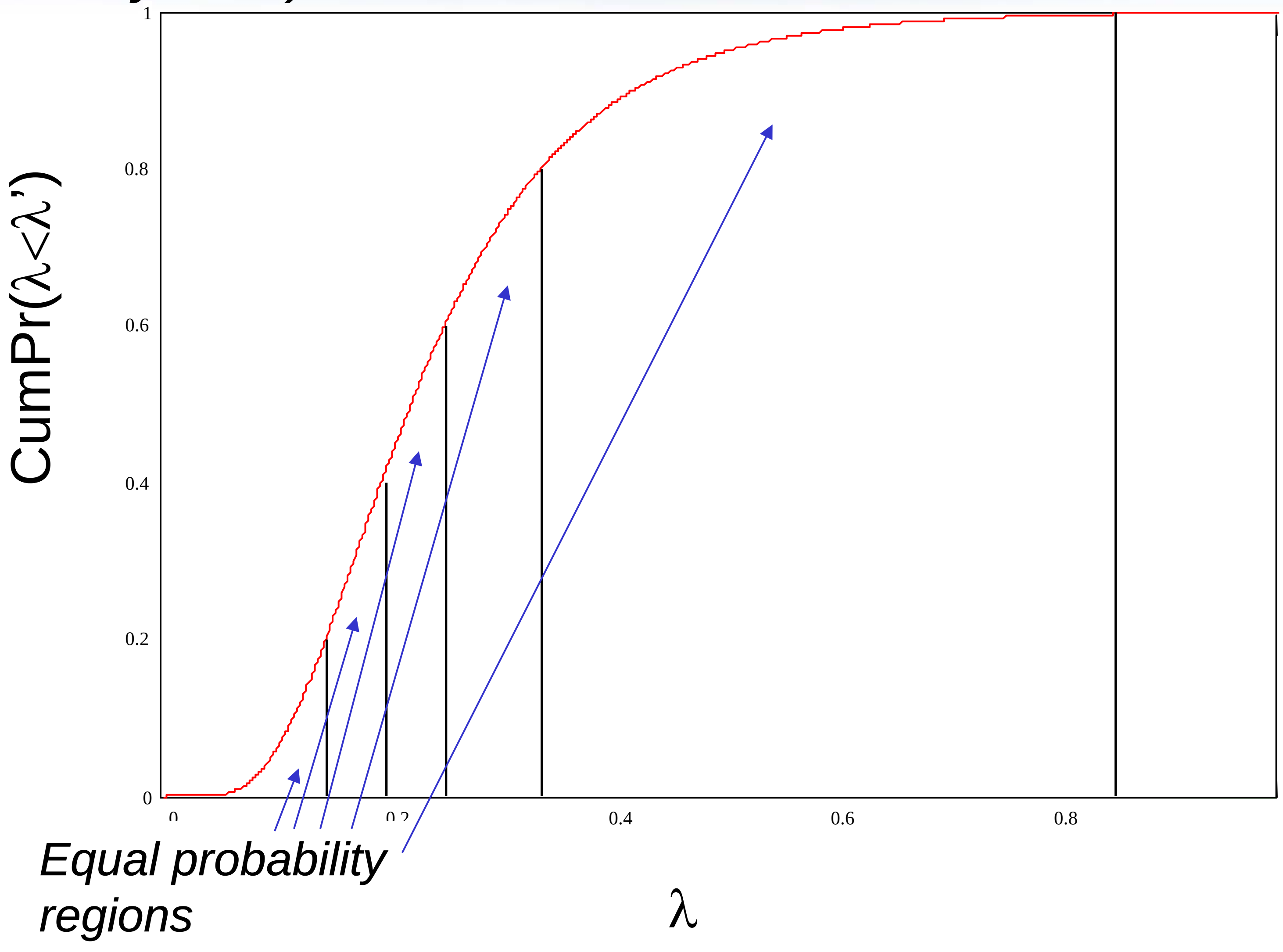
Simulation Techniques Used to Quantify Models

- Analytical methods simply not feasible
- Monte Carlo or Latin Hypercube are currently the only practical approaches to propagating uncertainty
  - Select random values from input parameter distributions, quantify model, repeat many times
    - repeating mathematical “experiment” over and over produces a frequency histogram on the output
- Quantification done step-wise
  - Distributions on intermediate results (e.g., CDF or PDS) are then inputs to subsequent steps

Example Monte Carlo Sampling (5 Samples) on input parameter λ

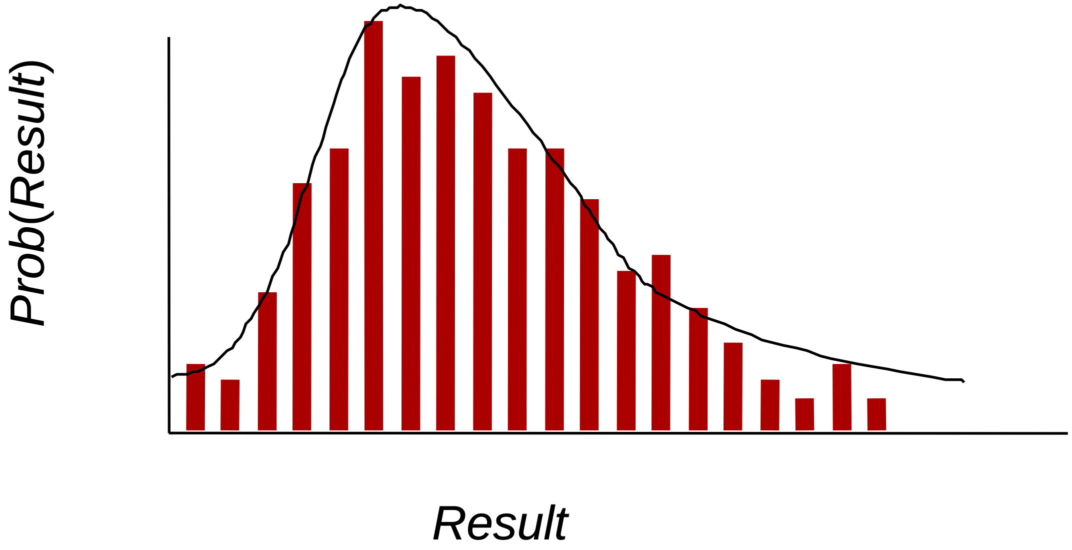


Latin Hypercube Sampling (one λ selected from each equal-probability area)



Propagation of Uncertainties

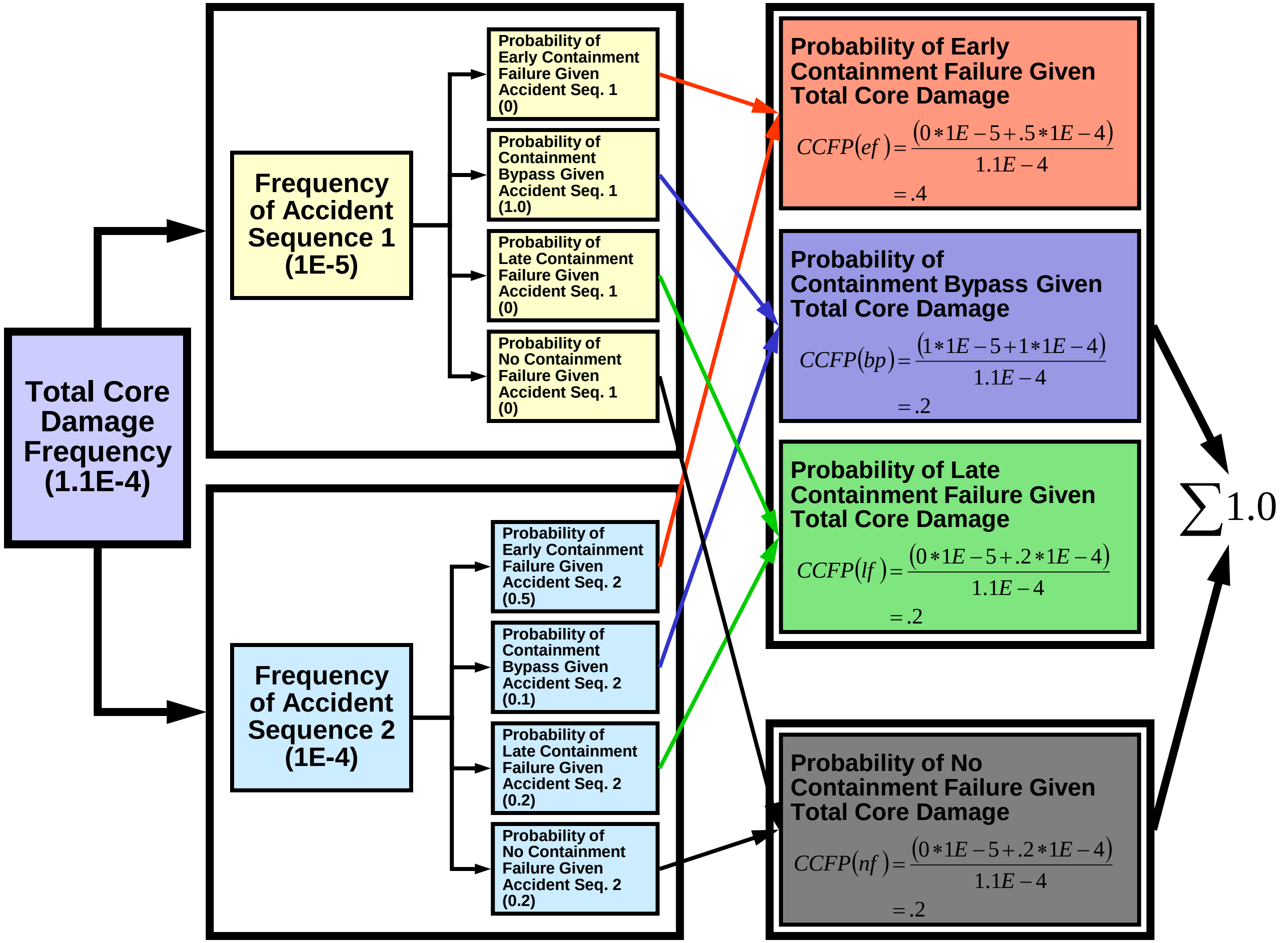
- Simulation Process (either Monte Carlo or Latin Hypercube)
  - Generates frequency histogram for Result = f(X, Y) by sampling from distributions for X and Y re-calculating result for each of simulation samples



Results Can Take Many Forms

- Level-1 Results
  - Core Damage Frequency or Plant Damage State Frequencies
- Level-2 Results
  - Containment Failure Frequency, Conditional Containment Failure Probability, Large Early Release Frequency
- Level-3 Results
  - Various health and financial consequence risk measures

CET Results for Each Accident Sequence Combined and Normalized



Two Measures Typically Cited for Assessing Containment Performance

Conditional Containment Failure Probability $= \text{CCFP} = \sum_{i=1}^{n} \frac{S_i}{CDF} C_i$

Containment Failure Frequency $= \text{CFF} = \sum_{i=1}^{n} S_i C_i$

S_i = *frequency for accident sequence, i*
C_i = *containment conditional failure probability given accident sequence , i*
n = *total number of accident sequences*

NUREG-1150 Presentation Bins

- Vessel Breach (VB), early (during core damage) containment failure (CF)
- VB, alpha, early CF (at VB)
- VB > 200 psi, early CF (at VB)
- VB < 200 psi, early CF (at VB)
- VB, late CF
- VB, basemat melt-thru, very late CF
- Bypass
- VB, no CF
- No VB, early CF (during core damage)
- No VB, no CF

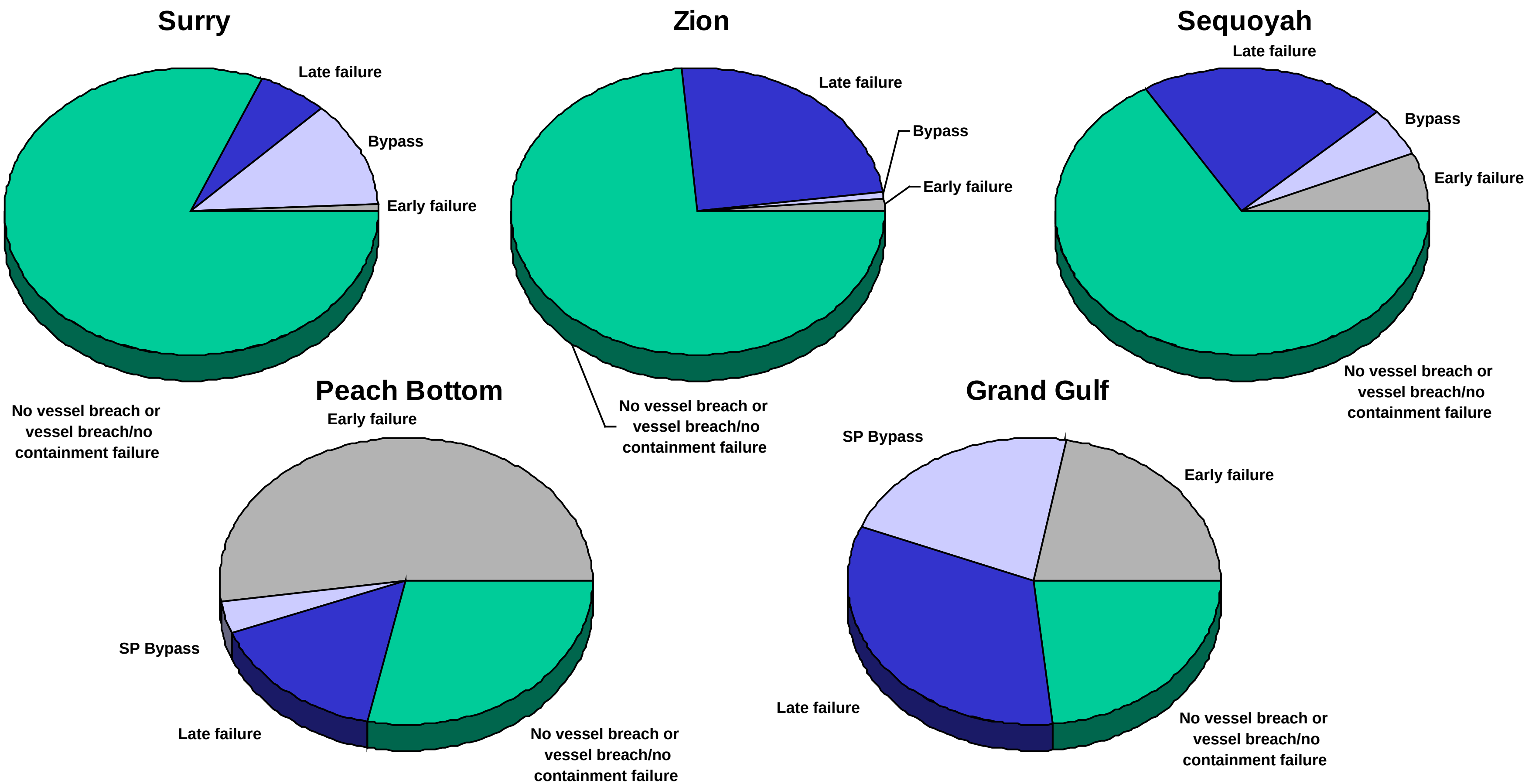
NUREG-1150 Sequoyah Accident Progression Bin Results for Summary PDSs

ACCIDENT PROGRESSION BIN	PLANT DAMAGE STATE (Mean Core Damage Frequency)					
	LOSP (1.38E−05)	ATWS (2.07E−06)	Transients (2.32E−06)	LOCAs (3.52E−05)	Bypass (2.39E−06)	Frequency Weighted Average (5.58E−05)
VB, early CF (during CD)	0.014	0.003		0.002		0.005
VB, alpha, early CF (at VB)	0.002	0.003		0.002		0.002
VB > 200 psi, early CF (at VB)	0.064	0.023	0.014	0.031		0.035
VB < 200 psi, early CF (at VB)	0.054	0.020	0.004	0.014		0.023
VB, late CF	0.153	0.001		0.001		0.038
VB, BMT, very late CF	0.065	0.151	0.039	0.260		0.171
Bypass	0.001	0.134	0.006		0.996	0.056
VB, No CF	0.200	0.471	0.137	0.301		0.269
No VB, early CF (during CD)	0.038	0.001	0.005	0.002		0.011
No VB	0.384	0.171	0.785	0.367		0.371

BMT = Basemat Meltthrough
CF = Containment Failure
VB = Vessel Breach
CD = Core Degradation

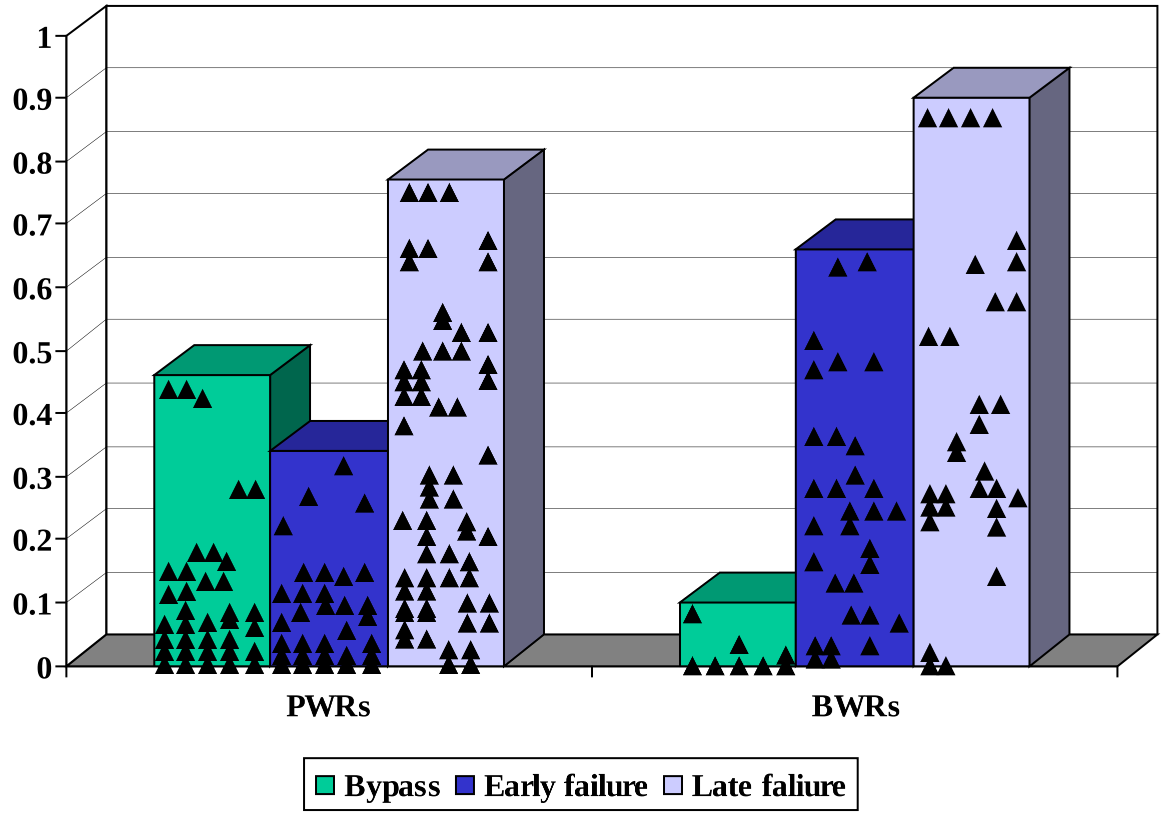
Sequoyah

NUREG-1150 Results Indicate BWR Early Containment Failures More Likely



NUREG-1150 relative probability of containment failure modes from internal events

IPEs Suggest that Late Failures Dominate in BWRs and PWRs



General Insights From Containment Response Analyses

- Large volumes of PWR containments are less likely to experience early structural failures than the smaller BWR pressure suppression containments.
- Probability of bypass is generally higher in PWRs because of higher operating pressures and use of steam generators
- Specific containment features as well as differing assumptions regarding containment loads lead to observed variability.

Session Review

- How are the Level-1 and Level-2 portions of a PRA linked?
- What are the two types of uncertainty?
- How is uncertainty propagated through the analysis?

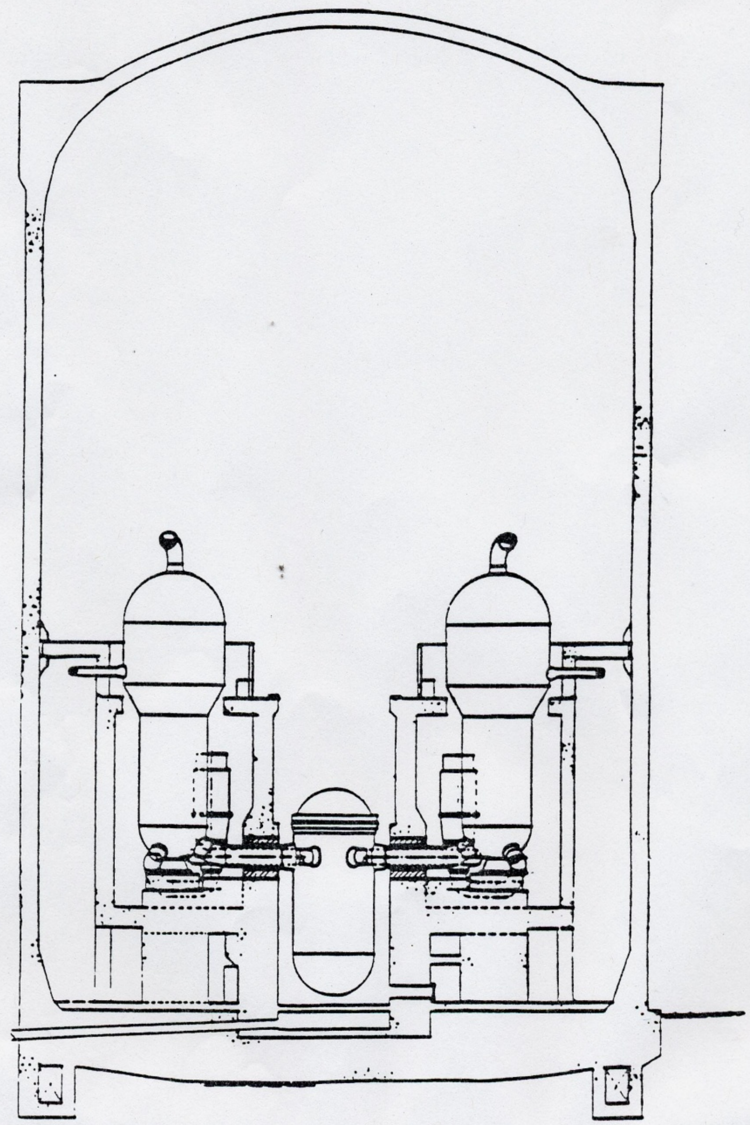
Accident Progression Analysis (P-300)

Example: Palisades IPE (Jan. 1993)

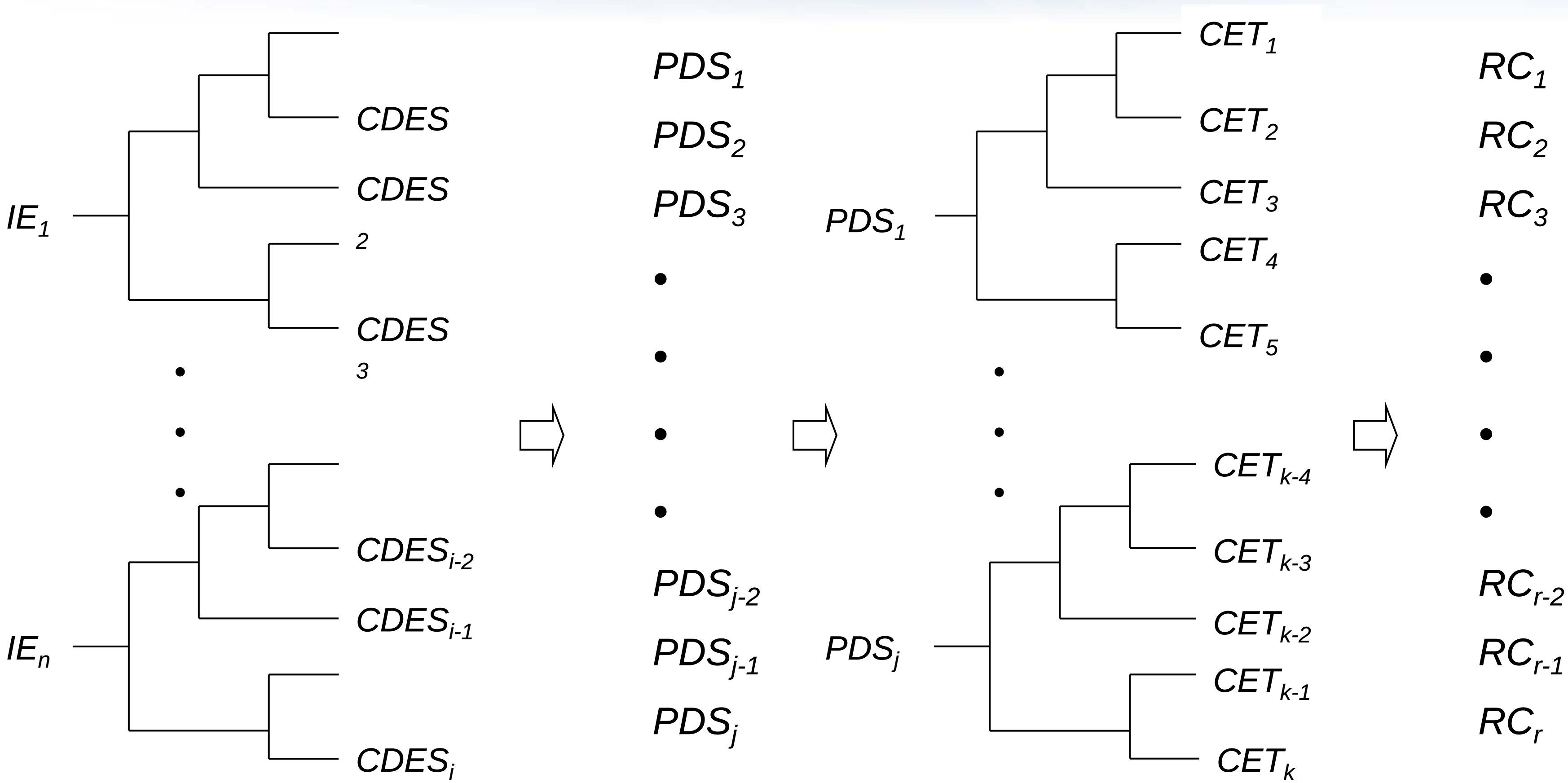
Example: Palisades IPE

- Two-loop Combustion Engineering (CE) 2530 MWt (780 MWe) PWR
  - Two steam generators (SGs)
  - Four reactor coolant pumps (RCPs)
  - Two power-operated relief valves (PORVs)
- Large dry pre-stressed concrete containment
  - Reinforced concrete cylinder (post-tensioned in three directions) with 1/4-in. carbon steel liner
  - Design basis capacity is 55 psig at 2830F
- Complete Level-2 PRA submitted as Individual Plant Examination (IPE) to NRC on January 29, 1993.

PALISADES NUCLEAR PLANT
GENERAL CONTAINMENT ARRANGEMENT



Palisades Level-2 PRA Analysis Process



$$\sum_{m=1}^{i} \Pr(CDES_m) = CDF$$

$$\sum_{n=1}^{j} \Pr(PDS_n) = CDF$$

$$\sum_{p=1}^{k} \Pr(CET_p) = CDF$$

$$\sum_{q=1}^{r} \Pr(RC_q) = CDF$$

Palisades IPE Used PDS Bridge Tree to Map CD Sequences Into PDS

- CET developed first, PDS-BT then developed to satisfy information needs of CET
- Total CDF conserved in binning to PDS's
  - i.e., Total CDF = $\Sigma_{m=1,i}$ PDS_m
- PDS-BT incorporated as an extension of the Level-1 core damage event tree
- PDS-BT primarily used as a sorting mechanism
  - Most branch choices dictated by previous events
  - Presence of water in containment was exception (PDS-BT top event SII)
    - In some CD sequences, operation of ECCS does not guarantee water in containment (i.e., ISLOCA, SGTR)

Palisades IPE PDS Characteristics

#	Characteristic	Description
1	Initiator	*Affects potential for containment bypass, fission product retention by the RCS, pressure of the RCS at vessel failure, etc.*
2	CD Time	*Time of fission product release and amount of warning time for offsite protective actions.*
3	Secondary Cooling	*Can affect late revaporization of fission products retained in the RCS*
4	Pressurizer PORV	*Affects RCS pressure during the core relocation/vessel failure phase of a CD sequence*
5	Containment Systems	*Affect long term integrity of containment. Can affect debris coolability, flammable gas behavior, fission product releases*

Palisades IPE PDS Character #1 (IE)

A1 - Large LOCA (d > 18 in.)

A2 - Medium LOCA (2 in. < d < 18 in.)

B - Small LOCA (1/2 in. < d < 2 in.)

C - Interfacing System LOCA

D - SGTR

T - Transient

Z - ATWS

Palisades IPE PDS Char. #'s 2, 3 & 4

2 Core Damage Timing
- E - Early CD
- L - Late CD

3 Secondary Cooling
- G - Secondary Cooling Available
- J - No Secondary Cooling

4 Pressurizer PORV
- M - PORV Available
- N - PORV Unavailable

Palisades IPE PDS Char. #5 (Cont. Sys.)

P - Containment sprays and air coolers available

Q - Cont. sprays avail. and cont. air coolers NOT avail.

R - Only cont. air coolers avail., RWST contents in cont.

S - Only cont. air coolers avail., RWST contents NOT in cont.

V - No cont. systems avail., RWST contents in cont.

W - No cont. systems avail., RWST contents NOT in cont.

X - Late (post VB) operation of only HPSI/LPSI

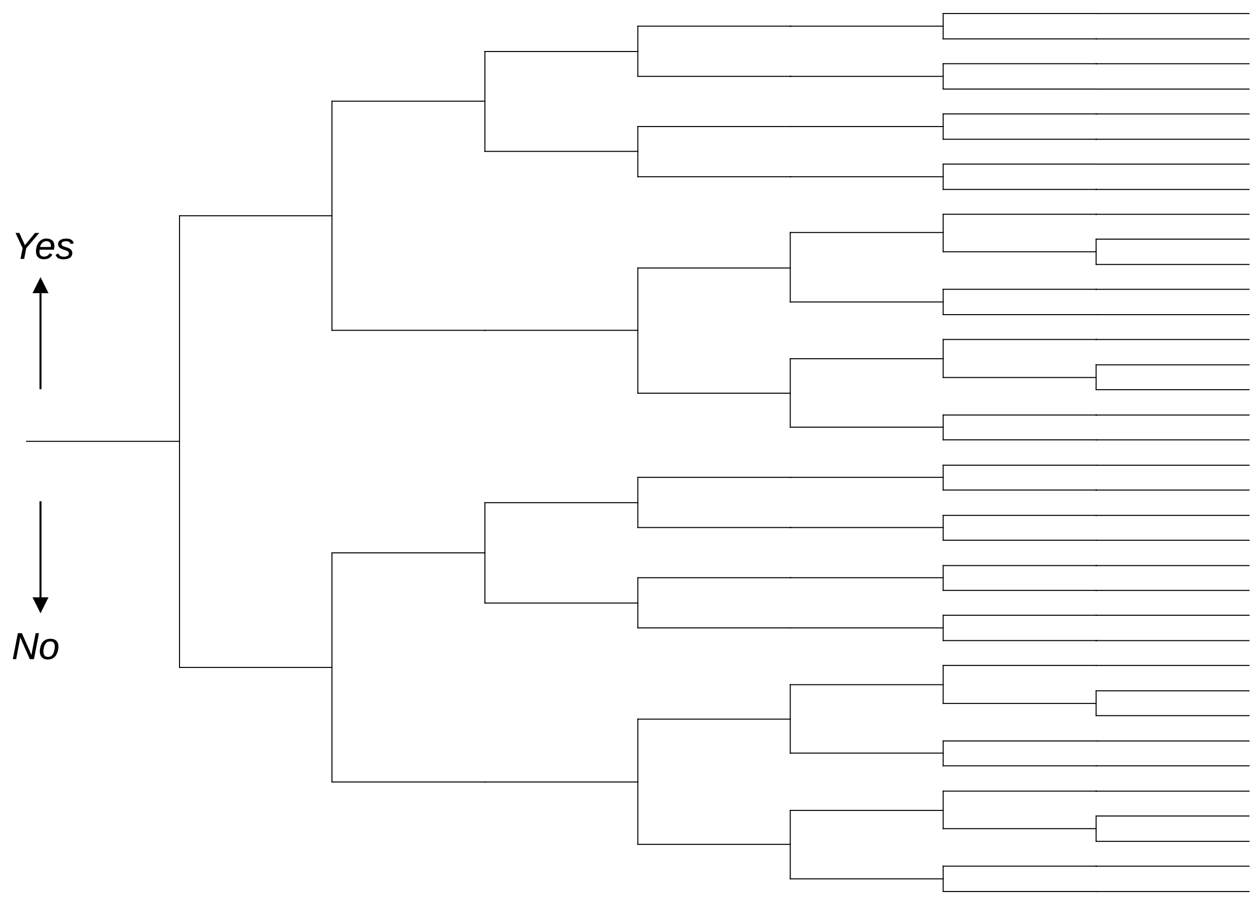
Palisades PDS's Grouped to Reduce Number of CET Analyses

- Initial development resulted in 392 possible PDS's
- IPE judged preemptive protective actions were unlikely
  - All core damage timing assumed to be early
  - Reduced number of possible PDS's to 196
- Illogical PDS's were also removed from the list (reduced number to 168)
- Truncation (at 1E-9) during the CD/PDS quantification further reduced the list to 70 PDS's
  - Still too many PDS's
- PDS's collapsed on PORV availability
  - For each remaining PDS PORV availability calculated by taking a weighted average (53 PDS's left)

Palisades IPE PDS Bridge Tree Top Events

Heading	Description
2ND	AFW available to both steam generators
CSI	Containment spray system available in injection mode
CSR	Containment spray system available in recirculation mode
PRV	One pressurizer PORV available to depressurize RCS
SII	RWST water is in containment
FC	Containment air coolers available
SIL	Safety injection available after vessel failure

Core Damage Sequence	AFW available to both S/Gs	Containment spray injection	Containment Spray Recirculation	PORV available	RWST water in containment	Containment fan coolers available	Safety injection after VB		
CD	2ND	CSI	CSR	PRV	SII	FC	SIL	#	PDS



#	PDS
1	_EGMP
2	_EGMQ
3	_EGNP
4	_EGNQ
5	_EGMR
6	_EGMV
7	_EGNR
8	_EGNV
9	_EGMR
10	_EGMX
11	_EGMV
12	_EGMS
13	_EGMW
14	_EGNR
15	_EGNX
16	_EGNV
17	_EGNS
18	_EGNW
19	_EJMP
20	_EJMQ
21	_EJNP
22	_EJNQ
23	_EJMR
24	_EJMV
25	_EJNR
26	_EJNV
27	_EJMR
28	_EJMX
29	_EJMV
30	_EJMS
31	_EJMW
32	_EJNR
33	_EJNX
34	_EJNV
35	_EJNS
36	_EJNW

PALISADES-IPE-BT - Palisades IPE (Jan. 1993) Plant Damage State Bridge Tree 2001/06/13 Page 3

Top 18 PDSs from Palisades IPE

PDS	Freq	PDS	Freq
BEGP	1.11E-5	BEGS	7.22E-7
TEJP	9.40E-6	TEJQ	3.70E-7
TEJW	9.02E-6	CEJW	3.70E-7
TEJV	6.89E-6	A2EGR	2.42E-7
ZEGP	4.20E-6	BEGV	2.33E-7
BEGR	2.97E-6	TEJS	3.32E-7
TEJR	2.42E-6	DEJR	1.10E-7
DEJP	1.33E-6	A2EGP	1.00E-7
DEJS	1.04E-6	A1EGR	9.72E-8

CET Top Event Quantification Focus on Probability of Containment Failure

- Need to know how strong is the containment structure
- Need to identify the likely failure location
- Need to identify the size of any potential containment failure

Palisades IPE Containment Structural Response and Failure Characterization

- Purpose
  - To establish best estimate probabilistic measure of containment fragility
  - Identify failure mode (i.e., leak or rupture) given a predicted failure due to quasi-static overpressure event
- Approach
  - Two dimensional axi-symmetric finite element analysis of the total containment structure
    - Provided detailed information on potential weak links (discontinuities)
  - Detailed analyses of the weak links

Containment Structural Evaluation Comprised Two Parts

- Palisades Finite Element Model (PFEM) mesh consisted of five major sections
  - dome, ring girder, cylinder wall, basemat and soil
  - Analysis performed by plant Engineer/Constructor (Bechtel)
- Leakage at major penetrations was evaluated using EPRI developed method (EPRI NP-6260-M)
  - Penetrations less than 24-inches diameter were judged not to constitute a weak link in a concrete containment
  - Electrical penetrations also judged to not be a concern (based on NUREG-1037 analysis)

Structural Evaluations Identified Potential Weak Links

- Global Weak Links (failure = catastrophic rupture)
  - Mid-Height Region of Cylindrical Wall
  - Apex Region of the Dome
  - Basemat-Cylindrical Wall Interface Region
- Local Weak Links (failure = minor loss of pressure)
  - Access Openings (including seals)
    - equipment hatch
    - escape lock
    - personnel air lock
  - Large pipe penetrations

Containment Fragility Curve Combination of Fragility Curves for Each Weak Link

- Fragility curve provides cumulative probability of containment failure as a function of internal pressure
  - Seven weak link fragility curves combined into composite (total containment) fragility curve
  - $PrF(p) = 1- \Pi_{i=1,n}[1-PrF_i(p)]$
  - where:
  - $PrF_i(p)$ = probability of failure mode i at pressure p
  - n = total number of failure modes
- Minimum median capacity of the Palisades containment at 95% confidence level was determined to be
  - 131 psig (0.90 MPa) or 2.38 times the design pressure of 55 psig

Palisades IPE CET Features

- CET and PDS's developed together such that PDS's contain ONLY plant system information, and CET addresses ONLY effect of severe accident physical processes
  - Plant system dependencies accounted for
  - CET focused on containment performance and fission product release
- Single, general-form CET
  - Consistent treatment of PDS's
  - Consistent binning of CET endstates into source terms

Only Dominant PDS's Used in CET Analysis

- Highest frequency PDS's analyzed until 99% of total frequency has been included
  - Highest 18 PDS contribute 99.16% of total frequency
    - Comprises all PDS with frequency greater than 1E-7
  - Most severe PDS frequency was increased to account for the missing 0.84% frequency
    - Total core damage frequency of 5.12E-5/yr is preserved

Palisades IPE CET Top Events

PDS - Plant Damage State

BYE - Early Cont. Bypass

CIS - Cont. Isolation

BYL - Late Cont. Bypass

RIV - Recovery after CD but before VB

UDD - Upward debris dispersal at VB

CAE - Early relocation of core debris to aux. bldg.

CIE - Cont. intact early

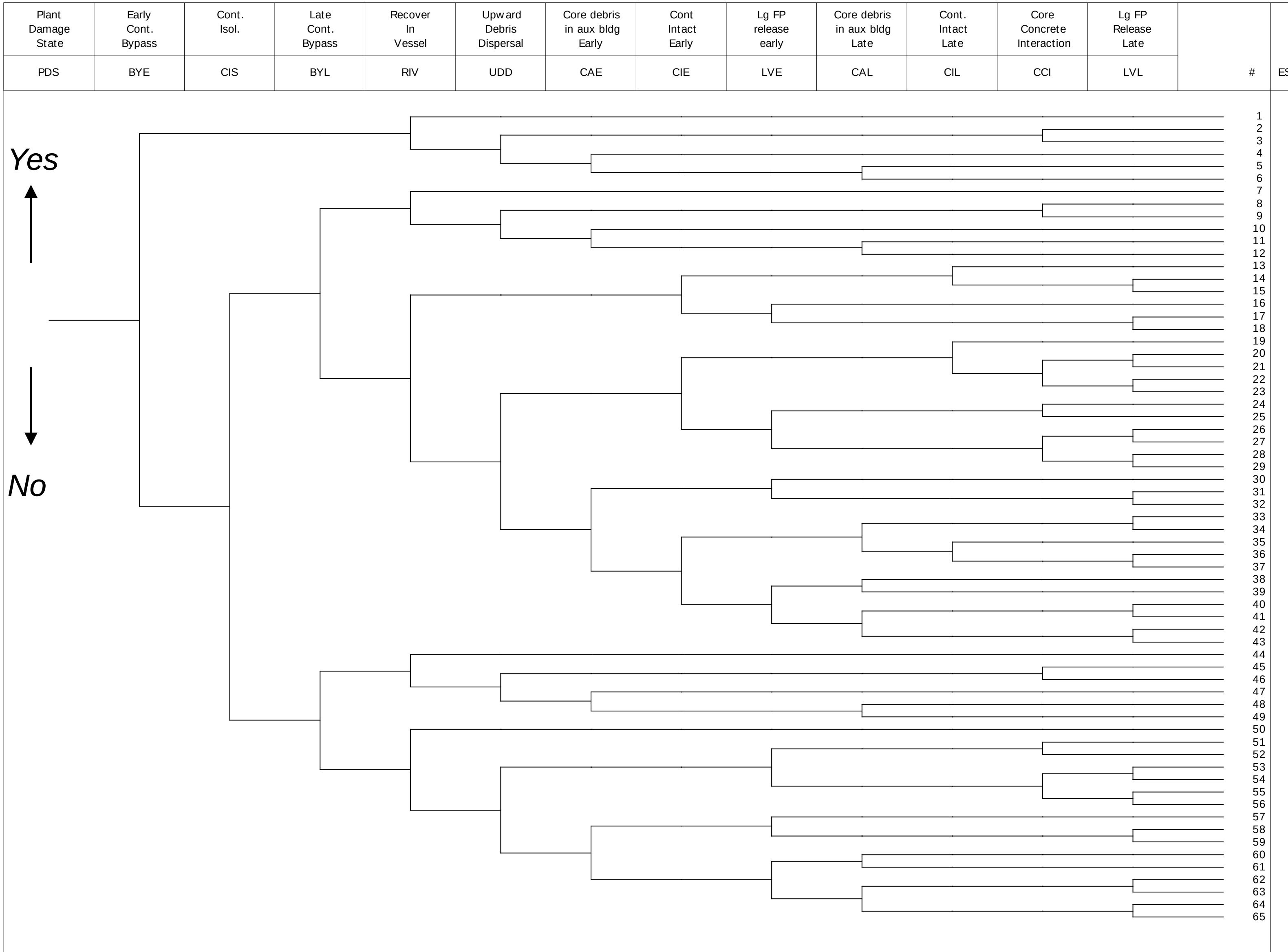
LVE - Large volatile fission product release early

CAL - Late relocation of core debris to aux. bldg.

CIL - Cont. intact late

CCI - Core concrete interaction resulting in large fission product release

LVL - Large volatile fission product release late



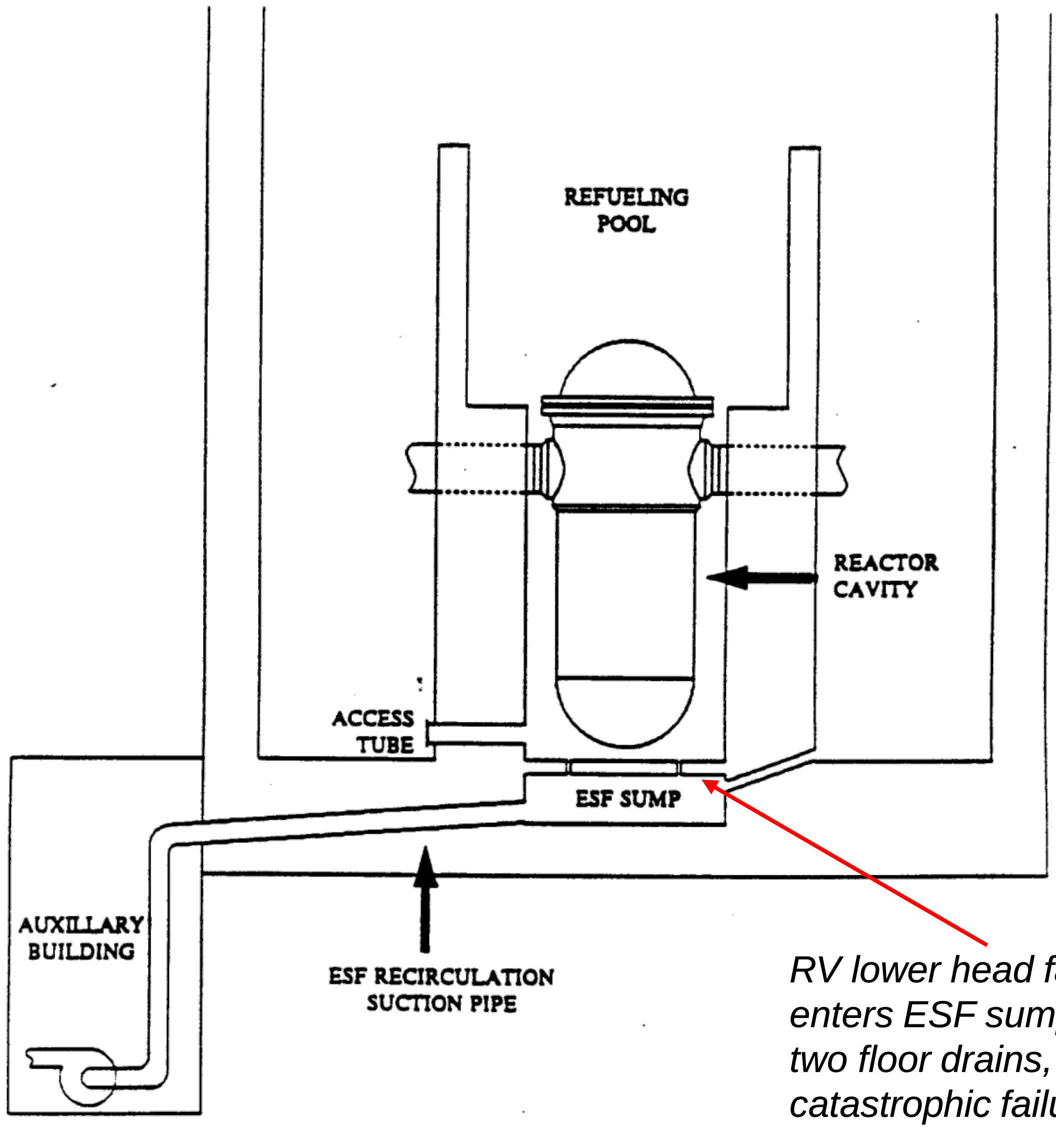
PALISADES-CET - Palisades IPE (Jan. 1993) Containment Event Tree 2001/06/14 Page 3

CET Top Events Modeled Using Fault Trees

- 93 pages of fault trees used to model 12 top events
  - Comprising about a hundred basic events (4 groups)
    - PDS dependent BEs (“house events”)
    - Recovery BEs
      - Recovery of containment systems or S/G cooling
    - Operator Action BEs
      - Operator open PORV to depressurize RCS
    - Phenomenological BEs
      - 45 events
        - Single event assigned different probabilities depending on context (boundary conditions)

CAE Top Event

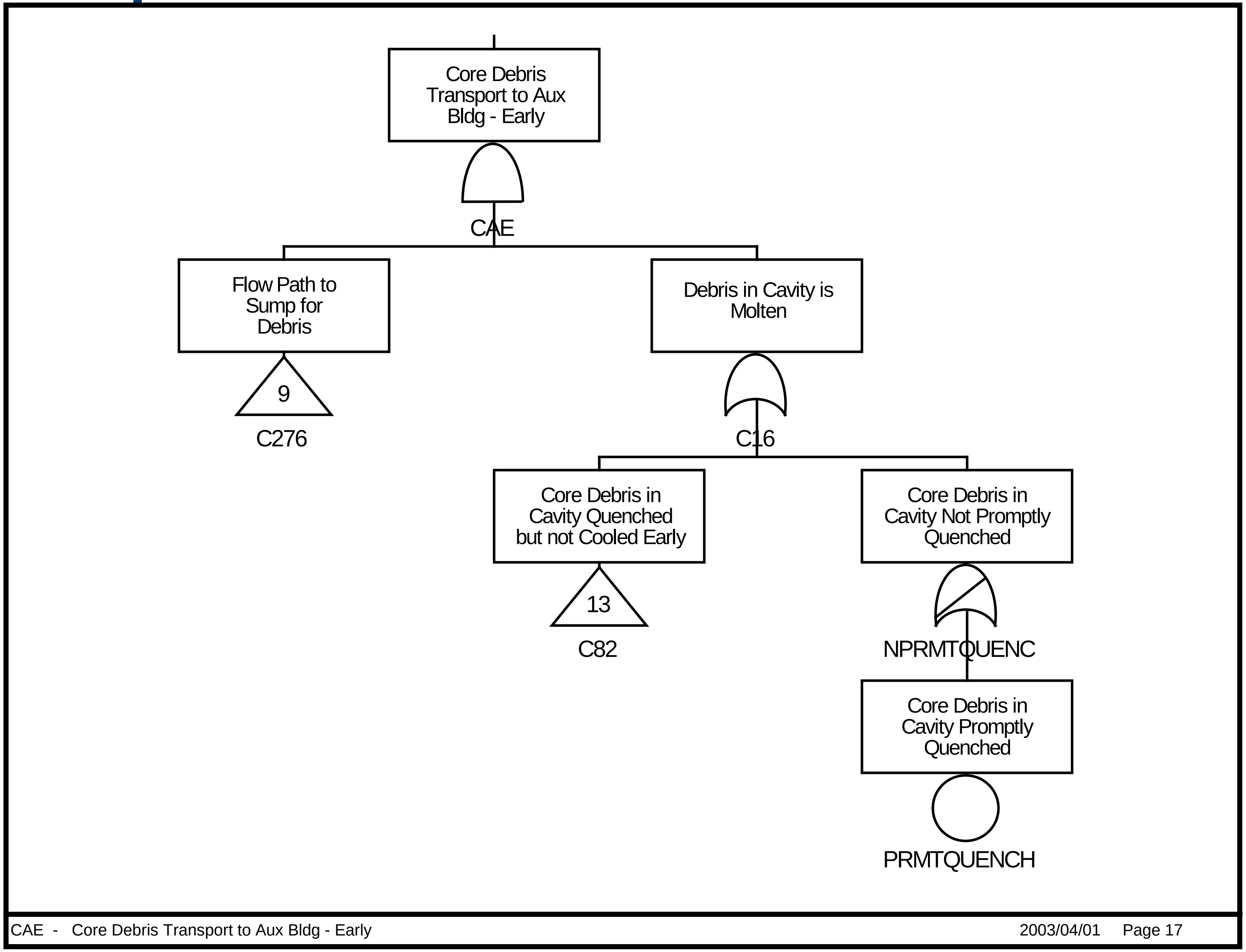
- Core debris enters Auxiliary building Early
  - Early: soon after vessel failure
- Core debris enters the auxiliary building via ESF (sump) recirculation line
  - Core debris falls to cavity floor, is not quenched and flows into sump via drain lines.
  - Core debris falls to cavity floor, is quenched, but not in a coolable geometry
    - Core debris reheats and flow into sump
  - RV fails at high pressure causing catastrophic failure of cavity floor.

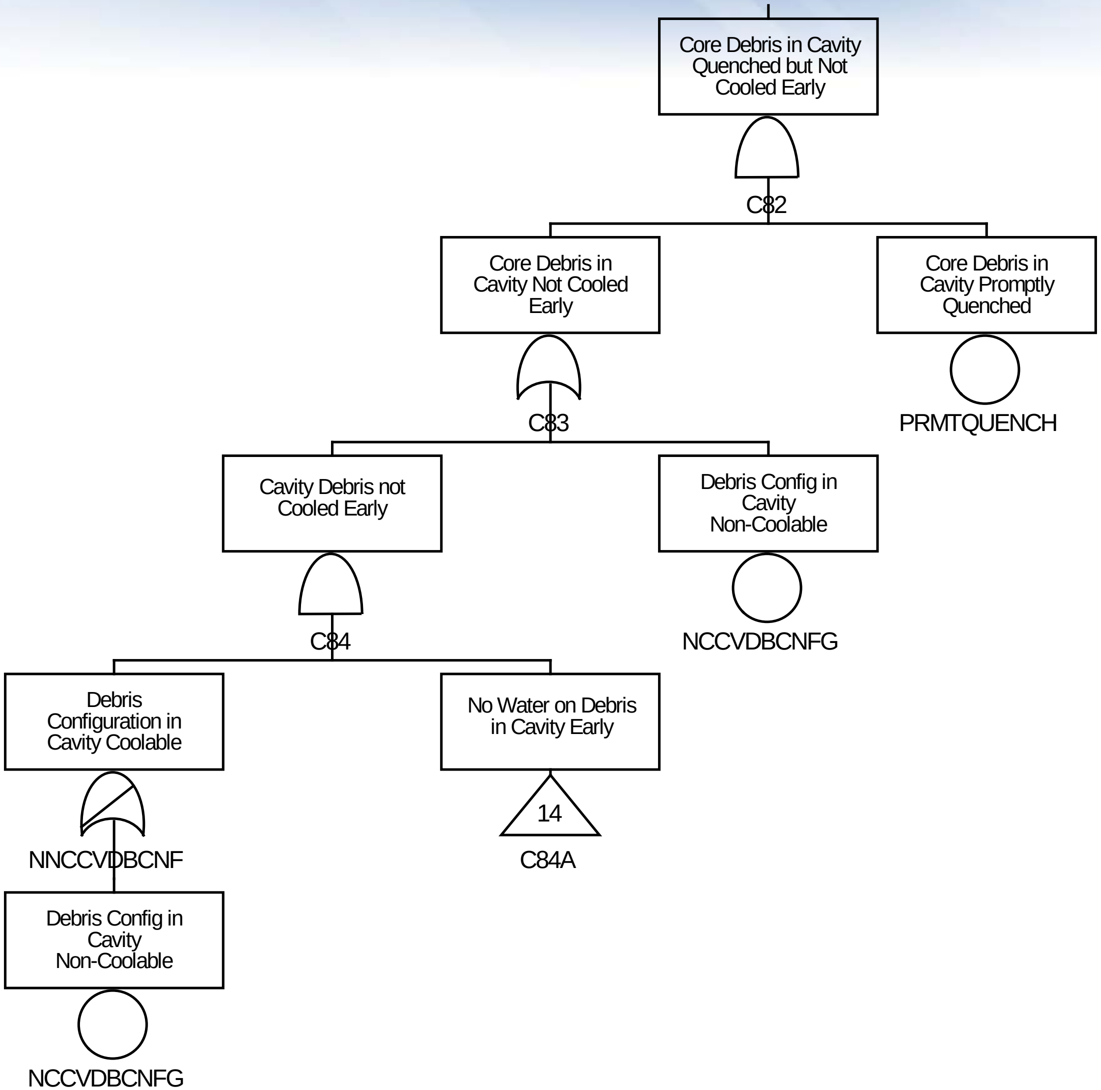


RV lower head fails, core debris enters ESF sump either via the two floor drains, or through catastrophic failure of cavity floor.

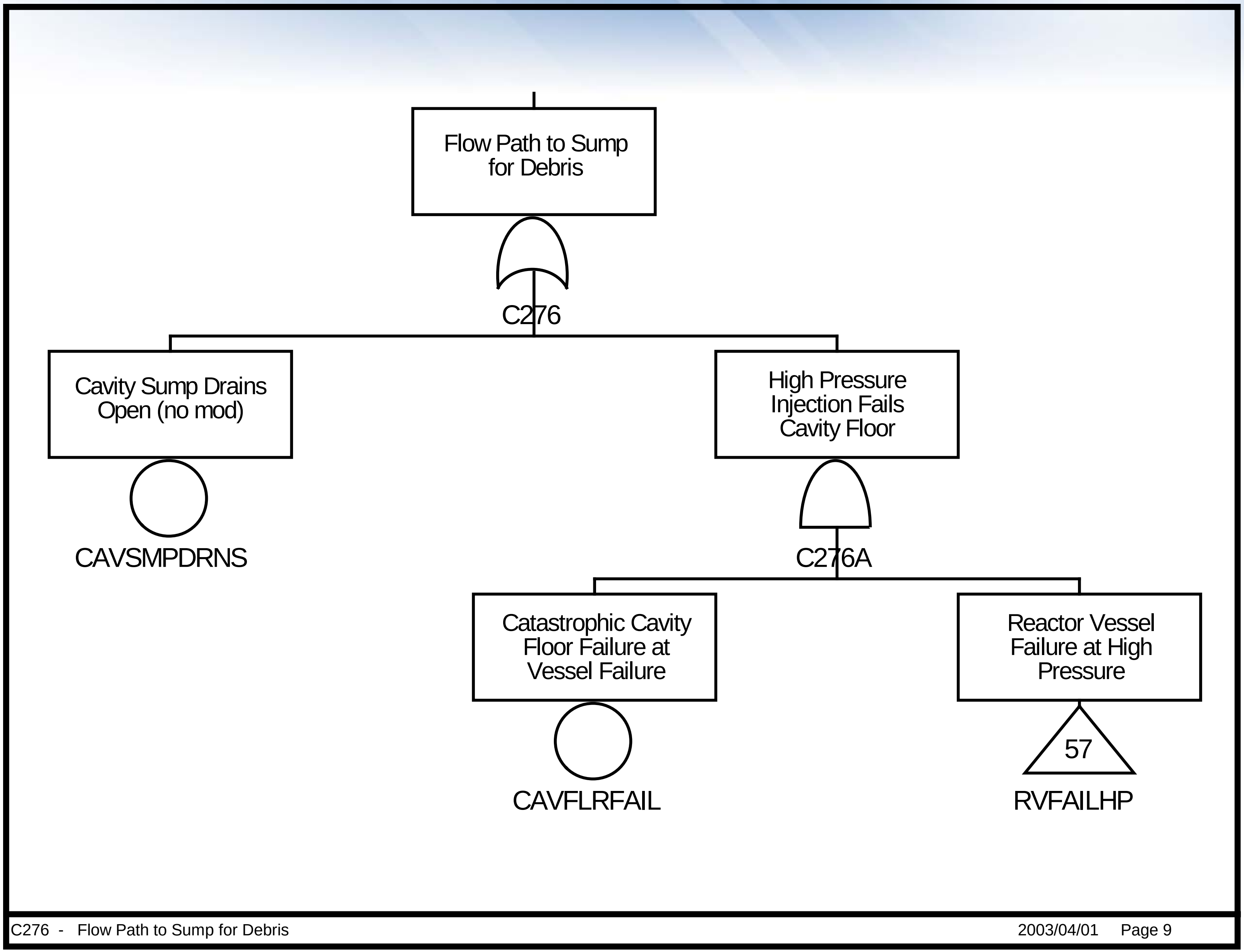
PALISADES REACTOR CAVITY-ENGINEERED SAFEGUARDS SUMP ARRANGEMENT

CAE Top Event Fault Tree





C82 - Core Debris in Cavity Quenched but not Cooled Early 2003/04/01 Page 13


Flow Path to Sump for Debris
C276
Cavity Sump Drains Open (no mod)
CAVSMPDRNS
High Pressure Injection Fails Cavity Floor
C276A
Catastrophic Cavity Floor Failure at Vessel Failure
CAVFLRFAIL
Reactor Vessel Failure at High Pressure
57
RVFAILHP
C276 - Flow Path to Sump for Debris
2003/04/01 Page 9

CAVFLRFAIL – Cavity Floor Fails

- Structural analysis estimates a failure pressure of 370 psid (2.55 MPa)
  - Assumed to be a mean value
  - Standard deviation of 10% assumed
- Analyses of Palisades severe accidents produced peak cavity pressure estimates for three class of PDS
  - High, medium, and low RCS pressures

CAVFLRFAIL – Cavity Floor Failure Probability Estimate

Probability of Cavity Floor Failure depends upon RCS Pressure (at time of RPV failure) – Estimated by convolution of peak cavity pressure distribution and floor failure pressure distribution.			
RCS Pressure (MPa)	RCS Pressure Class (at RPV failure)	Prob of Cavity Floor Failure	Applicable PDS
17.0	High	0.53	T w/o creep rupture
7.0	Medium	0.196	B and D
3.0	Low	2.71E-3	A1, A2, C, and T w/ creep rupture

Other BE Quantified in a Variety of Ways

Basic Event	Description	Comments	Prob.
CVFLOODSYS	RPV cavity flooding system fails	passive system consisting of drain lines and restricting orifices to direct water into cavity (engineering analysis)	1.65E-2
FLNGFAIL	Reactor Cavity Access Tube Blind Flange Failed by Debris	Failure probability depends on whether or not water is present on opposite side of flange (PDS dependent)	5E-3 (wet) 1.0 (dry)
HOTLEGFAIL	Induced failure (thermal creep) of RCS Hot Leg	CPMAAP analysis (RCS initially intact, SRV not stuck open)	0.402

BE Quantification (cont.)

Basic Event	Description	Comments	Prob.
SEALLOCA	Induced failure of RCP seals	Probability based on CEOG tests	1E-3
VFTIMELONG	Time to Vessel Failure sufficiently long to ensure low RCS pressure when lower RV head fails	Various potential failure mechanisms analyzed along with likelihood of necessary conditions	Depend on RCS pressure and whether cavity is flooded or dry (see next slide)

VFTIMELONG – Probability Estimates

	Containment System Status	
PDS Initiator	P or Q (Cavity Flooded)	R, S, V or W (Cavity Dry)
A1	0.99	0.95
A2	0.99	0.95
B	0.74	0.27
C	0.99	0.95
D	0.50	0.05
T (w/ induced failure)	0.75	0.56
T (w/o induced failure)	0.00	0.00

Values for each basic event documented for every PDS

PDS DESCRIPTOR - TEJP   COMPLETED BY: RGChristie   DATE - 08/27/92

PDS PROB = 9.400E-06   REVIEWED BY: ________   REV - 2

PDS DEPENDENT BASIC EVENTS -

ISLOCA	0.0	LBLOCA	0.0	MBLOCA	0.0
SBLOCA	0.0	SGTR	0.0	TRANSIENT	1.0
CSP	0.0	CAC	0.0	SECONDCOOL	1.0
SIRWT	1.0	SILATE	0.0		

RECOVERY BASIC EVENTS -

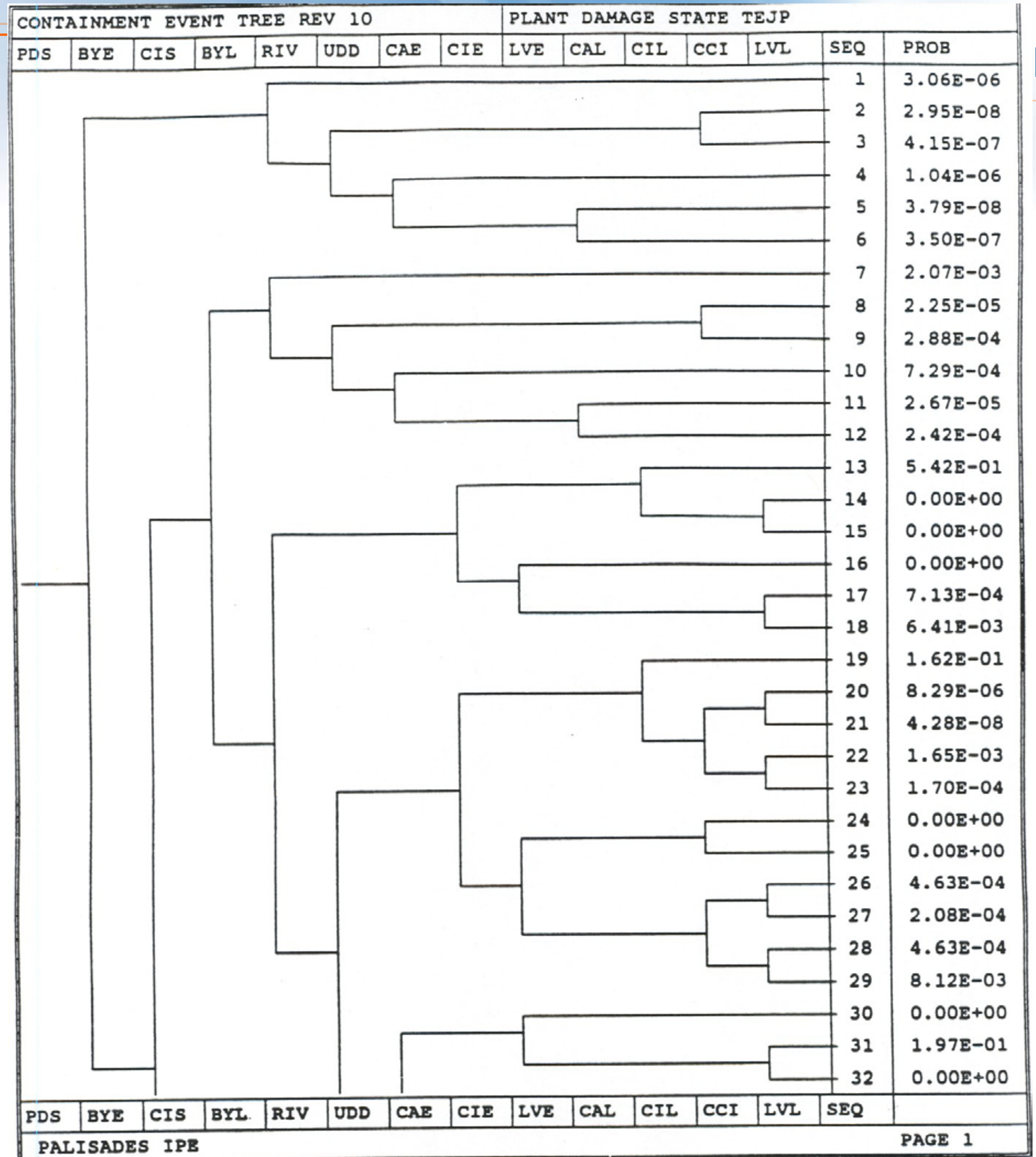
CACRECOV	0.0	CSPRECOV	0.0
CSPRECIV	0.0	SECCLRECOV	0.0

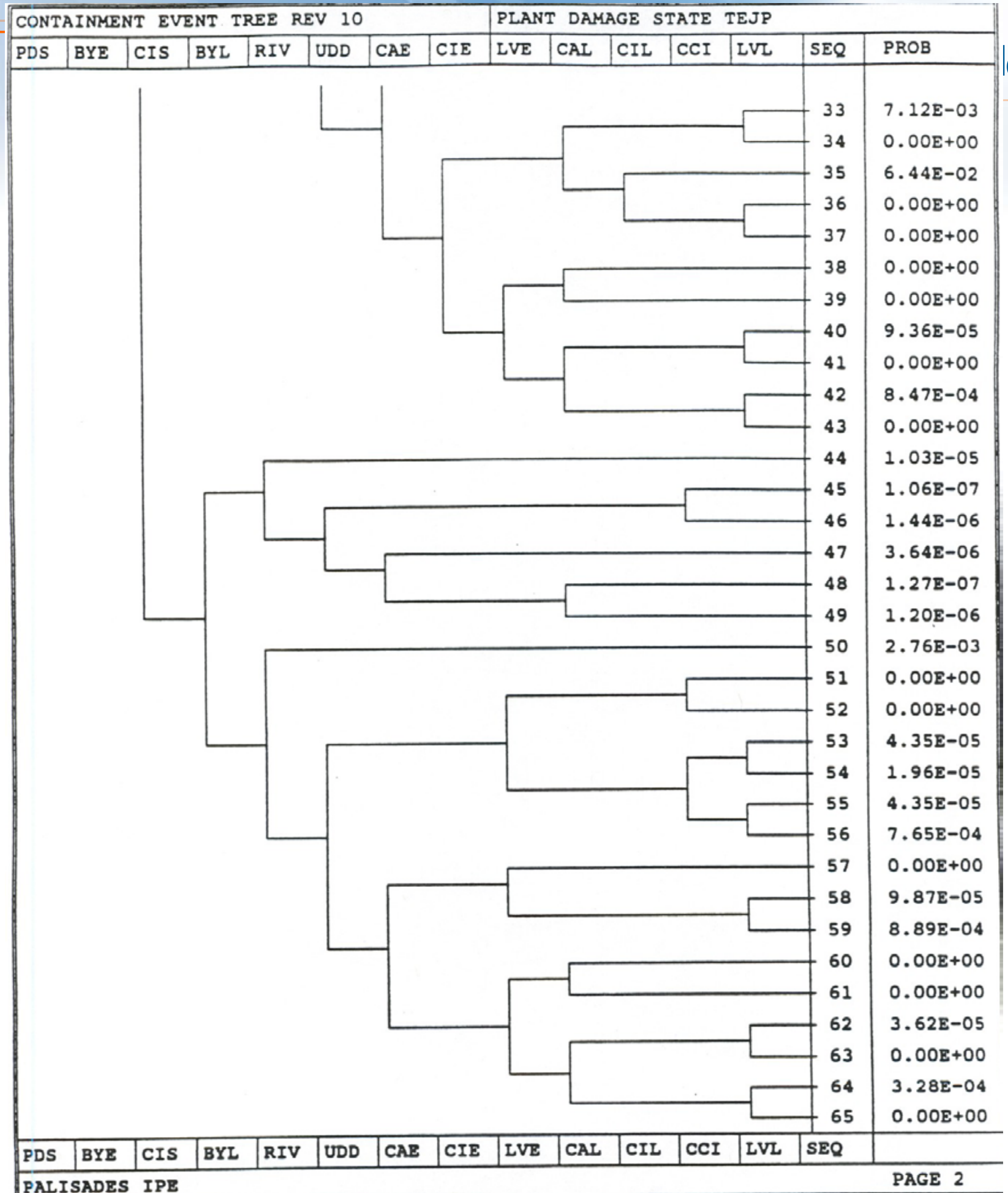
OPERATOR ACTION BASIC EVENTS -

OPDEPRESS = 0.0

PHENOMENOLOGICAL BASIC EVENTS -

CACFAIL	0.100	FLAMMGAS1	0.00116	PORVS	0.4149
CAVFLRFAIL	0.530	FLAMMGAS2	0.0	PRMTQUENCH	0.375
CAVITYFAIL	0.0215	FLNGFAIL	0.005	PZSRVFTC	0.475
CAVSMPDRNS	1.000	FRACMSLBIC	0.0002	REVAPTMNG	1.000
CAVTYRETEN	0.001	FRACMSLBOC	0.0001	RLFVLVOPEN	0.0
CIS	0.995	HOTLEGFAIL	0.402	SEALLOCA	0.001
CNCATKTMG1	1.000	INVSLSTEXP	0.0001	SECVLVOPEN	1.000
CNCATKTMG2	0.010	INVSSLH2	0.00792	SGTUBEFAIL	0.0034
CRDBAFTBD	0.050	MSLBPRESS	0.00087	SIFLOWPATH	1.000
CRIUMIMPNG	0.010	MSLBSGFAIL	0.050	STMSPIKE	0.178
CSPFAIL	0.100	MV504SPP	1.000	SURGEFAIL	0.025
CSPFAILRIV	0.050	NATCONV	0.500	VFTIMELONG	0.750
CVFLOODSYS	0.0165	NCCVDBCNFG	0.250	VSSLHTXFR	0.900
DCH	0.00992	NCUCDBCNFG	0.050	VSSLIMPNG	0.001
DRYOUTTMNG	1.000	PCSDEPRESS	0.730	VSSLTHRUST	0.00005
EXVSLSTEXP	0.005	PCSRETEN	0.550		


CONTAINMENT EVENT TREE REV 10
PLANT DAMAGE STATE TEJP
PDS BYE CIS BYL RIV UDD CAE CIE LVE CAL CIL CCI LVL SEQ PROB
1 3.06E-06
2 2.95E-08
3 4.15E-07
4 1.04E-06
5 3.79E-08
6 3.50E-07
7 2.07E-03
8 2.25E-05
9 2.88E-04
10 7.29E-04
11 2.67E-05
12 2.42E-04
13 5.42E-01
14 0.00E+00
15 0.00E+00
16 0.00E+00
17 7.13E-04
18 6.41E-03
19 1.62E-01
20 8.29E-06
21 4.28E-08
22 1.65E-03
23 1.70E-04
24 0.00E+00
25 0.00E+00
26 4.63E-04
27 2.08E-04
28 4.63E-04
29 8.12E-03
30 0.00E+00
31 1.97E-01
32 0.00E+00
PDS BYE CIS BYL RIV UDD CAE CIE LVE CAL CIL CCI LVL SEQ
PALISADES IPE
PAGE 1


CONTAINMENT EVENT TREE REV 10
PLANT DAMAGE STATE TEJP
PDS BYE CIS BYL RIV UDD CAE CIE LVE CAL CIL CCI LVL SEQ PROB
33 7.12E-03
34 0.00E+00
35 6.44E-02
36 0.00E+00
37 0.00E+00
38 0.00E+00
39 0.00E+00
40 9.36E-05
41 0.00E+00
42 8.47E-04
43 0.00E+00
44 1.03E-05
45 1.06E-07
46 1.44E-06
47 3.64E-06
48 1.27E-07
49 1.20E-06
50 2.76E-03
51 0.00E+00
52 0.00E+00
53 4.35E-05
54 1.96E-05
55 4.35E-05
56 7.65E-04
57 0.00E+00
58 9.87E-05
59 8.89E-04
60 0.00E+00
61 0.00E+00
62 3.62E-05
63 0.00E+00
64 3.28E-04
65 0.00E+00
PDS BYE CIS BYL RIV UDD CAE CIE LVE CAL CIL CCI LVL SEQ
PALISADES IPE
PAGE 2

CET Quantified for Each PDS (18)

- For each PDS:
  - CET basic events quantified
  - CET fault trees quantified
  - CET end states (65) quantified
- Generates a 18 x 65 matrix
- CET end state frequencies summed over 18 PDS
  - Total frequency of each containment-state/source-term
- Source terms generated for each of the 65 CET end states
  - CPMAAP (Consumers Power version of MAAP)

Initial CET End State Conditional Probability By PDS

	TEJQ	CEJW	A2EGR	BEGV	TEJS	DEJR	A2EGP	A1EGR	ZEGP
CET-01	3.06E-06	0.00E-00	0.00E-00	0.00E-00	0.00E-00	0.00E-00	0.00E-00	0.00E-00	3.99E-06
CET-02	2.95E-08	2.37E-02	0.00E-00	0.00E-00	1.04E-06	0.00E-00	0.00E-00	0.00E-00	1.86E-09
CET-03	4.15E-07	2.37E-02	0.00E-00	0.00E-00	1.04E-06	0.00E-00	0.00E-00	0.00E-00	4.16E-08
CET-04	1.04E-06	9.53E-01	0.00E-00	0.00E-00	2.87E-06	0.00E-00	0.00E-00	0.00E-00	6.77E-07
CET-05	3.79E-08	0.00E-00	0.00E-00	0.00E-00	0.00E-00	0.00E-00	0.00E-00	0.00E-00	2.39E-08
CET-06	3.49E-07	0.00E-00	0.00E-00	0.00E-00	0.00E-00	0.00E-00	0.00E-00	0.00E-00	2.17E-07
CET-07	2.07E-03	0.00E-00	0.00E-00	0.00E-00	0.00E-00	0.00E-00	0.00E-00	0.00E-00	2.70E-03
CET-08	2.25E-05	0.00E-00	0.00E-00	0.00E-00	7.06E-04	0.00E-00	0.00E-00	0.00E-00	2.33E-06
CET-09	2.88E-04	0.00E-00	0.00E-00	0.00E-00	7.06E-04	0.00E-00	0.00E-00	0.00E-00	3.00E-05
CET-10	7.29E-04	0.00E-00	0.00E-00	0.00E-00	1.97E-03	0.00E-00	0.00E-00	0.00E-00	4.82E-04
CET-11	2.69E-05	0.00E-00	0.00E-00	0.00E-00	0.00E-00	0.00E-00	0.00E-00	0.00E-00	1.67E-05
CET-12	2.42E-04	0.00E-00	0.00E-00	0.00E-00	0.00E-00	0.00E-00	0.00E-00	0.00E-00	1.51E-04
CET-13	5.42E-01	0.00E-00	0.00E-00	0.00E-00	0.00E-00	0.00E-00	8.18E-01	0.00E-00	7.05E-01
CET-14	0.00E-00	0.00E-00	0.00E-00	0.00E-00	0.00E-00	0.00E-00	0.00E-00	0.00E-00	0.00E-00
CET-15	0.00E-00	0.00E-00	0.00E-00	0.00E-00	0.00E-00	0.00E-00	0.00E-00	0.00E-00	0.00E-00
CET-16	0.00E-00	0.00E-00	0.00E-00	0.00E-00	0.00E-00	0.00E-00	0.00E-00	0.00E-00	0.00E-00
CET-17	7.13E-04	0.00E-00	0.00E-00	0.00E-00	0.00E-00	0.00E-00	9.20E-04	0.00E-00	4.17E-04
CET-18	6.41E-03	0.00E-00	0.00E-00	0.00E-00	0.00E-00	0.00E-00	8.28E-03	0.00E-00	8.86E-03
CET-19	1.48E-01	0.00E-00	4.12E-02	0.00E-00	4.01E-01	7.83E-01	7.62E-03	0.00E-00	9.34E-02
CET-20	8.21E-05	0.00E-00	2.31E-05	1.15E-03	2.25E-04	4.40E-04	3.89E-07	0.00E-00	2.15E-06
CET-21	4.28E-08	0.00E-00	0.00E-00	1.15E-03	0.00E-00	0.00E-00	2.01E-09	0.00E-00	2.65E-06
CET-22	1.63E-02	0.00E-00	4.60E-03	2.28E-01	4.48E-02	8.76E-02	7.74E-05	0.00E-00	4.27E-04
CET-23	1.70E-04	0.00E-00	0.00E-00	2.28E-01	0.00E-00	0.00E-00	7.96E-06	0.00E-00	6.20E-04
CET-24	0.00E-00	0.00E-00	0.00E-00	0.00E-00	0.00E-00	0.00E-00	0.00E-00	0.00E-00	0.00E-00
CET-25	0.00E-00	0.00E-00	0.00E-00	0.00E-00	0.00E-00	0.00E-00	0.00E-00	0.00E-00	0.00E-00
CET-26	4.63E-04	0.00E-00	7.12E-04	6.91E-03	9.66E-03	1.31E-02	1.77E-05	0.00E-00	1.20E-04
CET-27	2.08E-04	0.00E-00	0.00E-00	3.45E-03	0.00E-00	0.00E-00	5.49E-06	0.00E-00	2.66E-04
CET-28	4.63E-04	0.00E-00	7.12E-04	3.45E-03	9.66E-03	1.31E-02	1.22E-05	0.00E-00	1.20E-04
CET-29	8.12E-03	0.00E-00	0.00E-00	3.45E-03	0.00E-00	0.00E-00	2.15E-04	0.00E-00	4.82E-03
CET-30	0.00E-00	0.00E-00	8.15E-01	2.37E-01	2.37E-01	0.00E-00	0.00E-00	8.55E-01	0.00E-00
CET-31	1.97E-01	0.00E-00	0.00E-00	0.00E-00	2.90E-01	8.41E-02	1.17E-01	0.00E-00	5.91E-02
CET-32	0.00E-00	0.00E-00	0.00E-00	2.37E-01	0.00E-00	0.00E-00	0.00E-00	0.00E-00	7.22E-02
CET-33	7.16E-03	0.00E-00	1.32E-02	2.44E-02	0.00E-00	1.37E-03	2.22E-03	1.38E-02	2.03E-03
CET-34	0.00E-00	0.00E-00	0.00E-00	2.44E-02	0.00E-00	0.00E-00	0.00E-00	0.00E-00	2.48E-03
CET-35	6.44E-02	0.00E-00	1.19E-01	0.00E-00	0.00E-00	1.23E-02	4.00E-02	1.25E-01	4.08E-02
CET-36	0.00E-00	0.00E-00	0.00E-00	0.00E-00	0.00E-00	0.00E-00	0.00E-00	0.00E-00	0.00E-00
CET-37	0.00E-00	0.00E-00	0.00E-00	0.00E-00	0.00E-00	0.00E-00	0.00E-00	0.00E-00	0.00E-00
CET-38	0.00E-00	0.00E-00	1.40E-04	1.25E-04	0.00E-00	0.00E-00	0.00E-00	1.56E-04	0.00E-00
CET-39	0.00E-00	0.00E-00	1.26E-03	0.00E-00	0.00E-00	0.00E-00	0.00E-00	1.40E-03	0.00E-00
CET-40	9.41E-05	0.00E-00	0.00E-00	0.00E-00	0.00E-00	7.01E-06	2.50E-05	0.00E-00	2.66E-05
CET-41	0.00E-00	0.00E-00	0.00E-00	1.25E-04	0.00E-00	0.00E-00	0.00E-00	0.00E-00	3.26E-05
CET-42	8.47E-04	0.00E-00	0.00E-00	0.00E-00	0.00E-00	6.32E-05	4.50E-04	0.00E-00	2.41E-04
CET-43	0.00E-00	0.00E-00	0.00E-00	0.00E-00	0.00E-00	0.00E-00	0.00E-00	0.00E-00	2.95E-04
CET-44	1.04E-05	0.00E-00	0.00E-00	0.00E-00	0.00E-00	0.00E-00	0.00E-00	0.00E-00	1.35E-05
CET-45	1.07E-07	0.00E-00	0.00E-00	0.00E-00	3.54E-06	0.00E-00	0.00E-00	0.00E-00	7.73E-09
CET-46	1.44E-06	0.00E-00	0.00E-00	0.00E-00	3.54E-06	0.00E-00	0.00E-00	0.00E-00	1.39E-07
CET-47	3.64E-06	0.00E-00	0.00E-00	0.00E-00	9.87E-06	0.00E-00	0.00E-00	0.00E-00	2.39E-06
CET-48	1.29E-07	0.00E-00	0.00E-00	0.00E-00	0.00E-00	0.00E-00	0.00E-00	0.00E-00	7.69E-08
CET-49	1.21E-06	0.00E-00	0.00E-00	0.00E-00	0.00E-00	0.00E-00	0.00E-00	0.00E-00	7.46E-07
CET-50	2.76E-03	0.00E-00	0.00E-00	0.00E-00	0.00E-00	0.00E-00	4.16E-03	0.00E-00	3.59E-03
CET-51	0.00E-00	0.00E-00	1.19E-04	5.93E-04	6.50E-04	1.01E-03	0.00E-00	0.00E-00	0.00E-00
CET-52	0.00E-00	0.00E-00	1.19E-04	5.93E-04	6.50E-04	1.01E-03	0.00E-00	0.00E-00	0.00E-00
CET-53	4.36E-05	0.00E-00	1.19E-04	5.93E-04	1.17E-03	2.25E-03	1.99E-06	0.00E-00	1.13E-05
CET-54	1.96E-05	0.00E-00	0.00E-00	5.93E-04	0.00E-00	0.00E-00	8.92E-07	0.00E-00	2.50E-05
CET-55	4.36E-05	0.00E-00	1.19E-04	5.93E-04	1.17E-03	2.25E-03	1.99E-06	0.00E-00	1.13E-05
CET-56	7.65E-04	0.00E-00	0.00E-00	5.93E-04	0.00E-00	0.00E-00	3.50E-05	0.00E-00	4.54E-04
CET-57	0.00E-00	0.00E-00	4.09E-03	1.19E-03	1.19E-03	0.00E-00	0.00E-00	4.30E-03	0.00E-00
CET-58	9.88E-05	0.00E-00	0.00E-00	0.00E-00	1.46E-03	4.23E-04	5.87E-05	0.00E-00	2.97E-05
CET-59	8.89E-04	0.00E-00	0.00E-00	1.19E-03	0.00E-00	0.00E-00	5.29E-04	0.00E-00	6.30E-04
CET-60	0.00E-00	0.00E-00	6.70E-05	1.23E-04	0.00E-00	0.00E-00	0.00E-00	7.03E-05	0.00E-00
CET-61	0.00E-00	0.00E-00	6.03E-04	0.00E-00	0.00E-00	0.00E-00	0.00E-00	6.33E-04	0.00E-00
CET-62	3.64E-05	0.00E-00	0.00E-00	0.00E-00	0.00E-00	6.91E-06	1.13E-05	0.00E-00	1.03E-05
CET-63	0.00E-00	0.00E-00	0.00E-00	1.23E-04	0.00E-00	0.00E-00	0.00E-00	0.00E-00	1.26E-05
CET-64	3.28E-04	0.00E-00	0.00E-00	0.00E-00	0.00E-00	6.23E-05	2.03E-04	0.00E-00	9.34E-05
CET-65	0.00E-00	0.00E-00	0.00E-00	0.00E-00	0.00E-00	0.00E-00	0.00E-00	0.00E-00	1.14E-04
	1.000815	1.0004	1.001083	1.004794	1.002039	1.002092	0.999853	1.000359	0.999645

Initial CET End State Conditional Probability By PDS

	BEGP	TEJP	TEJW	TEJV	BEGR	TEJR	DEJP	DEJS	BEGS
CET-01	0.00E-00	3.06E-06	0.00E-00	0.00E-00	0.00E-00	0.00E-00	7.26E-02	0.00E-00	0.00E-00
CET-02	0.00E-00	2.95E-08	1.04E-06	1.04E-06	0.00E-00	1.04E-06	3.45E-03	3.69E-01	0.00E-00
CET-03	0.00E-00	4.15E-07	1.04E-06	1.04E-06	0.00E-00	1.04E-06	4.41E-02	3.69E-01	0.00E-00
CET-04	0.00E-00	1.04E-06	2.88E-06	2.86E-06	0.00E-00	2.46E-06	3.83E-02	8.06E-02	0.00E-00
CET-05	0.00E-00	3.79E-08	0.00E-00	6.92E-09	0.00E-00	4.04E-08	1.43E-03	0.00E-00	0.00E-00
CET-06	0.00E-00	3.50E-07	0.00E-00	0.00E-00	0.00E-00	3.64E-07	1.30E-02	0.00E-00	0.00E-00
CET-07	0.00E-00	2.07E-03	0.00E-00	0.00E-00	0.00E-00	0.00E-00	0.00E-00	0.00E-00	0.00E-00
CET-08	0.00E-00	2.25E-05	7.06E-04	7.06E-04	0.00E-00	7.06E-04	0.00E-00	0.00E-00	0.00E-00
CET-09	0.00E-00	2.88E-04	7.06E-04	7.06E-04	0.00E-00	7.06E-04	0.00E-00	0.00E-00	0.00E-00
CET-10	0.00E-00	7.29E-04	1.97E-03	1.96E-03	0.00E-00	1.69E-03	0.00E-00	0.00E-00	0.00E-00
CET-11	0.00E-00	2.67E-05	0.00E-00	5.54E-06	0.00E-00	2.77E-05	0.00E-00	0.00E-00	0.00E-00
CET-12	0.00E-00	2.42E-04	0.00E-00	0.00E-00	0.00E-00	2.49E-04	0.00E-00	0.00E-00	0.00E-00
CET-13	7.89E-01	5.42E-01	0.00E-00	0.00E-00	0.00E-00	0.00E-00	3.44E-01	0.00E-00	0.00E-00
CET-14	0.00E-00	0.00E-00	0.00E-00	0.00E-00	0.00E-00	0.00E-00	0.00E-00	0.00E-00	0.00E-00
CET-15	0.00E-00	0.00E-00	0.00E-00	0.00E-00	0.00E-00	0.00E-00	0.00E-00	0.00E-00	0.00E-00
CET-16	0.00E-00	0.00E-00	0.00E-00	0.00E-00	0.00E-00	0.00E-00	0.00E-00	0.00E-00	0.00E-00
CET-17	2.48E-04	7.13E-04	0.00E-00	0.00E-00	0.00E-00	0.00E-00	1.76E-04	0.00E-00	0.00E-00
CET-18	4.72E-03	6.41E-03	0.00E-00	0.00E-00	0.00E-00	0.00E-00	1.59E-03	0.00E-00	0.00E-00
CET-19	9.14E-03	1.62E-01	0.00E-00	0.00E-00	4.12E-01	3.96E-01	2.17E-01	1.42E-01	4.12E-01
CET-20	2.34E-07	8.29E-06	2.23E-03	2.22E-03	1.16E-04	2.22E-04	1.11E-05	7.96E-05	1.16E-04
CET-21	2.36E-07	4.28E-08	0.00E-00	0.00E-00	1.16E-04	0.00E-00	5.74E-08	0.00E-00	1.16E-04
CET-22	4.65E-05	1.65E-03	4.44E-01	4.42E-01	2.30E-02	4.43E-02	2.21E-03	1.58E-02	2.30E-02
CET-23	5.60E-05	1.70E-04	0.00E-00	0.00E-00	2.30E-02	0.00E-00	2.27E-04	0.00E-00	2.30E-02
CET-24	0.00E-00	0.00E-00	0.00E-00	0.00E-00	0.00E-00	0.00E-00	0.00E-00	0.00E-00	0.00E-00
CET-25	0.00E-00	0.00E-00	0.00E-00	0.00E-00	0.00E-00	0.00E-00	0.00E-00	0.00E-00	0.00E-00
CET-26	2.10E-05	4.63E-04	9.54E-03	1.07E-02	7.17E-03	1.24E-02	4.80E-04	2.37E-03	7.17E-03
CET-27	1.37E-05	2.08E-04	0.00E-00	0.00E-00	3.59E-03	0.00E-00	1.49E-04	0.00E-00	3.59E-03
CET-28	7.23E-06	4.63E-04	9.54E-03	1.07E-02	3.59E-03	1.24E-02	3.31E-04	2.37E-03	3.59E-03
CET-29	2.61E-04	8.12E-03	0.00E-00	0.00E-00	3.59E-03	0.00E-00	5.82E-03	0.00E-00	3.59E-03
CET-30	0.00E-00	0.00E-00	2.37E-01	2.36E-01	2.25E-01	2.04E-01	0.00E-00	0.00E-00	2.61E-01
CET-31	7.10E-02	1.97E-01	2.90E-01	2.89E-01	0.00E-00	2.49E-01	1.82E-01	1.77E-02	0.00E-00
CET-32	7.10E-02	0.00E-00	0.00E-00	0.00E-00	2.25E-01	0.00E-00	0.00E-00	0.00E-00	2.61E-01
CET-33	2.45E-03	7.12E-03	0.00E-00	1.47E-03	3.65E-03	7.31E-03	6.79E-03	0.00E-00	0.00E-00
CET-34	2.45E-03	0.00E-00	0.00E-00	0.00E-00	3.65E-03	0.00E-00	0.00E-00	0.00E-00	0.00E-00
CET-35	4.43E-02	6.44E-02	0.00E-00	0.00E-00	6.58E-02	6.58E-02	6.14E-02	0.00E-00	0.00E-00
CET-36	0.00E-00	0.00E-00	0.00E-00	0.00E-00	0.00E-00	0.00E-00	0.00E-00	0.00E-00	0.00E-00
CET-37	0.00E-00	0.00E-00	0.00E-00	0.00E-00	0.00E-00	0.00E-00	0.00E-00	0.00E-00	0.00E-00
CET-38	0.00E-00	0.00E-00	0.00E-00	3.39E-06	2.30E-05	4.32E-05	0.00E-00	0.00E-00	0.00E-00
CET-39	0.00E-00	0.00E-00	0.00E-00	0.00E-00	2.07E-04	3.89E-04	0.00E-00	0.00E-00	0.00E-00
CET-40	1.54E-05	9.36E-05	0.00E-00	4.15E-06	0.00E-00	5.28E-05	3.48E-05	0.00E-00	0.00E-00
CET-41	1.54E-05	0.00E-00	0.00E-00	0.00E-00	2.30E-05	0.00E-00	0.00E-00	0.00E-00	0.00E-00
CET-42	1.39E-04	8.47E-04	0.00E-00	0.00E-00	0.00E-00	4.76E-04	3.15E-04	0.00E-00	0.00E-00
CET-43	1.39E-04	0.00E-00	0.00E-00	0.00E-00	2.07E-04	0.00E-00	0.00E-00	0.00E-00	0.00E-00
CET-44	0.00E-00	1.03E-05	0.00E-00	0.00E-00	0.00E-00	0.00E-00	0.00E-00	0.00E-00	0.00E-00
CET-45	0.00E-00	1.06E-07	3.54E-06	3.54E-06	0.00E-00	3.53E-06	0.00E-00	0.00E-00	0.00E-00
CET-46	0.00E-00	1.44E-06	3.54E-06	3.54E-06	0.00E-00	3.53E-06	0.00E-00	0.00E-00	0.00E-00
CET-47	0.00E-00	3.64E-06	9.89E-06	9.85E-06	0.00E-00	8.48E-06	0.00E-00	0.00E-00	0.00E-00
CET-48	0.00E-00	1.27E-07	0.00E-00	2.06E-08	0.00E-00	1.30E-07	0.00E-00	0.00E-00	0.00E-00
CET-49	0.00E-00	1.20E-06	0.00E-00	0.00E-00	0.00E-00	1.24E-06	0.00E-00	0.00E-00	0.00E-00
CET-50	3.99E-03	2.76E-03	0.00E-00	0.00E-00	0.00E-00	0.00E-00	1.74E-03	0.00E-00	0.00E-00
CET-51	0.00E-00	0.00E-00	6.50E-04	6.50E-04	5.93E-04	6.50E-04	0.00E-00	0.00E-00	5.93E-04
CET-52	0.00E-00	0.00E-00	6.50E-04	6.50E-04	5.93E-04	6.50E-04	0.00E-00	0.00E-00	5.93E-04
CET-53	1.19E-06	4.35E-05	1.17E-03	1.17E-03	5.93E-04	1.17E-03	5.69E-05	4.08E-04	5.93E-04
CET-54	2.25E-06	1.96E-05	0.00E-00	0.00E-00	5.93E-04	0.00E-00	2.56E-05	0.00E-00	5.93E-04
CET-55	1.19E-06	4.35E-05	1.17E-03	1.17E-03	5.93E-04	1.17E-03	5.69E-05	4.08E-04	5.93E-04
CET-56	4.32E-05	7.65E-04	0.00E-00	0.00E-00	5.93E-04	0.00E-00	9.99E-04	0.00E-00	5.93E-04
CET-57	0.00E-00	0.00E-00	1.19E-03	1.19E-03	1.13E-03	1.02E-03	0.00E-00	0.00E-00	1.31E-03
CET-58	3.56E-05	9.87E-05	1.46E-03	1.45E-03	0.00E-00	1.25E-03	9.16E-05	8.90E-05	0.00E-00
CET-59	6.78E-04	8.89E-04	0.00E-00	0.00E-00	1.13E-03	0.00E-00	8.24E-04	0.00E-00	1.31E-03
CET-60	0.00E-00	0.00E-00	0.00E-00	3.34E-06	1.85E-05	1.67E-05	0.00E-00	0.00E-00	0.00E-00
CET-61	0.00E-00	0.00E-00	0.00E-00	0.00E-00	1.66E-04	1.51E-04	0.00E-00	0.00E-00	0.00E-00
CET-62	1.23E-05	3.62E-05	0.00E-00	4.08E-06	0.00E-00	2.04E-05	3.43E-05	0.00E-00	0.00E-00
CET-63	1.23E-05	0.00E-00	0.00E-00	0.00E-00	1.85E-05	0.00E-00	0.00E-00	0.00E-00	0.00E-00
CET-64	1.12E-04	3.28E-04	0.00E-00	0.00E-00	0.00E-00	1.84E-04	3.10E-04	0.00E-00	0.00E-00
CET-65	1.12E-04	0.00E-00	0.00E-00	0.00E-00	1.66E-04	0.00E-00	0.00E-00	0.00E-00	0.00E-00
	1.000023	1.000049	1.002004	1.001784	1.005919	1.002076	0.999552	0.999825	1.00435

CET ES	Aggregated Freq (/yr)	Important PDS Contributors
1	1.20E-8	DEJP(100%[a])
2	8.47E-7	DEJS(99%)
3	8.53E-7	DEJS(99%)
4	4.18E-7	CEJW(55%) DEJS(44%)
10[b]	4.44E-8	TEJW(39%) TEJV(29%) TEJP(15%) TEJR(9%)
18	1.54E-7	BEGP(40%) TEJP(36%) ZEGP(22%)
22	7.31E-6	TEJW(53%) TEJV(40%)
23	1.43E-7	BEGR(46%) BEGV(36%) BEGS(11%) ZEGP(5%)
26	2.29E-7	TEJW(36%) TEJV(30%) TEJR(13%) BEGR(7%)
28	2.15E-7	TEJW(39%) TEJV(33%) TEJR(14%) BEGR(4%)
29	1.18E-7	TEJP(65%) ZEGP(25%) BEGR(4%)
30	5.43E-6	TEJW(36%) TEJV(27%) BEGR(14%) TEJR(8%) A2EGR(4%) BEGS(4%)
31	8.25E-6	TEJW(30%) TEJV(23%) TEJP(22%) BEGP(11%) TEJR(7%) ZEGP(3%)
32	2.01E-6	BEGP(40%) BEGR(34%) ZEGP(13%) BEGS(10%
33	1.54E-7	TEJP(40%) BEGP(20%) TEJR(10%) BEGR(8%) TEJV(6%) ZEGP(5%)
57[c]	2.73E-8	TEJW(36%) TEJV(27%) BEGR(14%) TEJR(8%)
total	2.62E-5 2.68E-5	Sum of dominant CET ES Total containment failure frequency

a. Contribution to ES frequency
b. Containment bypass
c. Containment isolation failure

Source Terms Calculated Using CPMAAP

- 41 cases selected for CPMAAP analysis
  - various combinations of PDS and CET-ES from list of dominant contributors to containment failure
    - For example:
    - DEJP-01 – SGTR with recovery in-vessel
    - DEJS-02 – SGTR with stuck open secondary SRV, upward debris dispersal and CCI in upper containment
    - CEJW-04 – ISLOCA outside containment
    - TEJW-10 – Blackout with creep induced SGTR
    - A1EGR-30 – LBLOCA with core to aux early and a large volatile release early
    - BEGP-31 – SBLOCA with core to aux early and late revaporization form aux building and CCI in aux bldg

CET ES	Aggregated Freq (/yr)	Important PDS Contributors
1	1.20E-8	DEJP(100%[a])
2	8.47E-7	DEJS(99%)
3	8.53E-7	DEJS(99%)
4	4.18E-7	CEJW(55%) DEJS(44%)
10[b]	4.44E-8	TEJW(39%) TEJV(29%) TEJP(15%) TEJR(9%)
18	1.54E-7	BEGP(40%) TEJP(36%) ZEGP(22%)
22	7.31E-6	TEJW(53%) TEJV(40%)
23	1.43E-7	BEGR(46%) BEGV(36%) BEGS(11%) ZEGP(5%)
26	2.29E-7	TEJW(36%) TEJV(30%) TEJR(13%) BEGR(7%)
28	2.15E-7	TEJW(39%) TEJV(33%) TEJR(14%) BEGR(4%)
29	1.18E-7	TEJP(65%) ZEGP(25%) BEGR(4%)
30	5.43E-6	TEJW(36%) TEJV(27%) BEGR(14%) TEJR(8%) A2EGR(4%) BEGS(4%)
31	8.25E-6	TEJW(30%) TEJV(23%) TEJP(22%) BEGP(11%) TEJR(7%) ZEGP(3%)
32	2.01E-6	BEGP(40%) BEGR(34%) ZEGP(13%) BEGS(10%
33	1.54E-7	TEJP(40%) BEGP(20%) TEJR(10%) BEGR(8%) TEJV(6%) ZEGP(5%)
57[c]	2.73E-8	TEJW(36%) TEJV(27%) BEGR(14%) TEJR(8%)
total	2.62E-5 2.68E-5	Sum of dominant CET ES Total containment failure frequency

a. Contribution to ES frequency
b. Containment bypass
c. Containment isolation failure

Calculated Source Terms from CPMAAP (examples)

PDS-ES	Nobel Gas	I	Cs	Te	Sr	Mo	La	Ce	Ba	Time of release (hr)	Warning Time (hr)	Release Duration (hr)
DEJP-01	0.03	0.01	0.01	<1E-5	6E-5	3E-5	4E-6	1E-5	6E-4	25	3.6	2.0
DEJS-02	0.97	0.30	0.29	1E-3	1E-3	0.04	3E-5	1E-5	0.01	27	5.7	2.0
DEJS-03	0.97	0.30	0.29	1E-3	1E-3	0.04	3E-5	1E-5	0.01	27	5.7	2.0
CEJW-04	1.0	0.92	0.92	0.45	0.03	0.25	0.01	4E-4	0.10	1.3	0.9	2.0
A2EGR-32	0.49	0.02	0.02	<1E-5	2E-4	8E-3	<1E-5	<1E-5	2E-3	4.0	3.0	1.0

Typically, multiple PDSs selected for each ES/CPMAAP calculation with “worst-case” eventually selected to represent particular CET-ES.

Accident Progression Analysis (P-300)

11. NUREG/CR-6595, Rev.1

Session Objectives

- To understand the simple LERF analysis approach developed under NRC sponsorship
  - NUREG/CR-6595, Rev. 1, Oct. 2004
    - An Approach for Estimating the Frequencies of Various Containment Failure Modes and Bypass Events

NUREG/CR-6595 Approach

- Simplified approach to supplement Level-1 PRA used in risk-informed decision-making
- Relies upon Level-1 information to estimate containment failure frequencies
- Aimed at Regulatory Guide 1.174 acceptance guidelines
  - Quick, approximate estimate of LERF for screening against 1.174 guidelines

Containment Failure Modes

- Early structural failure
- Containment bypass
- Containment isolation failure
- Late structural failure
- Containment venting

LERF Models developed for PWRs and BWRs

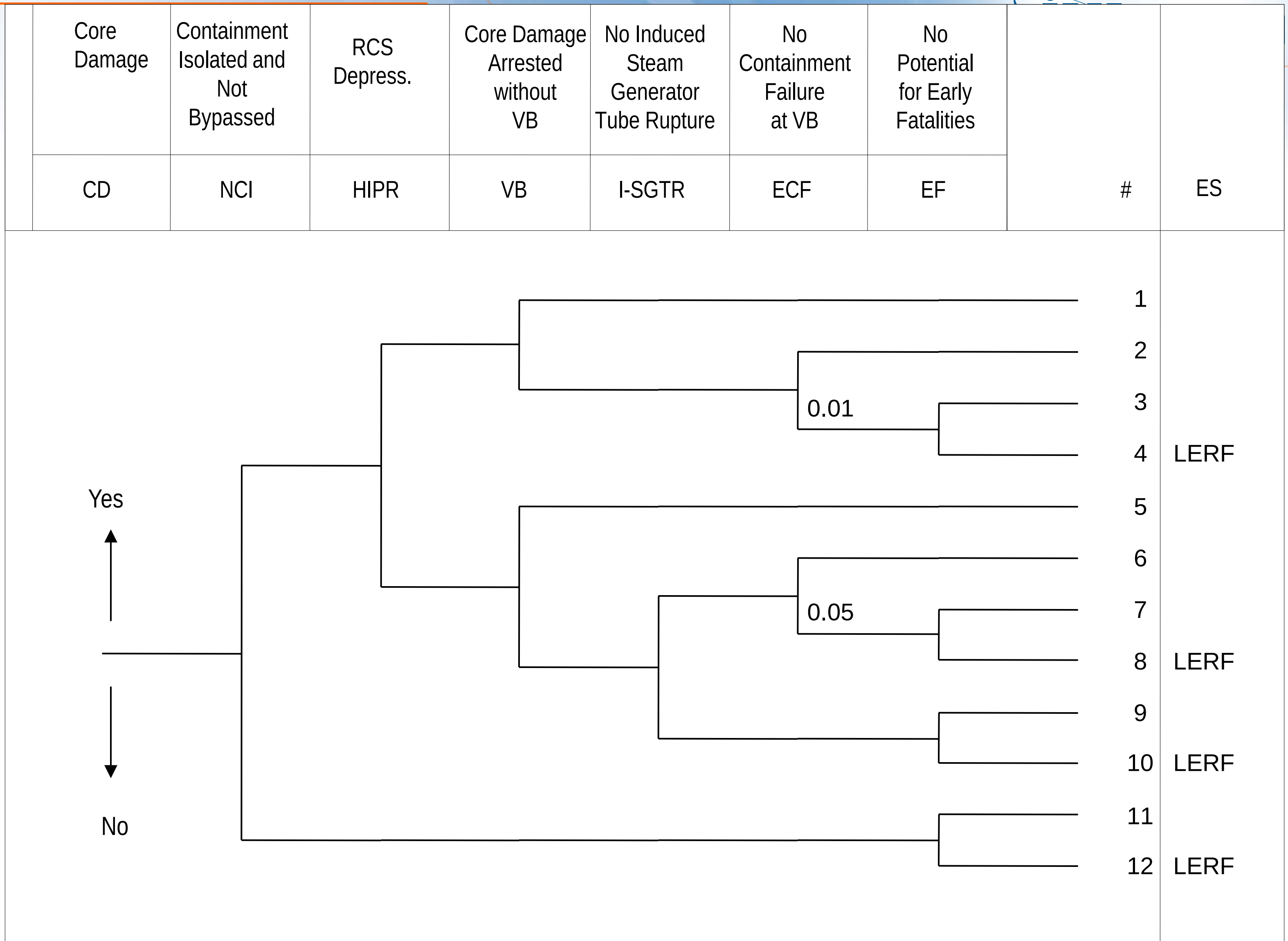
- PWRs with a large volume containment
- PWRs with an ice condenser containment
- BWRs with a Mark I containment
- BWRs with a Mark II containment
- BWRs with a Mark III containment

CET from NUREG/CR-6595 (LERF)

- Focus is on early loss of containment integrity
- Includes 5 CETs:
  - PWR large dry (and subatmospheric), and ice condenser containments
  - BWR Mark-I, Mark-II and Mark-II containments
- Simplified, high-level models intended to provide reasonable, somewhat bounding estimates of LERF for most plants
  - first step in scoping study for comparing plant-specific analysis to RG-1.174 acceptance criteria

LERF CET for PWR Large Dry Containment

- Also encompasses subatmospheric containments
  - Both rely on large volumes and relatively high design pressures to mitigate consequences
- Initiating Event is Core Damage (CD) - Frequency and characteristics of CD sequences from Level-1 analysis
- Most split fractions determined from Level-1 PRA supplemented by additional analysis and information
  - Generic estimates provided only for probability of early containment failure



LERF-LGDRY - PWR Large Dry LERF CET (NUREG/CR-6595)

NCI - No Containment Isolation (Nor Containment Bypass)

- Includes:
  - Failure of containment to isolate
  - Interfacing system LOCA
  - SGTR Initiating Event
  - ATWS (pressure-)induced SGTR or RCS pipe failure
  - Loss of containment heat removal
    - i.e., containment failure before core damage
- Quantified using Level-1 information

HIPR - RCS Not Depressurized

- Top event identifies pressure in reactor vessel at time of core damage (for subsequent evaluation of likelihood of containment failure)
- Dependent on
  - Level-1 initiating event (i.e., small LOCA - RCS at high pressure, medium and large LOCAs - RCS at low pressure)
  - Likelihood of operator initiating depressurization
  - Likelihood of temperature-induced RCS pressure boundary failure after core damage

VB - Vessel Breach

- Addresses possibility of recovery of coolant injection after uncovery of top of active fuel (i.e., Level-1 CD state) but before vessel failure
  - Recovery of electric power - typically based on probability of recovering offsite power (Level-1 analysis)
  - Depressurization of RCS by operators - if low pressure systems are available

I-SGTR - Induced Steam Generator Tube Rupture

- Creep failure (thermally induced) of SG tubes during core oxidation
- Depends on status of S/G secondary side
  - Not likely if steam-side remains pressurized
- Typically assessed with plant-specific calculations that track relevant phenomena and compute creep damage to multiple RCS components to determine likely failure point
  - Surge line, hot leg, S/G tube failures all possible

ECF - Early Containment Failure

- Containment failure at vessel breach, depends on:
  - RCS pressure
  - Amount and temp. of core debris exiting vessel
  - Size of hole in vessel
  - Amount of water in cavity
  - Configuration in cavity
  - Operability of containment sprays
  - Structural capacity of containment building
- In simplified treatment, only RCS pressure explicitly considered

ECF - Low Pressure RCS

- ECF given Low Pressure Vessel Failure Includes:
  - In-vessel steam explosion
  - Rapid steam generation from core debris contacting water in the cavity
  - Hydrogen combustion
- Conditional probability of ECF estimated at 0.01
  - based on previous PRAs

ECF - High Pressure RCS

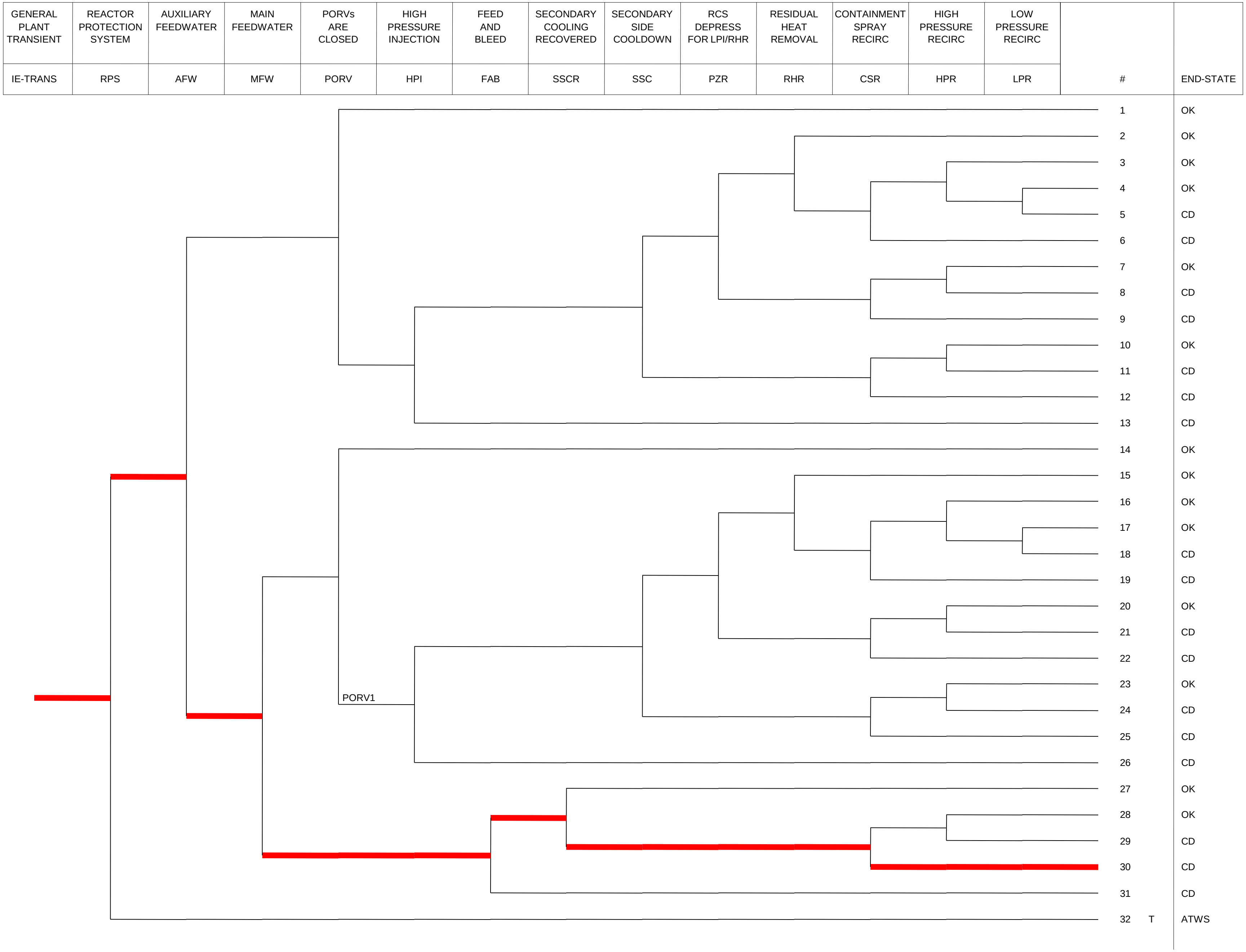
- ECF given high pressure vessel failure Includes:
  - High Pressure Melt Ejection (HPME)
    - Direct Containment Heating (DCH)
    - Hydrogen combustion
  - In-vessel steam explosions (less likely compared to low pressure RCS case)
- Conditional probability of ECF estimated at 0.05
  - based on previous PRA and research

EF - Early Fatalities

- Given loss of containment integrity
  - depends on magnitude and timing of radionuclide release
  - Sequence-specific (timing of start of core damage, vessel failure)
  - CET path specific (timing of containment failure)
  - Plant/site-specific (timing of declaration of site emergency, initiation of evacuation, and time needed for evacuation)

Example Core Damage Sequence

- Transient IE with failure of all secondary side cooling (AFW & MFW), success of feed and bleed, but failure of recirculation from containment sump
  - Sequence #30 (see next slide)
    - Sequence frequency = 1E-6/year


GENERAL PLANT TRANSIENT
REACTOR PROTECTION SYSTEM
AUXILIARY FEEDWATER
MAIN FEEDWATER
PORVs ARE CLOSED
HIGH PRESSURE INJECTION
FEED AND BLEED
SECONDARY COOLING RECOVERED
SECONDARY SIDE COOLDOWN
RCS DEPRESS FOR LPI/RHR
RESIDUAL HEAT REMOVAL
CONTAINMENT SPRAY RECIRC
HIGH PRESSURE RECIRC
LOW PRESSURE RECIRC
IE-TRANS
RPS
AFW
MFW
PORV
HPI
FAB
SSCR
SSC
PZR
RHR
CSR
HPR
LPR
#
END-STATE
PORV1
1 OK
2 OK
3 OK
4 OK
5 CD
6 CD
7 OK
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Analyze CD Sequence for LERF

Failure ID – Success event description

NCI – Containment isolated and not bypassed

HIPR – RCS depressurized

VB – Core damage arrested without vessel breach

I-SGTR – No induced steam generator tube rupture

ECF – No containment failure at vessel breach

EF – No potential for early fatalities

NCI – Containment Not-Isolated or Bypassed

- Two issues to address
  - Is initiating event a bypass
  - Does containment isolation fail
- Transient – Not a bypass IE
- Containment Isolation
  - Addressed by system model (fault tree)
    - Signal to isolate containment (auto/manual)
    - Hardware success
    - Assume fault tree model yields $P_{fail} = 0.01$

HIPR – RCS Not Depressurized

- Two considerations
  - Effects of IE and subsequent plant response
- Transient – RCS at high pressure (at least initially)
- Subsequent plant response
  - Secondary side cooling fails
  - Feed and bleed cooling success
    - RCS not depressurized
  - Possible induced failure of reactor coolant pressure boundary (RCPB)

HIPR – RCS Remains at High Press.

- Severe accident progression can induce a failure of the reactor coolant pressure boundary before CD
  - In order of likelihood
    - Surge line failure
    - Hot leg failure
    - Induced SGTR
  - This event does NOT include I-SGTR (considered later)
    - CD sequence determined, operator initiated, and induced surge line or hot let failures
  - Addressed through T/H code calculations
    - Assume analysis results in an induced RCS failure probability estimate of 25%
    - Therefore, P(HIPR) = 0.75

VB – CD Not Arrested Before VB

- System analysis issue
  - Primarily for loss of offsite power sequences
    - Recovering offsite power can recover coolant injection and/or core cooling
  - Possibly recover (after start of CD) from hardware failures
    - Non-safety coolant injection or core cooling options
  - Assume probability of non-recovery = 0.9

I-SGTR – Induced SGTR

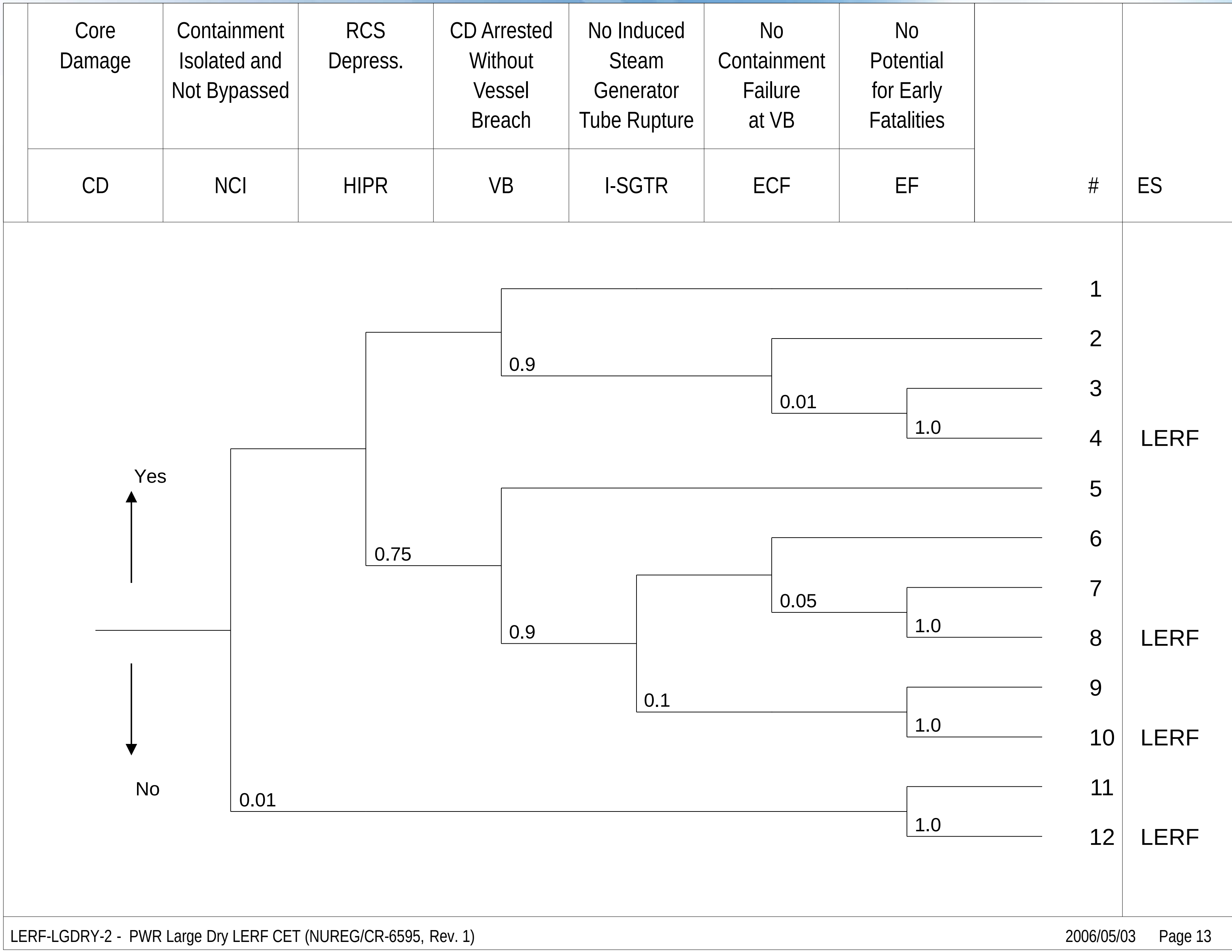
- No secondary side cooling (secondary side depressurized) makes an I-SGTR a concern
  – SGTR can be either temperature-induced (creep) or pressure-induced (typically an ATWS issue)
  – Dependent on state of SG tubes (flaws)
    - Crack depth (% through wall)
    - Flaw size distributions vary greatly among plants and crack size difficult to determine
- Conditional probability of I-SGTR given HIPR (i.e., no other induced RCS failure or depressurization)
  – I-SGTR is the worst case induced RCS failure
    - Containment bypass
  – Assume P(I-SGTR|HIPR) = 10%

ECF – Containment Failure at VB

- Does containment structure survive loads resulting from vessel failure?
  - Yes = No large release
  - No = Potential for large release
- Depends on many factors
  - Simplified (NUREG/CR-6595, Rev.1)
    - Only depends on RCS pressure at VB
      - Low Pressure RCS: ECF = 0.01
      - High Pressure RCS: ECF = 0.05

EF – Potential for Early Fatalities

- Depends on magnitude and timing of release
- Typically comprises all early failures and bypass
- However, can discriminate on release size
  - Threshold of greater than 2.5% to 10% iodine
  - Requires estimation of source term
- Assume all early failures and bypass
  - Split fraction = 1.0


Core Damage
Containment Isolated and Not Bypassed
RCS Depress.
CD Arrested Without Vessel Breach
No Induced Steam Generator Tube Rupture
No Containment Failure at VB
No Potential for Early Fatalities
CD
NCI
HIPR
VB
I-SGTR
ECF
EF
#
ES
Yes
No
0.75
0.9
0.01
0.9
0.01
1.0
0.9
0.05
1.0
0.1
1.0
1.0
1
2
3
4 LERF
5
6
7
8 LERF
9
10 LERF
11
12 LERF
LERF-LGDRY-2 - PWR Large Dry LERF CET (NUREG/CR-6595, Rev. 1)
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LERF CET for PWR Ice Condenser Containment

- Similar to that for large dry
  - Additional top event for H2 igniters
- Initiating Event is Core Damage (CD) - Frequency and characteristics of CD sequences from Level-1 analysis
- Most split fractions determined from Level-1 PRA supplemented by additional analysis and information
  - Generic estimates provided only for probability of early containment failure

Most Top Events Same as Lg Dry

- Core Damage
- Containment Isolated or Not Bypassed
- *H2 Igniters - not in Lg Dry*
- RCS Depressurized
- CD arrested before VB
- No I-SGTR
- *No Cont. Failure at or before VB*
  - *Treated differently from Lg Dry*
- No Potential for Early Fatalities

H2 Igniters Operating Before TAF

- Uncovering Top of Active Fuel results in oxidation of zircaloy clad – releasing hydrogen
- Igniters require ac power
  - Some plants might have dedicated backup power
- Igniters usually started manually
- Detailed system model – desired approach
  - Availability of ac power and human action to actuate – reasonable approximation

Likelihood of Containment Failure Depends on Many Factors

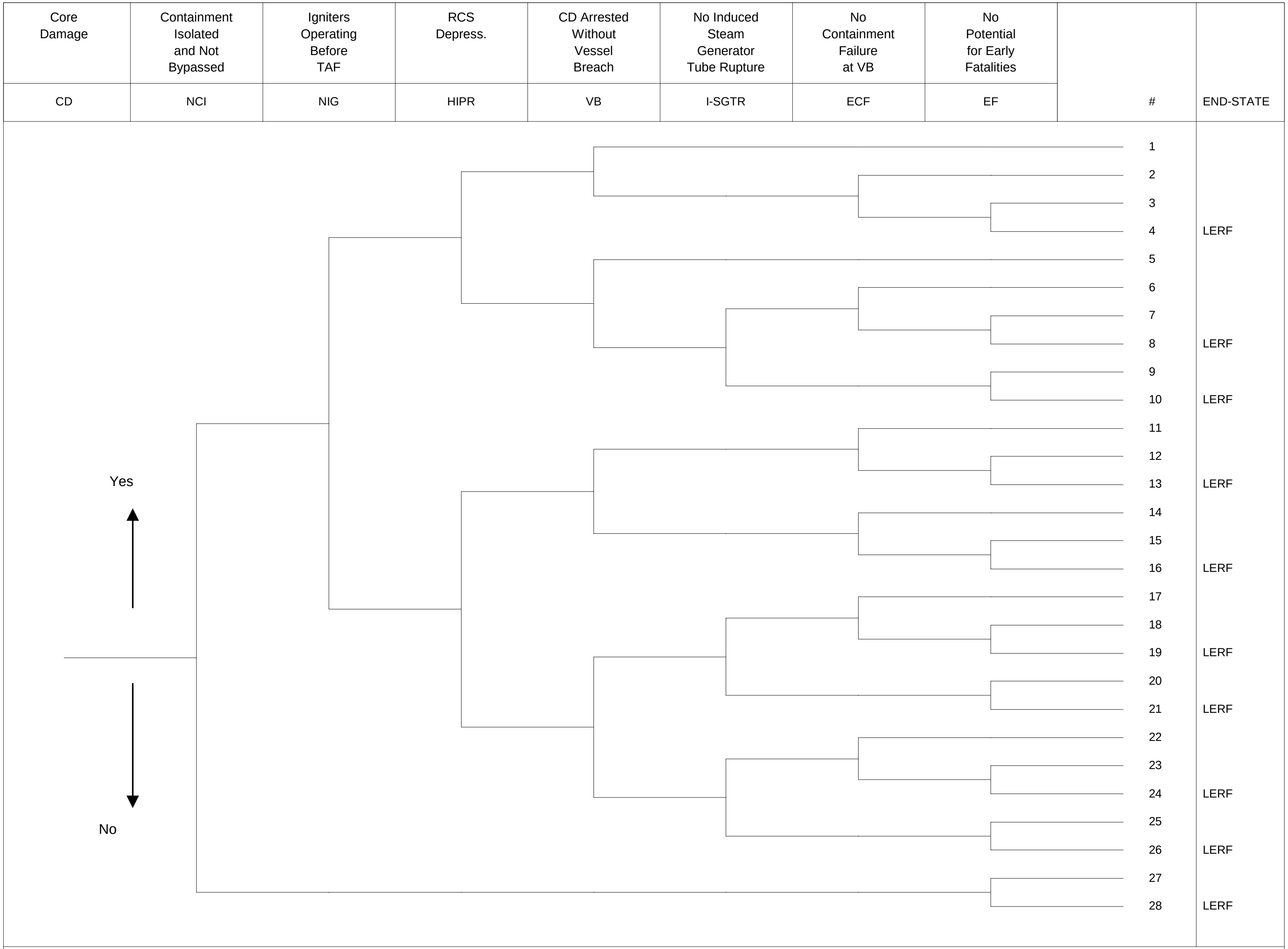
- RCS pressure
- Amount and temp of core debris exiting vessel
- Size of vessel failure
- Operability of containment sprays
- Operation of igniters
- Amount of ice at time of VB
- Amount of water in vessel cavity
- Configuration of cavity
- Structural strength of containment building

No Cont. Failure at or before VB

- In simplified treatment only RCS pressure and igniters are explicitly accounted for
  - If no Igniters and no VB then Prob. of CF = 0.04
    - i.e., prob. of CF before VB
    - If igniters operating, no CF before VB
  - If no igniters and
    - VB at low pressure (non-DCH) then Prob. of CF = 0.97
    - VB at high pressure then Prob. of CF = 1.0

No Cont. Failure at or before VB (cont.)

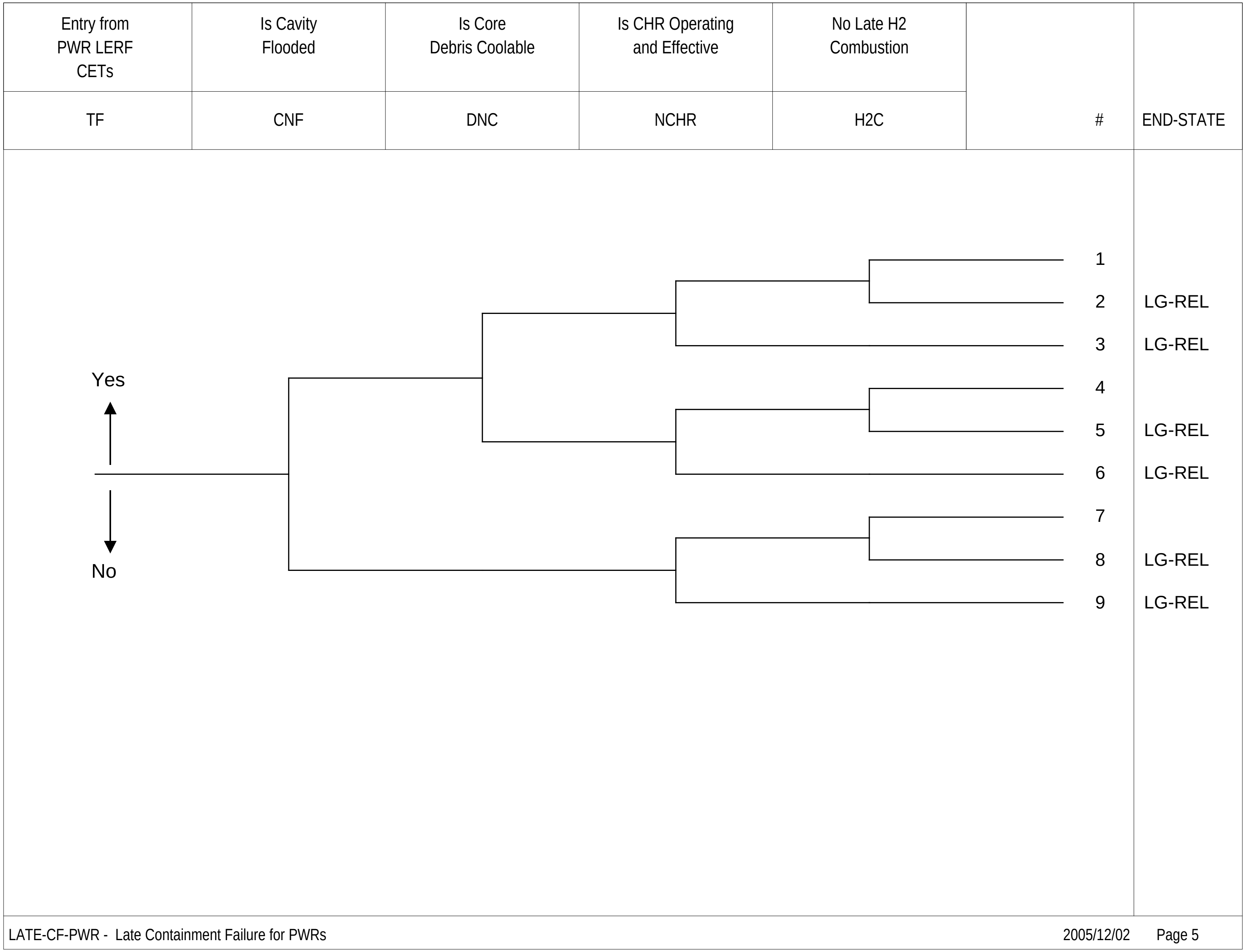
- If igniters are operating, CF still possible from other causes
  - VB at low pressure, prob. of CF = 0.01
    - In-vessel steam explosion
    - Ex-vessel steam generation
  - VB at high pressure, prob. of CF = 0.05
    - HPME can result in direct impingement of corium on containment wall



LERF-ICECOND - PWR Ice Condenser LERF CET (NUREG/CR-6595, Rev. 1) 2006/04/26 Page 10

PWR Late Containment Failure

- To address accidents where evacuation not effective
  - E.g., Seismic and high wind
- Considers only
  - Core concrete interaction
  - H2 combustion (Ice Condenser only)
  - Basemat penetration not included
    - Assumed to result in “small” release


Entry from PWR LERF CETs
Is Cavity Flooded
Is Core Debris Coolable
Is CHR Operating and Effective
No Late H2 Combustion
TF
CNF
DNC
NCHR
H2C
#
END-STATE
Yes
No
1
2 LG-REL
3 LG-REL
4
5 LG-REL
6 LG-REL
7
8 LG-REL
9 LG-REL
LATE-CF-PWR - Late Containment Failure for PWRs
2005/12/02
Page 5

Flooded Cavity

- Core debris falling on a dry vs flooded cavity
  - Flooded cavity produces steam
  - Dry cavity results in core concrete interaction
    - Produces noncondensibles
- Debris can be cooled in a flooded cavity if it is in a coolable geometry

Core Debris Coolable

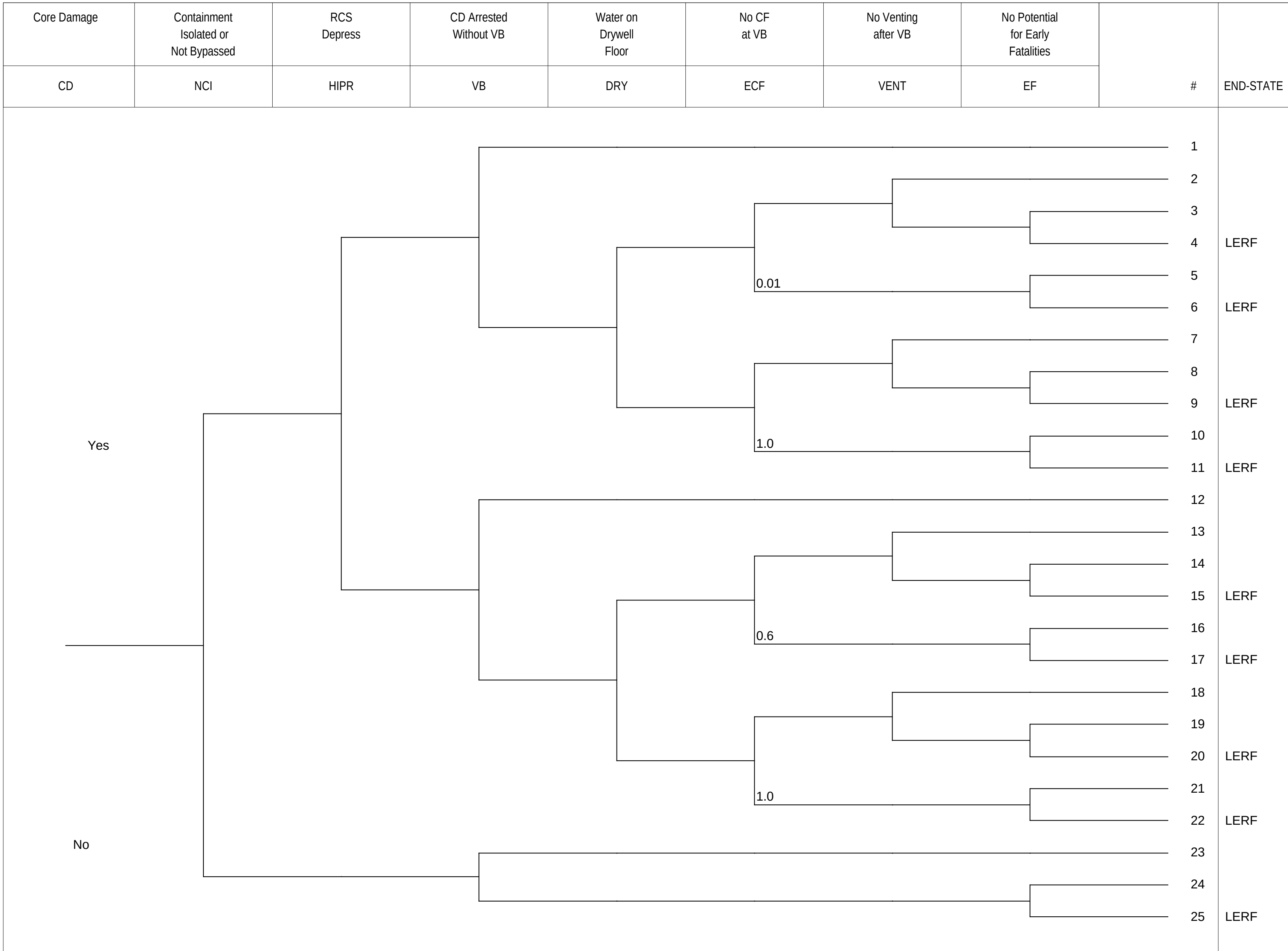
- Debris fragments (or forms very thin bed) then it is coolable
  - If coolable & water available then steam is produced
- Not coolable or no water then CCI produces noncondensible and combustible gases

CHR Operating and Effective

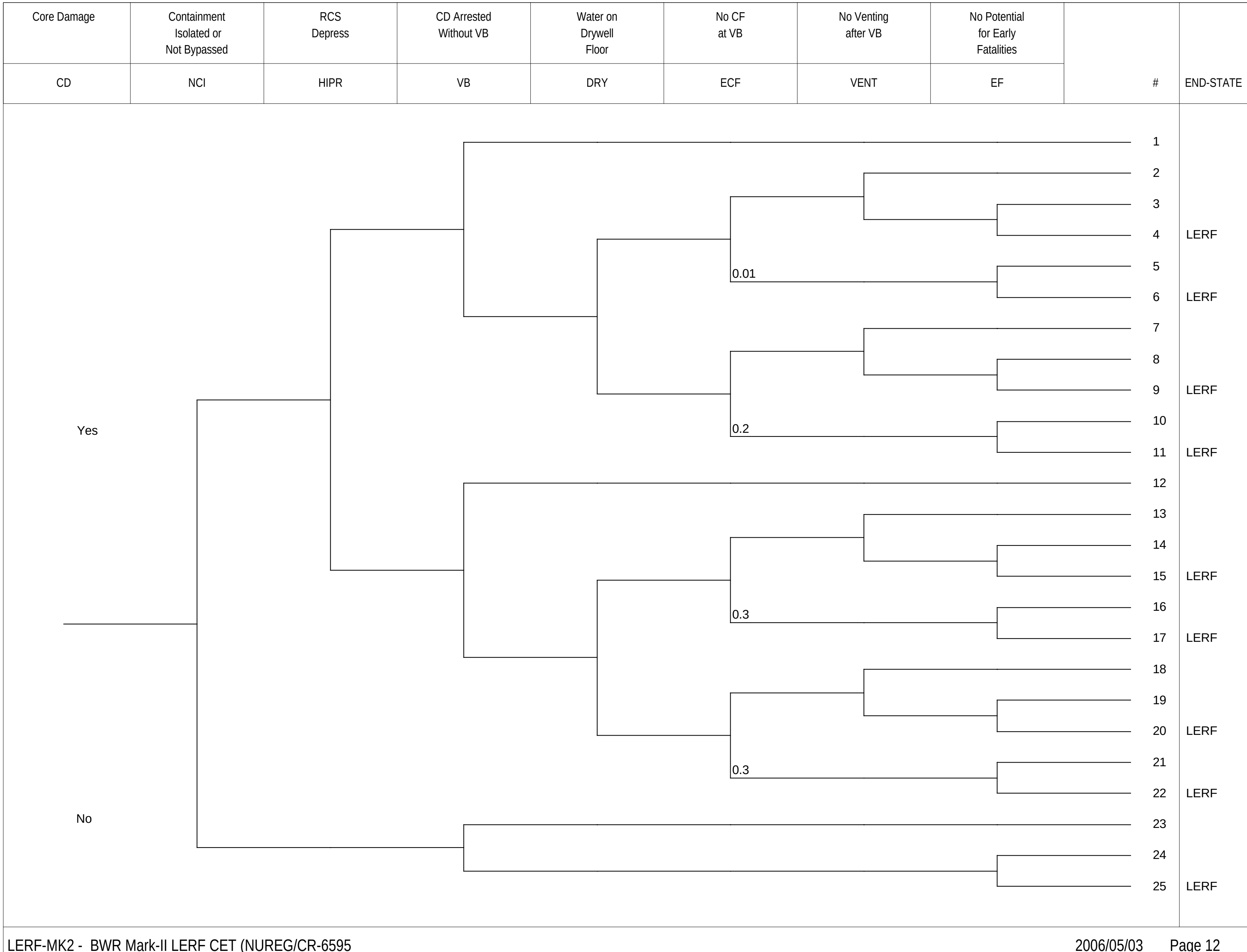
- Long-term operation of containment heat removal
  - Containment sprays
  - Two questions: Operating? Effective?
- If cavity is dry
  - Then core concrete interaction produces noncondensible and combustible gases
    - If CHR was operating, continued operation is questionable
- If cavity is flooded
  - Is core debris coolable?
    - Yes and CHR operating - then CCI does not occur and late CF prevented
    - Yes but CHR not operating – then eventual CF probable
    - No - then CCI occurs and CF probable

Late H2 Combustion

- Applies only to Ice Condenser
- Are igniters available?
- Did H2 combustion occur early?
- If No and No
  - Then CF (late H2 combustion) = 1.0
- If igniters are available
  - Then Late H2 Combustion = 0.0



LERF-MK1 - BWR Mark-I LERF CET (NUREG/CR-6595) 2006/05/03 Page 11


Core Damage
Containment Isolated or Not Bypassed
RCS Depress
CD Arrested Without VB
Water on Drywell Floor
No CF at VB
No Venting after VB
No Potential for Early Fatalities
CD
NCI
HIPR
VB
DRY
ECF
VENT
EF
#
END-STATE
Yes
No
0.01
0.2
0.3
0.3
1
2
3
4 LERF
5
6 LERF
7
8
9 LERF
10
11 LERF
12
13
14
15 LERF
16
17 LERF
18
19
20 LERF
21
22 LERF
23
24
25 LERF
LERF-MK2 - BWR Mark-II LERF CET (NUREG/CR-6595
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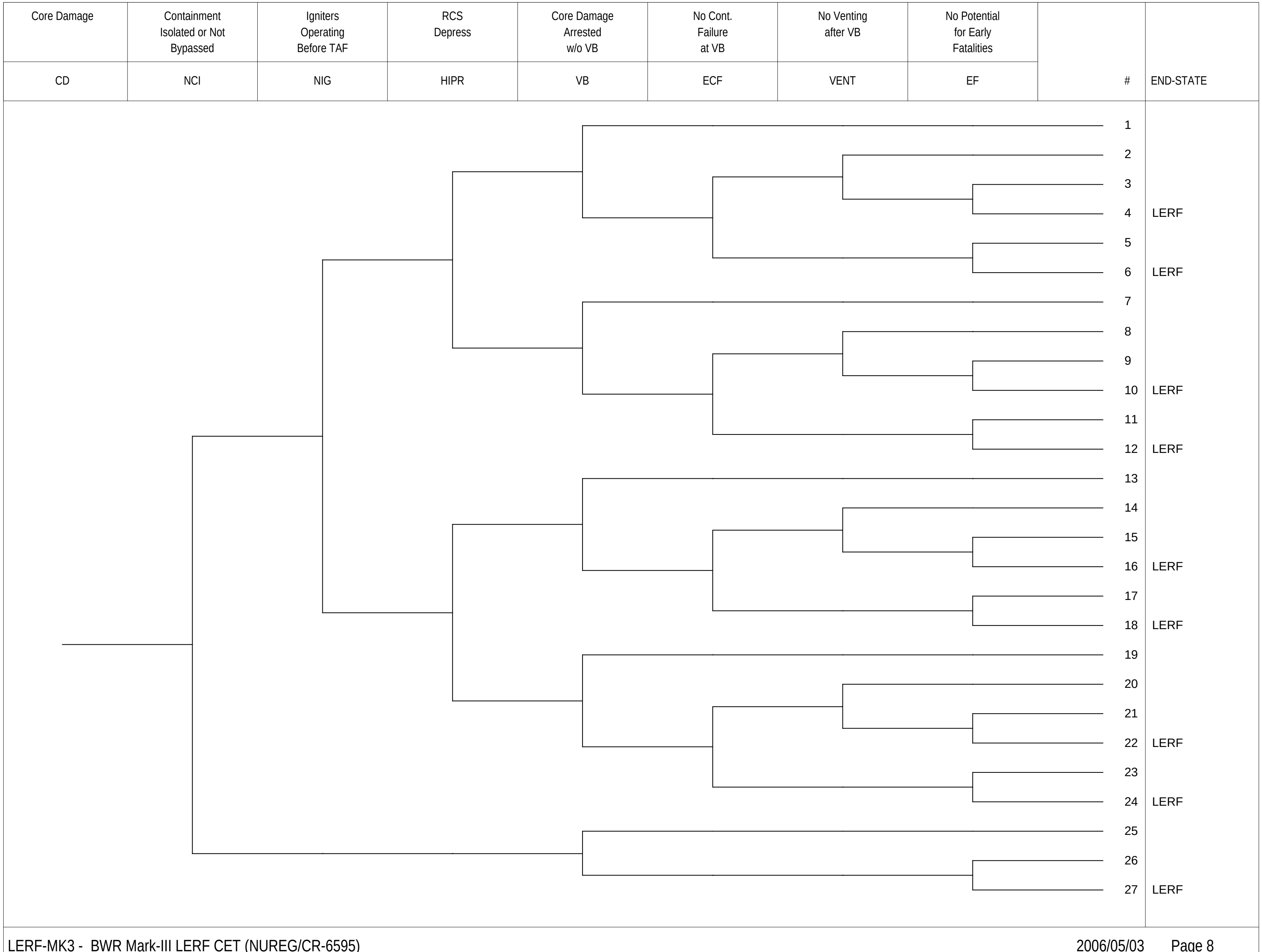
Core Damage
Containment Isolated or Not Bypassed
Igniters Operating Before TAF
RCS Depress
Core Damage Arrested w/o VB
No Cont. Failure at VB
No Venting after VB
No Potential for Early Fatalities
CD
NCI
NIG
HIPR
VB
ECF
VENT
EF
#
END-STATE
1
2
3
4 LERF
5
6 LERF
7
8
9
10 LERF
11
12 LERF
13
14
15
16 LERF
17
18 LERF
19
20
21
22 LERF
23
24 LERF
25
26
27 LERF
LERF-MK3 - BWR Mark-III LERF CET (NUREG/CR-6595)
2006/05/03
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Accident Progression Analysis (P-300)

Review

April 2015

www.inl.gov

Idaho National Laboratory

Review Questions

1. Why do a level-2 Analysis?
2. What are the major events of interest in a level-2 analysis?
3. What severe accident progression issues are important to vessel failure probability?
4. What severe accident progression issues are important to containment failure probability?
5. What are the major LWR containment types?

Review Questions (cont.)

6. What are some characteristics/design-features of each containment type (that are important from a severe accident analysis perspective)?
7. List the time frames of interest with respect to containment failure?
8. Each containment type incorporates a design feature to mitigate the hydrogen combustion failure mode. What are they?