Accident Progression Analysis (P-300)

Introduction

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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 *typically* defined as the uncovering of the top of active fuel (TAF)**
  - **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

- Risk-Informed Regulation and Review of PRA Basic concepts
- Overview of Level-1/2/3 PRA
- LWR Containment Designs
- Phenomena Affecting Vessel Integrity
- Phenomena Affecting Containment Integrity
- Containment Event Tree Development
- Phenomenological Modeling Capabilities
- Radionuclide Release and Transport
- Level-2 PRA Integration and Quantification
- Example Level-2 Analysis

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, An Approach for Estimating the Frequencies of Various Containment Failure Modes and Bypass Events, January 1999.
  – 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
- CFF Containment Failure Frequency
- CHF Critical Heat Flux
- CHR Containment Heat Removal
- CRD Control Rod Drive
- CS Cutset
- 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 Acciden
- 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
- 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)

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Accident Progression Analysis

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

May 10-12, 2005 - Rockville, MD

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 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

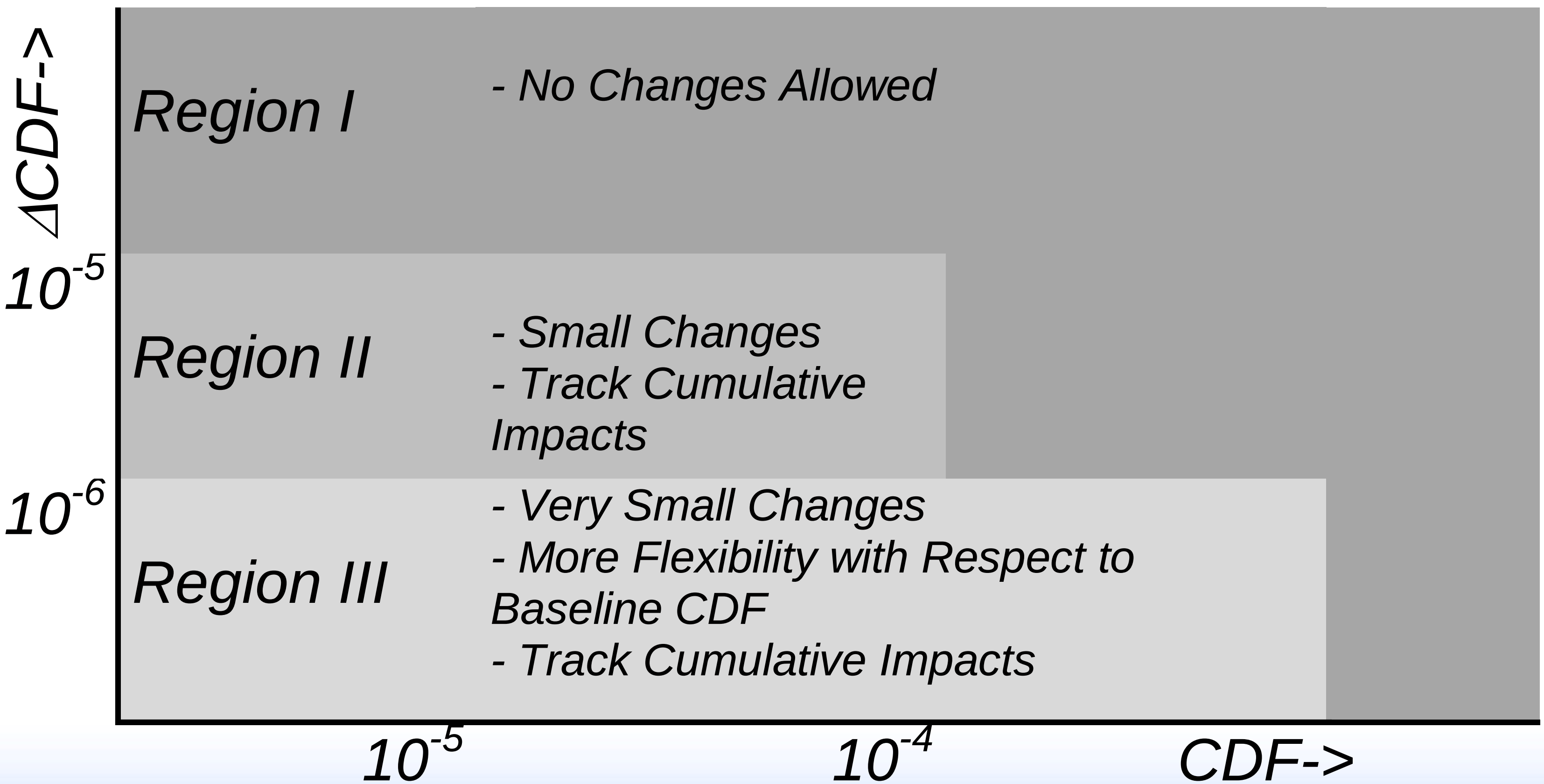
  #4. Proposed increases in CDF or risk are small and consistent with Commission's Safety Goal Policy Statement

  - Use of CDF <u>and</u> LERF as bases for PRA acceptance guidelines is an acceptable approach to addressing Principle 4.

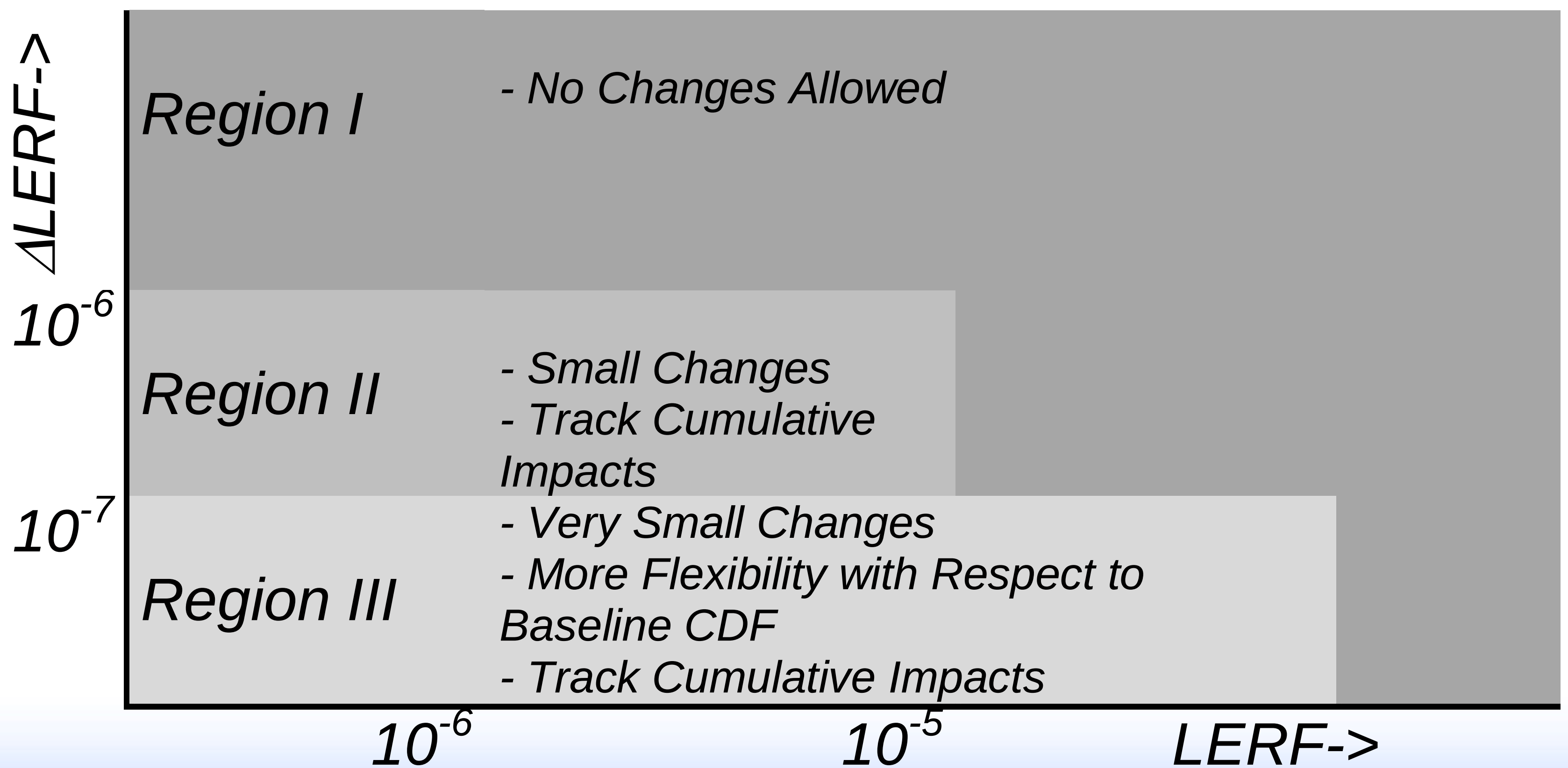
Large Early Release Frequency (LERF)

- 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 - uncovery of top of active fuel (UTAF - common definition but not universal), i.e., beginning of core degradation
- 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**

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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)**

F = $\lambda\varphi$

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

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

LEVEL 2

ACCIDENT PROGRESSION ANALYSIS

Models Progression of Severe Accident (APET or CET)

SOURCE TERM ANALYSIS

Parametric Information About Fission Product Transport and Removal

> *Frequency of containment failure and release of radioactive material*

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

> *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)

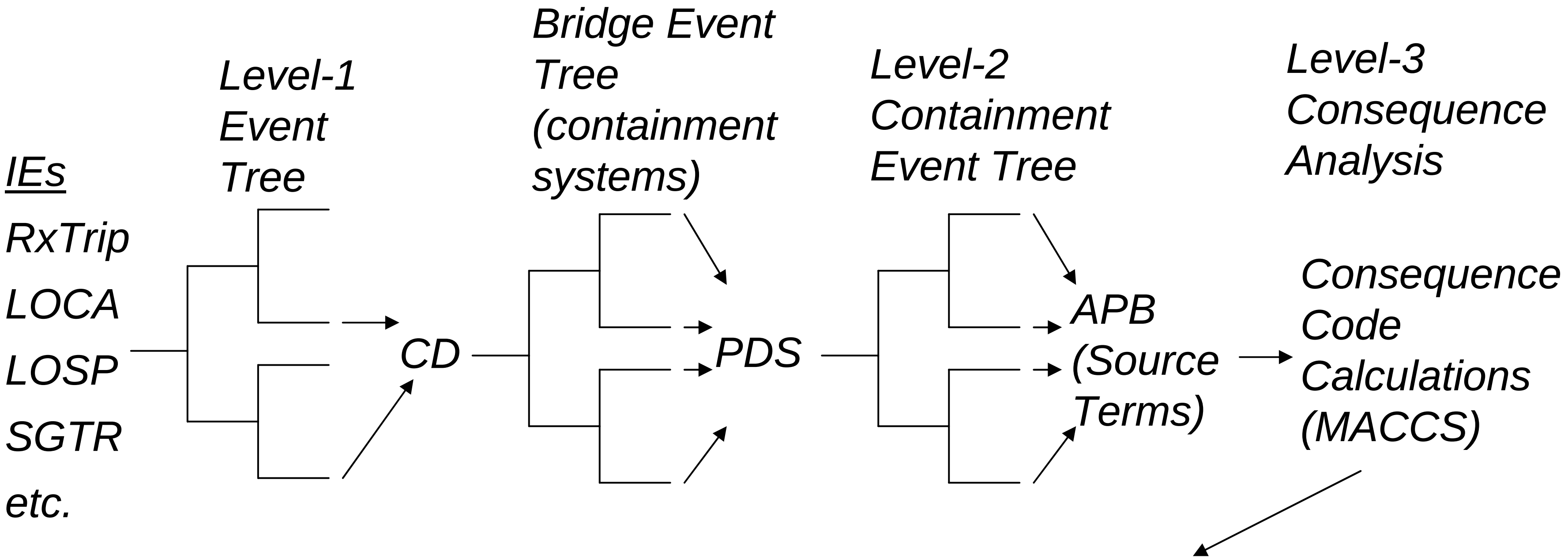
2. Overview of PRA

May 10-12, 2005 - Rockville, MD

Session Objectives

- **To understand the PRA framework**
  - **Level-1, -2 and –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.*

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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>

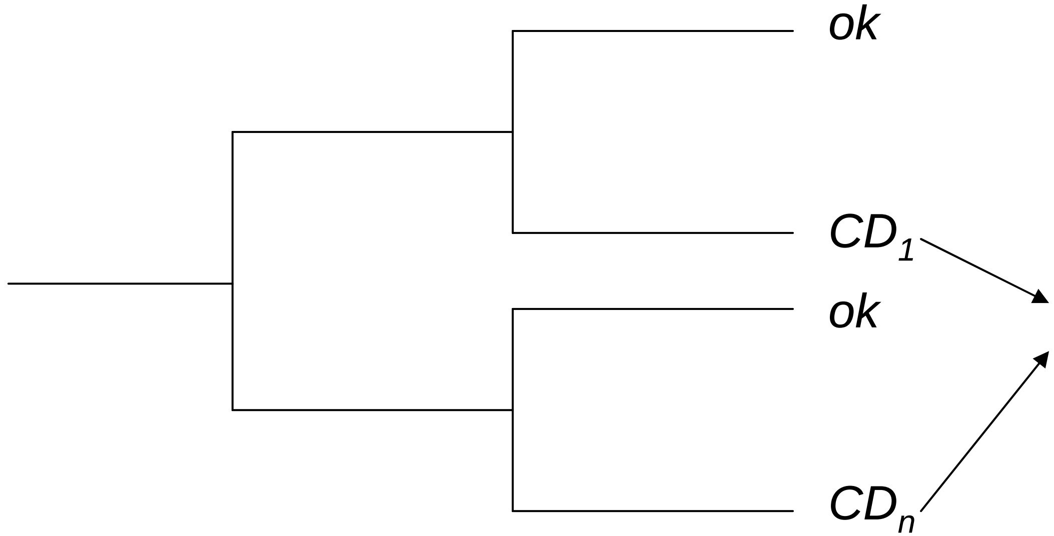
RxTrip

LOCA

LOSP

SGTR

etc.



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

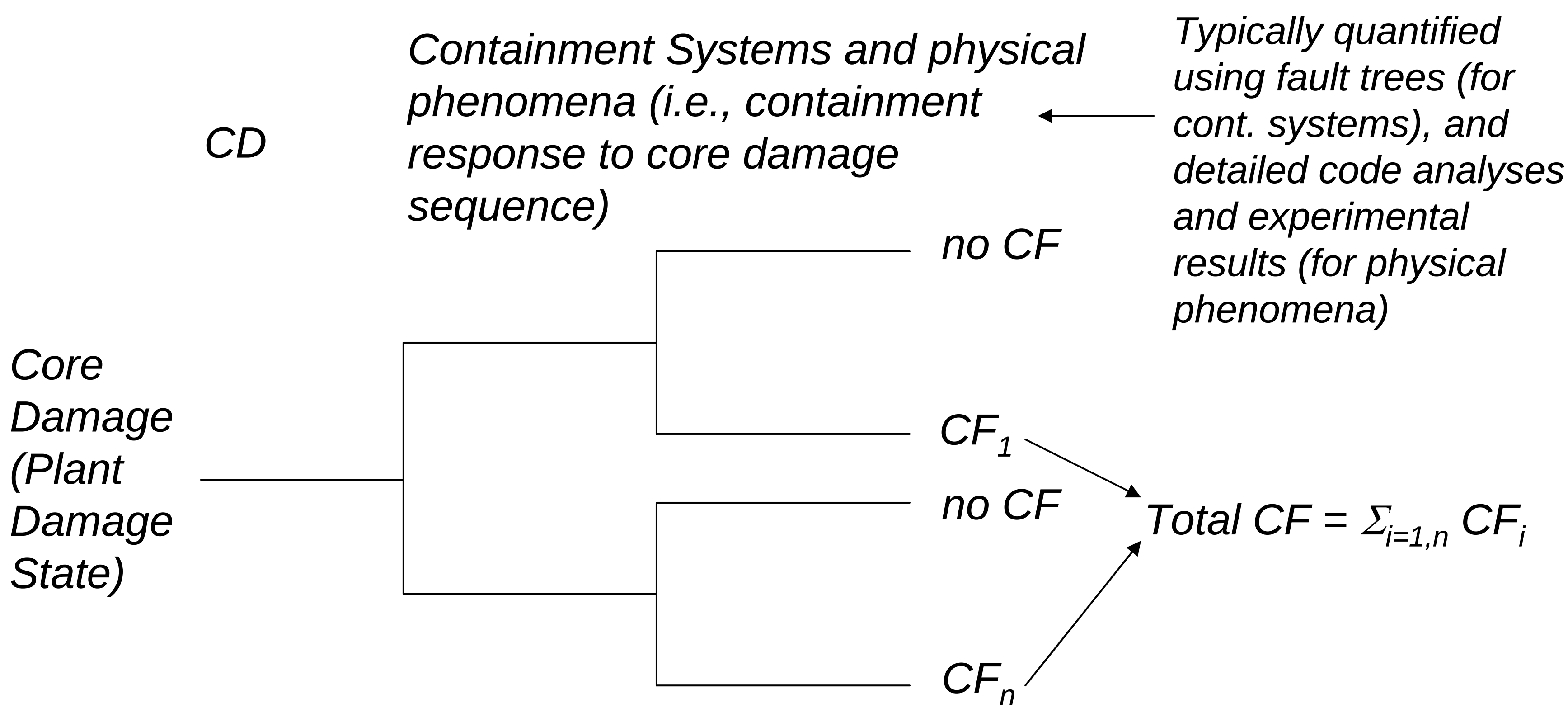
Purpose of Level-2 PRA Analysis

- 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 when coolant level drops below top of the active fuel in the reactor core)
  - 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)



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)

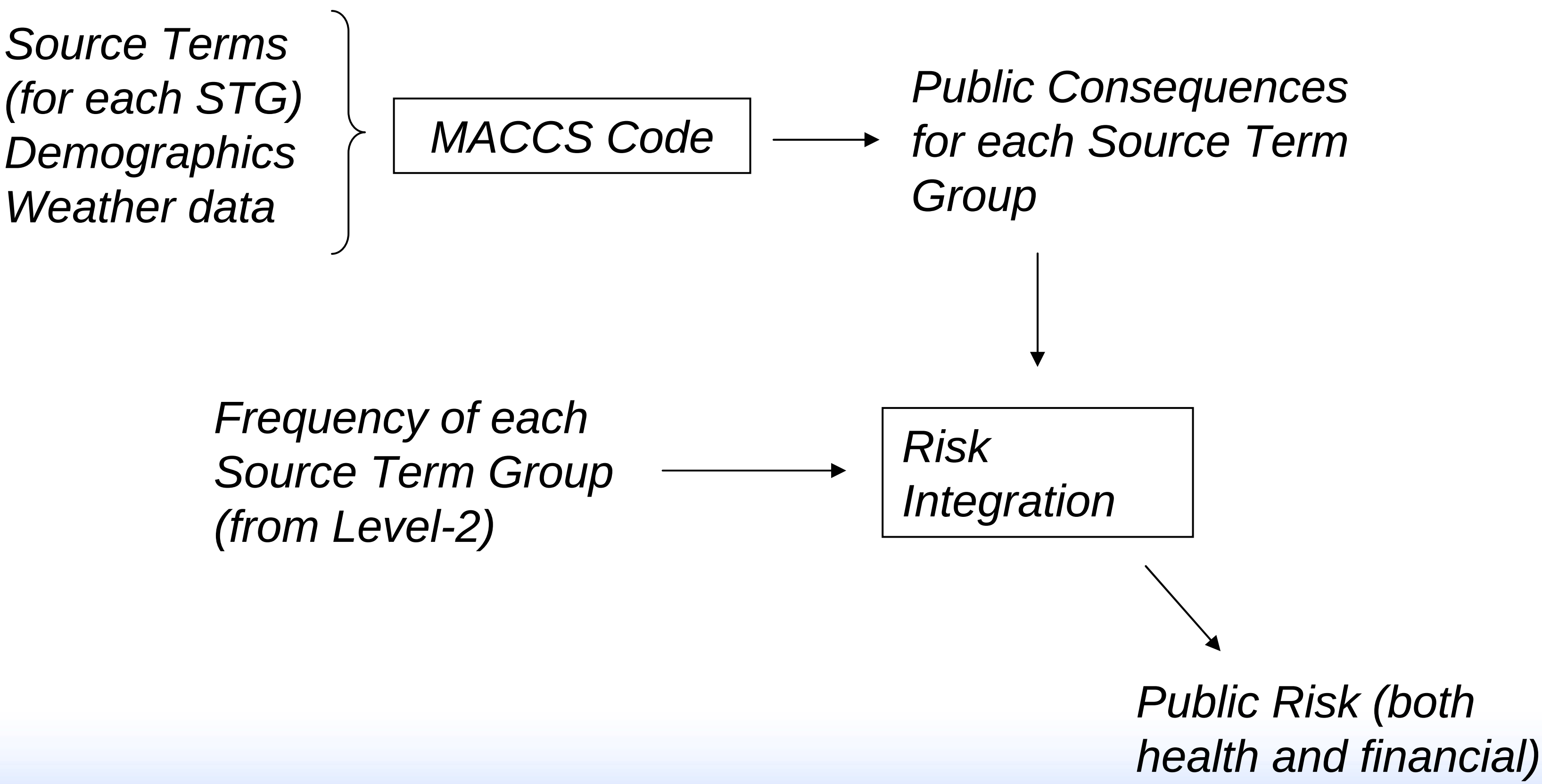
Purpose of Level-3 PRA Analysis

- 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



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:

$$P(a|b) \neq P(a)$$

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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**

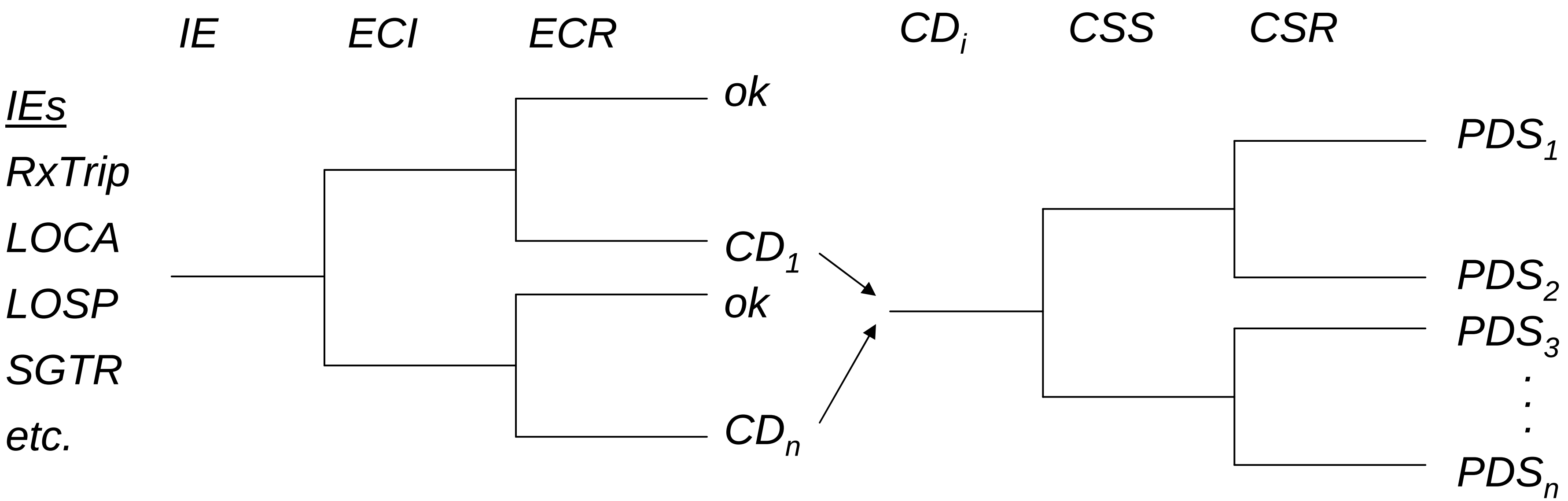
Plant Damage States (PDS) Framework Used As Input to Level-2 (from Level-1)

- 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

Level-1 Event Tree

Bridge Event Tree Appends Containment System Models to Level-1 ET



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

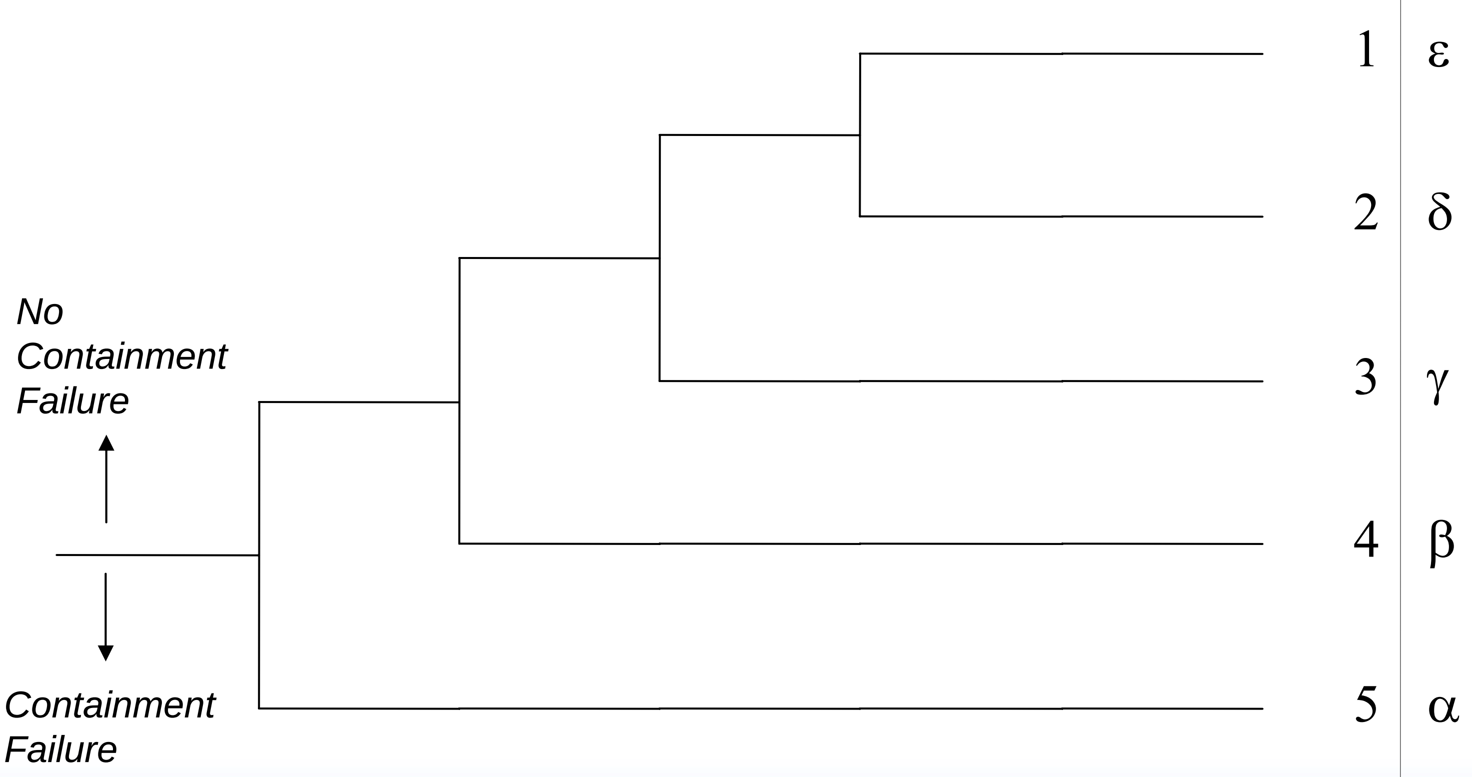
Level-2 Analysis Assesses Containment Response to Each PDS

- 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

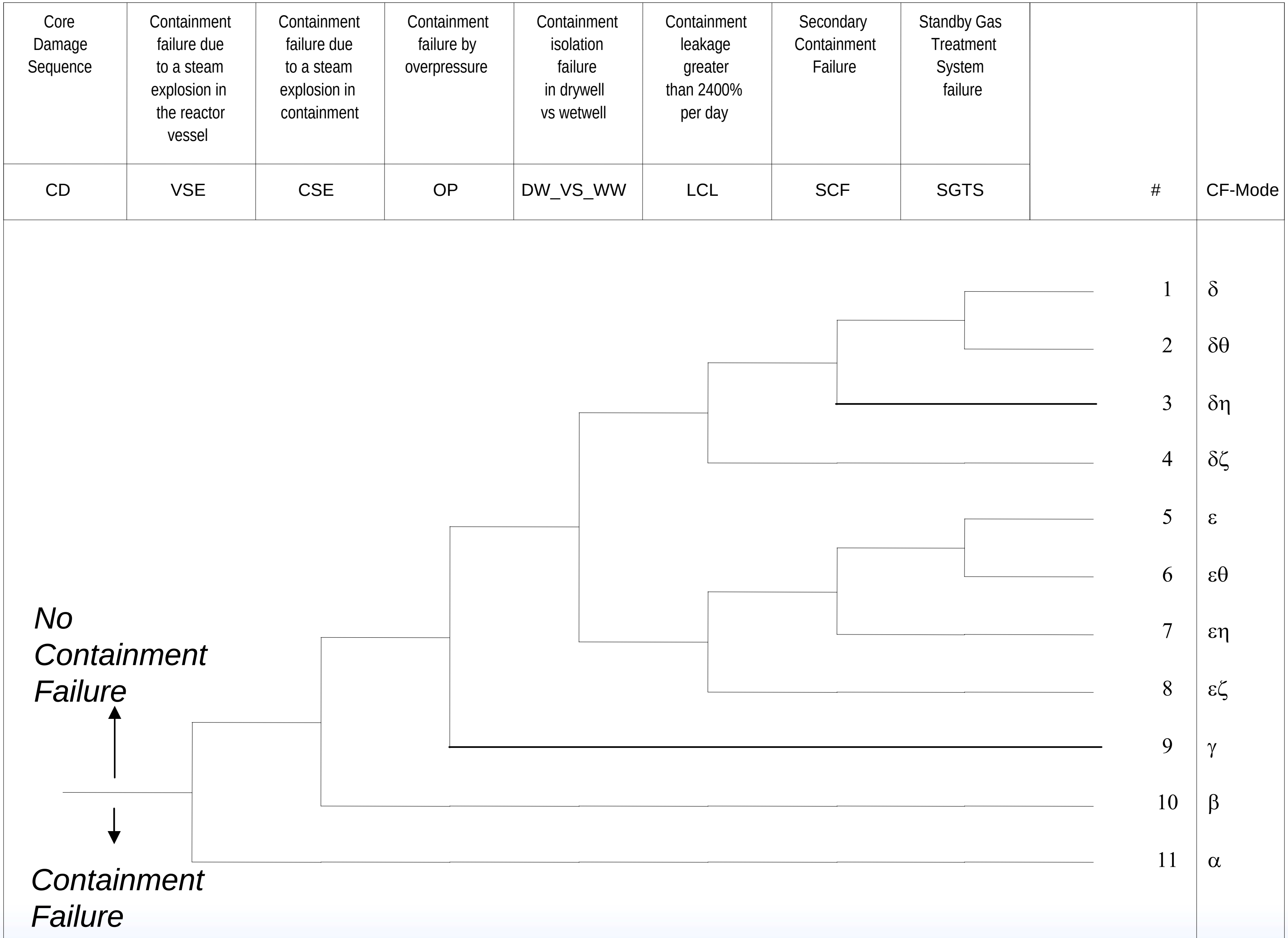
Two General Techniques for Level-2 Modeling

- 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) 2001/05/24 Page 4



WASH-1400-BWR-CET - BWR Containment Event Tree (WASH-1400) 2001/05/24 Page 3

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

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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 H_2 produced by core concrete interaction to H_2 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

May 10-12, 2005 - Rockville, MD

Session Objectives

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

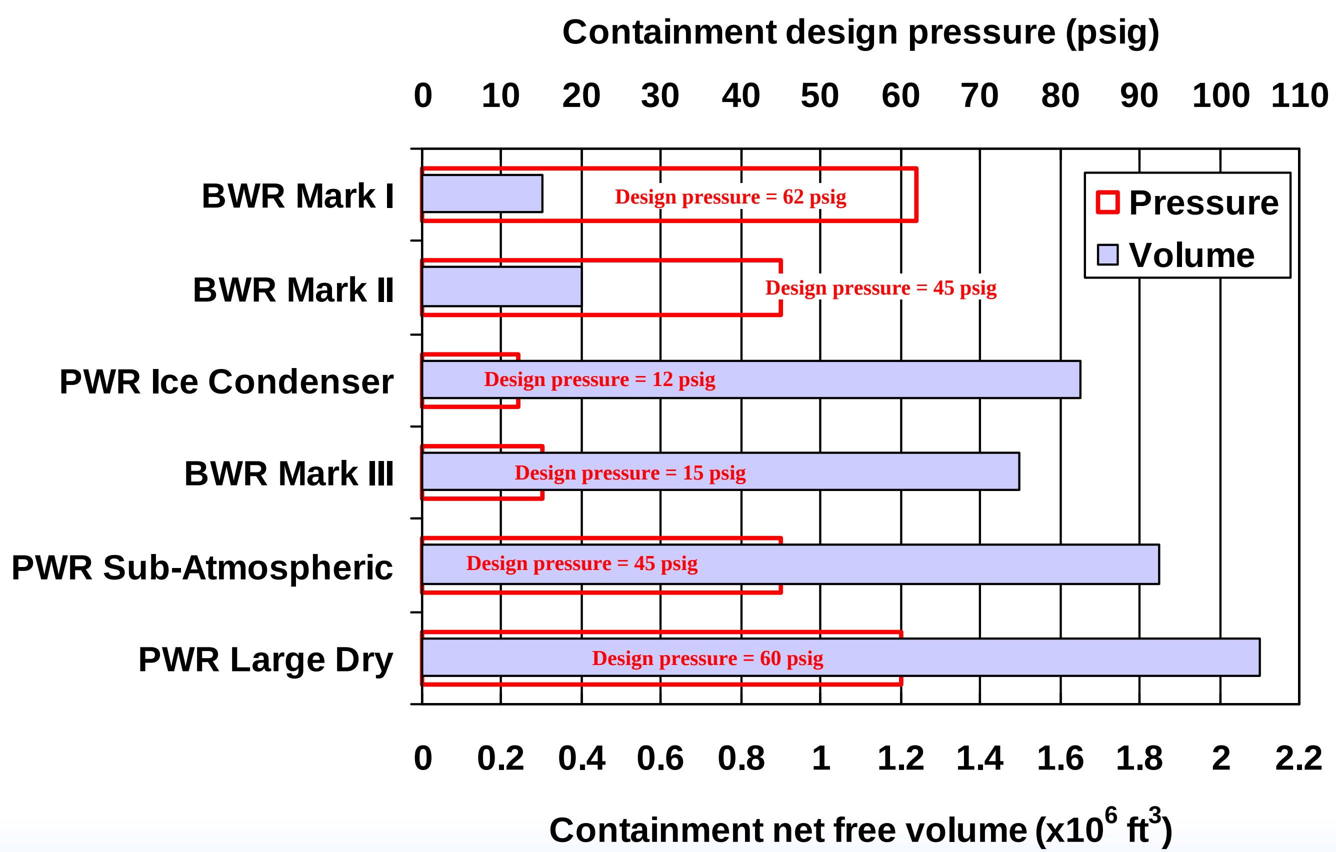
Six Major Types of LWR Containment Designs

- **Boiling Water Reactors (BWRs)**
  - **Mark I (e.g., Peach Bottom 2&3, Cooper)**
  - **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)**
- **Design variations within each group**

Significantly Larger Number of Dry Containments

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

Containment Free Volumes and Design Pressures Differ



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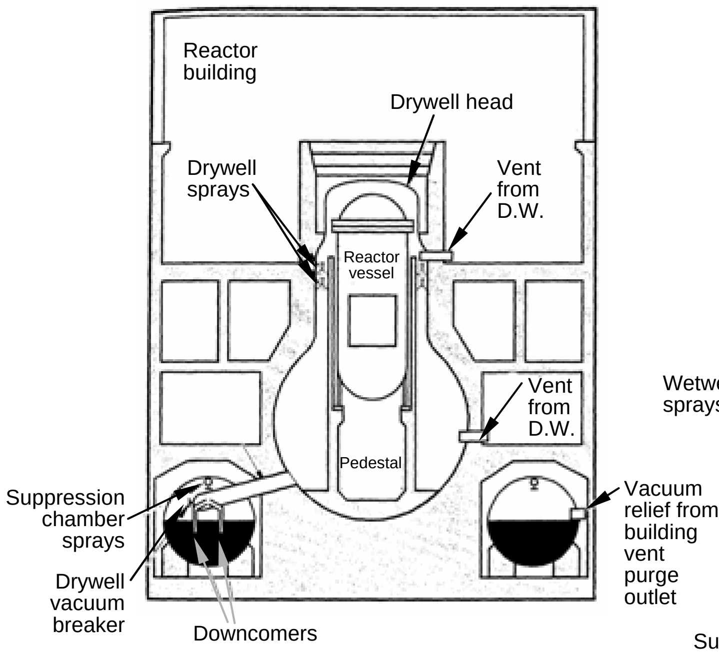
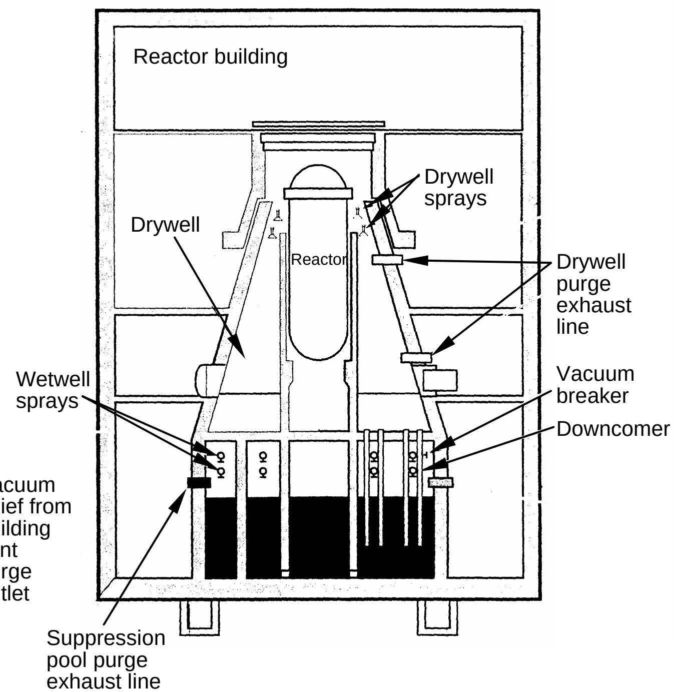
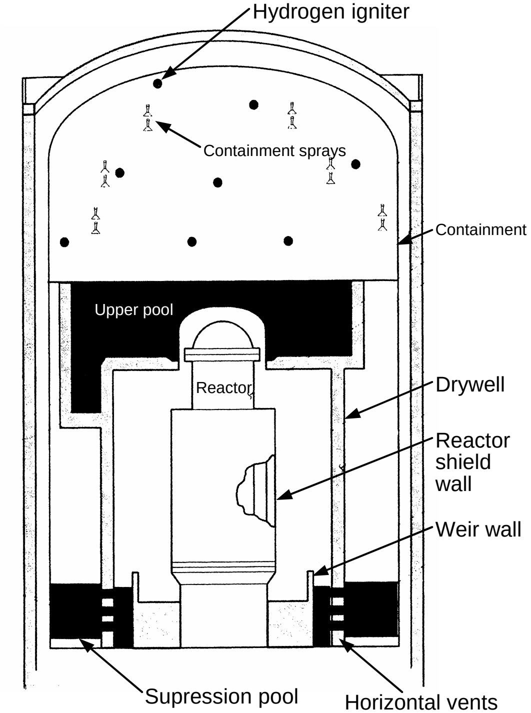
BWR Containment Designs Differ



Mark I

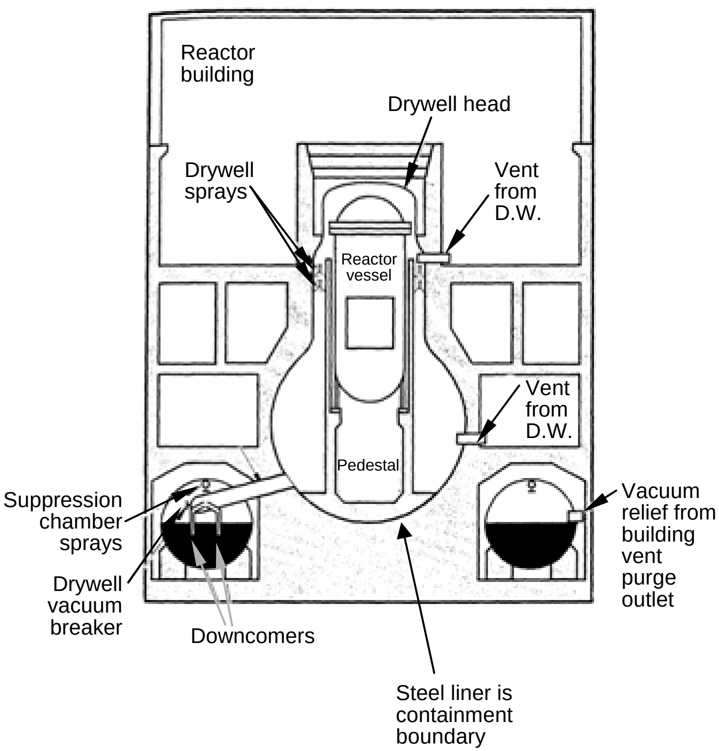


Mark II



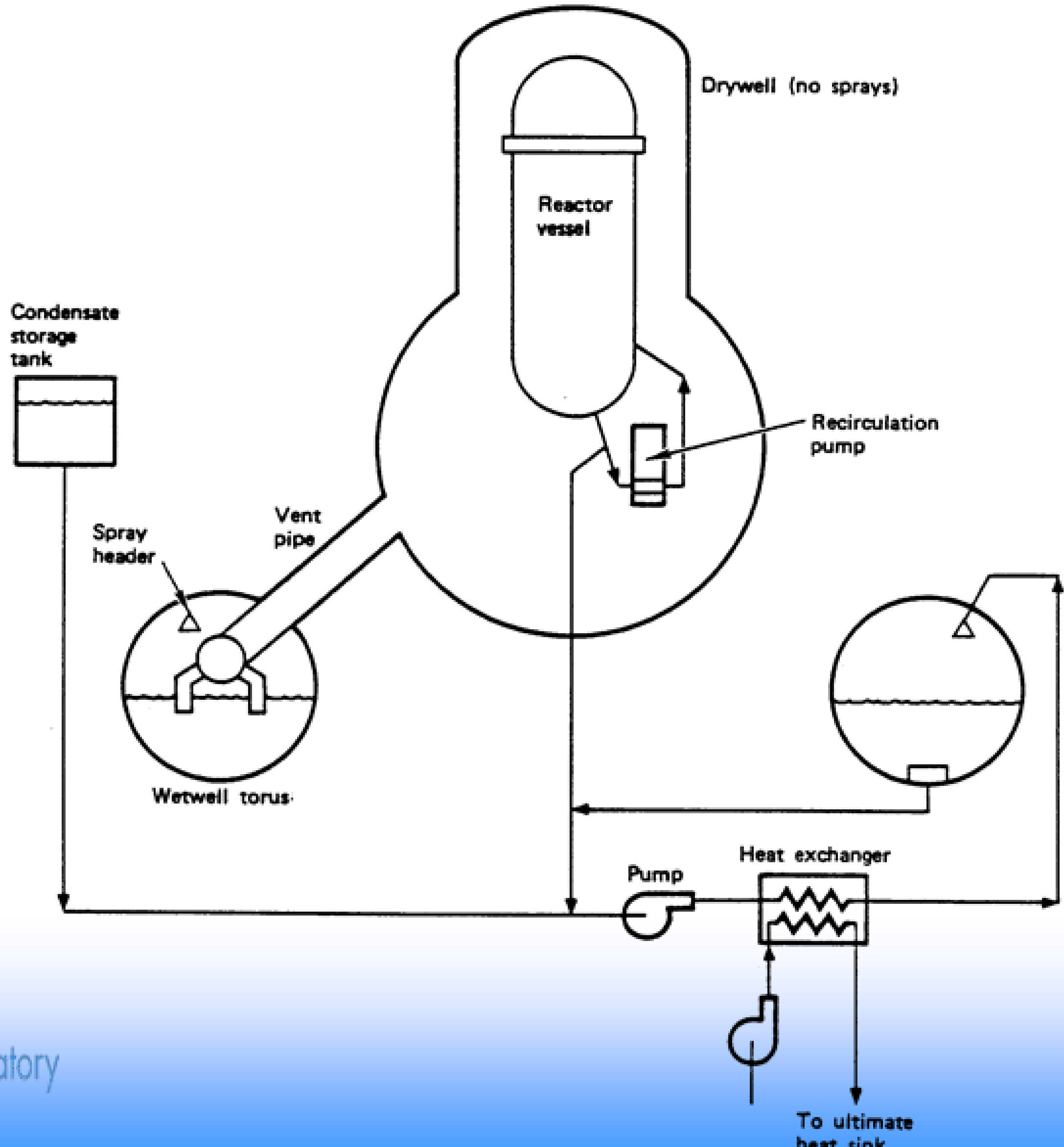
Mark III

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 (H_2) combustion

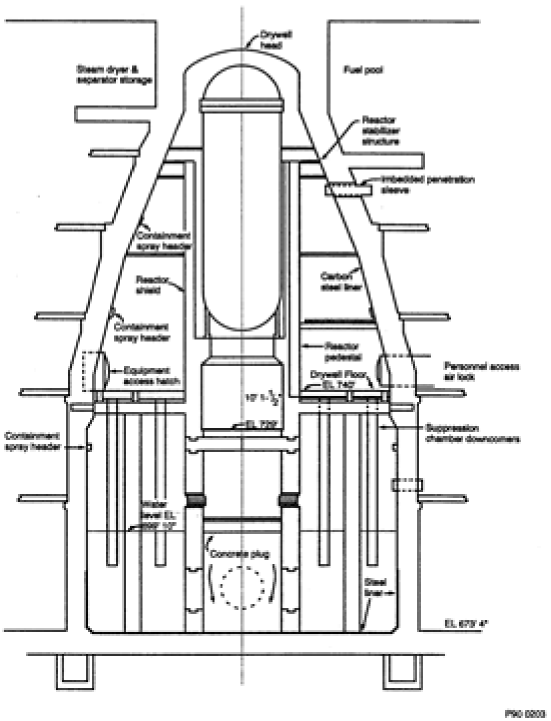
Mark I Containment Heat Removal Relies Primarily on Suppression Pool Water

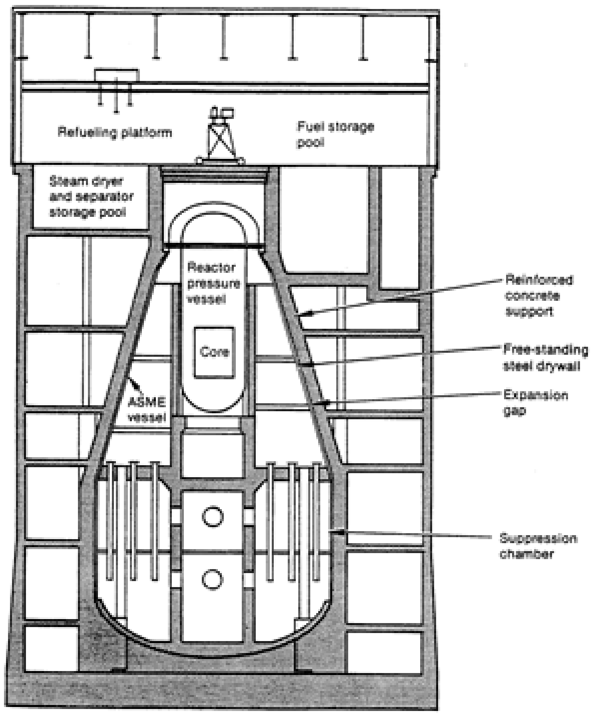


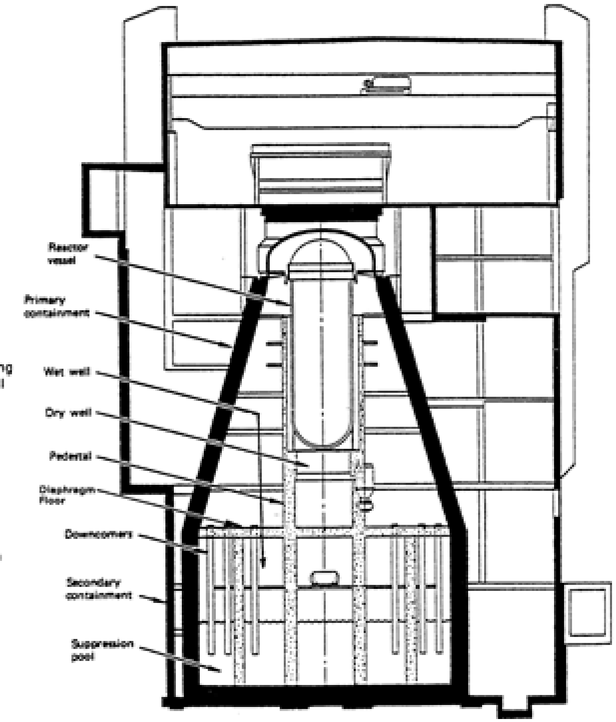
Mark II Design More Unified than Mark I Design

- **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 H_2 combustion**

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







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

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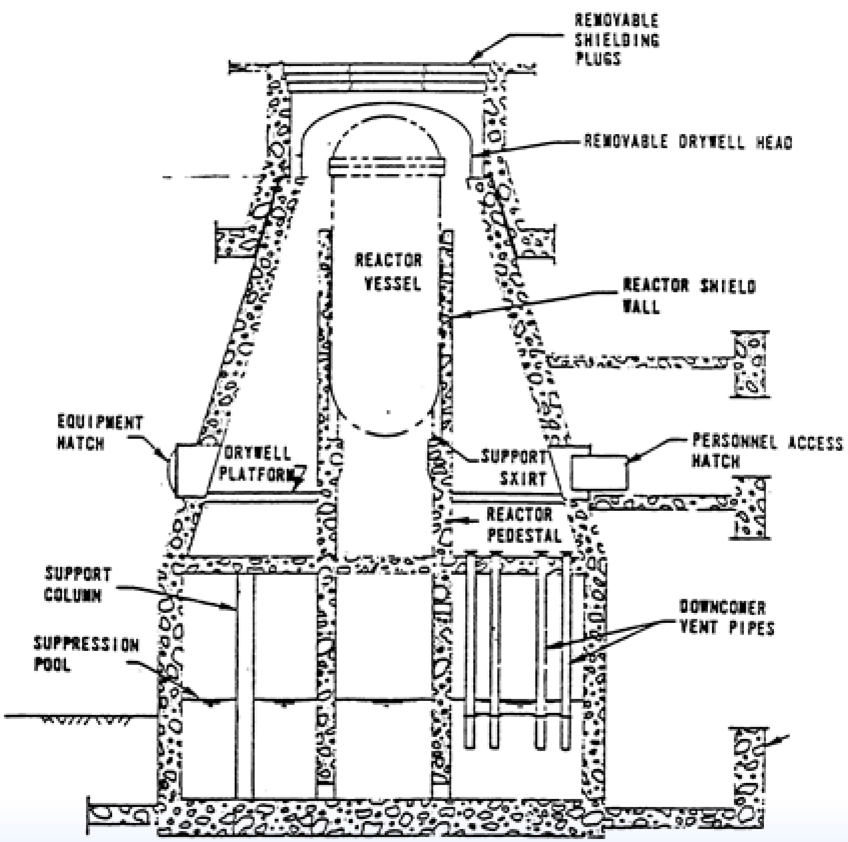
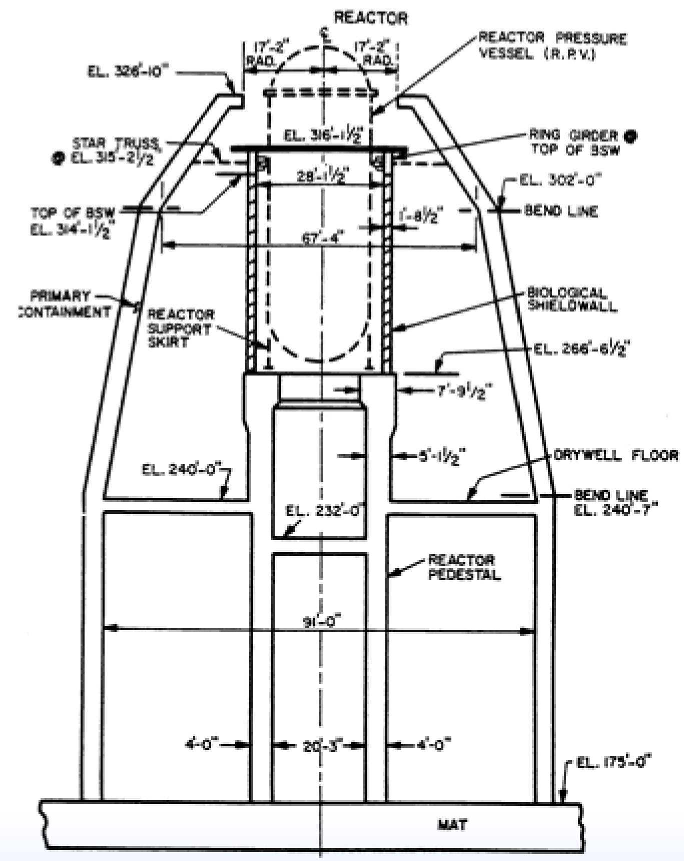
Mark II Design More Unified than Mark I Design (continued)



Susquehanna Units 1 & 2



Nine Mile Point 2

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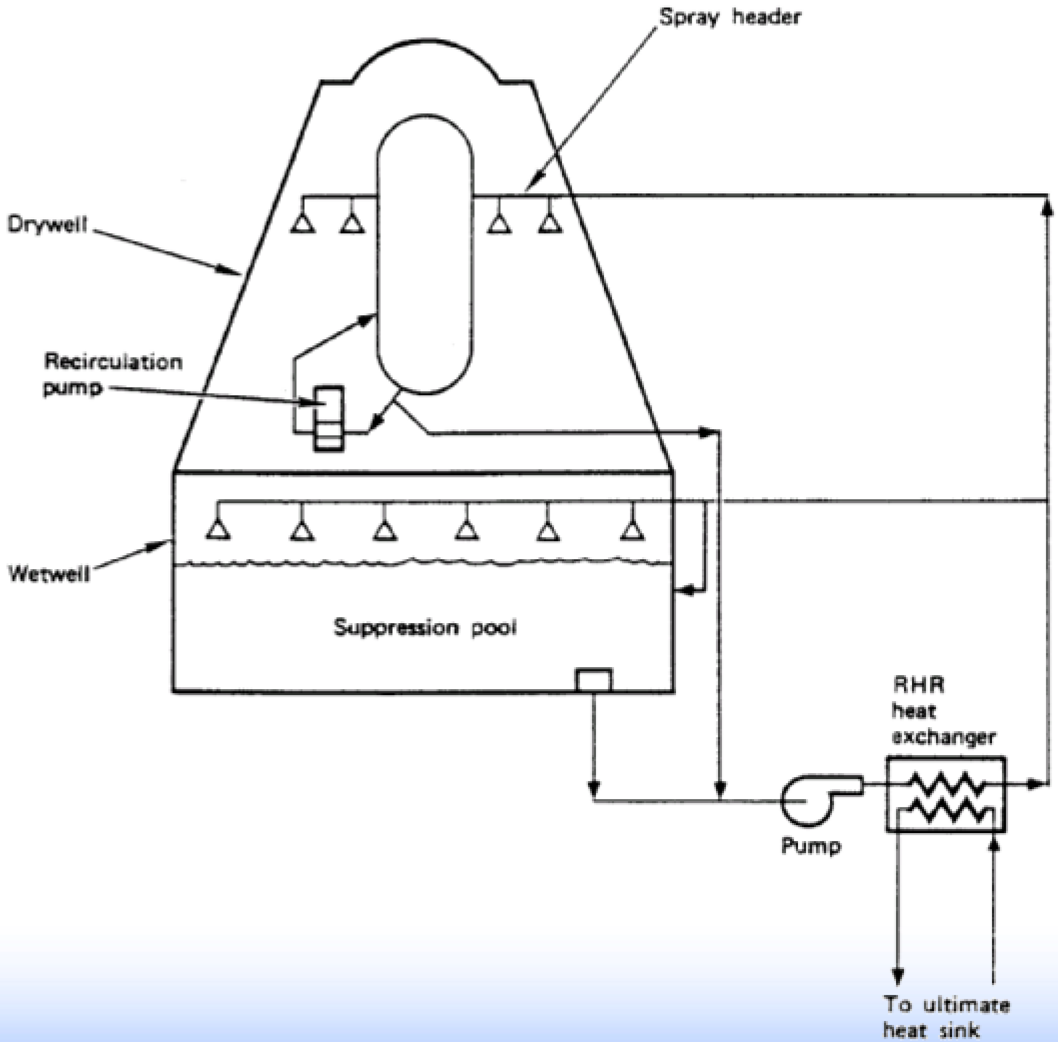
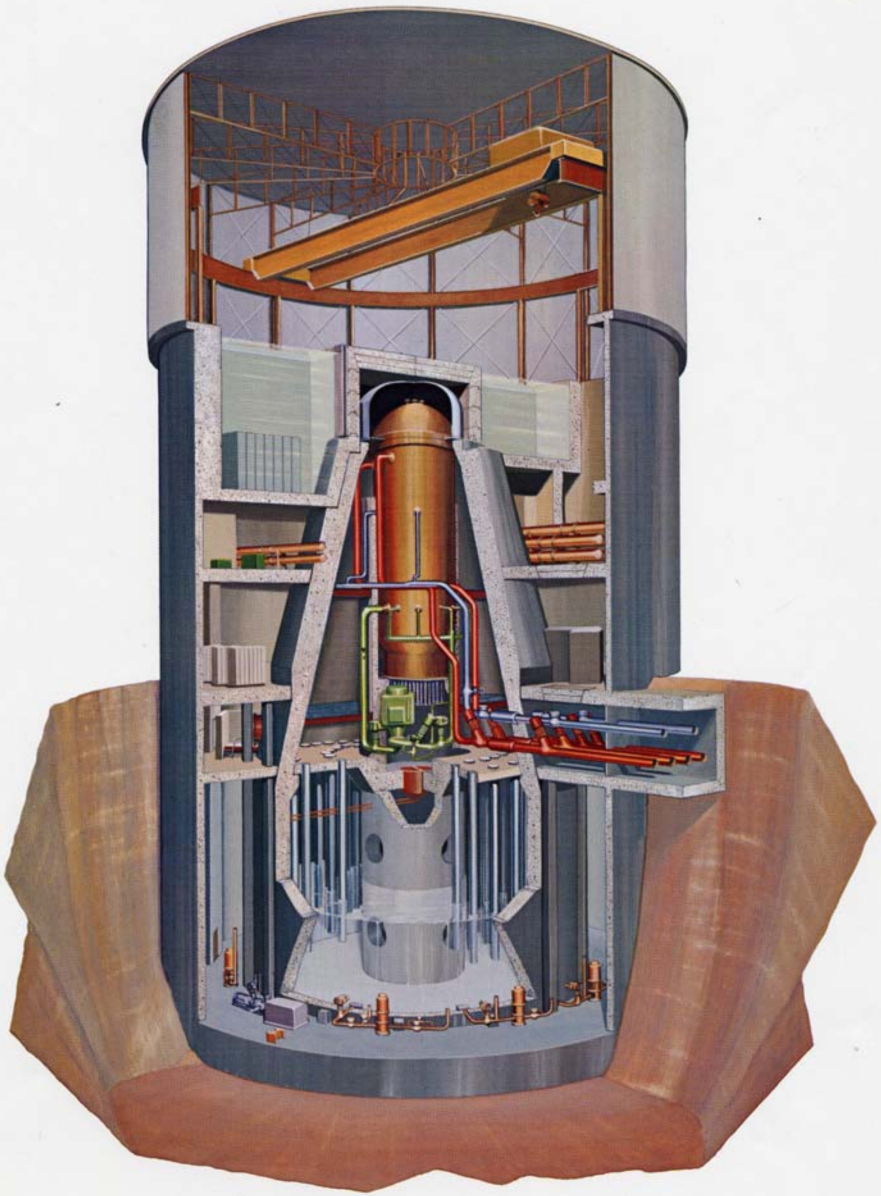
Containment Heat Removal for Mark II Containment



BWR CONCRETE CONTAINMENT



GENERAL ELECTRIC

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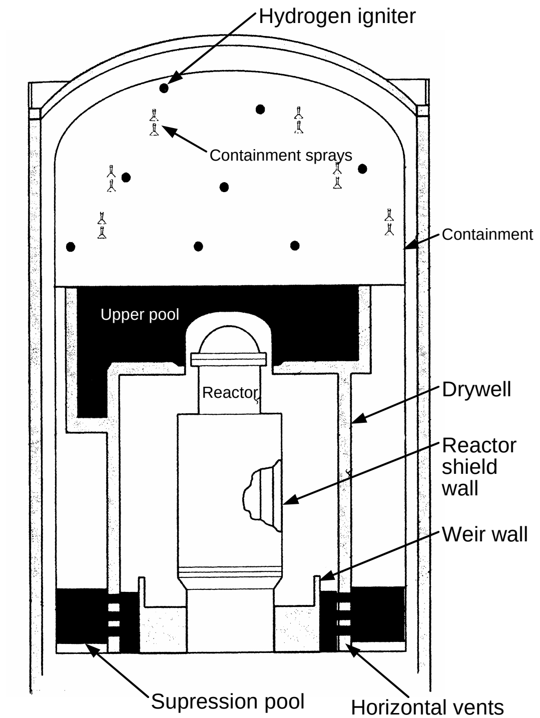
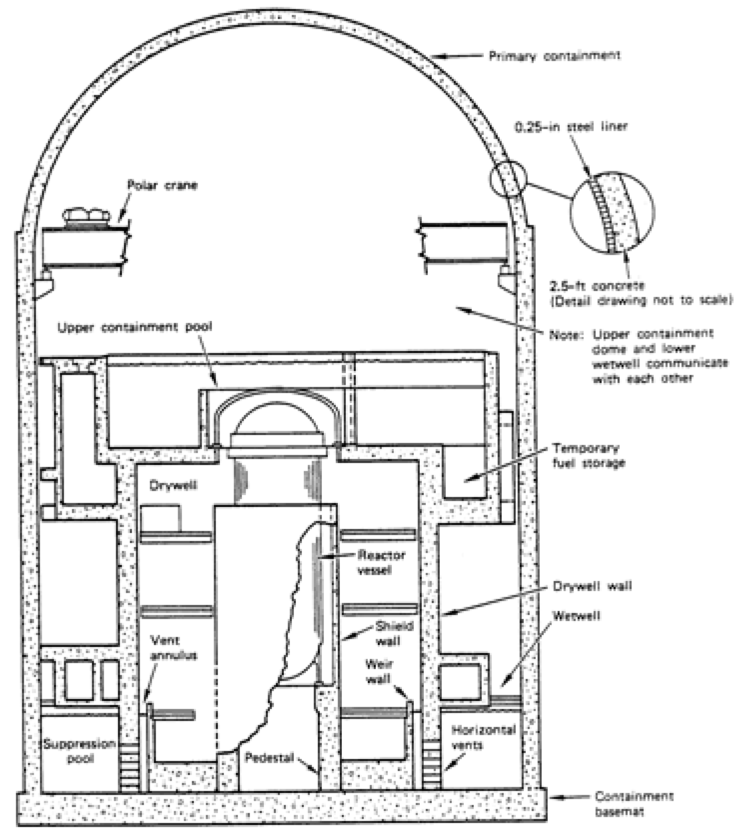
Mark III Dramatically Differs from Mark I and II Designs

- 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

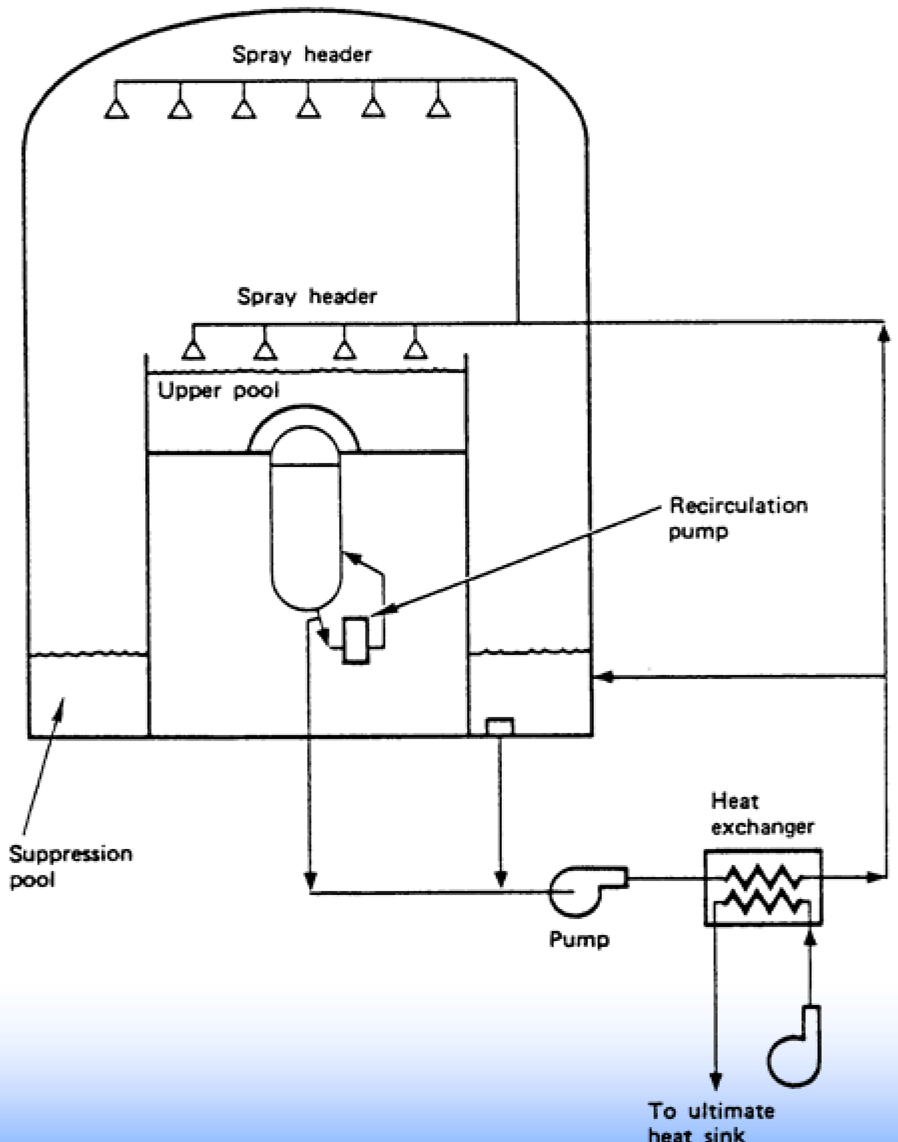




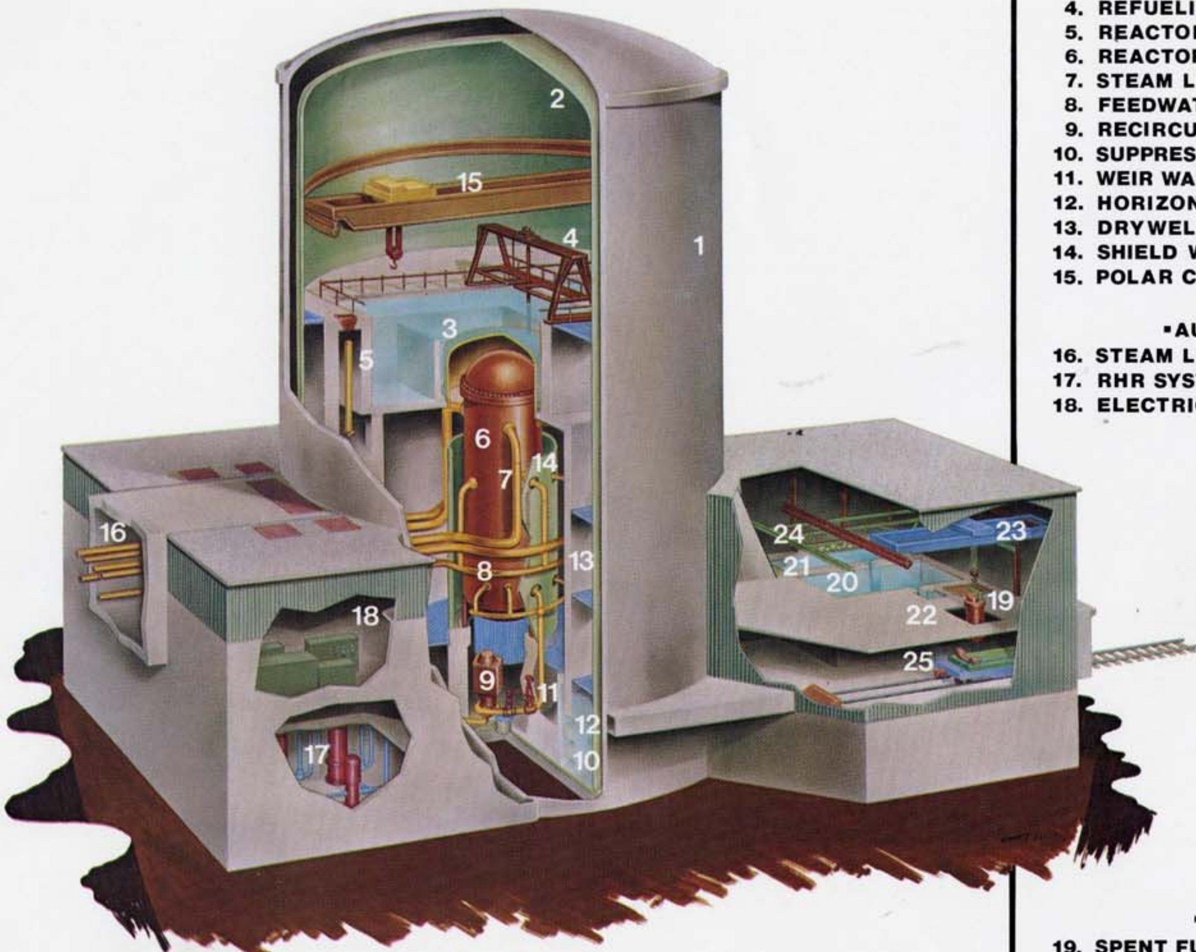
Free standing steel structure

Reinforced concrete

Mark III Containment Heat Removal Accomplished via Sprays and Suppression Pool



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

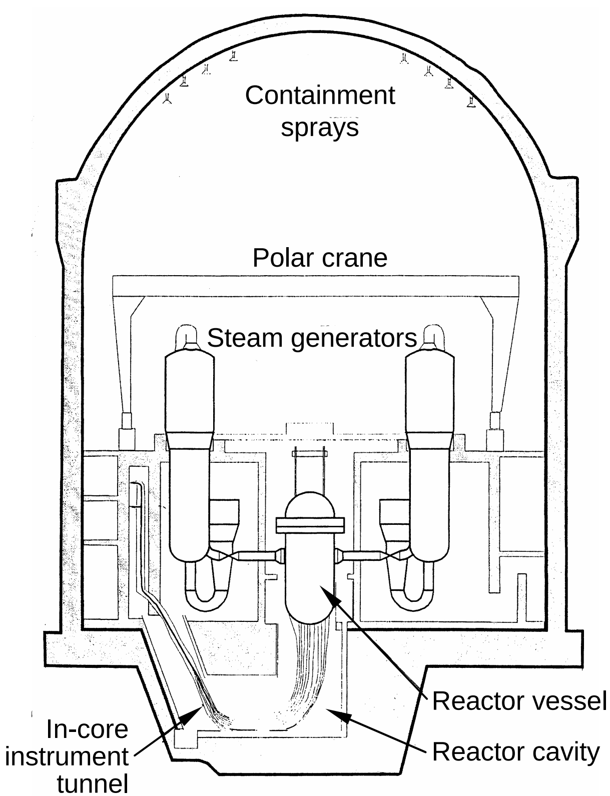
▪FUEL BUILDING▪

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

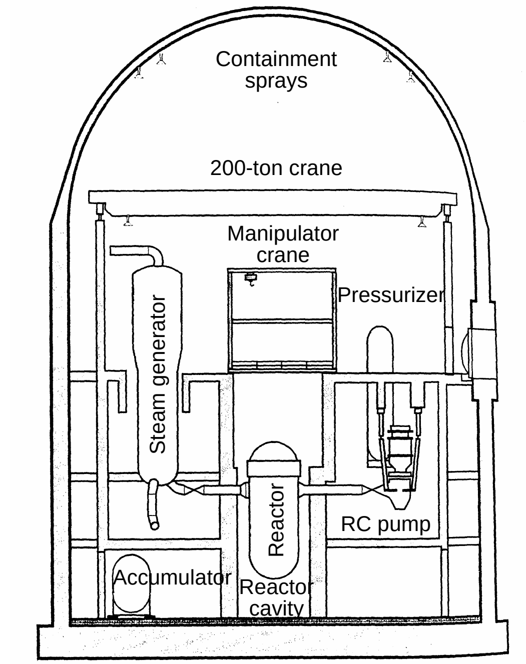
GENERAL ELECTRIC

GEZ-4385.1

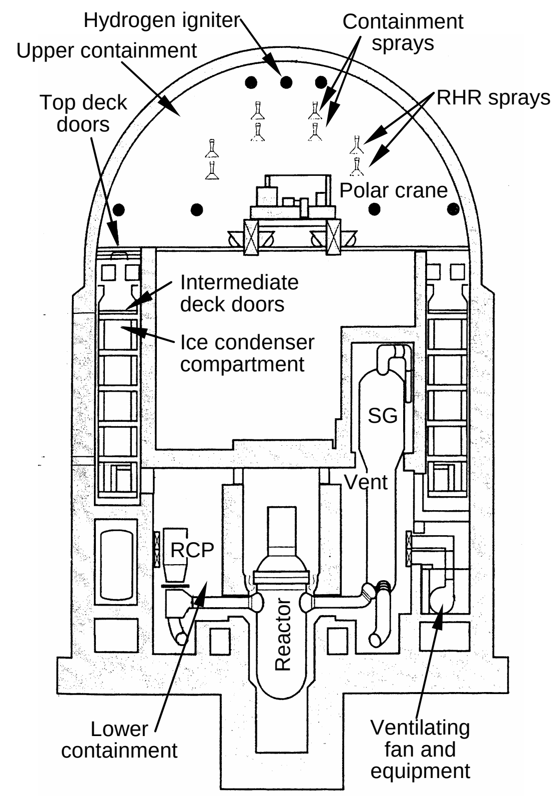
PWR Containment Designs Differ



Large dry



Subatmospheric

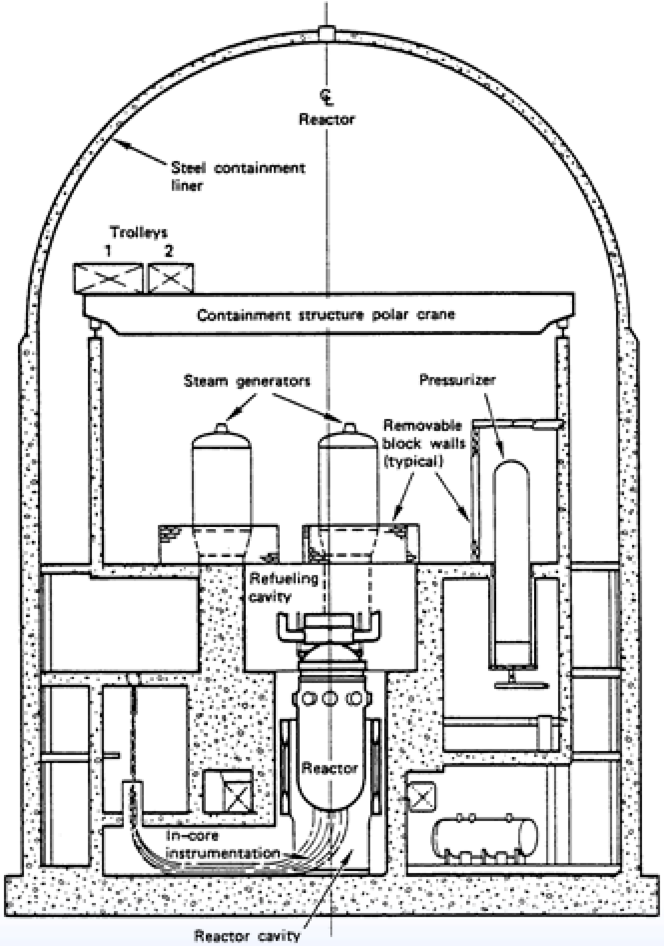


Ice condenser

Diverse Types of Large Dry Containments

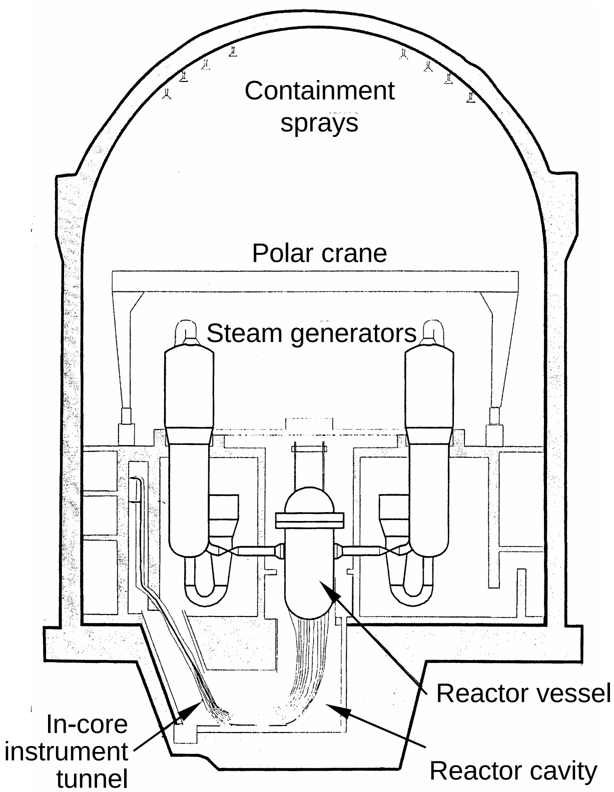
- Rely on large internal volume and structural strength (i.e., no passive pressure suppression system)
  - greater diversity of designs compared to other types
- Represents largest containment design group
  - includes a small subset (about 7) subatmospheric containment designs
- Most use reinforced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

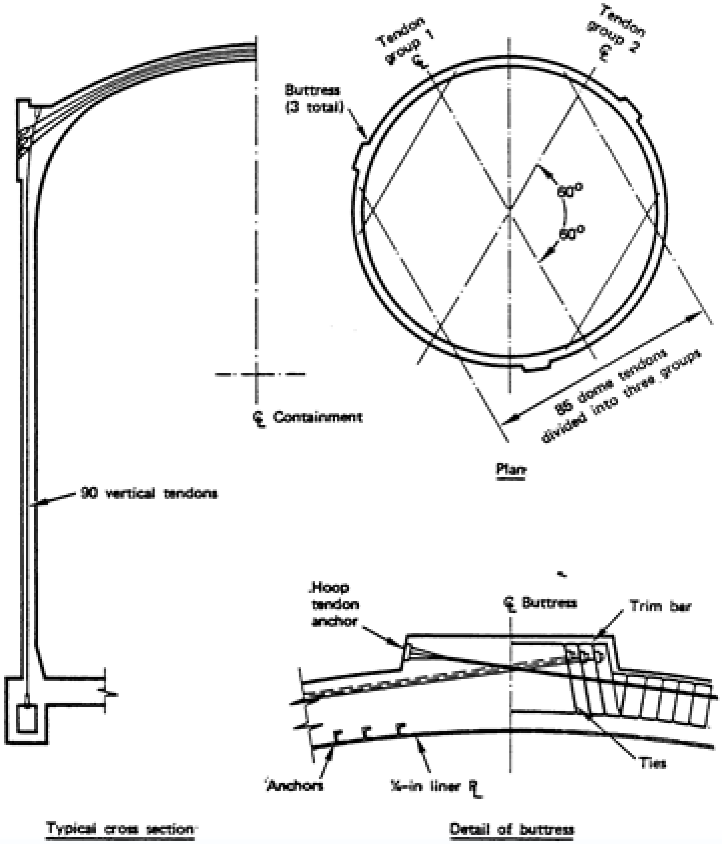
Diverse Types of Large Dry Containments (continued)



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

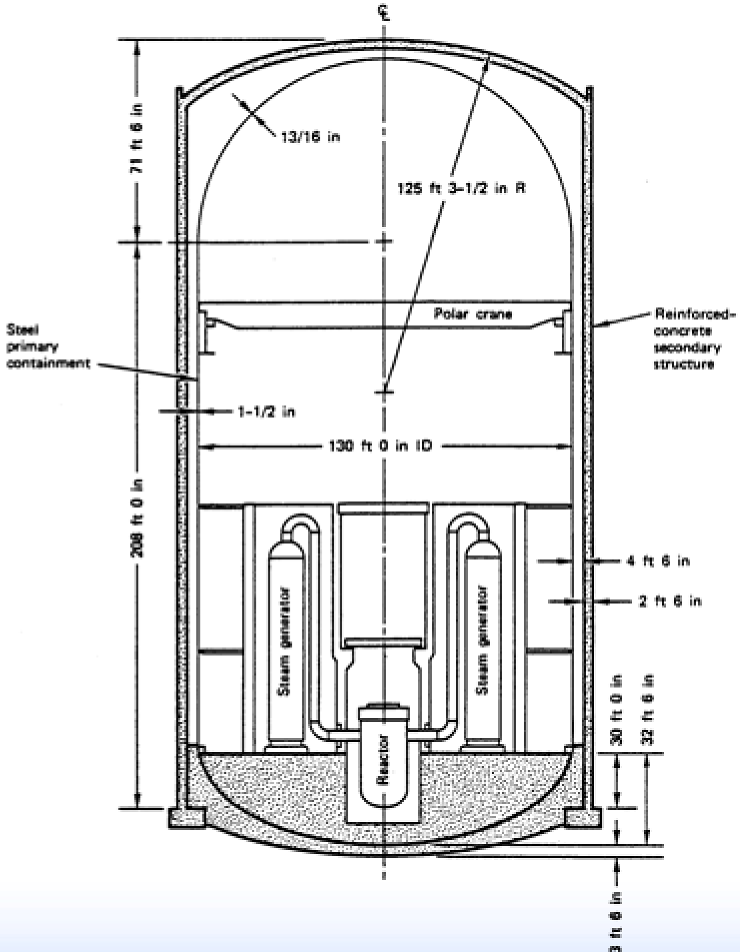
Diverse Types of Large Dry Containments (continued)





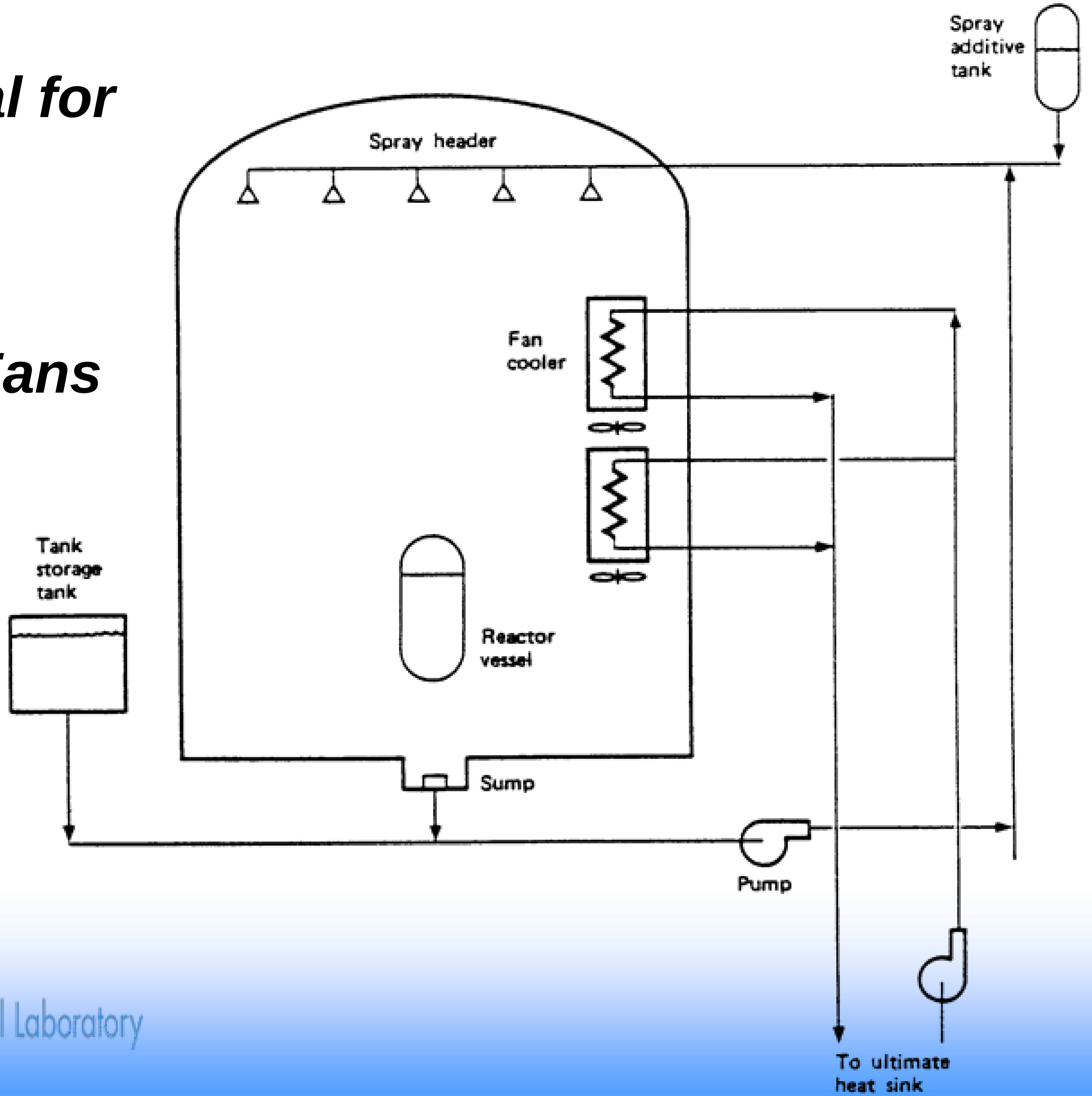
Large Dry Pre-stressed (or Post-tensioned) Concrete
e.g., Palisades
(This is the most common containment design)

Diverse Types of Large Dry Containments (continued)



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

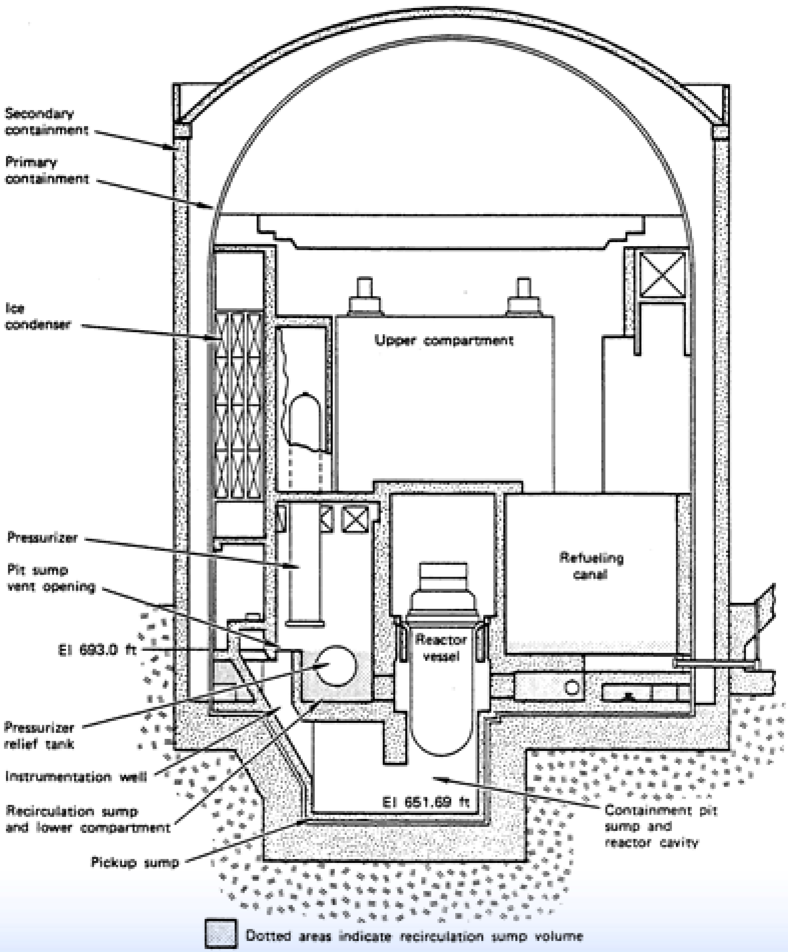
Containment Heat Removal for Large Dry Containment Design Uses Sprays and Fans Coolers

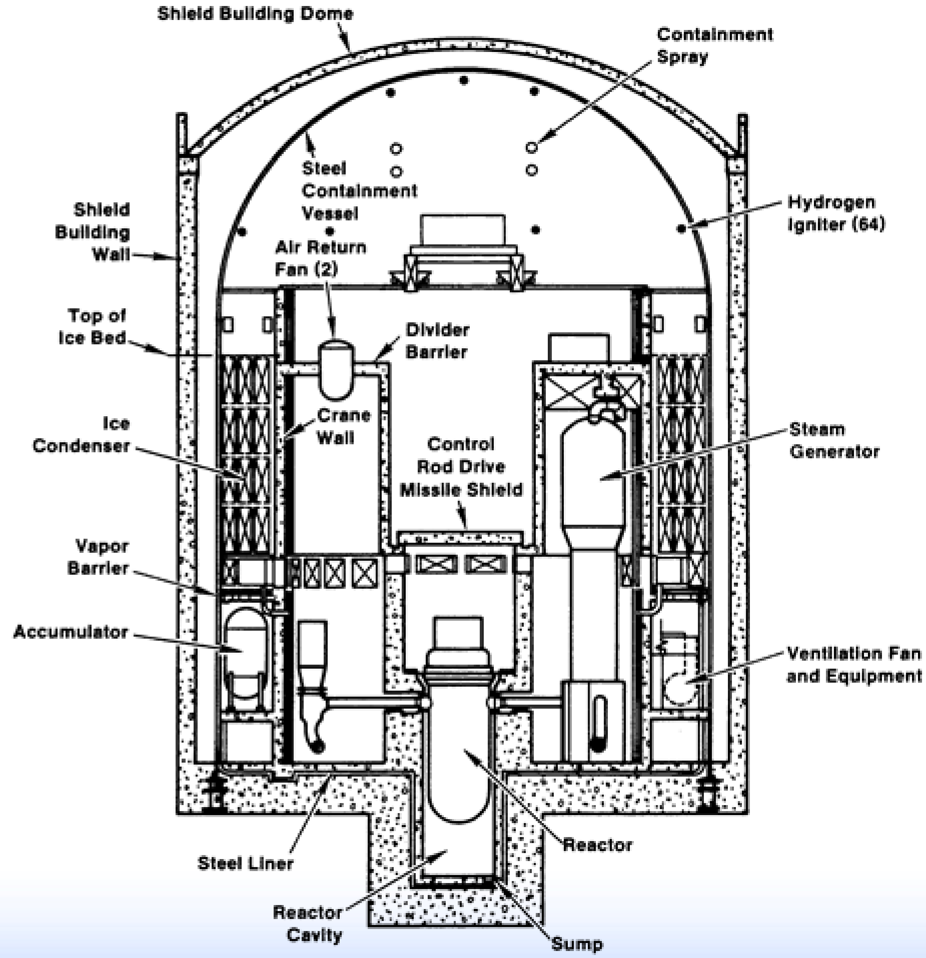


Less Diversity in Ice Condenser Containments

- 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

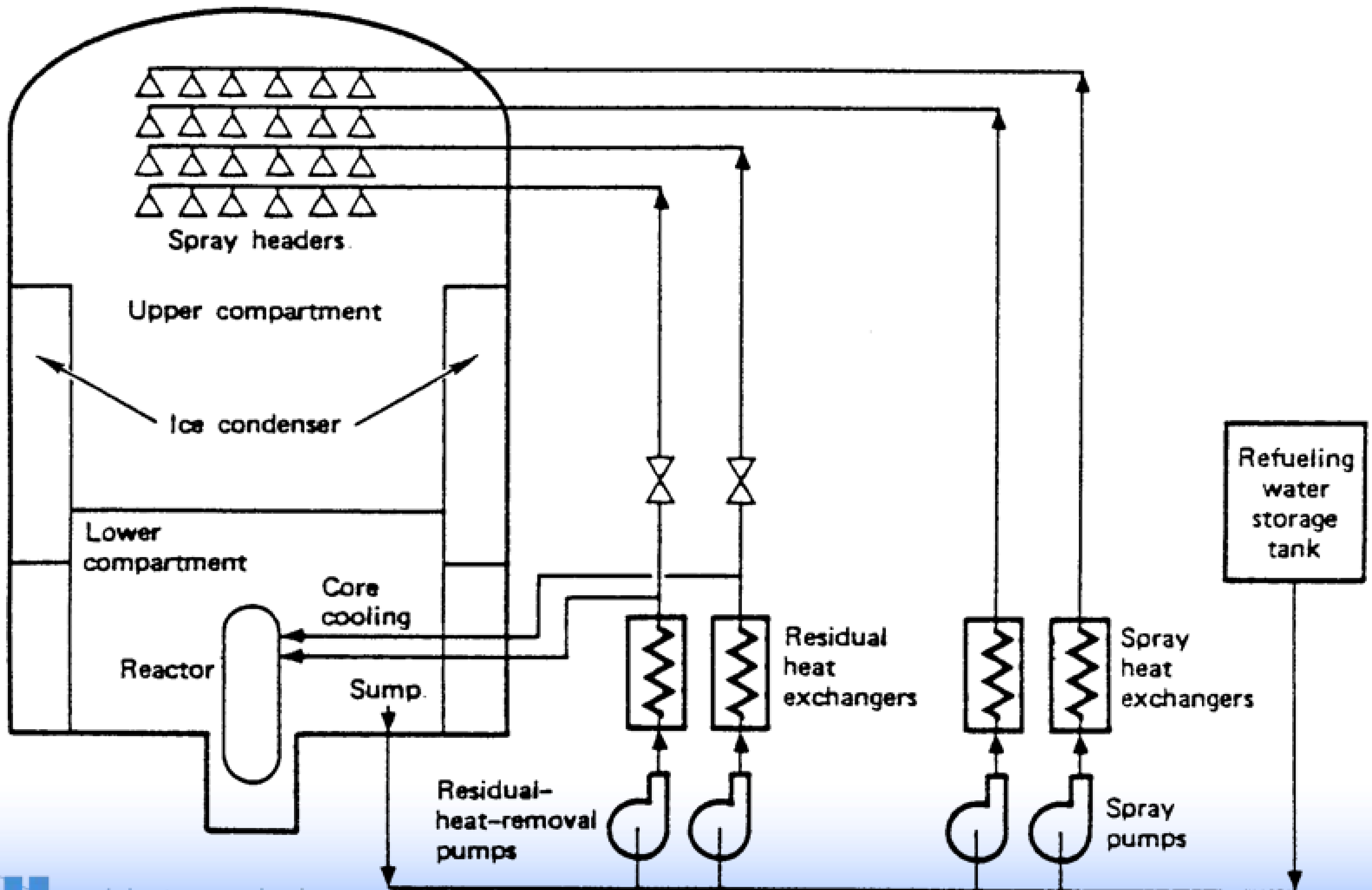
Less Diversity in Ice Condenser Containments (continued)





Sequoyah

CHR for IC Design Uses Sprays and Ice Condensers



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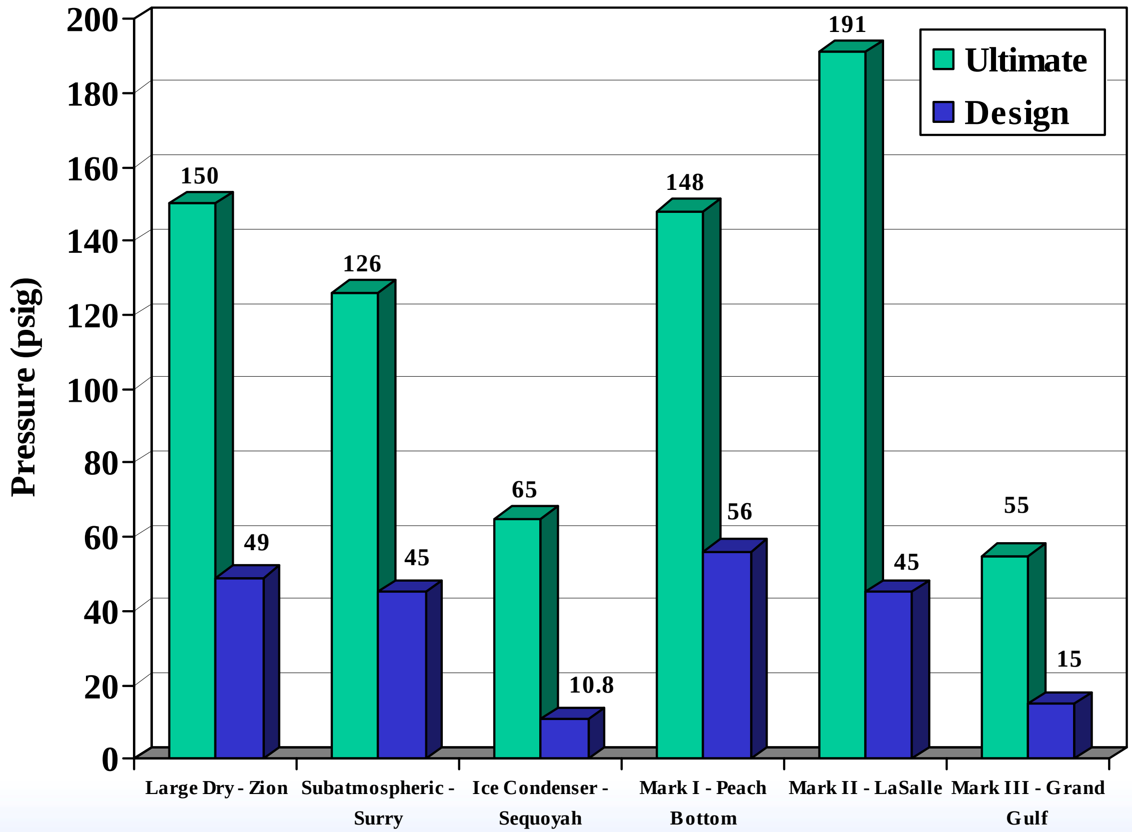
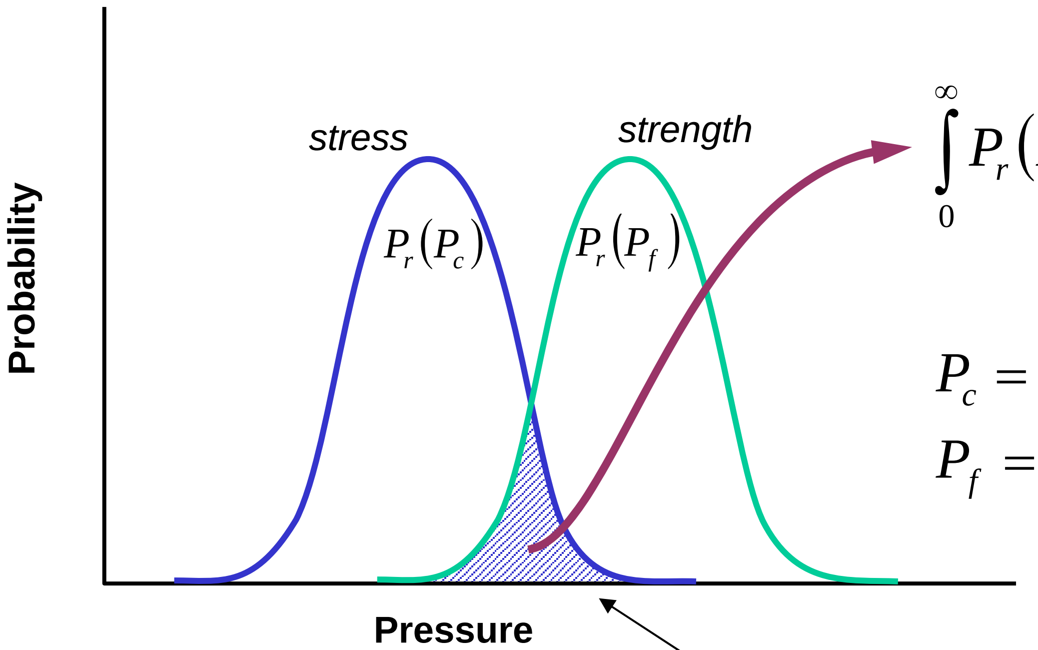
Severe Accidents Pose Several Challenges to Containment Integrity

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

Containment Failure Pressures Significantly Higher than Design Pressures



Conditional Probability for Containment Failure for Each Sequence Calculated Probabilistically



overlap of two curves represents the probability of containment failing

Conditional containment failure probability (CCFP) =

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

$$P_c = Peak\ containment\ pressure$$

$$P_f = Containment\ failure\ pressure$$

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**

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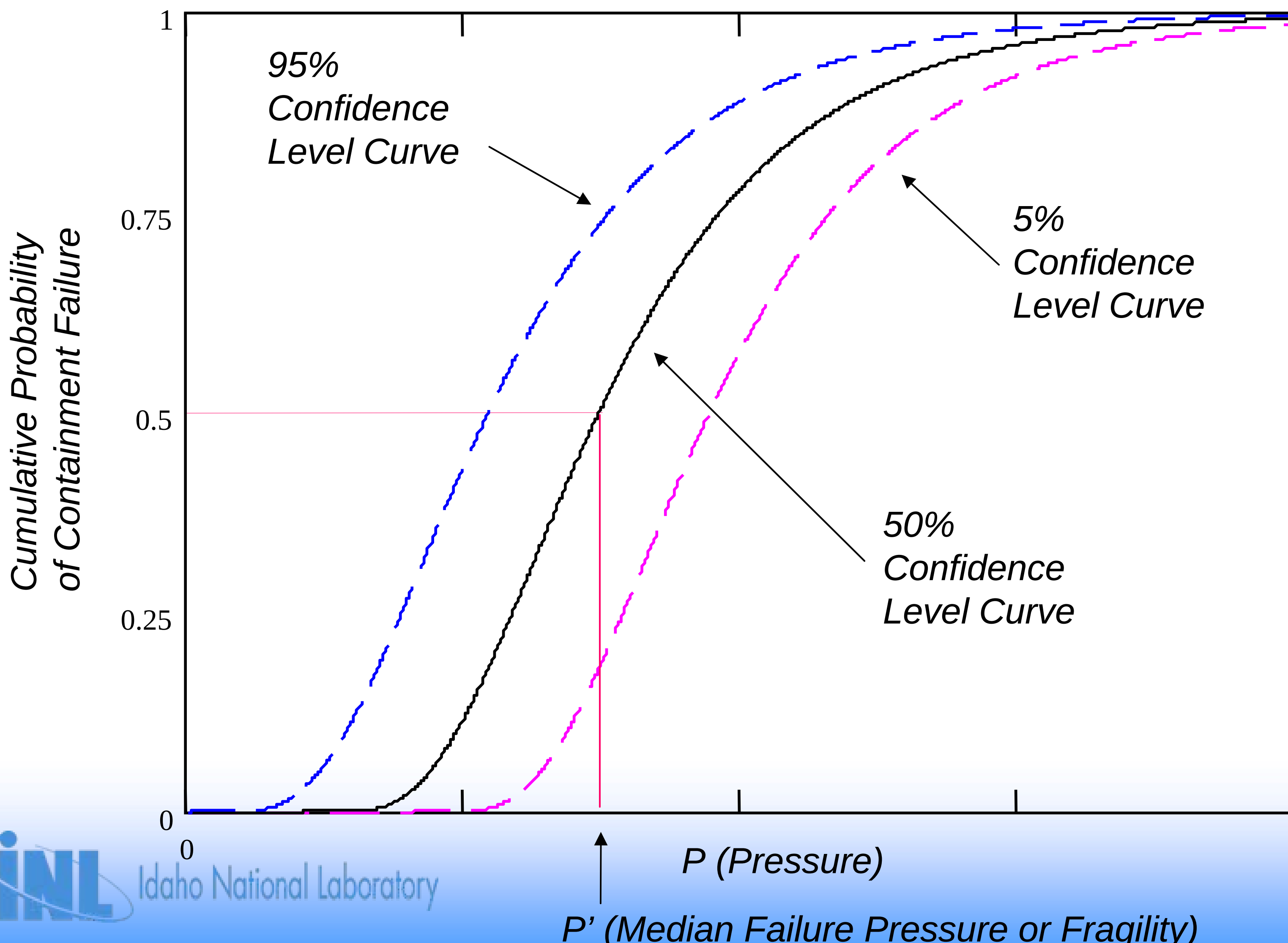
Containment Fragility Curves at Different Confidence Levels



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

Session Review

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

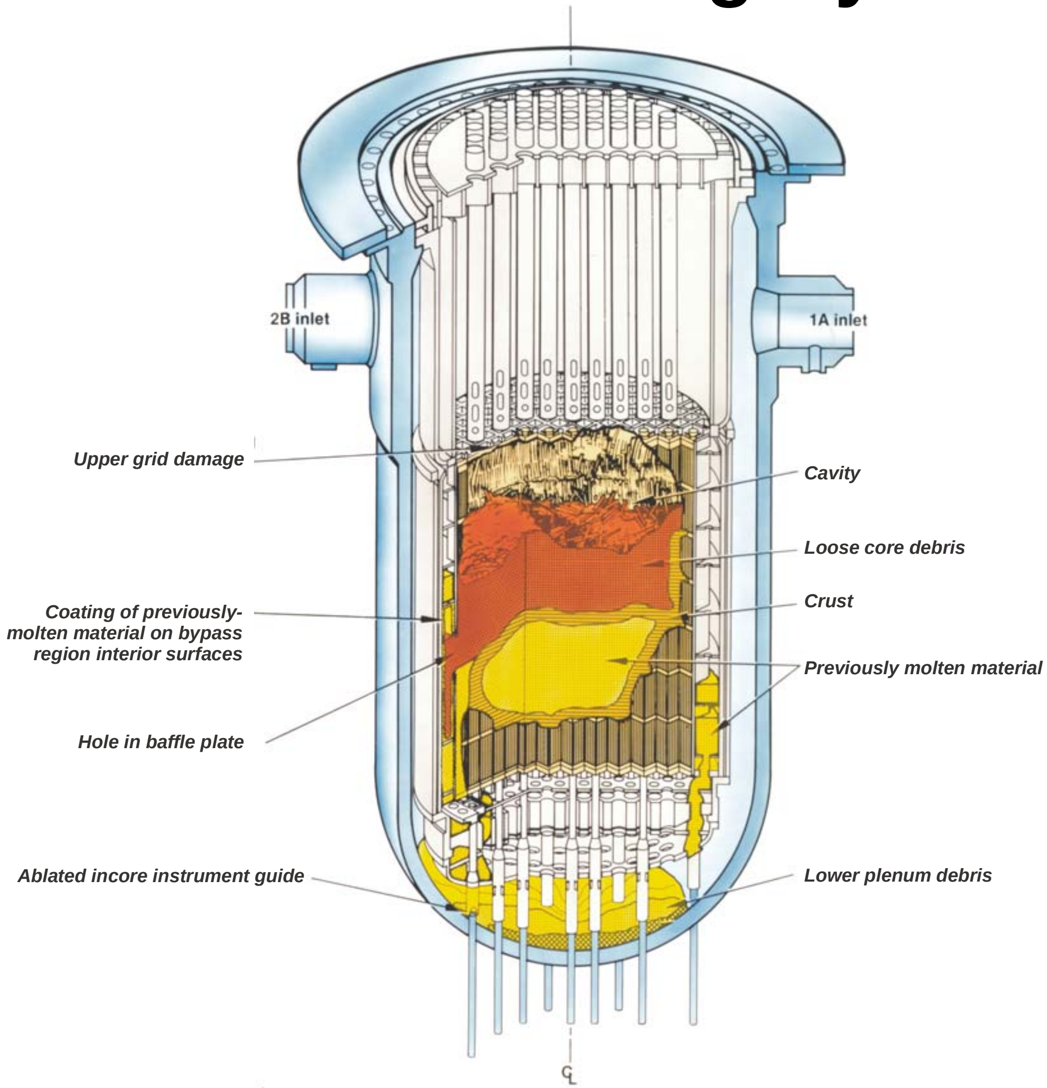
4. Phenomena Affecting Vessel Integrity

- Introduction
- Design
- Failure Modes
- Debris Heat Loads
- Failure Mitigation Measures
- Case Study and Problems
- Study Questions
- References

Objectives

- Identify various vessel failure modes and understand their likelihood in various reactor designs and accident scenarios.
- Describe various endstates for relocated debris in the reactor vessel and discuss their impact on vessel heat load.
- Discuss various mechanisms or actions that may prevent vessel failure.

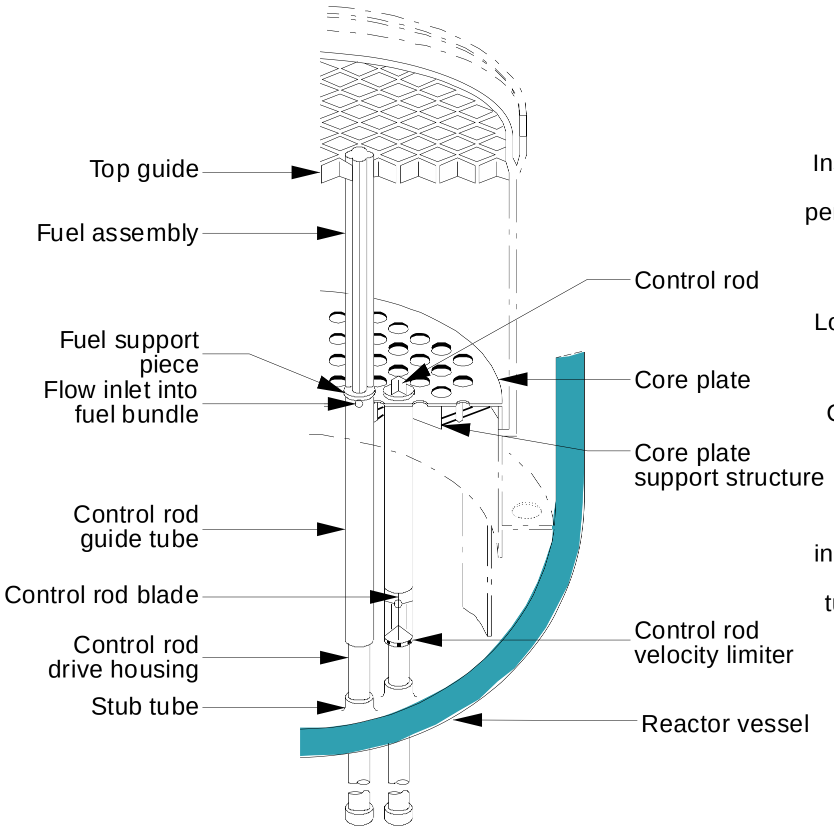
TMI-2 Event Presented Several Challenges to Vessel integrity



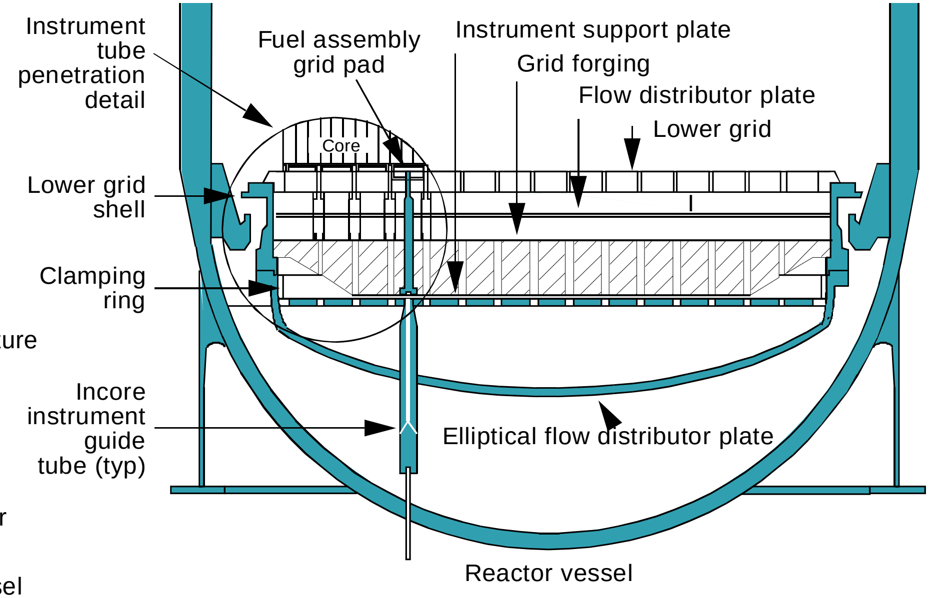
LWR Design Affects Severe Accident Response

Design Feature		Impact
Masses **Uranium Dioxide** **Zirconium** **Steel**	 BWRs have at least 50% more. BWRs have at least 100% more. BWRs have at least 20% more.	 Potential for larger relocation masses. Potential for more hydrogen production. Relocated materials have higher steel content.
Vessel Isolation	MSIV closes to isolate BWR vessel.	No natural circulation in external loops.
Power Distribution	Average power factors in peripheral regions of BWRs significantly lower.	Significant time lag between heatup in central and peripheral core regions.
Depressurization	Mandated operator actions increase potential for BWR accidents to be depressurized.	Increased potential for early core uncovery.
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.

LWR Lower Head Designs Differ



BWR



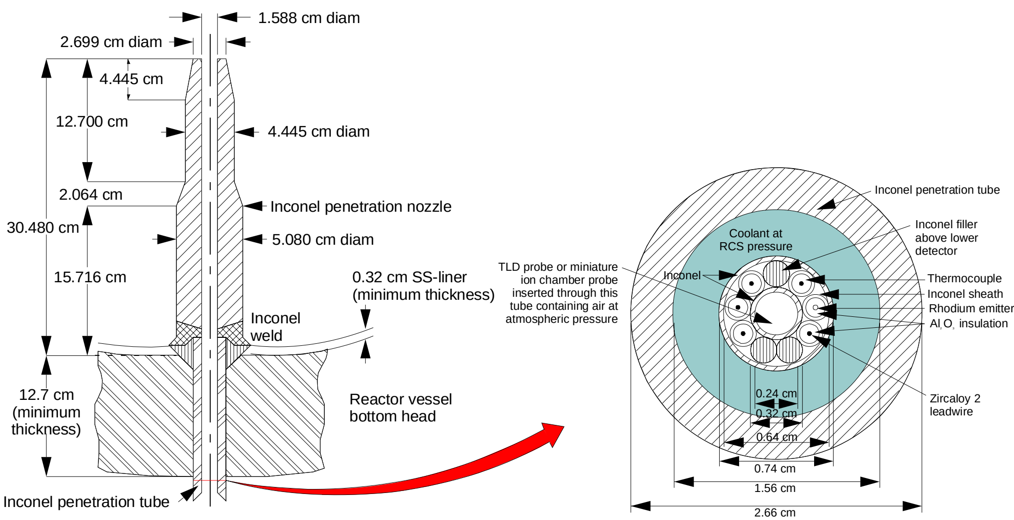
PWR

M801-WHT-293-22

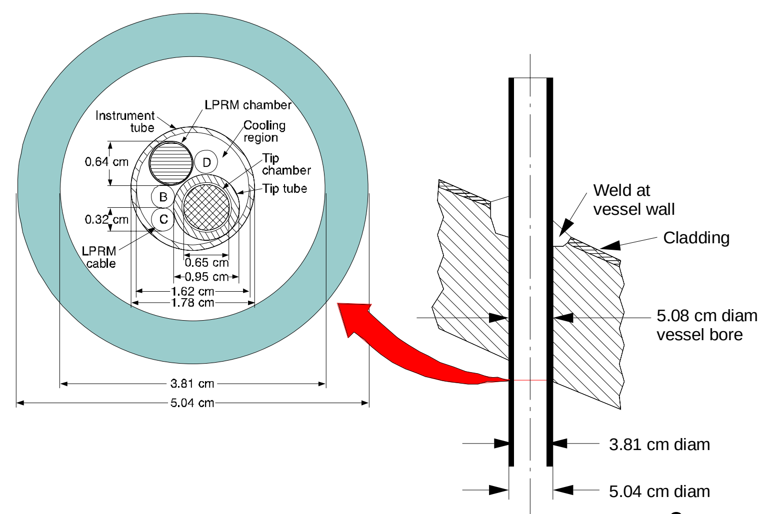
Representative vessel lower head designs

Wide Variation of LWR Instrument Tube Designs

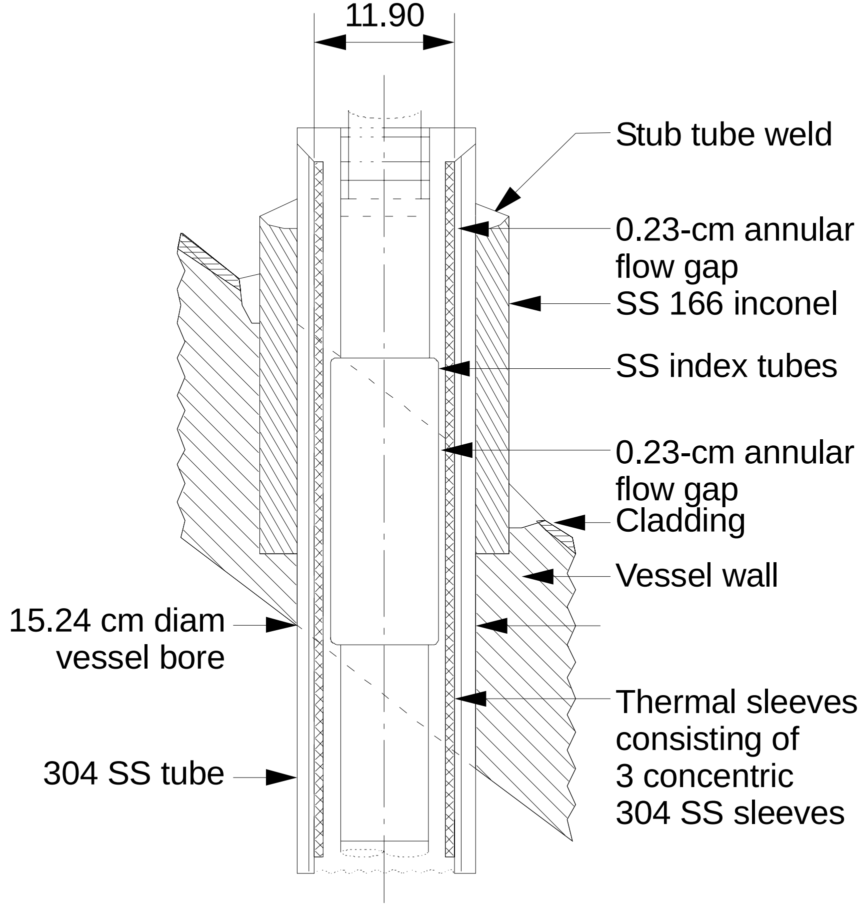
Typical B&W Instrument Tube



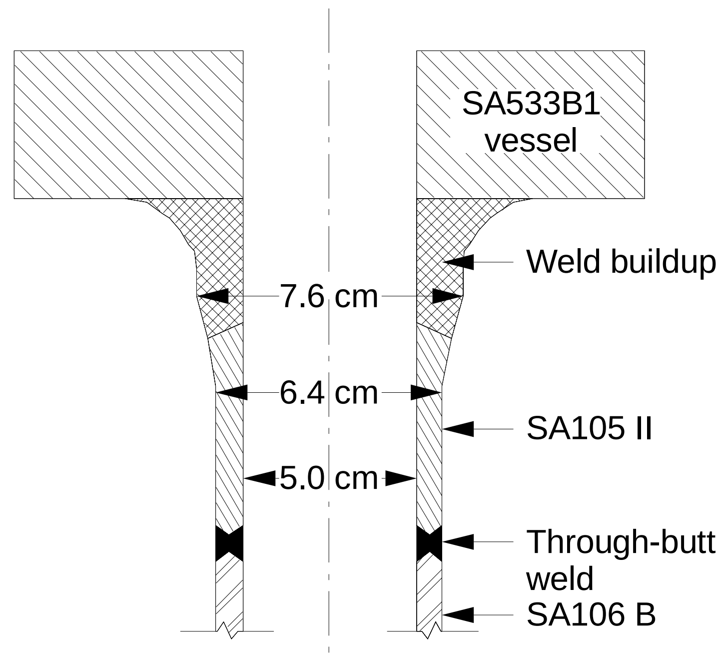
Typical GE Instrument Tube



BWR Vessels Also Penetrated by CRD Assemblies and Drain Line

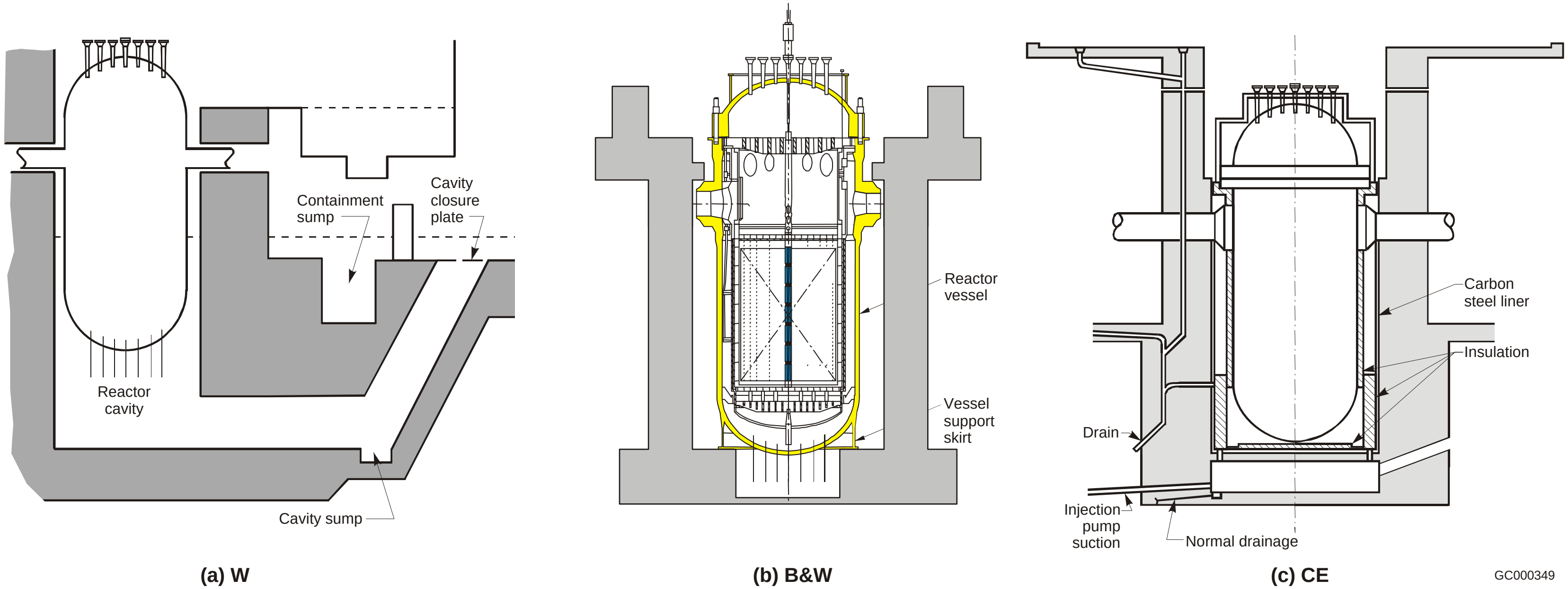


Typical GE CRD Assembly Penetration



Typical GE Drain Line Nozzle Penetration

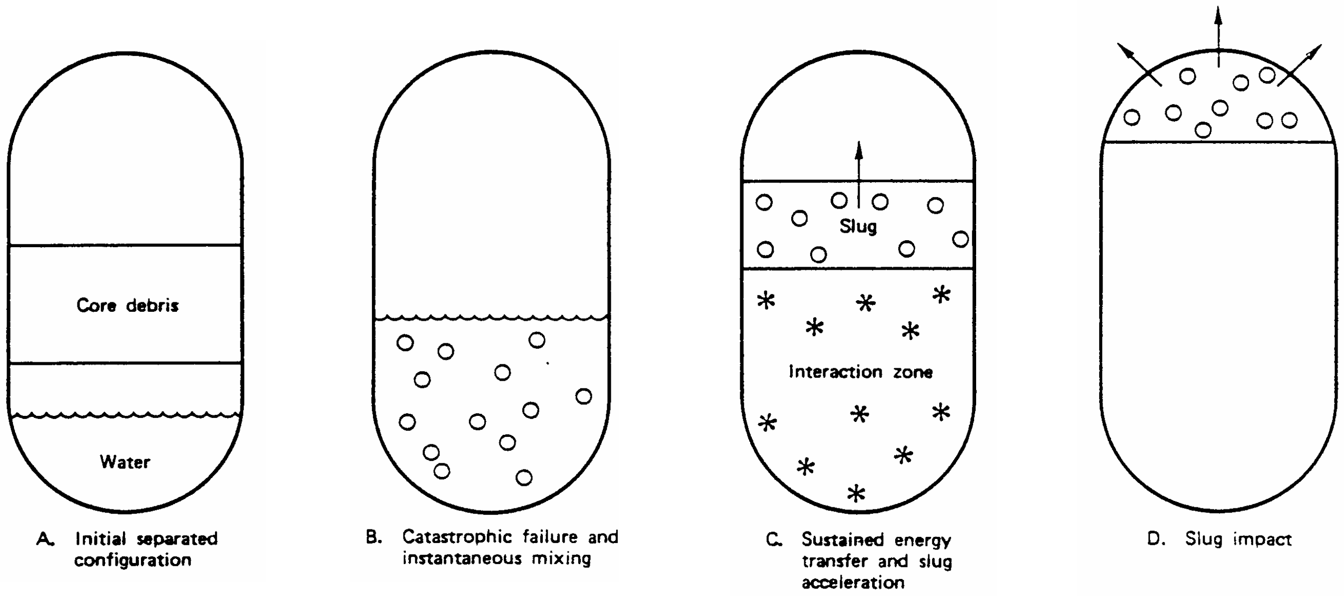
Insulation, Supports, and Cavities for Lower Heads Differ



Failure Modes Dominate at Different Time Periods

Time Regime	Failure Mode
Early	In-vessel steam explosion
Late	Tube failures Vessel failures

In-vessel Steam Explosion Issues



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

NUREG-1150 addressed issue as sensitivity study.

- 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.
- Sensitivity studies also performed assuming probability density function (pdf) derived by "averaging" published frequency estimates from representative researchers.

Recent findings suggest lower probability for steam explosion.

- Experimental results indicate:
  - At low pressure (0.1 MPa), limited fuel mass expected to participate in energetic FCI
  - At higher pressures (1-2 MPa), explosion difficult to trigger
- All eleven SERG-2 experts estimated low probabilities for early containment failure due to in-vessel steam explosion (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

NUREG-1150 quantified lower head vessel failure mode using expert elicitation.

- Aggregate distributions derived from pdfs provided by three experts
  - Several cases considered (varied pressure, availability of upper head injection, and accumulator injection)
  - Experts based pdfs on available code calculation results, TMI-2 data, and severe fuel damage test data
- Wide variation in expert opinion

NUREG-1150 quantified lower head vessel failure mode using expert elicitation. (continued)

- For all cases, aggregated pdfs suggest:
  - penetration failure with high pressure melt ejection (HPME) more likely
  - less than 60% of core ejected with average value of 30%
- Several experts assumed initially small failure (2.5 to 5 cm diameter) that rapidly ablated

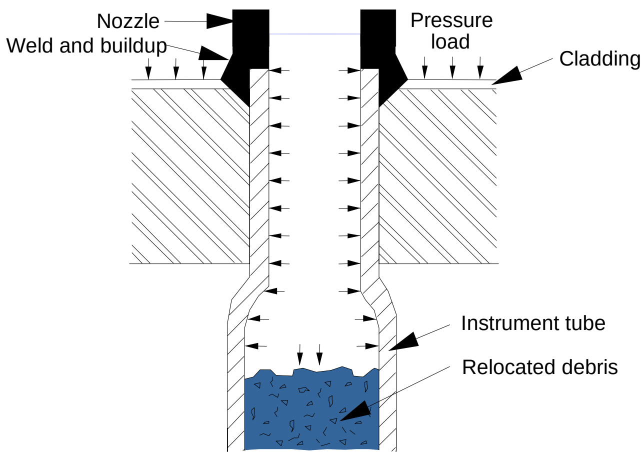
Vessel lower head failure identified as key uncertainty.

- Codes typically assumed early penetration failure (with subsequent depressurization) or global vessel failure based on temperature criterion
- Vessel failure mode and timing significantly affects subsequent accident progression
- Singled out as area with major uncertainty in Special Committee Review for NUREG-1150.

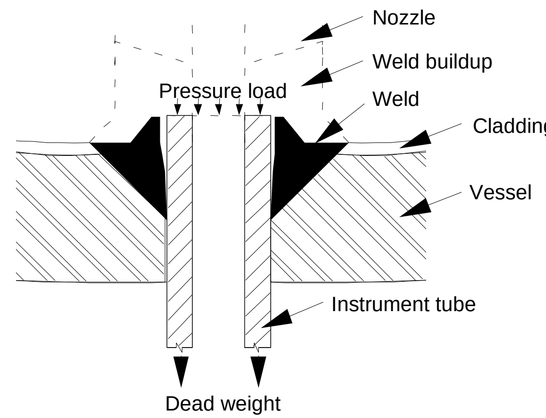
Subsequent research provided data and improved tools for predicting vessel failure.

Program	Focus	Heat Loads	Vessel	Pressure
NRC Lower Head Failure Program (INL)	Tools and high temperature 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)
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 carbon steel vessel with penetrations	High (3-15 MPa)
NRC Lower Head Failure Tests (SNL)	Failure data for well-defined heat loads	Localized and global heat loads	1/5th scale (with and without penetrations)	Low (<0.4 MPa)

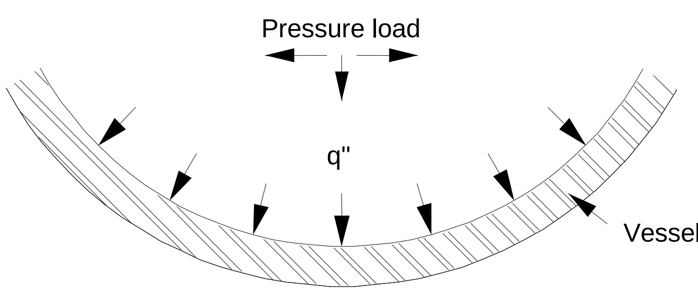
Subsequent Research Considered Wider Range of Failure Mechanisms



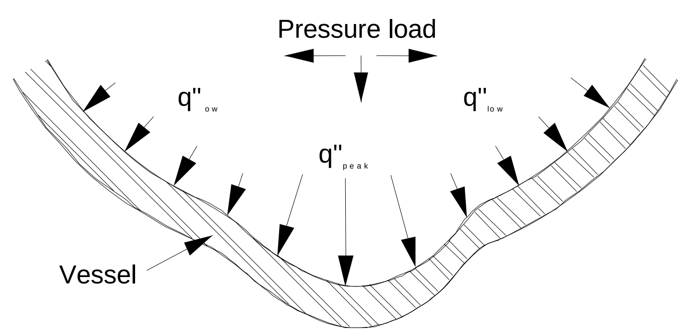
Tube rupture



Weld failure/tube ejection



Global vessel failure

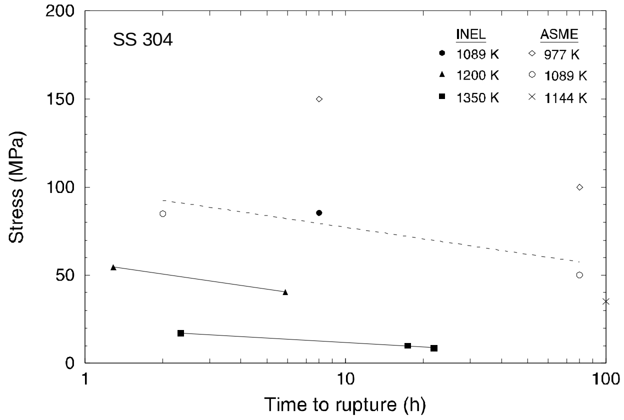


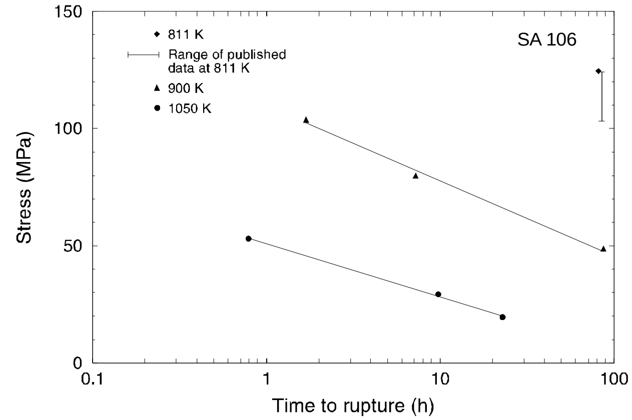
Localized vessel failure

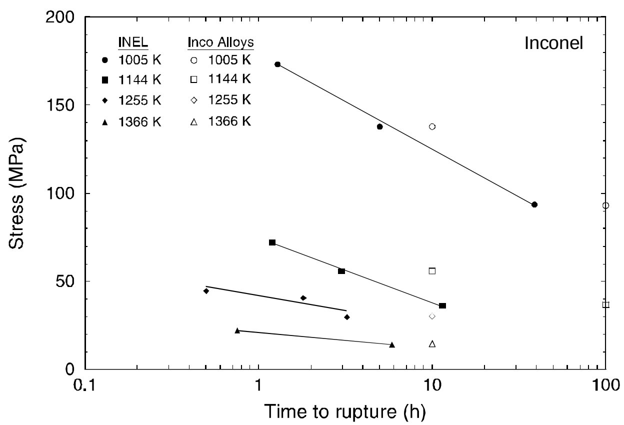
INL Lower Head Failure Program First Comprehensive Study of Vessel Failure Mechanisms

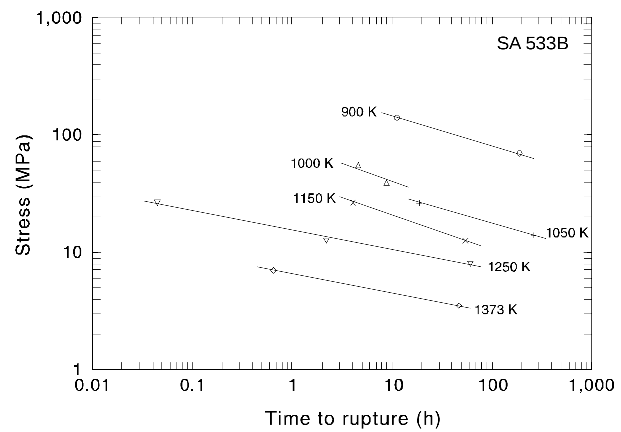
- Developed method to determine which failure mode occurs first in various accident scenarios and reactor designs
- Obtained high temperature creep and tensile data for LWR vessel and penetration materials
- Applied methods to obtain insights related to vessel failure for range of accident conditions and reactor designs

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









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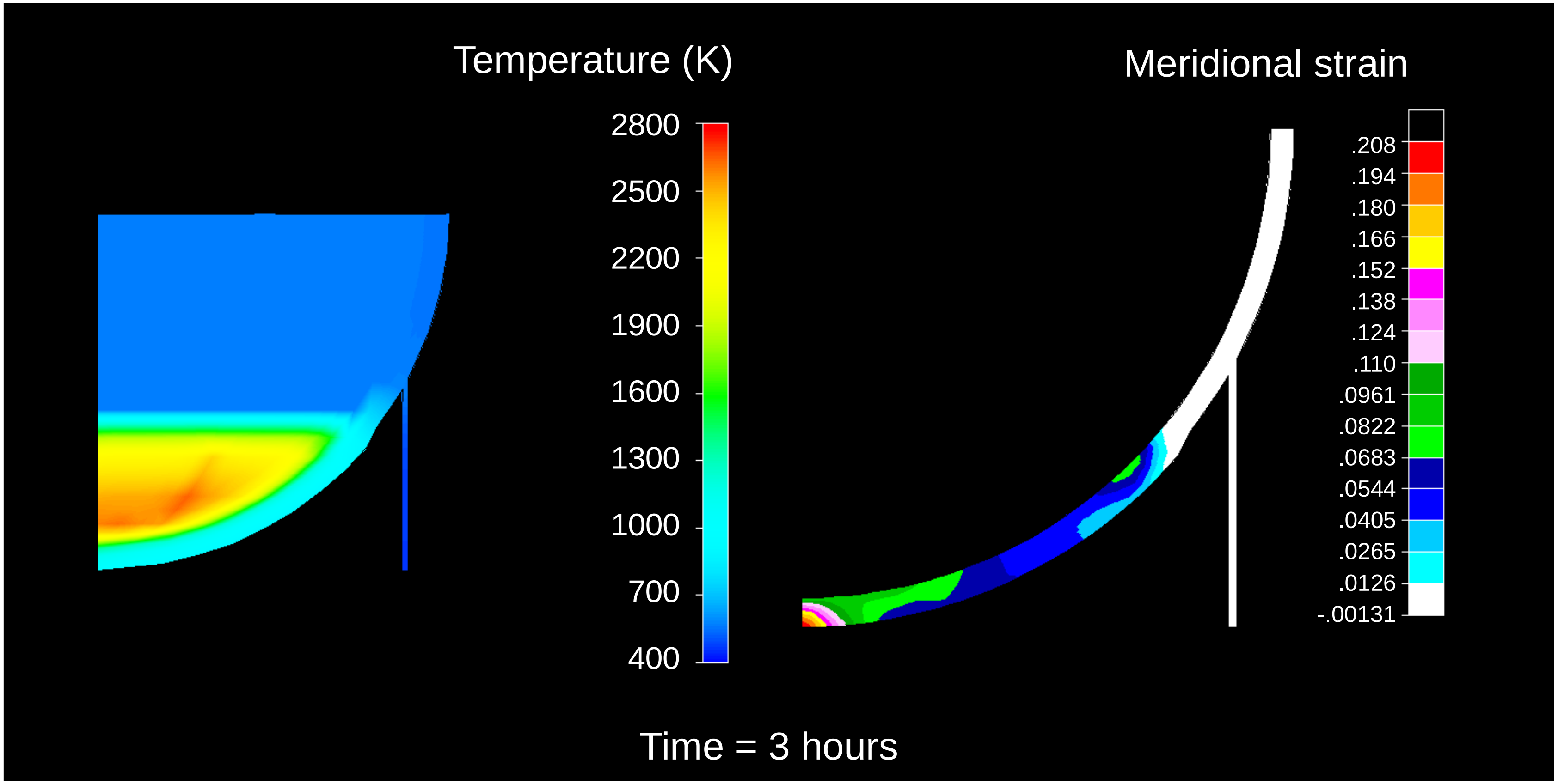
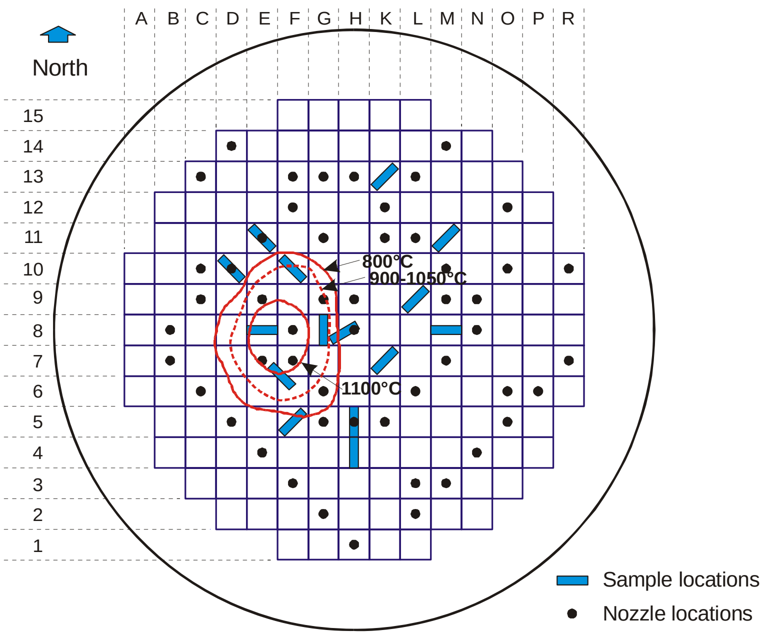
FEM Calculations Performed Using Linked PATRAN-SCDAP/RELAP5-ABAQUS Codes



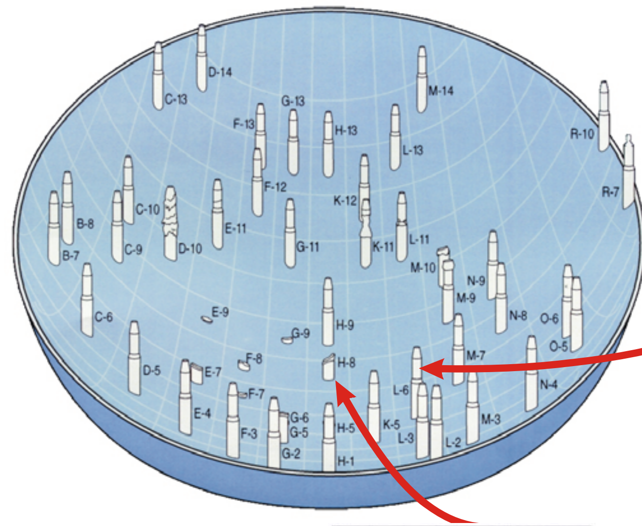
INL Study Provided Key Insights Related to Vessel Failure

- In-vessel tube melting predicted for most severe accident scenarios
- Heat load, pressure, and vessel design govern which failure mode most likely to occur:
  – Pressure-induced failure for BWR vessel governed by drain line failure
  – Pressure-induced failure for PWR vessel governed by localized or global vessel

OECD-sponsored TMI-2 Vessel Investigation Program Provided Initial Data for Model Assessment



GC00 0342



Nozzle H8
(2-3/4 in. tall)

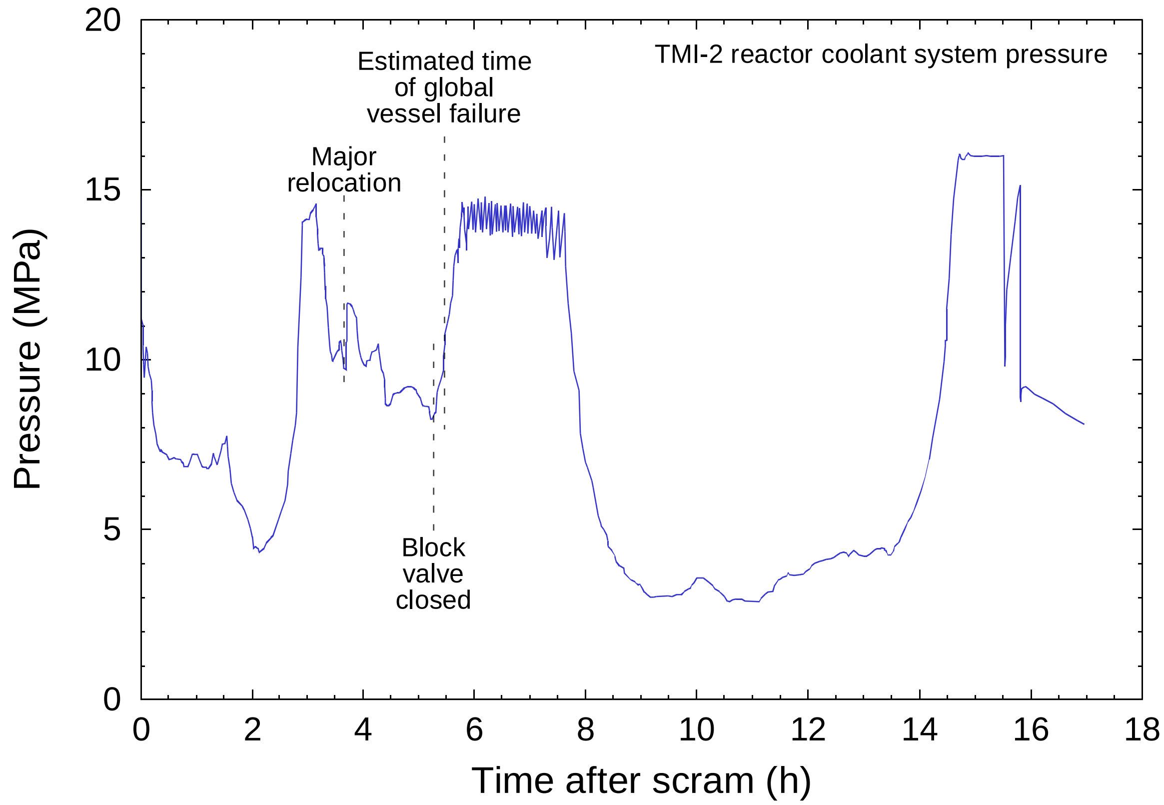
Nozzle L6
(9-3/8 in. tall)

GC00 0343

TMI-2 VIP Analysis Results Provided Insights About Vessel Failure

- Large margin-to-failure estimated for tube rupture and tube ejection failure modes
  – melt traveling below lower head insufficient to heat tube
  – thermal and pressure load on lower head insufficient to fail weld
- Localized "hot spot" not due to jet impingement
- Global or localized creep rupture estimated as failure mode with least margin for TMI-2 scenario
- Additional margin, not currently considered in severe accident analyses codes, exists in LWR lower heads.

Best Estimate Models Erroneously Predict TMI-2 Vessel Failure

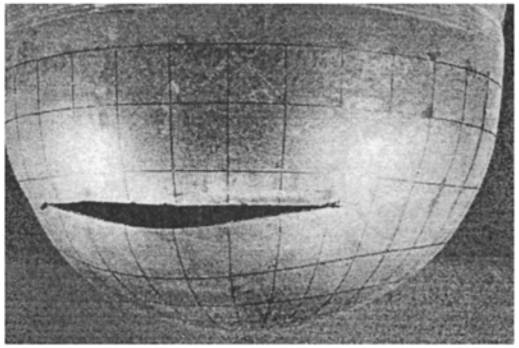


SNL Lower Head Failure Experiments Provide Assessment Data

One-fifth scale, SA533B1 carbon steel vessel failure tests performed to:

- assess and make recommendations for RPV creep rupture models
- identify effects of heat flux distribution, pressure, penetrations, and weldments on vessel failure

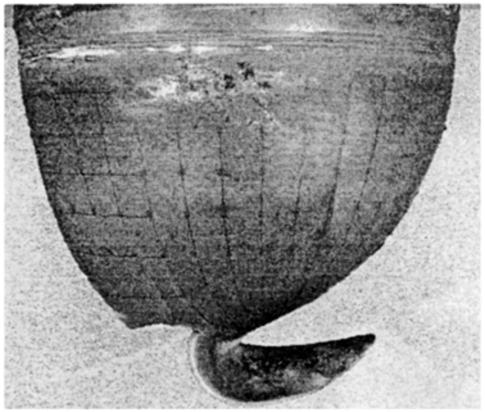
LHF Tests Provide Unique Data



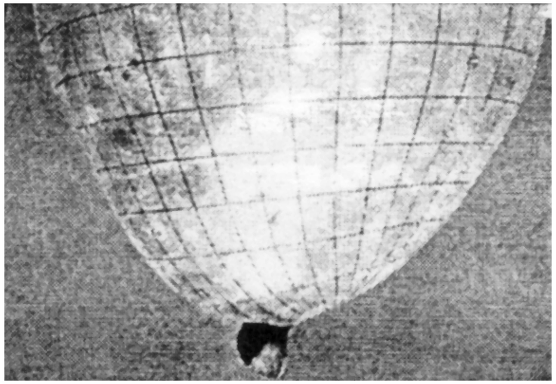
LHF-3

LHF-5

Pressure history affects potential for tearing (LHF-3) versus unzipping (LHF-5) global failure



LHF-1



LHF-6

Importance of non-uniform wall thickness (LHF-1) and localized hot spots (LHF-6) on vessel failure

Summary

- Research results suggest in-vessel steam explosions leading to α-mode failure are not important from a risk perspective
- Recent assessments and experiments provide key insights about potential for other failure modes:
  - Importance of RCS pressure, vessel manufacturing irregularities, and mass, composition, decay heat distribution and melt fraction of relocated debris
  - Penetration failures unlikely at pressures above 2 MPa
  - Experimental data and analyses suggest localized and global vessel failures more likely

Debris Heat Loads Impact Quantification of Several Events

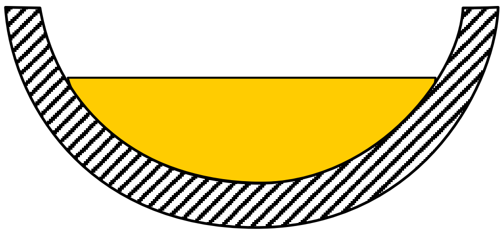
- Debris heat loads significantly impact mode and timing of vessel failure and potential for subsequent containment failure.
- Information needed to address key questions:
  - What type of debris endstates may occur?
  - How does debris endstate affect vessel heat loads?
  - What phenomena affect debris coolability?
  - How does endstate affect melt fraction, oxidation fraction, and composition of release if vessel fails?

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

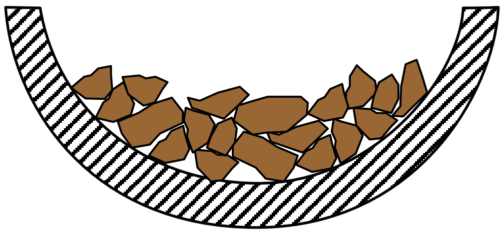
- Three experts asked to evaluate three cases (medium to high pressure, with and without injection)
- Available code calculations, TMI-2 post-accident examinations, and severe fuel damage tests used to derive pdfs for
  - mass ejection rate
  - melt temperature
  - oxidation fraction of released melt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

Several Debris Configurations Possible

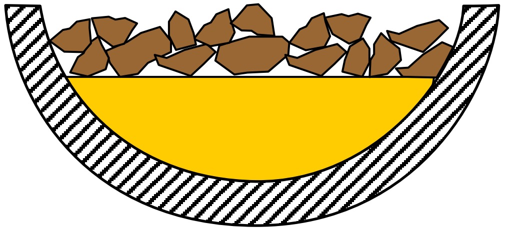
Molten pool



Fragmented rubble

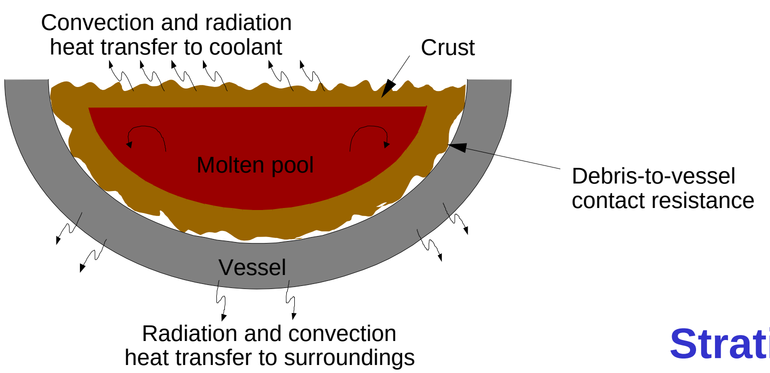


Molten pool beneath fragmented rubble

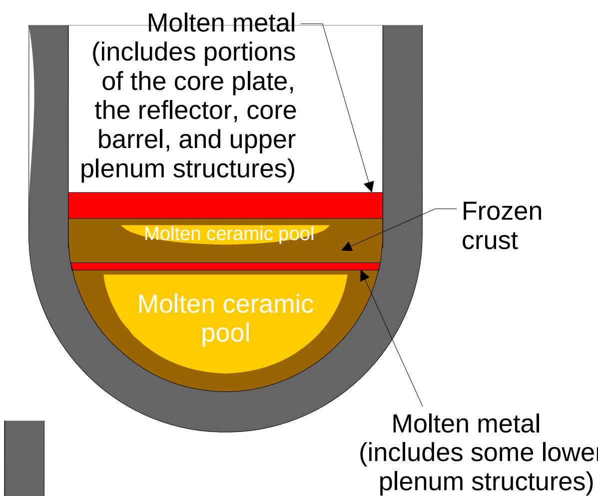


Several Debris Configurations Possible (continued)

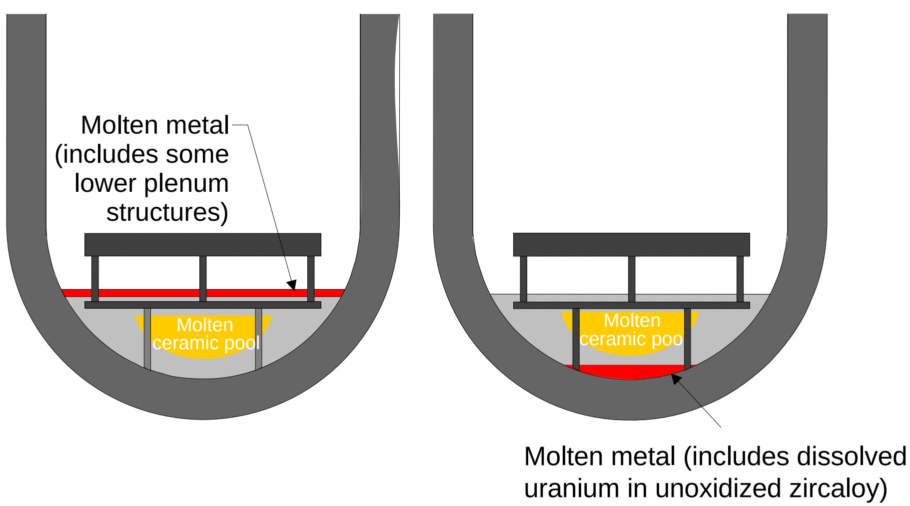
Homogeneous



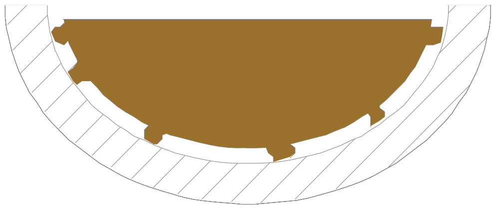
Sandwiched



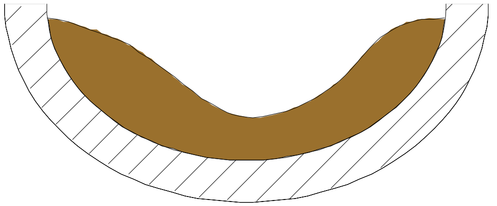
Stratified



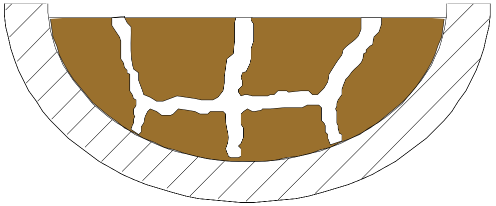
Evidence Suggests Enhanced Cooling Possible As Corium Solidifies



Intermittent debris-to-vessel gap



Enhanced upper surface corium surface area

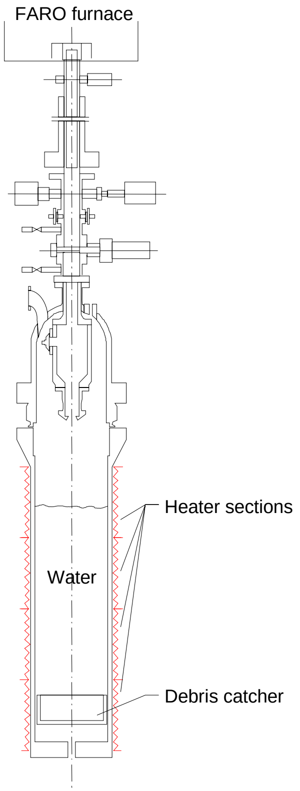


Interconnected corium cracks

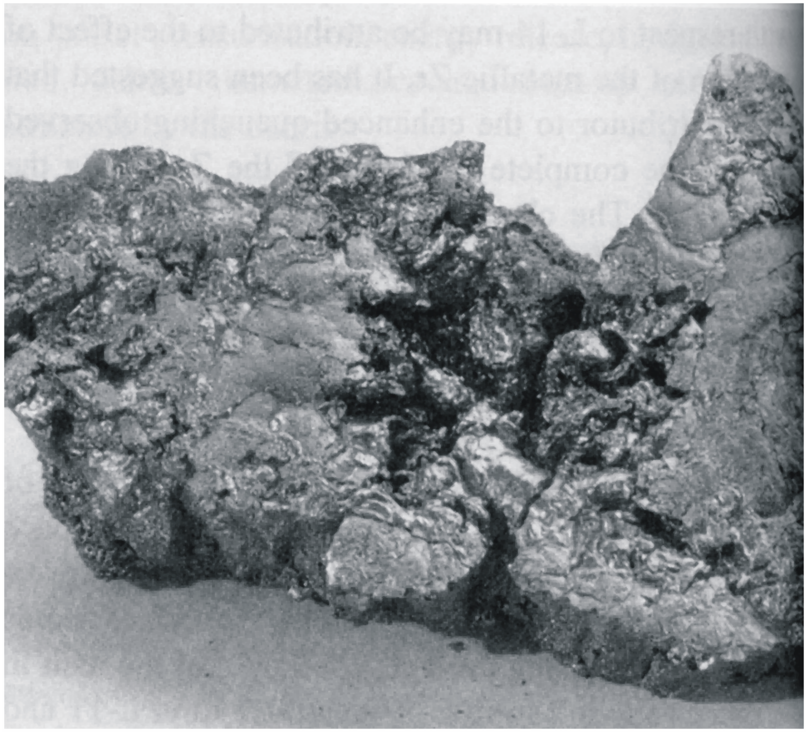
Wide Range of Investigations Provide Insights about Heat Load from Relocated Corium

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)
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)
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)
NUPEC COTELS Tests	Melt/water interactions, debris cooling morphology	UO_2, ZrO_2, Zr, SS	SS hemispherical vessel	Water	Low

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

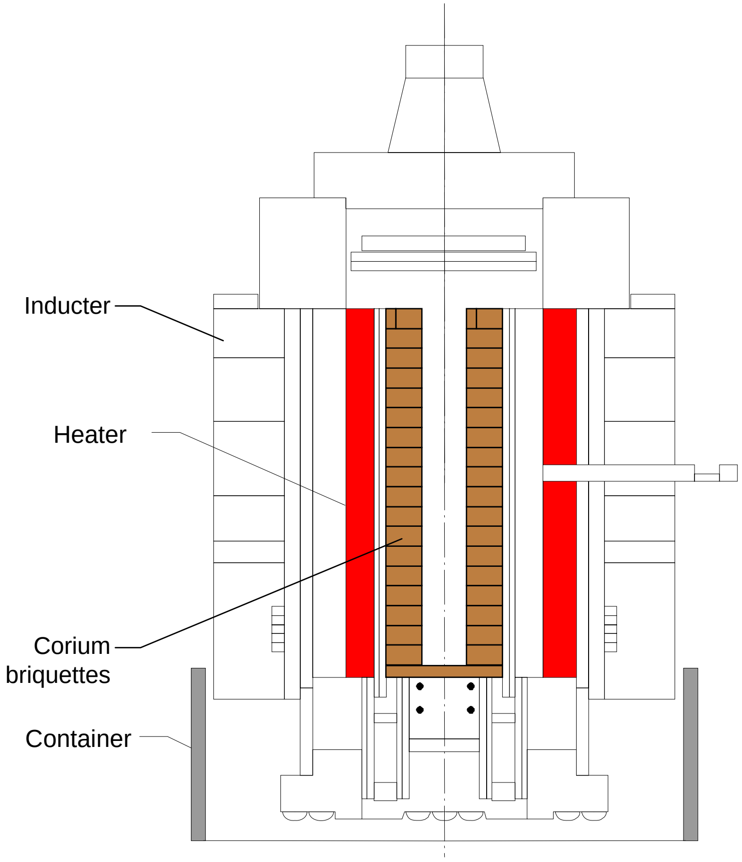


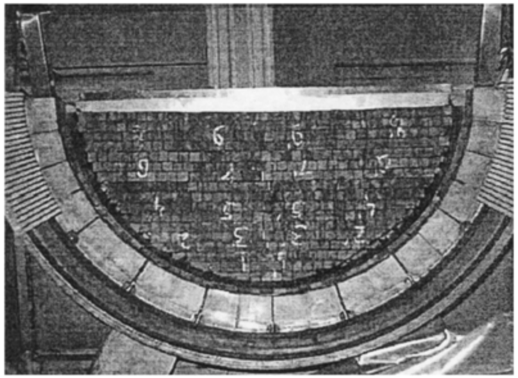
FARO



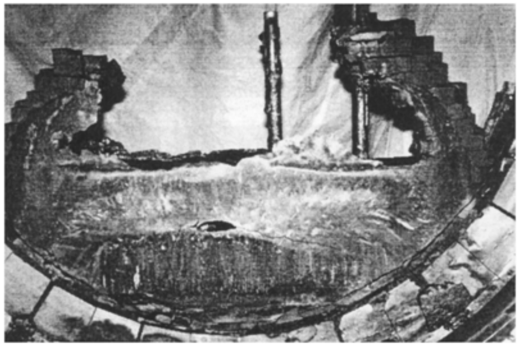
- **Furrows observed in relocated debris**
- **Intermittent contact between relocated debris and test plate**

RASPLAV provides insights about stratification in relocated molten corium materials





Before



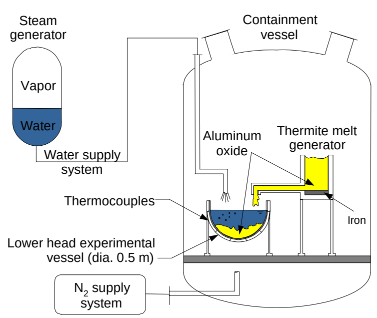
After

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)

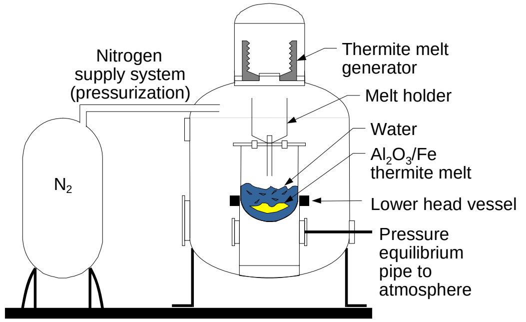
Simulant Material Investigations Also Provide Insight About Heat Load From Relocated Corium

Organization/Country	Phenomena Investigated	Materials: Debris	Materials: Vessel	Materials: Coolant	Pressure
KAERI/Korea SONATA IV - LAVA	Gap	Al_2O_3 (w and w/o Fe)	Carbon steel vessel. Some tests with penetrations.	Water.	$\leq$ 2.0 MPa for initial tests. ΔP_{ves} = 1.8 MPa
JAERI/Japan "ALPHA"	Gap, crack, enhanced area	Al_2O_3	SS-lined carbon steel vessel.	Water.	$\leq$ 1.6 MPa ΔP_{ves}= 0
FAI/USA In-Vessel Cooling Experiment	Gap	Al_2O_3 (w and w/o Fe)	Carbon steel vessel w and w/o insulation. Some tests with penetrations.	Water. Some tests w/o water addition.	~ 3.1 MPa ΔP_{ves}= 3.0 MPa
RIT/Sweden "FOREVER"	Gap	$CaO-B_2O_3$ or $CaO-WO_3$ with electrical heating for $q''' = 1\ MW/m^3$	Carbon steel vessel. Some tests with penetration.	Water. Some tests w/o water addition.	$\leq$ 4 MPa ΔP_{ves} = 2 MPa

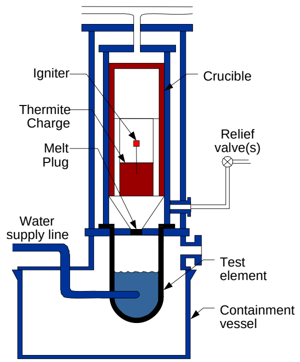
Various Approaches Also Investigate Coolability with Simulant Debris



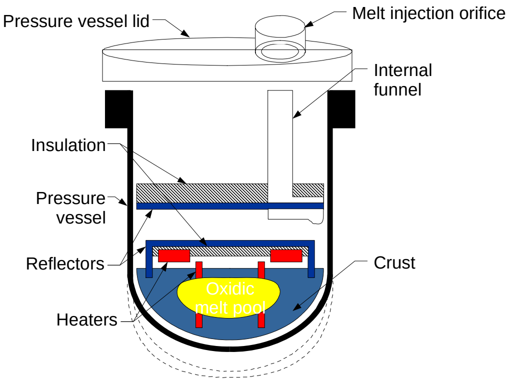
~1/10th scale JAERI ALPHA tests measure crack and gap formation and surface area enhancement



~1/10th scale KAERI SONATA-IV LAVA tests measure debris-to-vessel gap formation



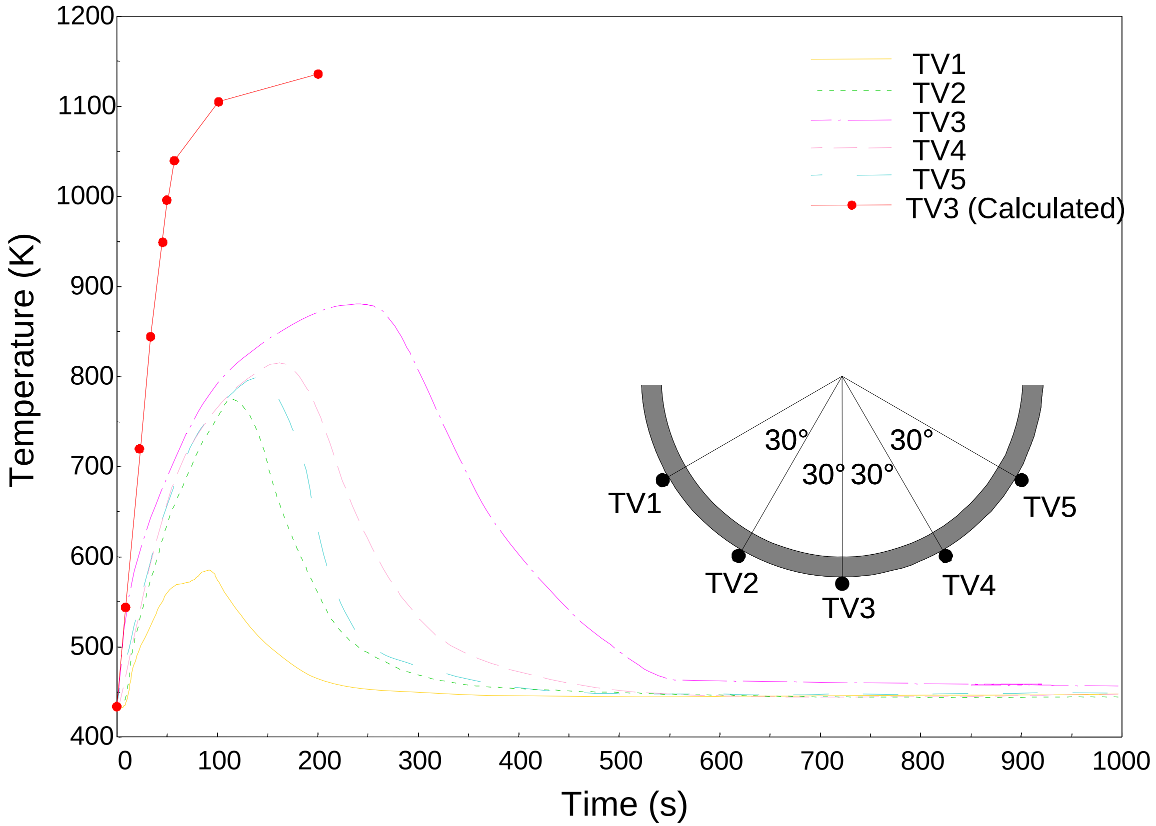
~1/10th scale FAI in-vessel tests include insulation and penetrations



~1/10th scale RIT FOREVER facility features sustained debris heating

* Facilities not to scale

Simulant ALPHA Test Results Suggest Enhanced Gaps Form and Provide Significant Cooling



Differences between simulant and prototypic material properties require validation with prototypic materials.

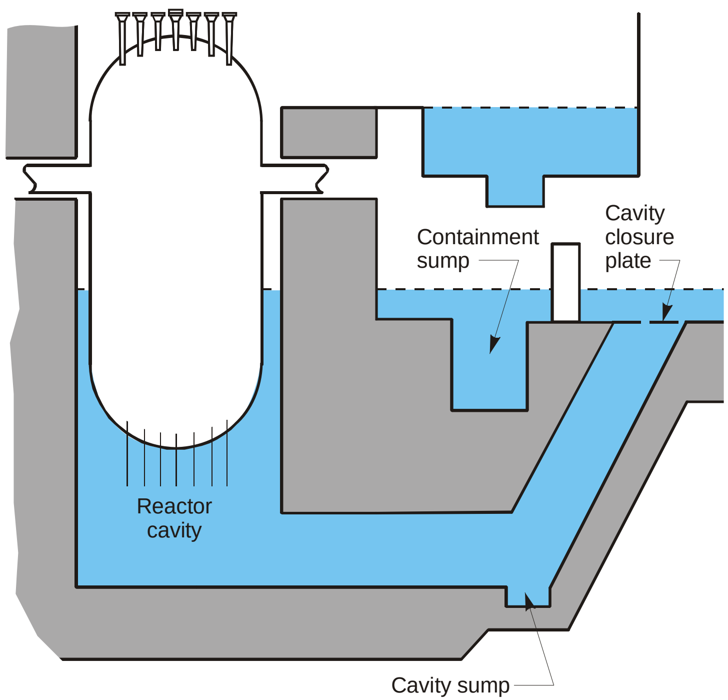
Summary

- Experimental data suggest range of debris states 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
  - Enhanced vessel/insulation arrangement
  - Enhanced vessel coatings
- RCS Depressurization Mechanisms
  - RCP seal leakage
  - Induced ex-vessel piping failure or Steam Generator Tube Rupture
  - Safety valve failure to close
  - Intentional depressurization through Pilot Operated Relief Valves (PORVs)

Requirements for Successful ERVC



- 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 for Several Plants

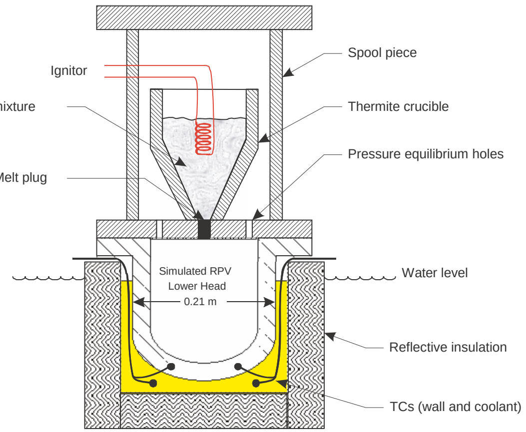
- In many Individual Plant Examinations (IPEs), cavity flooding assumed to preclude vessel failure and reduce event consequences
  - Westinghouse vessels (Zion, Bryon, etc.) penetrated by instrumentation tubes that travel through reactor cavity
  - CE vessels (Palisades, etc.) without lower head instrumentation tubes and insulation
- 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 (modifications to enhance ERVC being implemented into plant)
- Proposed for many advanced reactor designs, such as Westinghouse AP600, AP1000, Korean APR1400, and SWR 1000

Key Issues for Assessing ERVC Viability

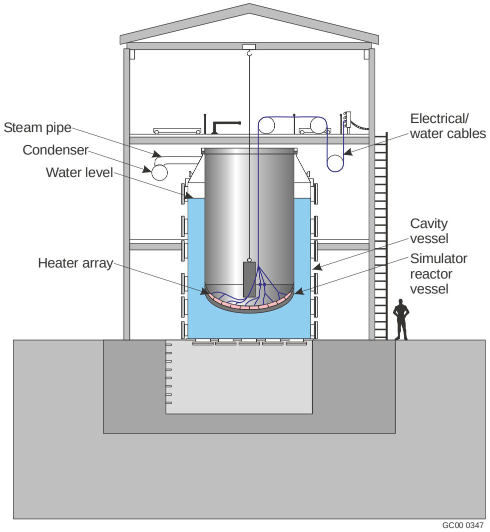
- What is time required to fill reactor cavity?
- What is heat transfer from lower head?
- Does insulation surround lower head?
- Are structures, such as penetrations or support skirt, attached to lower head?

Various Approaches used to Investigate ERVC

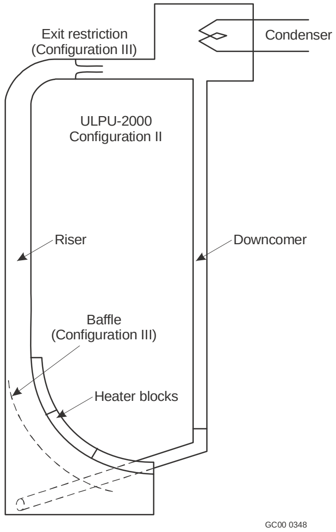
FAI

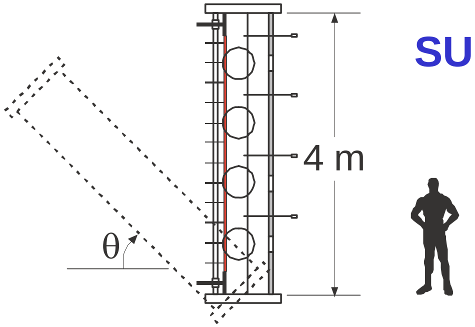


CYBL



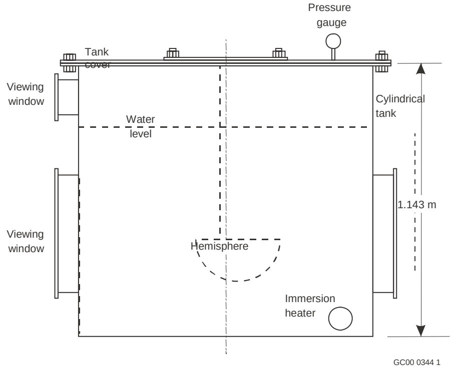
ULPU





SULTAN

SBLB

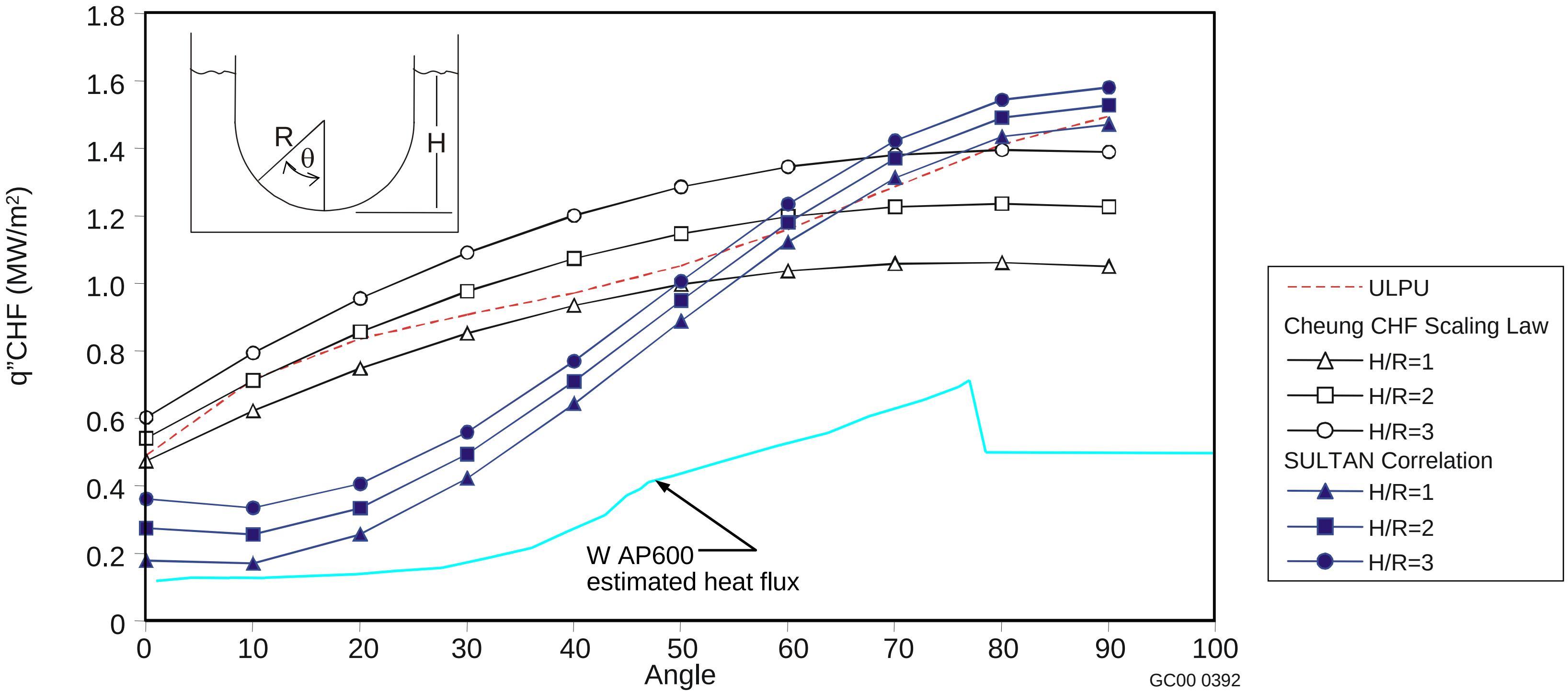


Drawings not to scale

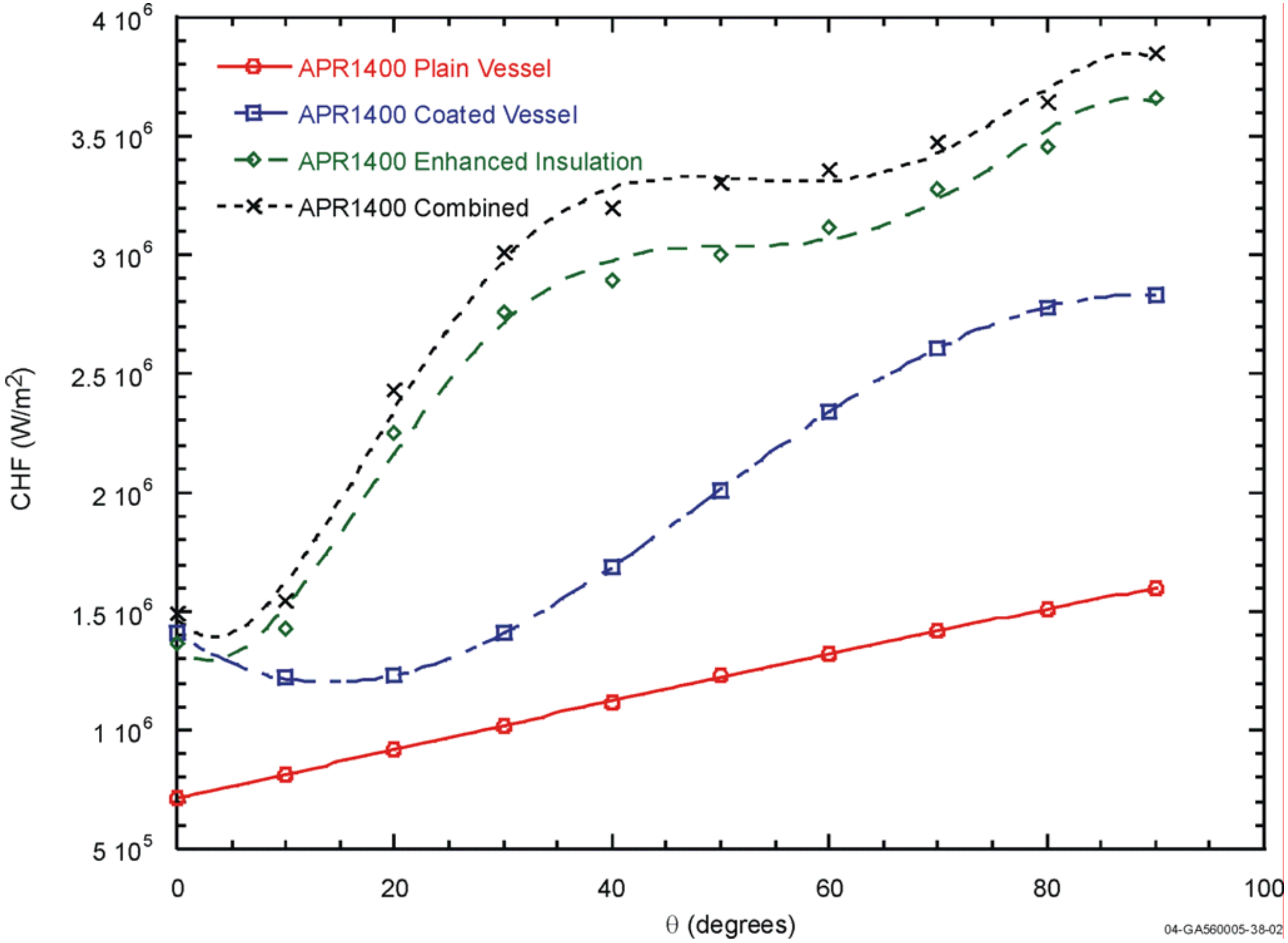
Various Approaches Used to Investigate ERVC (continued)

Program	Description	Subcooling (^oC)	Critical Heat Flux (kW/m^2)
FAI Quench	Quench tests with pipe cap welded to cylinder (0.21 m OD)	0	>1000 (Not observed)
SNL CYBL	SS heated torispherical tank (3.7 m OD/6.8 m high)	0	> 200 (Not observed)
UCSB ULPU	SS heated 2D full-scale slice (2 m outer radius)	0-14	~500 to 1500
CEA SULTAN	SS electrical heating of a flat plate (15 cm wide/4 m long)	0-50	~500 to 1500
Penn State SBLB	Quench and SS heated hemisphere (0.305 m OD)	0-10	~400 to 2000

Similar Trends Predicted With Correlations Obtained for Vessels Without Insulation



IVR Margin Increased With Vessel Coatings and Enhanced Insulation/Vessel Configuration



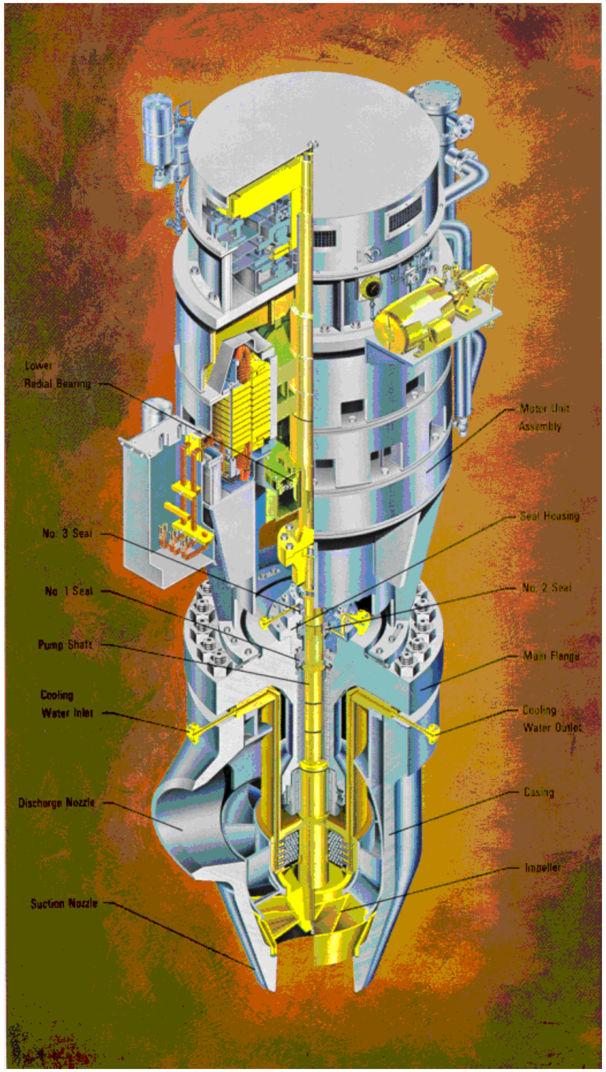
RCS Depressurization Issues

- Which RCS depressurization mechanisms may occur prior to vessel failure?
- How do various depressurization mechanisms affect subsequent accident progression, such as potential for vessel failure, High Pressure Melt Ejection (HPME), Direct Containment Heating (DCH), or containment bypass?

Failure Mitigation Measures

RCP Seal Leakage Increases During SBO

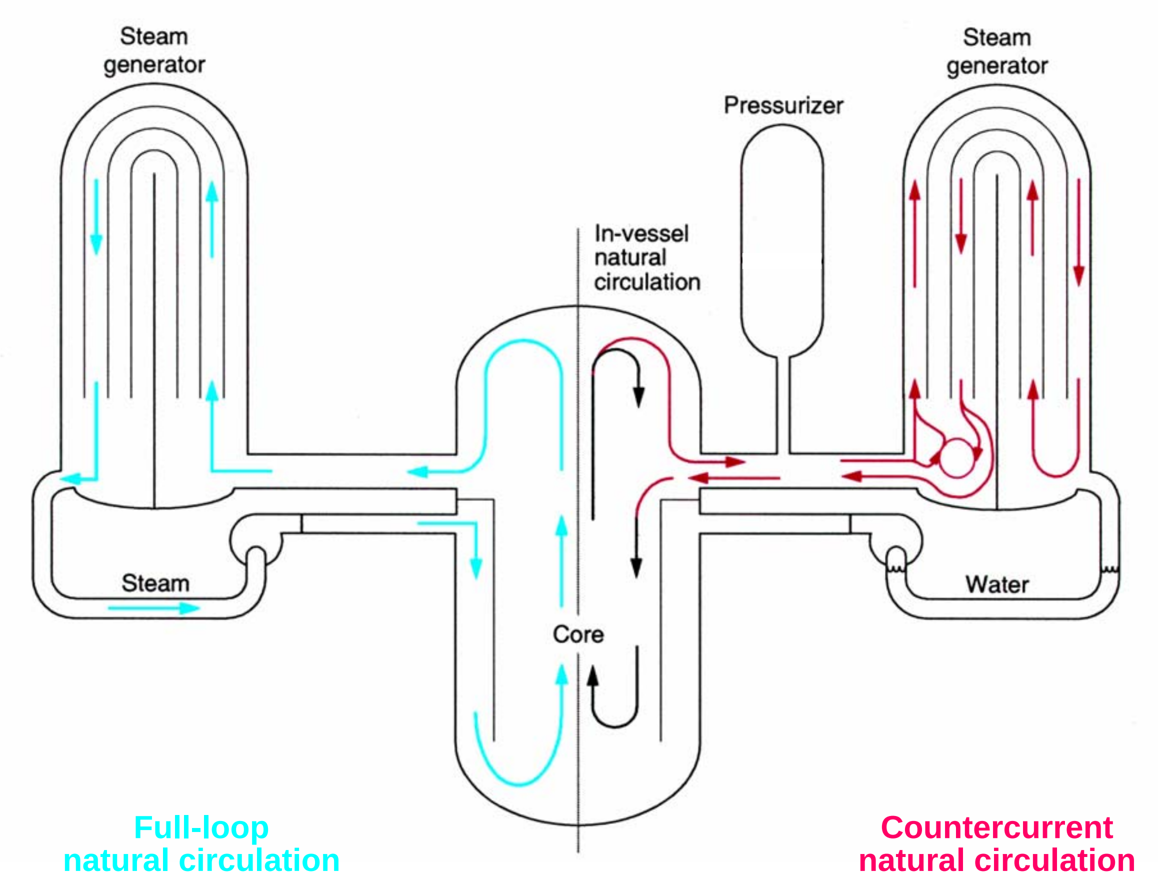
- During SBO, RCPs drive motors de-energize and RCPs lose seal injection and seal cooling water.
- Once seal cooling water purges, high temperature RCS water flows through and degrades pump seals.
- Seal leakage rates increase (from 3 gpm per pump to 21 to 480 gpm per pump) in Westinghouse pumps with older seal designs
- Increased leakage contributes to depletion of RCS inventory and core uncovery.



NUREG-1150 quantification still valid if RCP seals not upgraded.

- Westinghouse and AECL data used to identify and measure flowrates associated with various combinations of three failure mechanisms
  - seal ring binding
  - seal ring popping open
  - elastomer O-ring failure
- Probability of various failure combinations quantified using expert opinion
- Improved elastomers developed that are less susceptible to failure (must determine if installed in plant)

Natural Circulation Heating Affects RCS Piping Integrity



NUREG-1150 quantified induced ex-vessel piping failure with expert opinion.

- Aggregate distributions derived from individual pdfs provided by three experts for three cases:
  - Case 1: TMLB' sequence
  - Case 2: Seal LOCA w/o auxiliary feedwater
  - Case 3: Seal LOCA with auxiliary feedwater
- Experts derived pdfs for hot leg or surge line failure using results from available code calculations and experiments.
  - Aggregate pdf for Case 1 indicates hot leg LOCA very likely (mean value of 0.72 in Surry pdf).
  - Experts agreed hot leg failure unlikely for Cases 2 and 3 (mean value of < 0.03 in Surry pdf).
- NUREG-1150 model neglected time-zero seal failures identified in expert elicitation.

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

- Calculations performed for wide spectrum of SBLOCAs (no seal leaks, 250 gpm/pump and 480 gpm/pump) 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

S/R5 RCS Failure Predictions Based on Larson-Miller Creep Rupture Theory

- Rupture time given by

$$t_r = 10^{\left[\frac{P}{T} + C_1\right]}$$

where $P = C_2 \log\sigma + C_3$

T = absolute temperature

C_1, C_2, C_3 = empirically derived constants

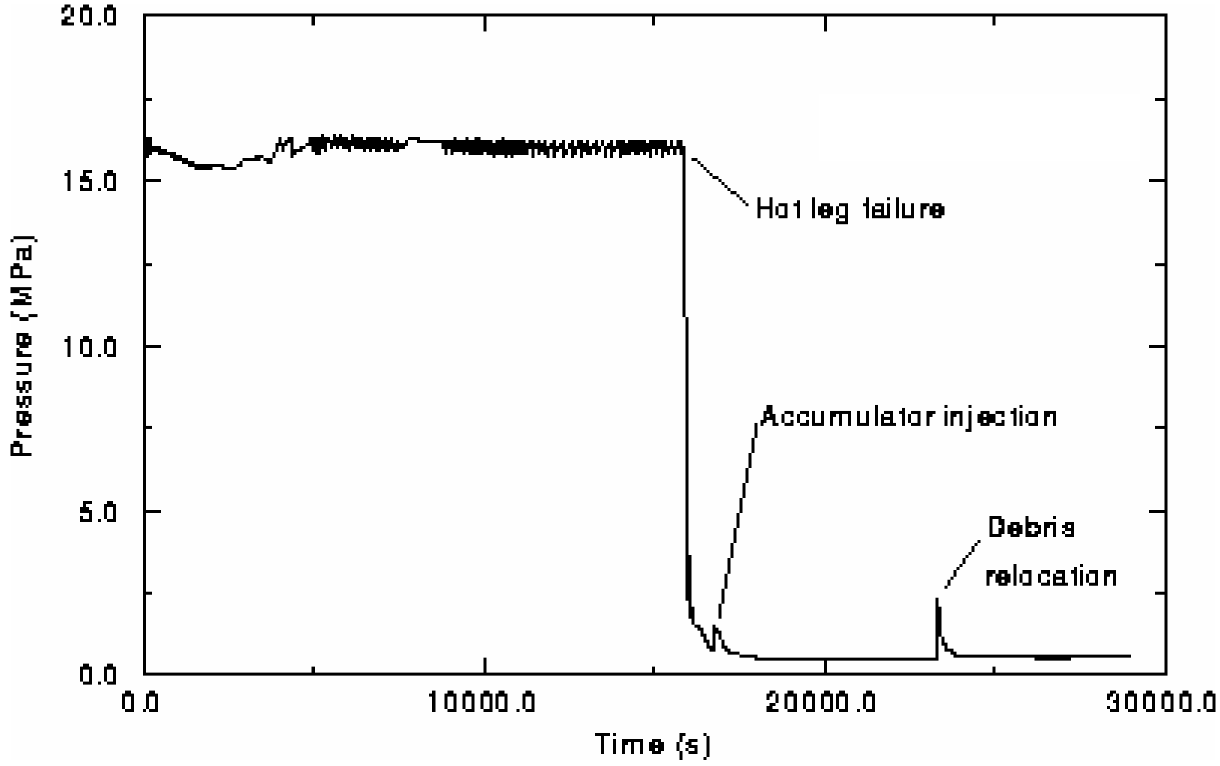
- Creep damage index given by

$$CDI(t + \Delta t) = CDI(t) + \frac{\Delta t}{t_r}$$

where t = current problem time

Δt = time step

Surge Line or Hot Leg Failure Predicted Prior to Major Relocation



Steam generator tube failures allow direct release to environment during SBO events

- Station blackout (SBO) accidents significantly contribute to core damage sequences with high reactor coolant system (RCS) pressures and dry steam generator (SG) secondaries
- Such conditions threaten integrity of SG tubes (and other RCS pressure boundaries)
- SG tube analyses critical because of associated radioactive release potential

NUREG-1150 quantified SGTR potential using expert elicitation.

- Experts asked to assume TMLB' case
  - RCS pressure at or near PORV setpoint value (if RCS pressure low, experts estimated zero SGTR probability)
  - Steam generators dry
  - Most core flow exits PORV through hot leg containing pressurizer
- Experts derived pdfs using available code calculations and operational experience.
  - Older calculations rarely estimate tube temperatures
  - Little relevant operational experience

NUREG-1150 quantified SGTR potential using expert elicitation. (continued)

- If SG tube defects neglected, experts indicated that:
  - SGTR frequency correlated with hot leg failure frequency
  - SGTR failure occurs after hot leg or surge line failure
- Experts disagreed on impact of tube defects on failure.

NUREG-1570 Estimates Low SGTR Risk

- NUREG-1570 documents NRC NRR and RES staff working group results
- Estimated containment bypass frequency for Surry cases with and without flawed steam generator tubes.
  - For cases with unflawed tubes, estimated frequency of 1.7×10^{-8}/RY, due to temperature-induced SGTR as a result of RCP seal leakage with concurrent loop seal clearing SG depressurization
  - For flawed tubes, predicted frequency of 3.9×10^{-6}/RY (**factor of four lower** than predicted in NUREG-1150)

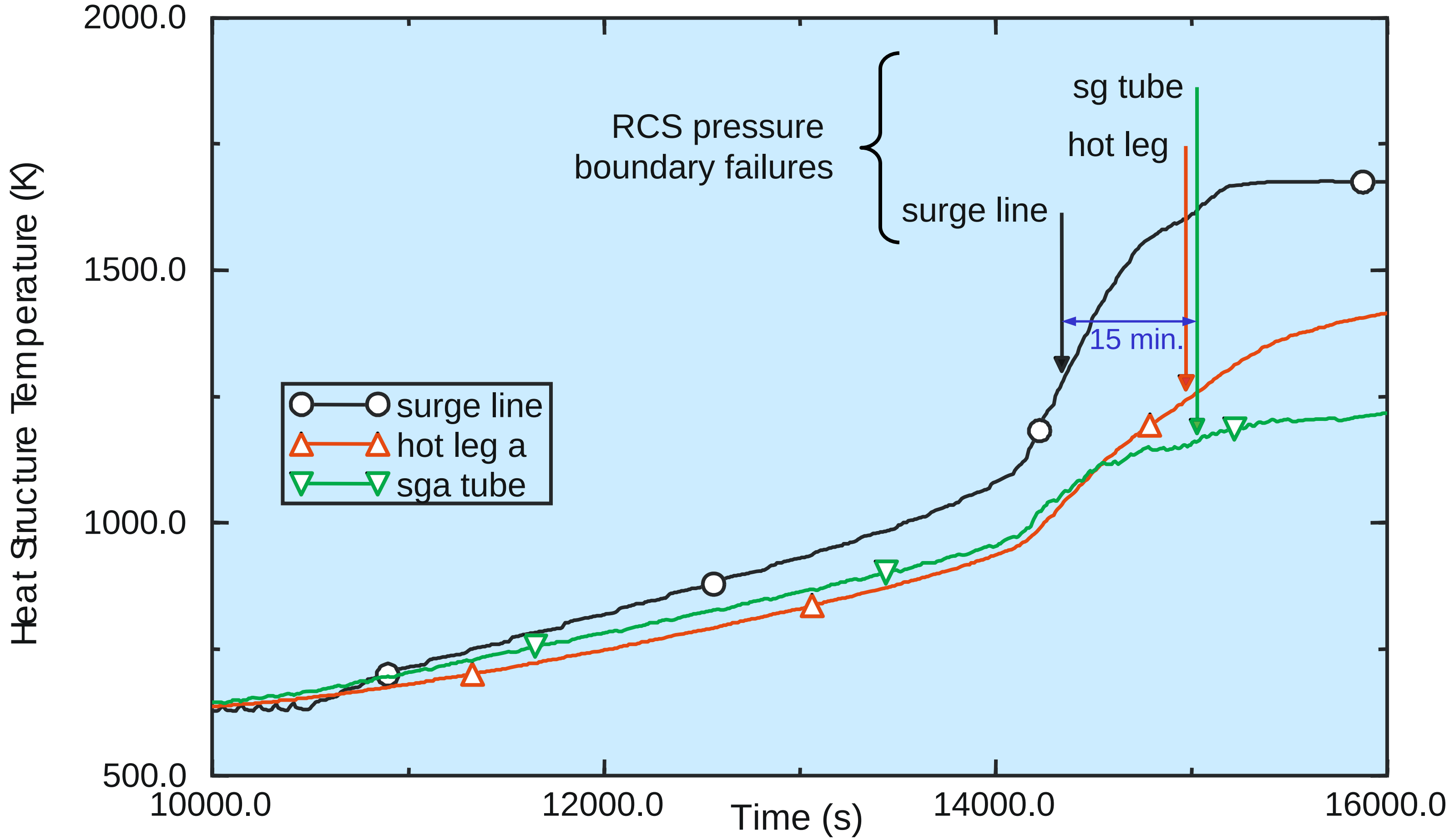
NUREG-1570 Estimates Low SGTR Risk (continued)

- Developed creep-based tube failure model (based on ANL testing)
- Difficult to extrapolate to other designs
  - Superheated gases not expected to reach B&W Once-Through Steam Generator (OTSG) tube bundles if loop seal isn't cleared
  - Tube flaw distributions plant-specific
  - Estimated frequencies may range from 10^{-7} to 10^{-5} in U-tube designs.

S/R5 Calculations Provide Key Insights

- Range of PWRs (Zion, Surry, Calvert Cliffs, ANO-2, and Oconee) with unflawed SG tubes for SBOs with and without SG depressurization considered.
- Key insights:
  - Cases with depressurized secondary side (via operator action, stuck-open MSSV, stuck-open ADV with failure to isolate, or failure to isolate steam flow to AFW pump) present more serious challenges to SG tubes.
  - Larson-Miller creep rupture model predicted surge line or hot leg failures prior to SG tube failures
  - RCS depressurization and accumulator injection following surge line/hot leg failure preclude SGTR

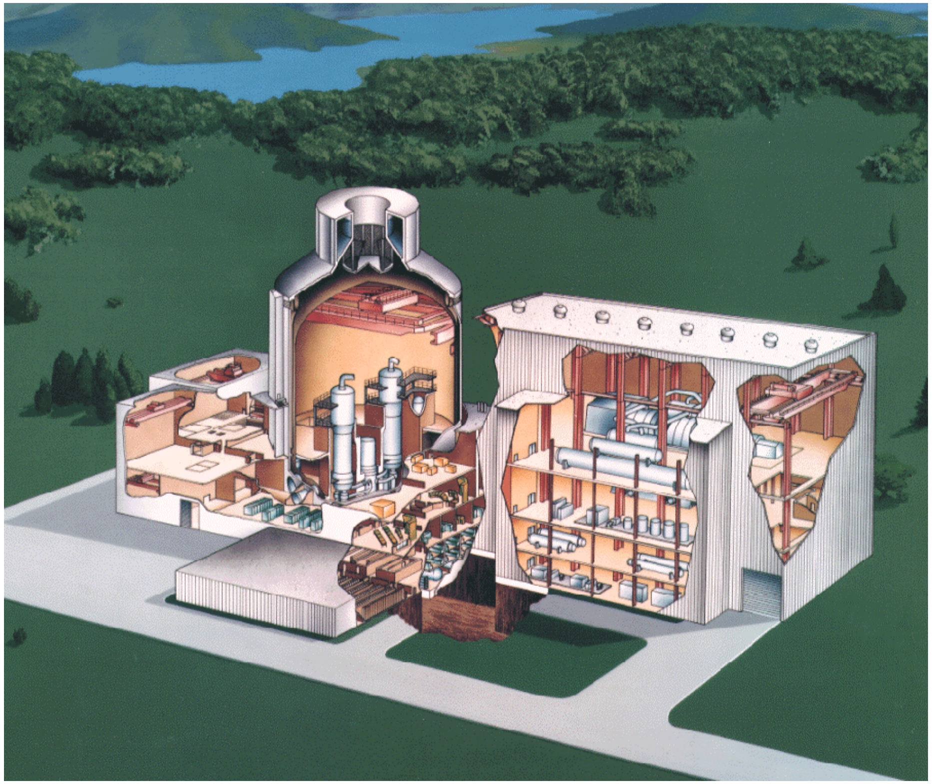
Margins Between Surge Line Failure and SGTR are Typically Small



Summary

- NUREG-1150 RCP seal leakage quantification still valid if RCP seals not upgraded
- Significant advances in understanding SGTR since NUREG-1150
  - For flawed tubes, Surry estimated SGTR frequency decreased by factor of four
  - Frequency estimates plant-specific (10^{-8} to 10^{-5} for U-tube SG, zero probability for B&W OTSGs if loop seal isn't cleared)
- Recent SCDAP/RELAP5 calculations suggest induced RCS piping failure prior to core relocation for wide spectrum of SBLOCAs
  - For unflawed SG tubes, hot leg or surge line failure predicted to occur prior to SGTR
  - Uncertainties in predicting natural circulation flows, heat transfer, and piping failure may reduce difference in failure time predictions.

Case Study: AP600 ERVC Submittal



Westinghouse Advanced PWR 600 MWe (AP600) focussed on simplicity

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

ERVC Central to Westinghouse AP600 Severe Accident Treatment

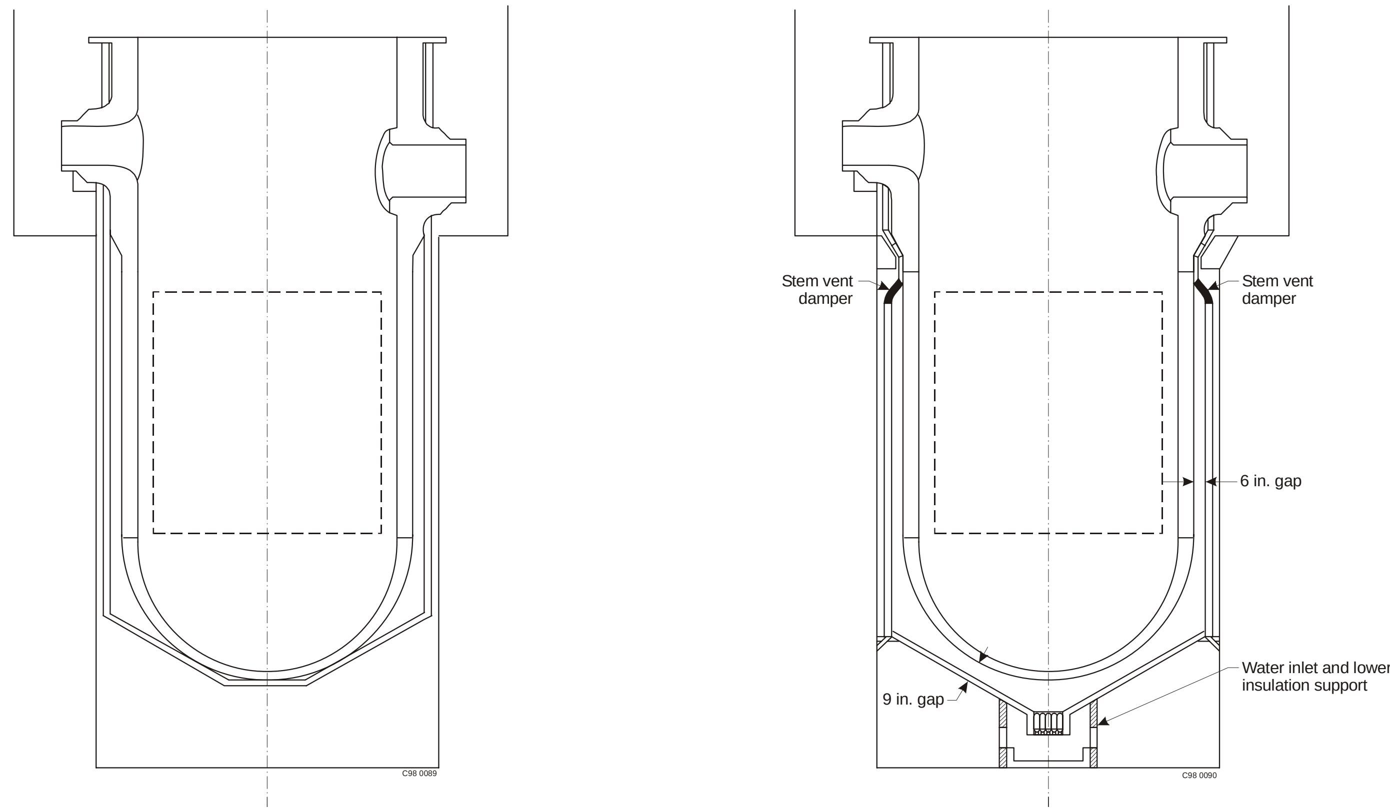
"In-vessel retention (IVR) of core debris by cooling from the outside is a severe accident mitigation attribute of the AP600 design. With the reactor vessel intact and debris retained in the lower head, there is no need to examine phenomena that may occur as a result of core debris being relocated to the reactor cavity."

- Westinghouse AP600 Probabilistic Risk Assessment

AP600 Designed to Rely on ERVC for IVR

- Low power density fuel
- Increased reliability of RCS depressurization system
- No lower head penetrations
- Vessel outer surface treatment promotes wetability
- Improved reliability of cavity flooding system
- In-containment Refueling Water Storage Tank (RWST)
- Modified insulation increases water ingress and steam egress rates
  - Increased RPV wall to vessel insulation gap
  - Structurally reinforced
  - Vessel designed to withstand thermal shock associated with one accident flooding

AP600 Insulation Evolved to Ensure ERVC



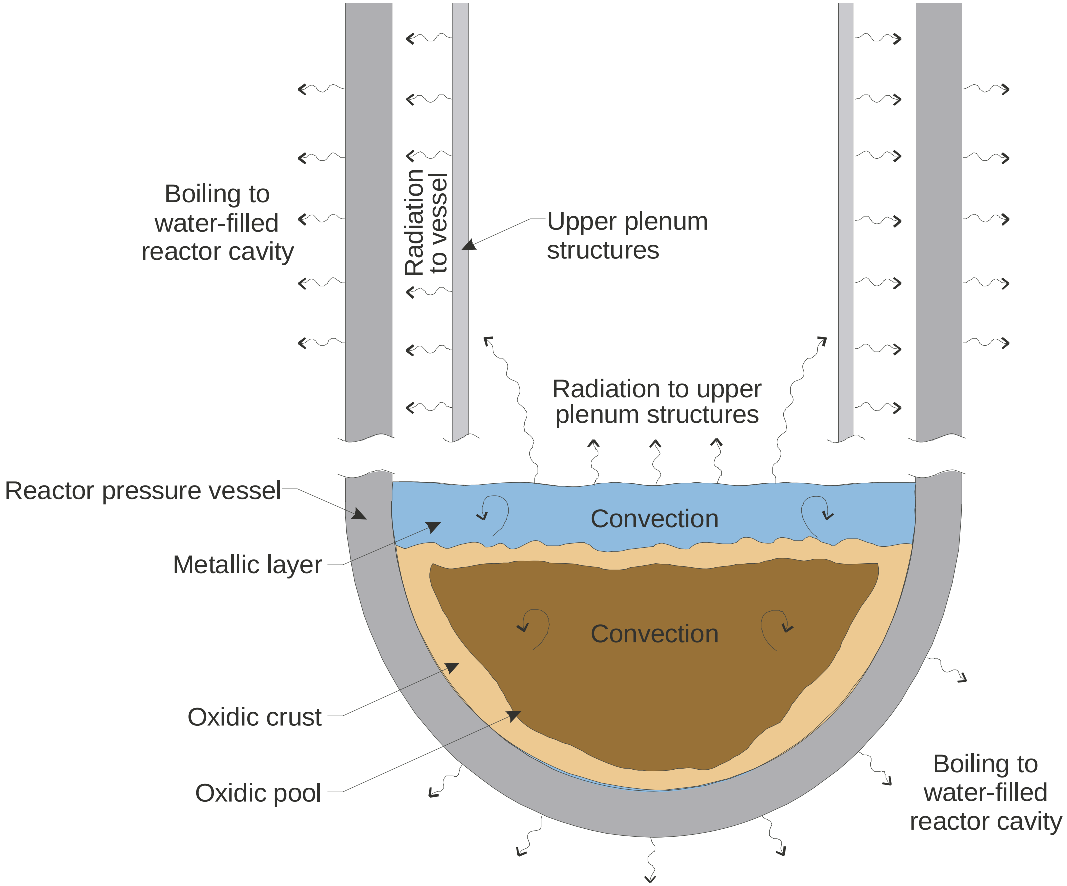
(a) Original AP600 insulation design

(b) Revised AP600 insulation design

ROAAM Analysis Concluded AP600 IVR Successful

- Concluded IVR successful after proving two assertions for their assumed debris conditions
  - ➔ Assertion 1 **The vessel remains intact if heat fluxes are at or below CHF**
  - ➔ Assertion 2 **Heat fluxes to the vessel are below CHF**
- Thermal and structural analyses with supporting experimental data applied for UCSB-assumed conditions
- Analyses based on numerous assumptions that UCSB considered "reasonable"
- Peer review used to validate analytical approach and input assumptions

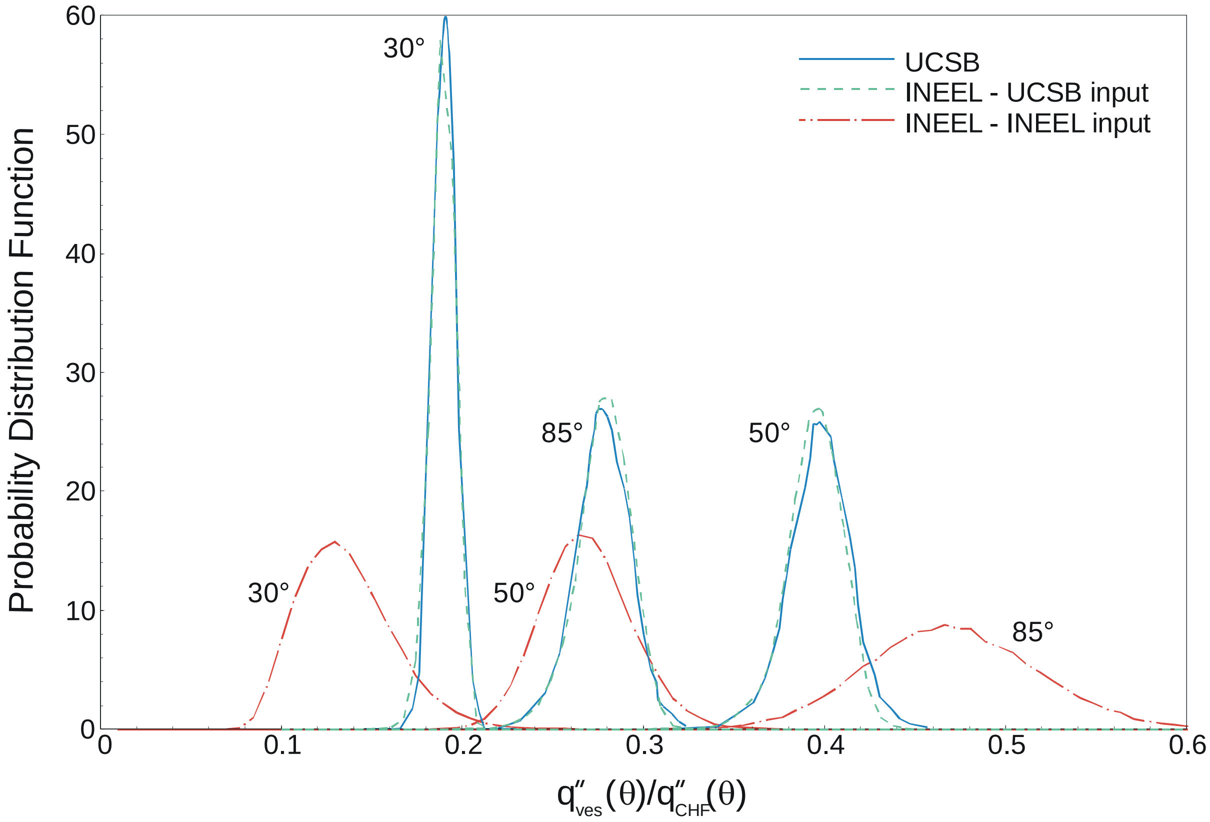
FInal Bounding State (FIBS) Debris Configuration Key UCSB Assumption



NRC-sponsored INL Review Reassessed Potential for AP600 IVR

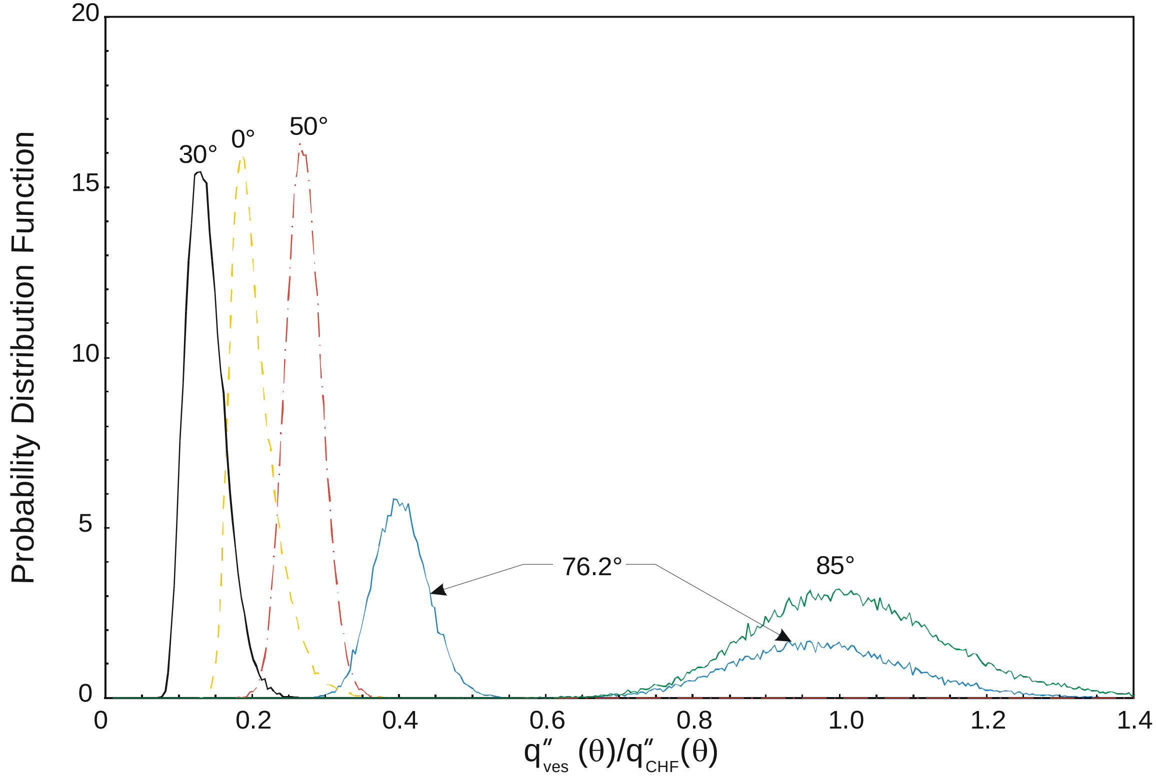
Input	UCSB	INL
Ceramic pool convection correlations and uncertainty	UCSB 1/8-scale Mini-ACOPO correlations with no uncertainty	UCSB 1/2-scale ACOPO correlations and uncertainty
Critical Heat Flux correlation and uncertainty	ULPU lower bound data with no uncertainty	Scaled Penn State SBLB correlations and uncertainty
Metallic layer heat transfer uncertainties	None	Variability between data and selected correlations
Decay power uncertainty	None, curve corresponds to 2σ lower bound for ANS standard one-group curve	Values from 1979 ANS 5.1 standard considering 3-group behavior
Metallic layer heat source	None	Fraction of fission products retained in metallic layer
Melt relocation time	Figure 7.7	Figure shifted forward by one hour based on severe accident analysis code results
Material properties	Single values; some have uncertainties	Temperature and/or composition-dependent values with uncertainties; incorporated additional experimental data

INL Reassessment Confirms IVR Successful for UCSB-Assumed FIBS



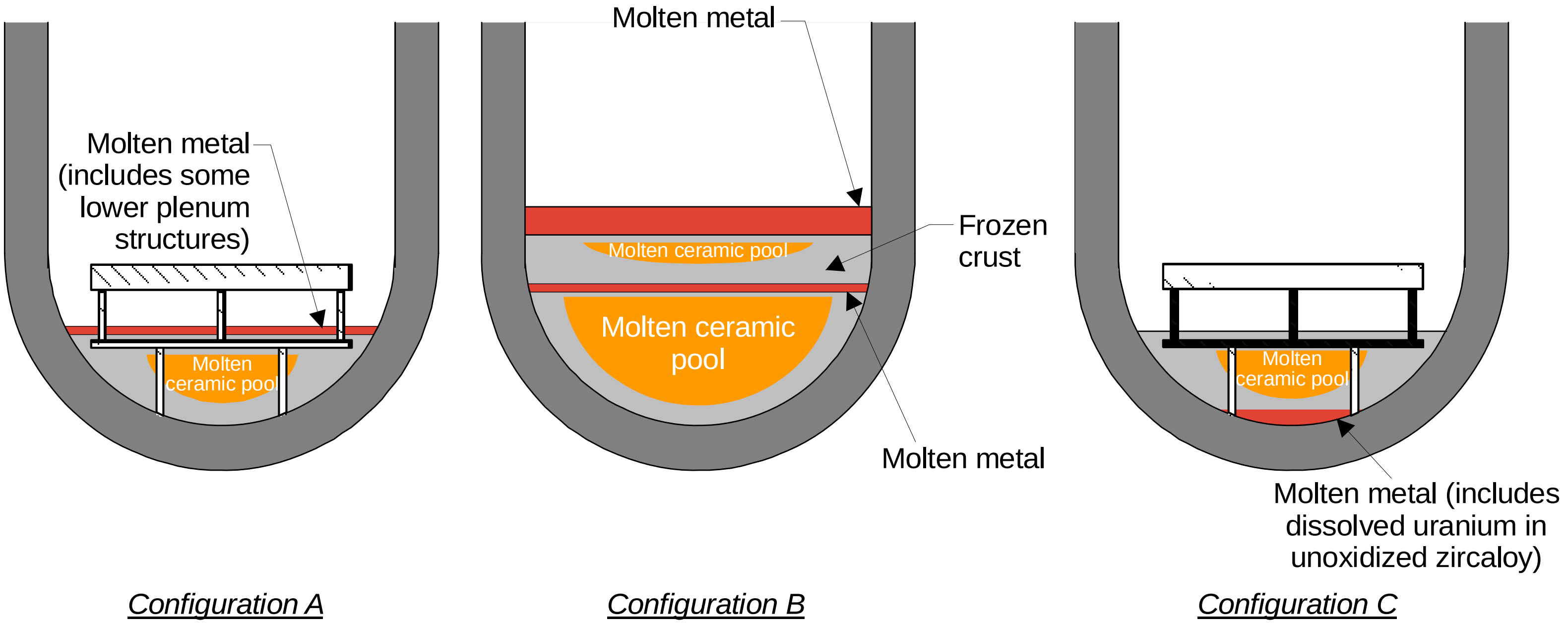
Significantly different failure margins predicted with INL input

Metallic Layer Mass Uncertainties Reduce Metallic Layer Failure Margin

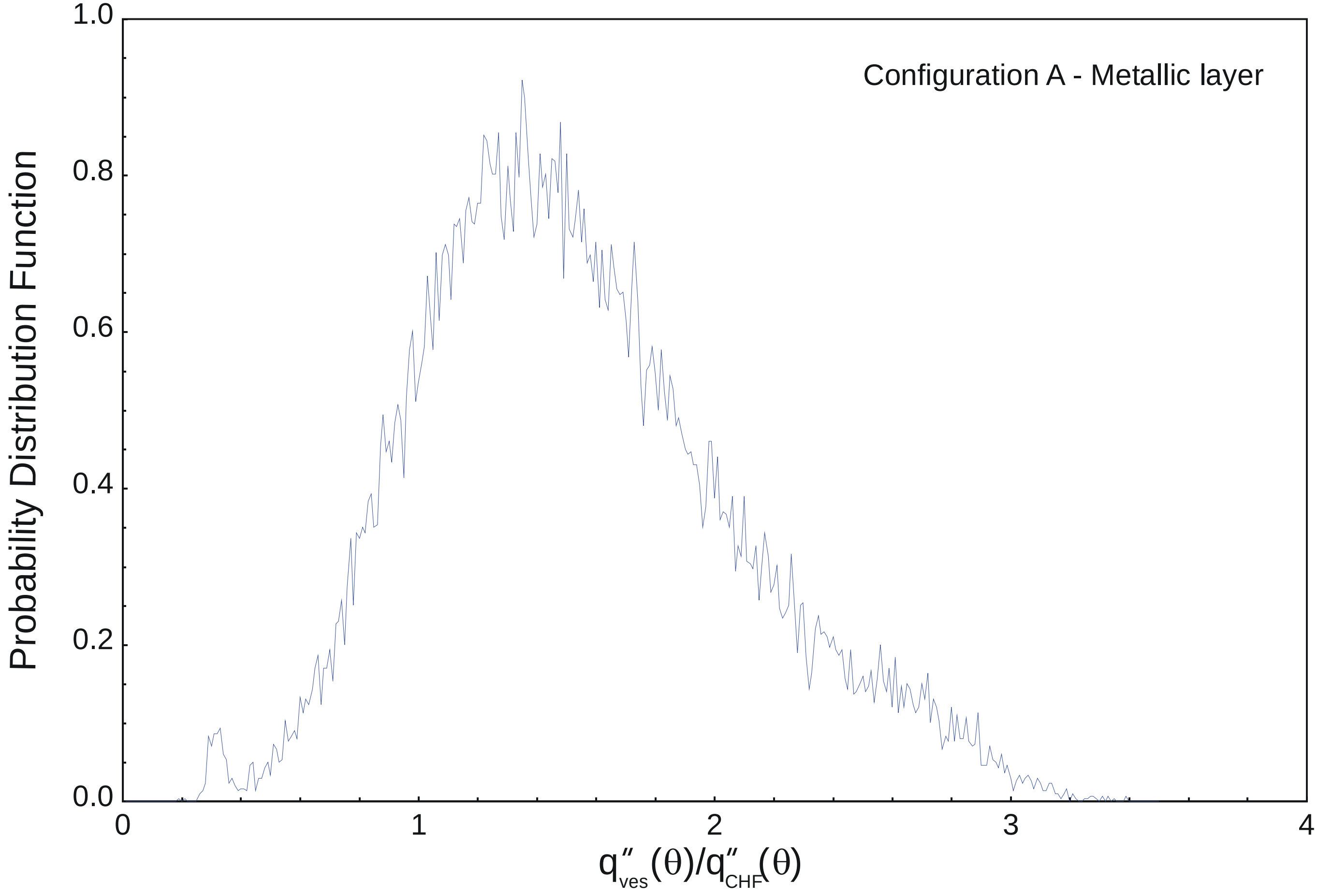


Factor of 4 reduction in metallic layer mass results in 52% probability of exceeding CHF at 85°.

Alternate Debris Configurations Also Considered



INL Assessment Indicate Heat Fluxes Exceed CHF for Configurations A, B, and C



INL Review Suggests AP600 ROAAM IVR Analysis not Conclusive

- INL calculations confirm that AP600-like vessel remains intact for UCSB-assumed "bounding" debris endstate
  - smaller failure margins predicted at locations near metallic layer
  - some phenomenological uncertainties significantly reduce, if not eliminate, failure margins
- INL assessments of alternate debris configurations indicate heat fluxes exceed CHF

AP600 Design Certification Not Impacted by Failure to Demonstrate Successful ERVC

- INL applied bounding approach to determine maximum increase in AP600 plant risk if all cavity flooding scenarios lead to vessel failure.
  - Data insufficient to estimate probability of various proposed debris configurations.
  - Conservatively assumed ERVC failure resulted in vessel and containment failure.
- AP600 plant risk still below Westinghouse design goal
  - Factor of 6 below 1 x 10^{-6} per year design goal from EPRI ALWR Requirements Document.
  - Sum of probabilities of events with releases that lead to doses exceeding 25 rem increases by factor of 20 to 1.6 x 10^{-7} per year

NRC Staff and ACRS Concur That UCSB IVR Study not Conclusive

"The analysis of in-vessel retention performed for the AP600 fails to demonstrate convincingly that vessel failure during a core melt is extremely unlikely. …The NRC staff ... has concluded that the possibility of reactor vessel penetration cannot be excluded. We agree with the staff's conclusion. ...Even discounting retention within the vessel and assuming containment vulnerability, the AP600 poses low risks to the public relative to existing reactors..."

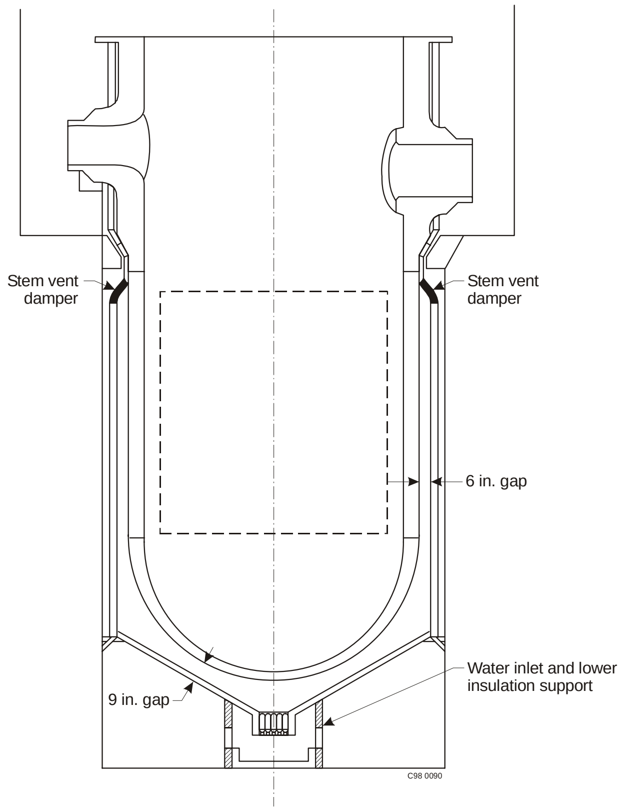
- letter from R. L. Seale, Chairman, ACRS, to L. J. Callan, EDO, US NRC, dated June 15, 1998

Problem 1: How would AP600 analysis change if power level were increased to 1200 MWe?

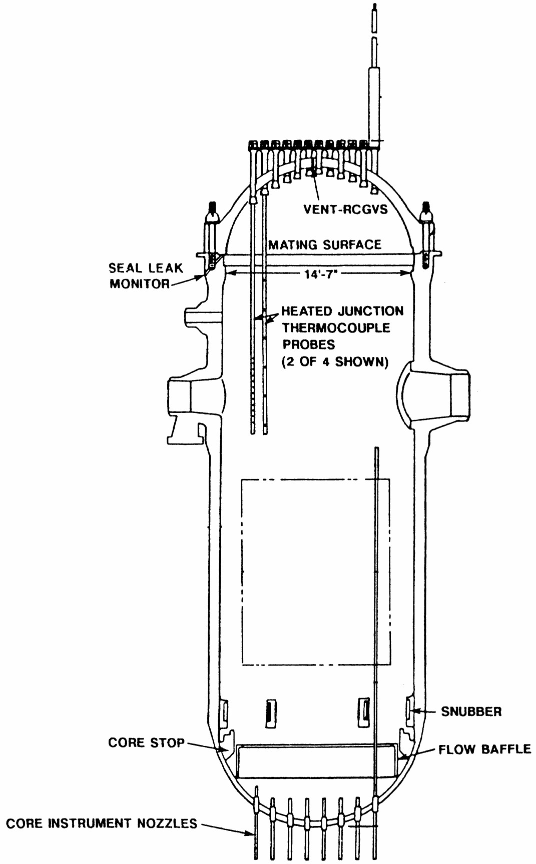
Comparison of AP600 and Large PWR Design Parameters

Parameter	AP600	Large PWR
Power, MWe	600	1200
Vessel diameter, m	4	4.4
Mass of Relocated Materials , kg		
UO_2	75,900	102,000
Zr	19,200	29,100
SS	70,000	58,100
Melt relocation time, seconds	1.62×10^4	1.7×10^4
Decay Power Density (at relocation time), MW/m^3	1.3	1.5
Upper Plenum Structure Surface Area, m^2	75.36	88.0
Minimum insulation –to-vessel thickness, in	6	4

Problem 2: How would AP600 analysis change if the CE System 80+ plant were evaluated?



AP600



CE System 80+

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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- *Papers presented at the OECD/CSNI Special Meeting on In-vessel Debris Coolability and Lower Head Integrity,* Paris, France, November 1996.
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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.

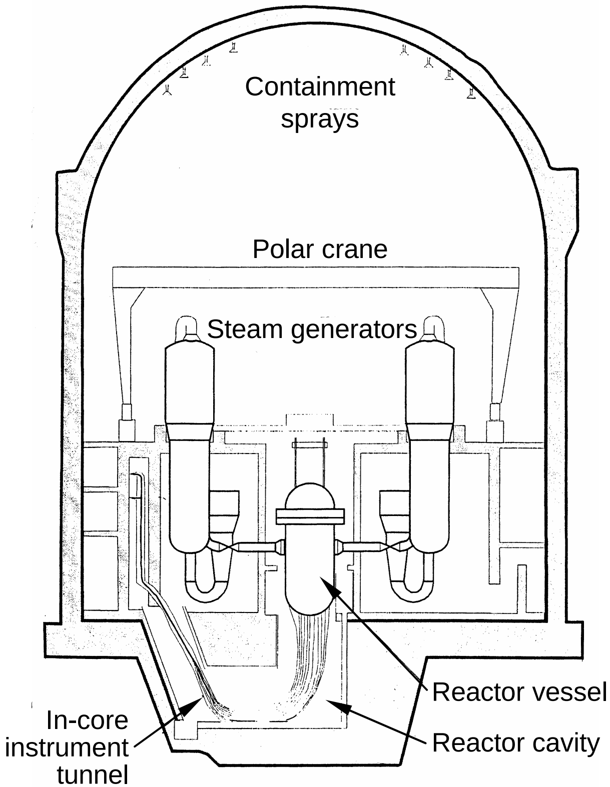
Several Challenges to Containment Integrity

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

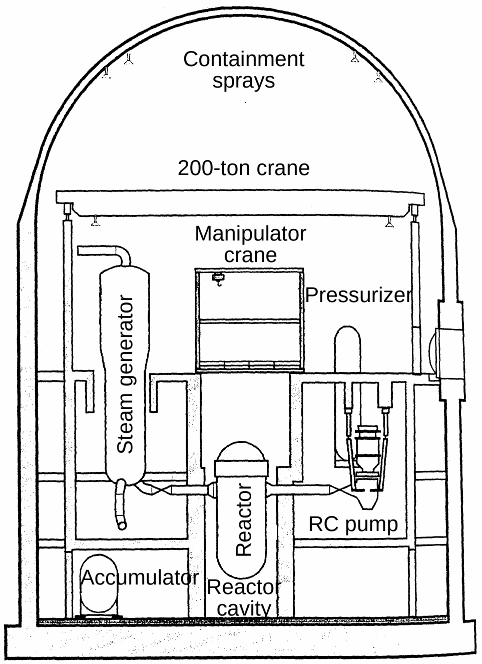
Challenges Dominate at Different Time Periods

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

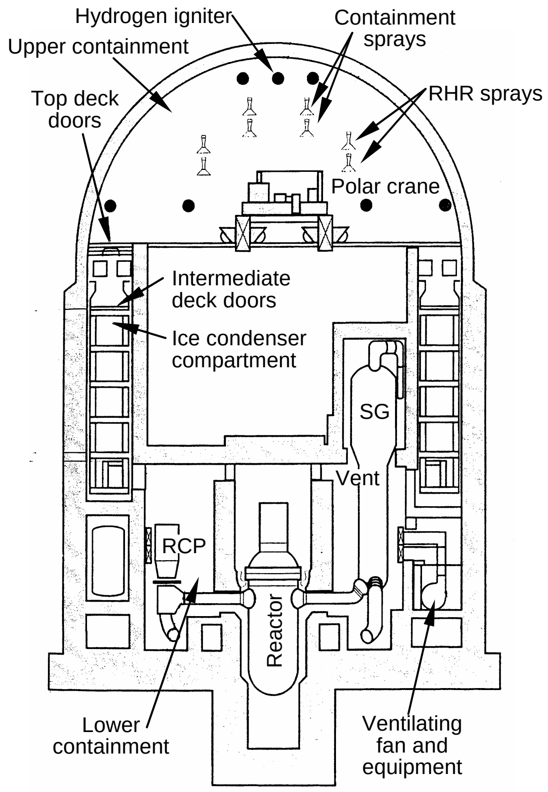
PWR Containment Designs Differ



Large dry

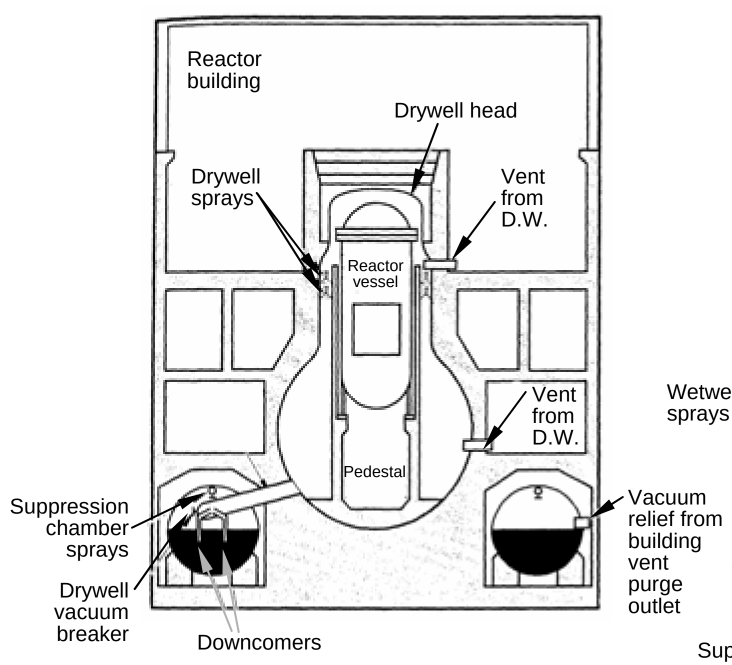


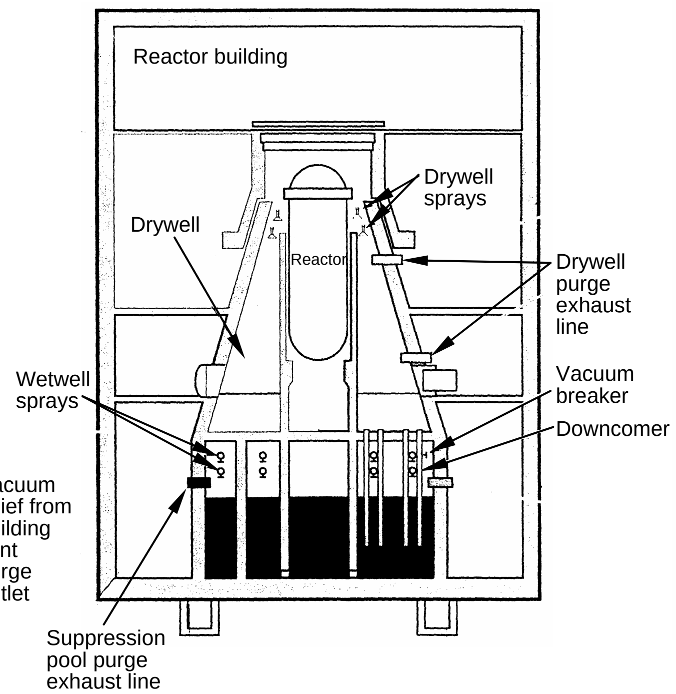
Subatmospheric

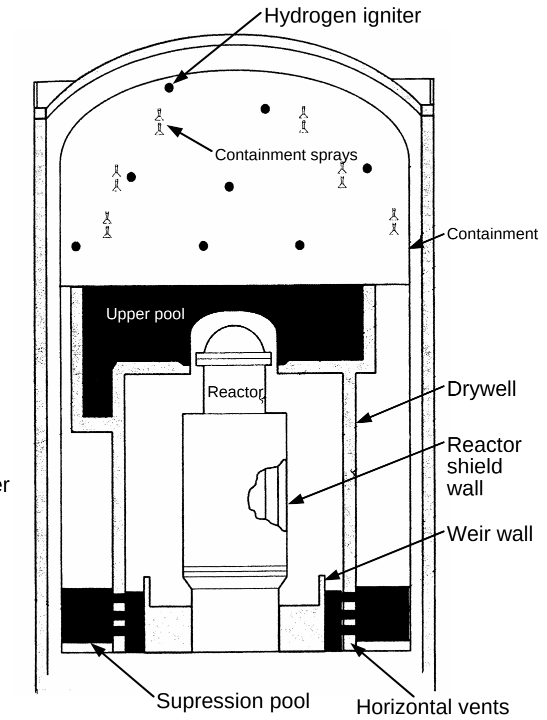


Ice condenser

BWR Containment Designs Differ

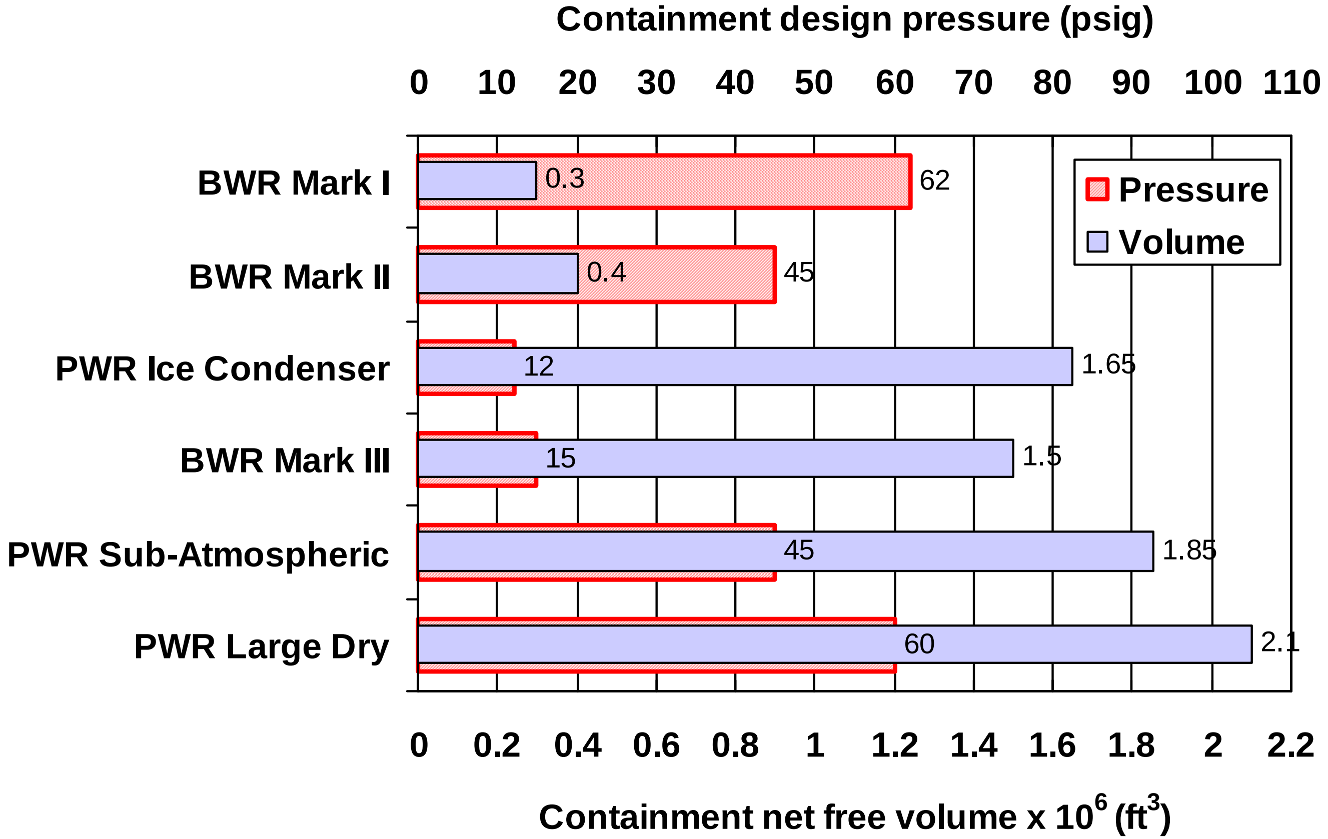




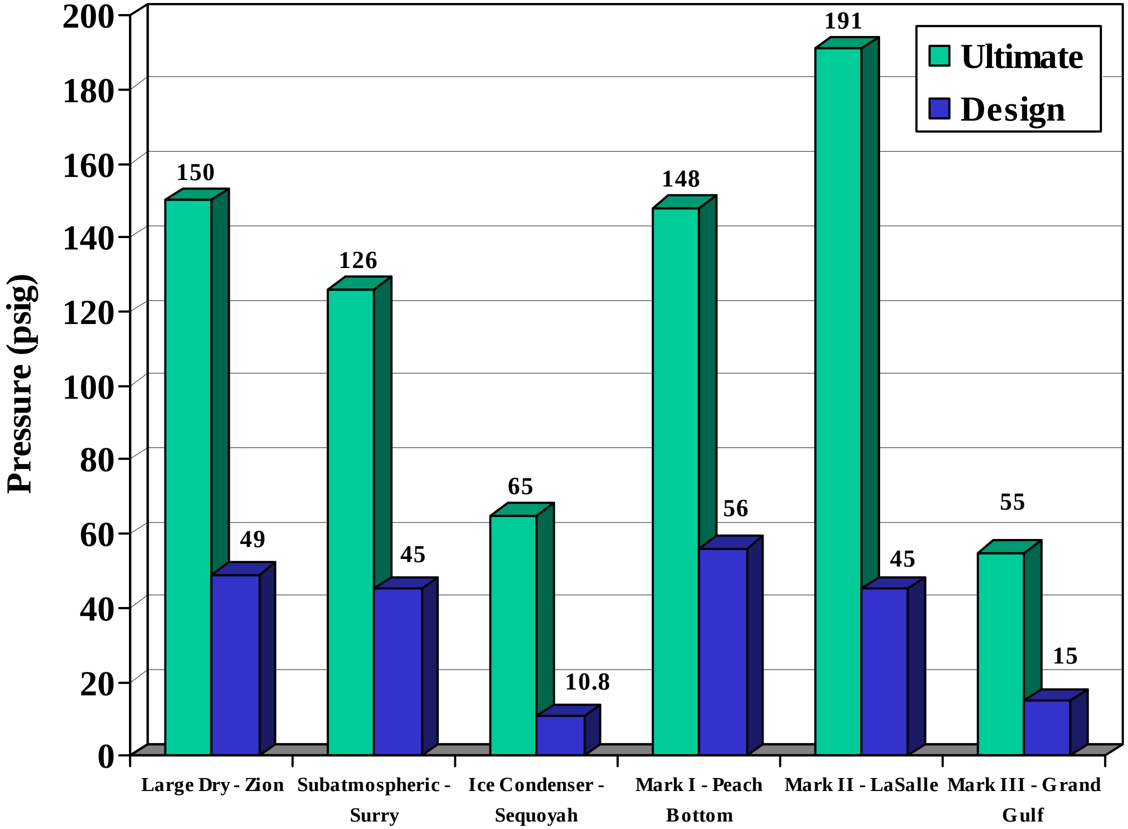


Mark I **Mark II** **Mark III**

Containment Free Volumes and Design Pressures Differ



Failure Pressures Significantly Higher than Design Pressures



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?

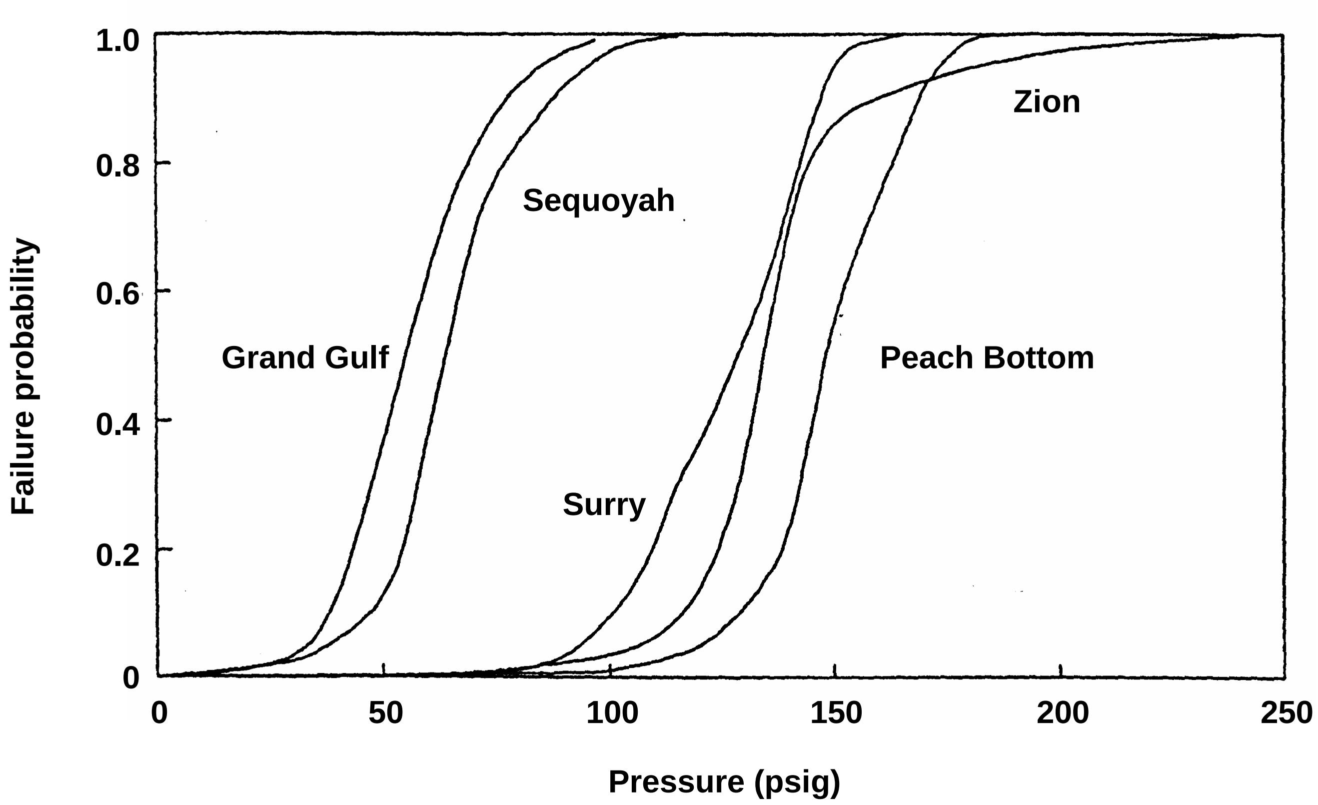
Containment Structural Response Characterized with Fragility Curve

- Probabilistic description of internal pressure capacity of containment structure
- Cumulative probability of failure as a function of internal pressure (assumed static and uniform)

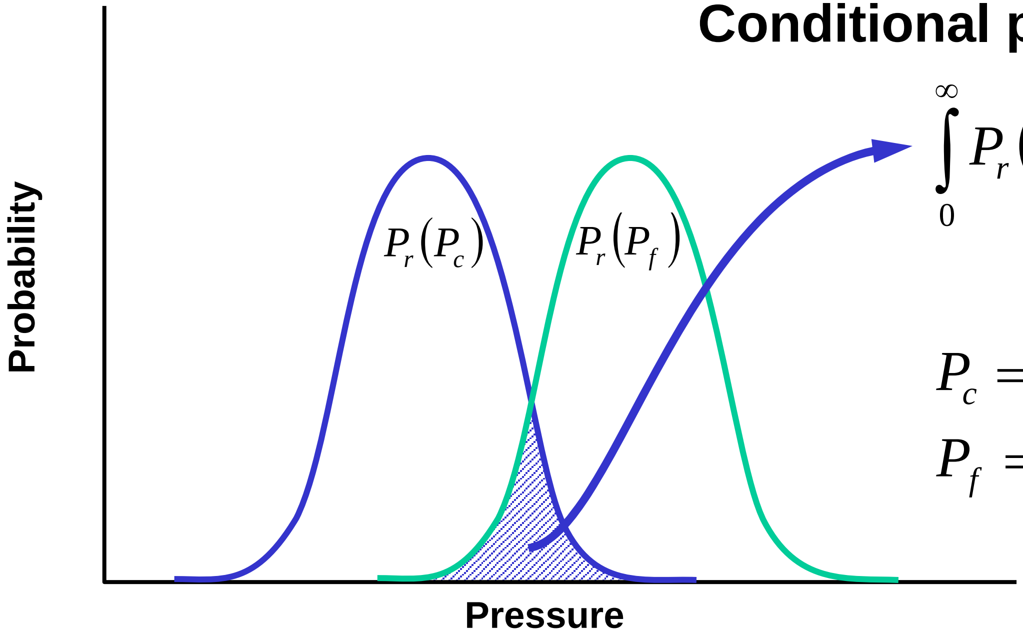
Containment Structural Response Characterized with Fragility Curve (continued)

- Combines individual fragility curves for different failure mechanisms
  - membrane failure in hoop direction
  - membrane failure in meridial direction
  - radial shear failure at cylinder to basemat or dome to cylinder discontinuity
  - bending failure in basemat
  - shear failure in basemat
  - shear failure in shell at penetrations
  - membrane, bending or shear failure in penetrations.
- Formulation assumes independence among different failure modes

NUREG-1150 Fragility Curves Suggest Mark III and Ice Condenser Containment More Susceptible



Conditional Probability for Containment Failure for Each Sequence Calculated Probabilistically



Conditional probability of failure =

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

$$P_c = Peak\ containment\ pressure$$

$$P_f = Containment\ failure\ pressure$$

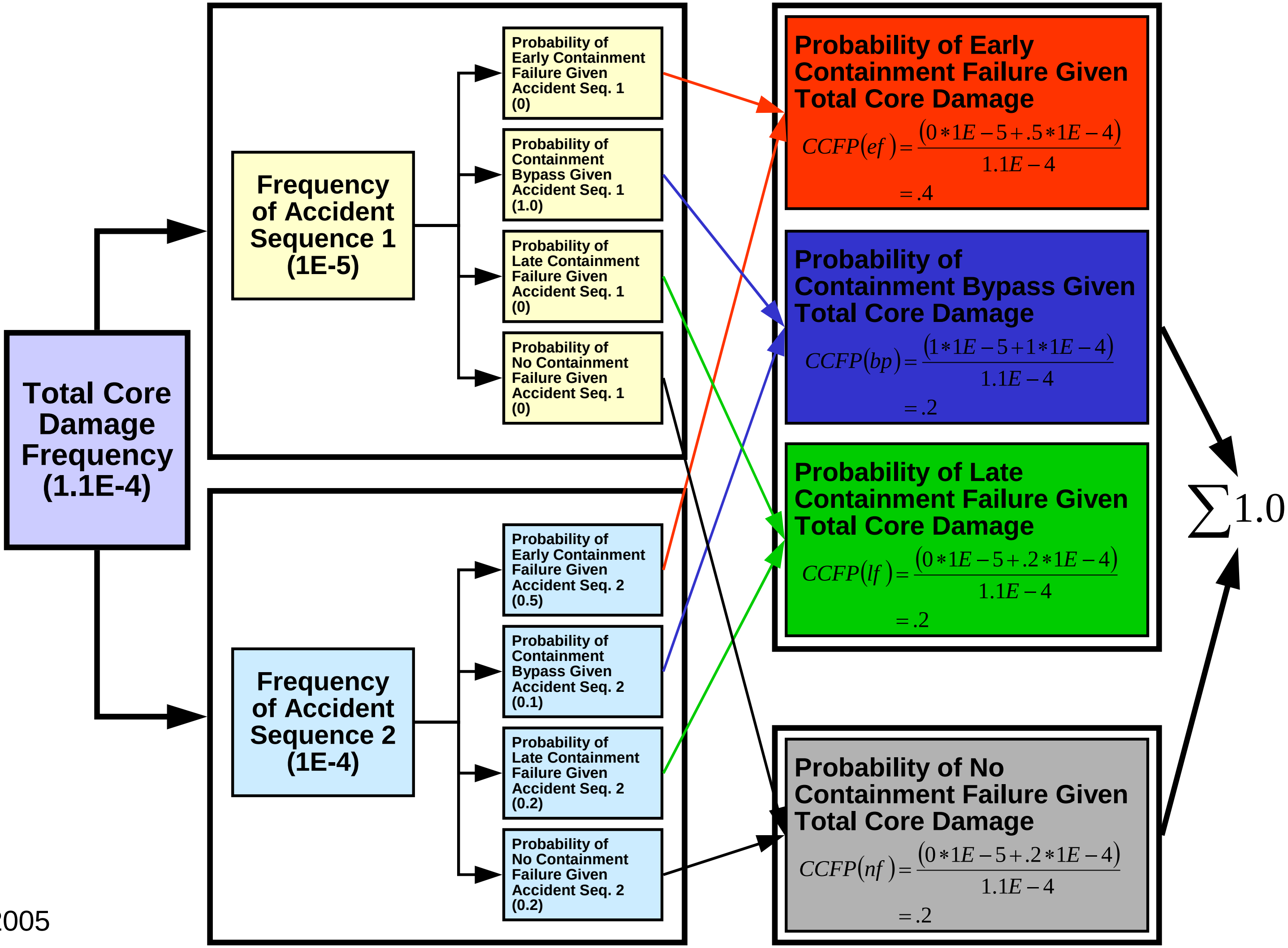
Two Measures Typically Cited for Assessing Containment Performance

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

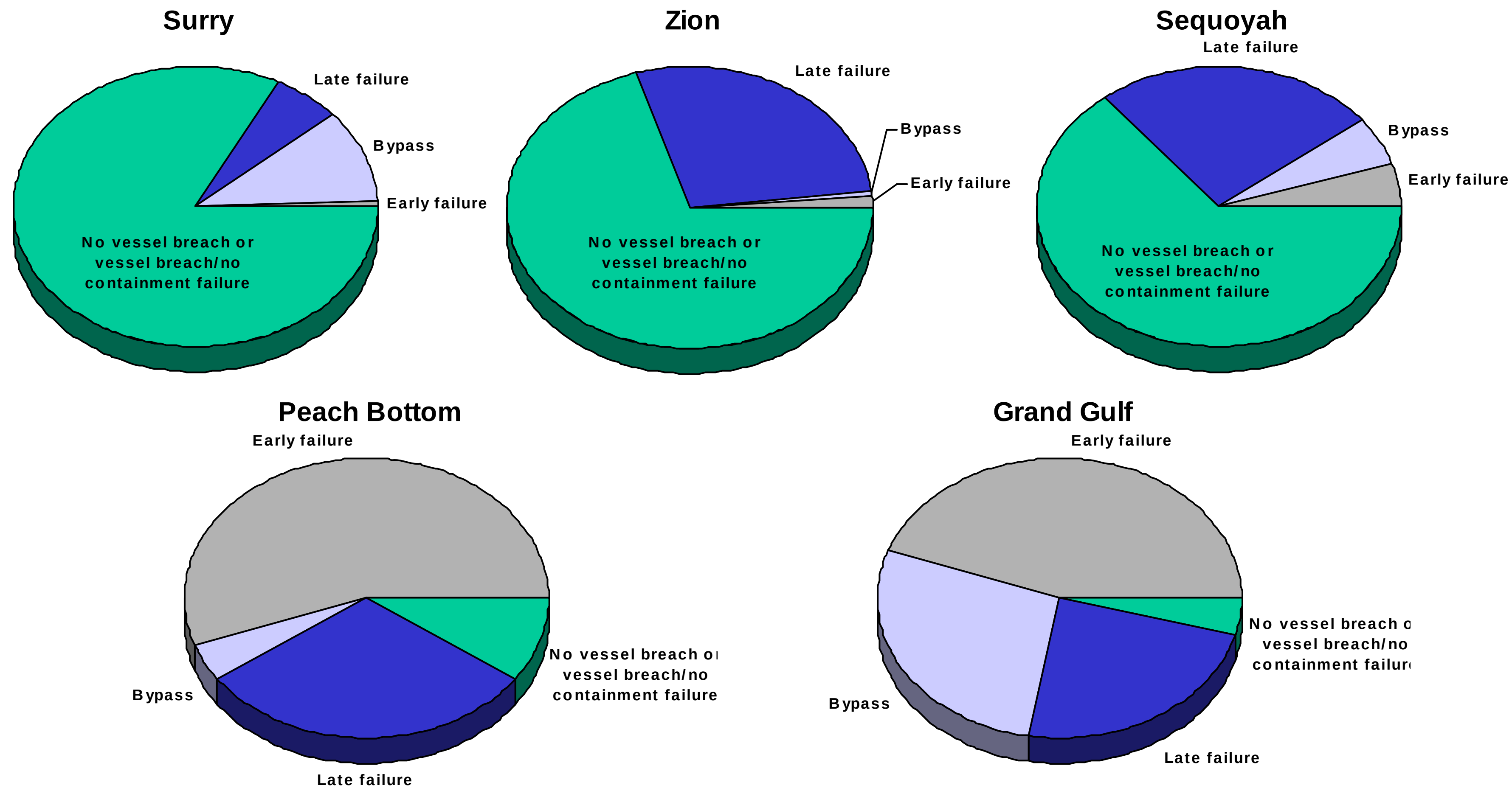
Containment Failure Frequency = CFF = $\sum_{i=1}^{n} S_i C_i$

S_i => frequency for accident sequence, i
C_i => containment conditional failure probability given accident sequence, i
n => total number of accident sequences
CDF => core damage frequency

Containment Event Trees Quantified by Propagating through Various Accident Events

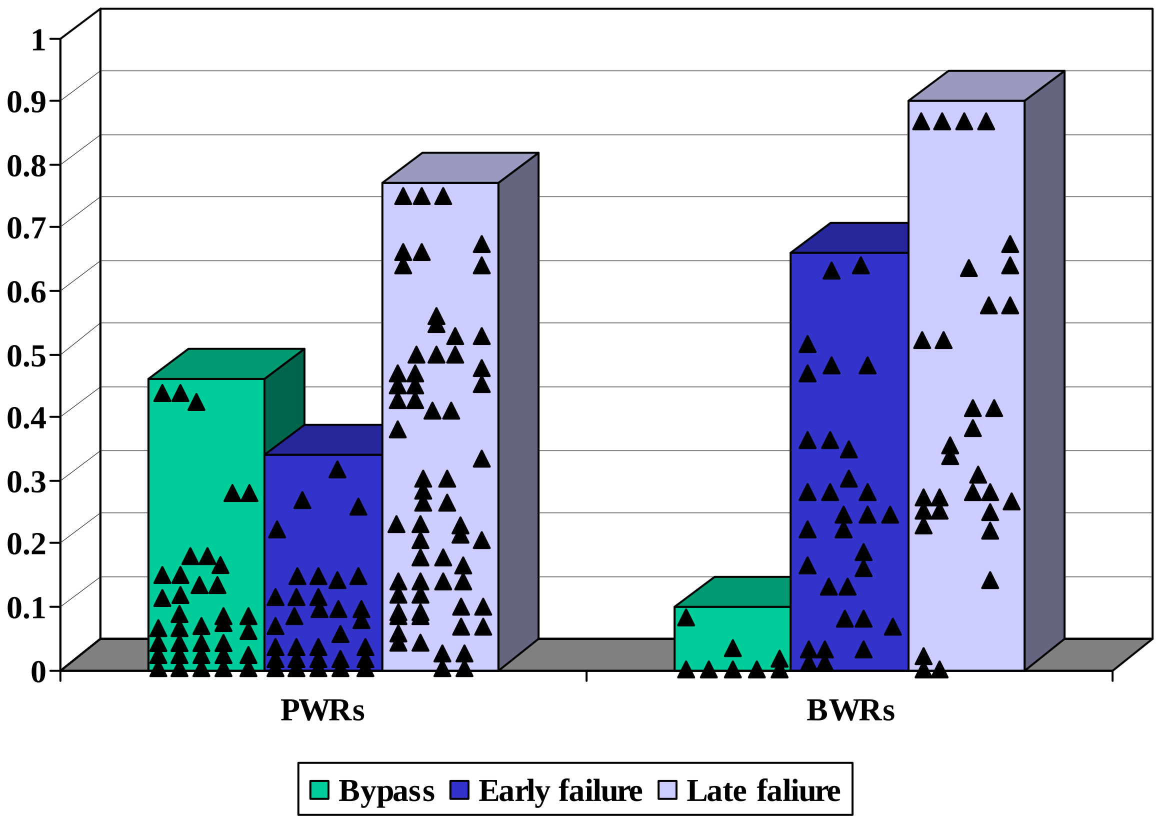


NUREG-1150 Results Indicate BWR Early Containment Failures More Likely



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

More Recently completed Individual Plant Examinations (IPEs) Suggest Late Failures Dominate



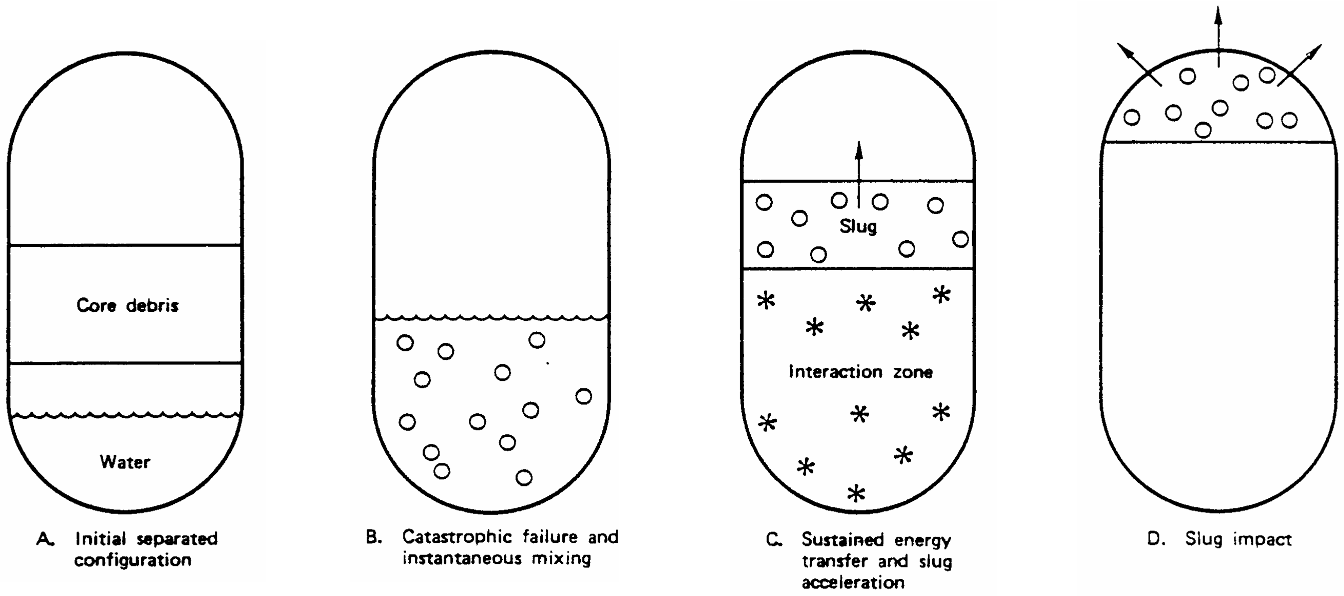
General Insights from IPE Containment Response Analyses

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

Key Phenomena Challenging Containment Integrity

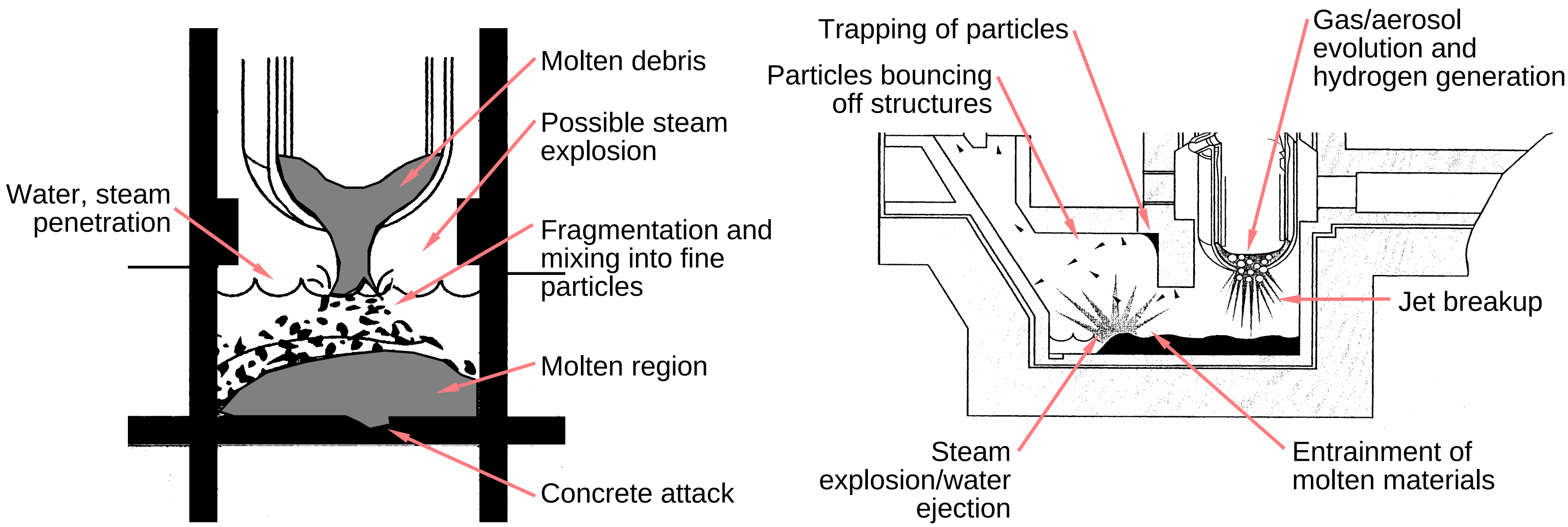
- In-vessel steam explosions
- Ex-vessel steam explosions
- Direct containment heating (DCH)
- Molten core concrete interactions (MCCI)
- Hydrogen combustion
- Meltthrough

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.

NUREG-1150 indicates ex-vessel SEs of concern for BWR Mark I and Mark II plants.

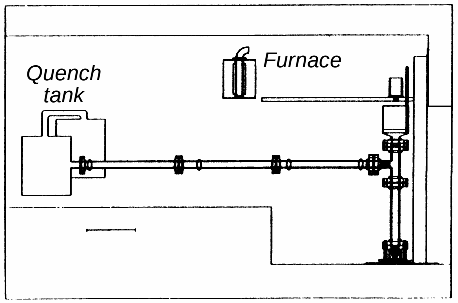
- PWR containment analyses indicate ex-vessel steam explosions not significant contributor to containment failure
- BWR analyses suggest drywell failures from ex-vessel steam explosions significant contributors to Mark I and Mark II containment failure.

Note many IPEs found ex-vessel steam explosions unimportant contributors for BWRs and PWRs

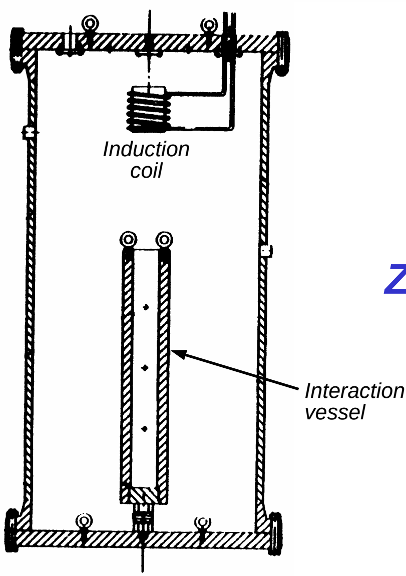
Recent Experimental Data Provides Key Insights about Steam Explosions

Facility/ ***Location***	**Pheonomena Investigated**	**Test Section Diameter (mm)**	**Melt Jet Diameter (mm)**	**Water Depth (m)**	**System Pressure (MPa)**	**Melt Composition and Mass (kg)**
FARO/ ***ISPRA***	Integral tests investigating premixing, quenching, propagation, and FCI energetics	700	100	0.1-5.0	0.1 – 5.0	UO_2-ZrO_2 (w/ and w/o Zr & SS) , 18-250
KROTOS/ ***ISPRA***	Smaller scale tests investigating premixing, quenching, propagation, and FCI energetics	95-200	30-50	1.0	0.1 - 1.0	UO_2-ZrO_2 Al_2O_3 1.4- 6.0
WFCI/ ***Univ. Wisconsin***	Conditions needed for energetic FCI	87-200	30	1.0	0.1	Sn 0.89- 4.5 FeO, Fe_3O_4
ZREX/ ***ANL***	Effects of chemical augmentation (due to the presence of metals in the melt) on FCI	100	25 – 50	1.0	0.1	Zr (w/ and w/o ZrO_2) 0.2 – 1.0
TROI/ ***KAERI***	Integral tests investigating premixing, quenching, propagation, and FCI energetics	600	~38 to 50	0.67	0.1 to 2.0	ZrO_2 and UO_2-ZrO_2 5 to 13.7

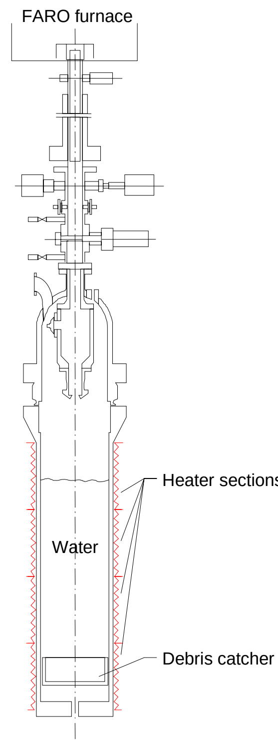
Recent Experimental Data Provides Key Insights about Steam Explosions (continued)



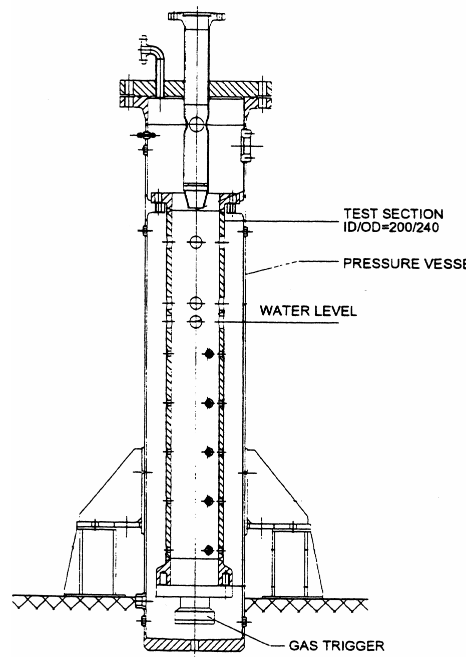
WFCI



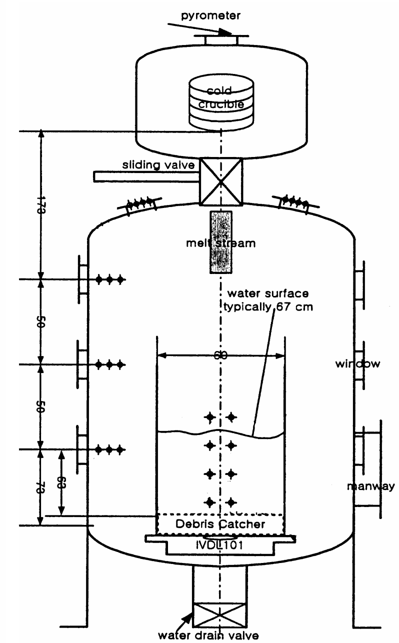
ZREX



FARO



KRYTOS

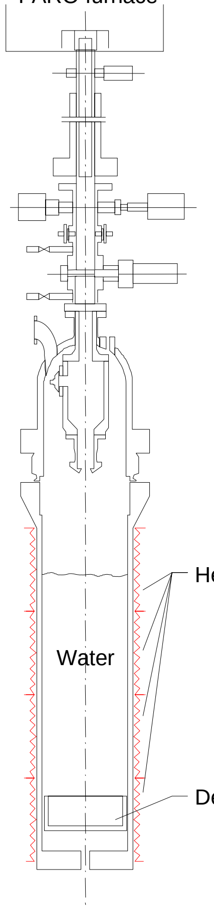


TROI

Key Parameters for Evaluating Ex-Vessel Steam Explosion Potential

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

Prototypic Large-scale FARO Data Suggest Steam Explosions Less Likely

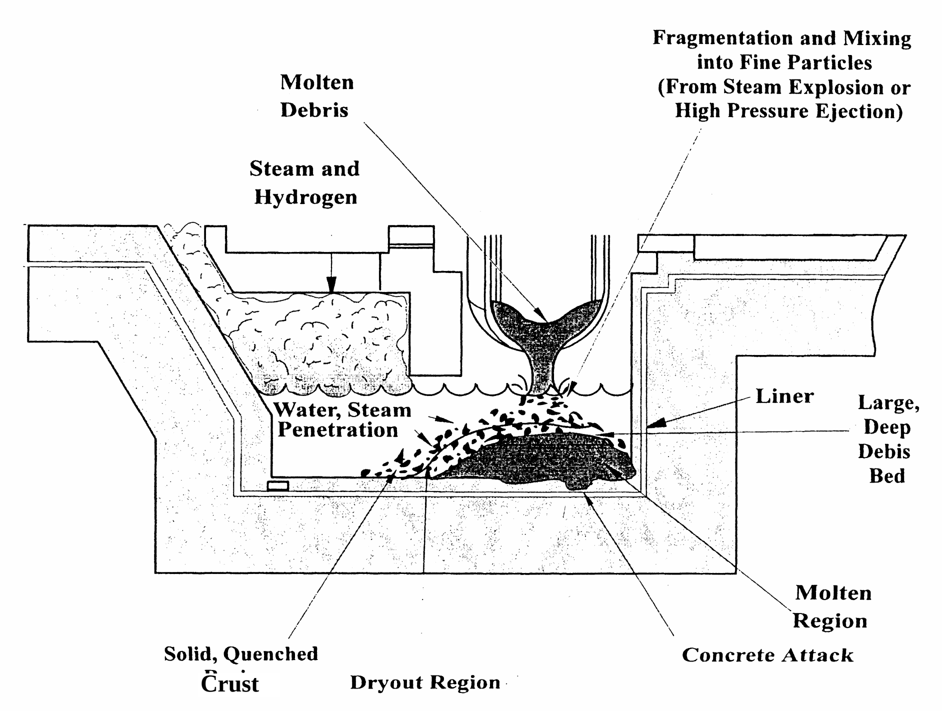


- In tests with UO_2, ZrO_2, and Zr, complete fragmentation occurred
- In tests with UO_2 and ZrO_2, relocated materials consisted of a “cake” with an overlying layer of fragmented debris
- Mean particle size of fragmented debris ranged from 3.4 to 4.8 mm
- Gap occurs between “cake” and test plate
- No energetic steam explosions observed in tests simulating in-vessel conditions.

Recent Findings Suggest Lower Probability for Steam Explosions

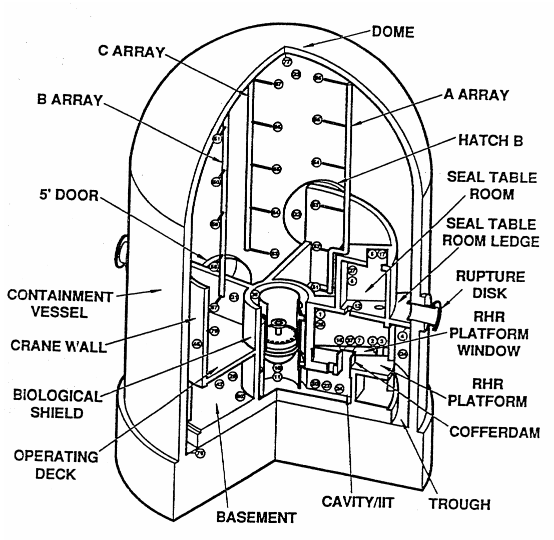
- Experimental results indicate:
  - At low pressure (0.1 MPa), limited fuel mass expected to participate in energetic FCI
  - At higher pressures (1-2 MPa), explosion difficult to trigger
- All eleven SERG-2 experts estimated low probabilities for α-mode failure
  - Low conversion energy
  - Lower explosivity of corium
  - Intervening structures
- Nine of eleven SERG-2 experts declared issue of α-mode failure induced by steam explosion resolved from risk perspective

Direct Containment Heating (DCH) Issues

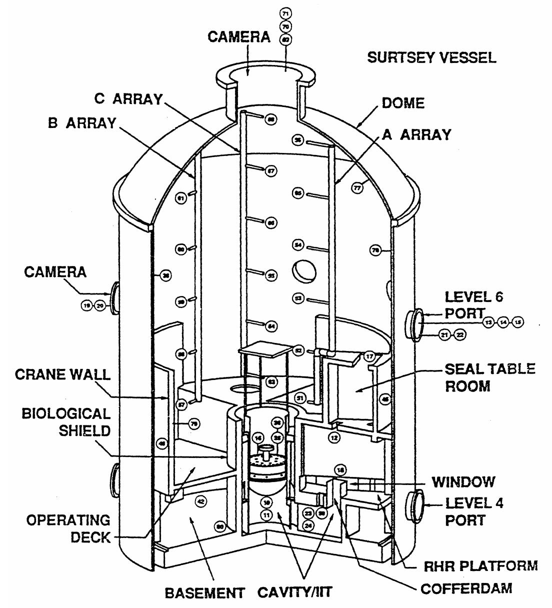


- Is sufficient melt entrained as vessel depressurizes?
- Does sufficient heat transfer, oxidation, and/or hydrogen combustion occur to threaten containment integrity?

Unique Experimental Facilities Provide Insights About Potential for DCH



CTTF



SURTSEY

Facility capabilities allowed measurement of :

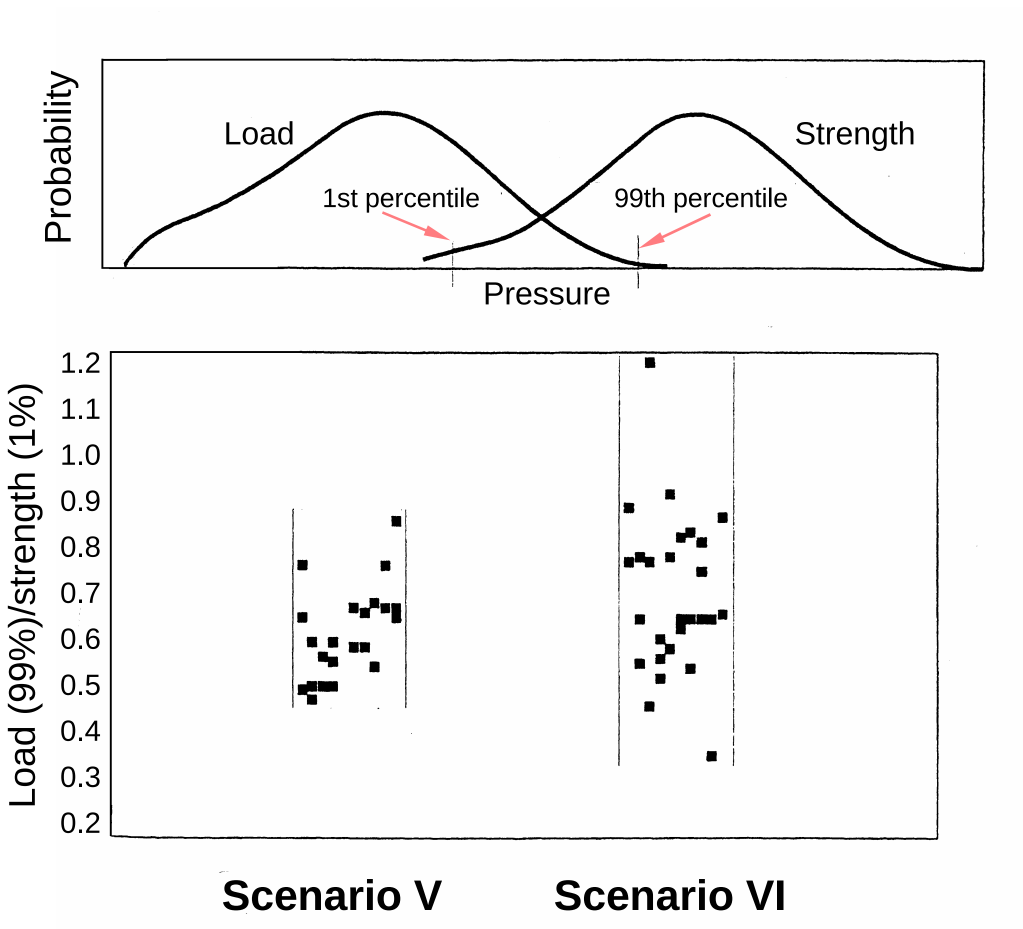
- Pressure load
- Hydrogen combustion
- Containment compartment geometry effect
- Post-test debris distribution
- Effectiveness of safety equipment

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.)

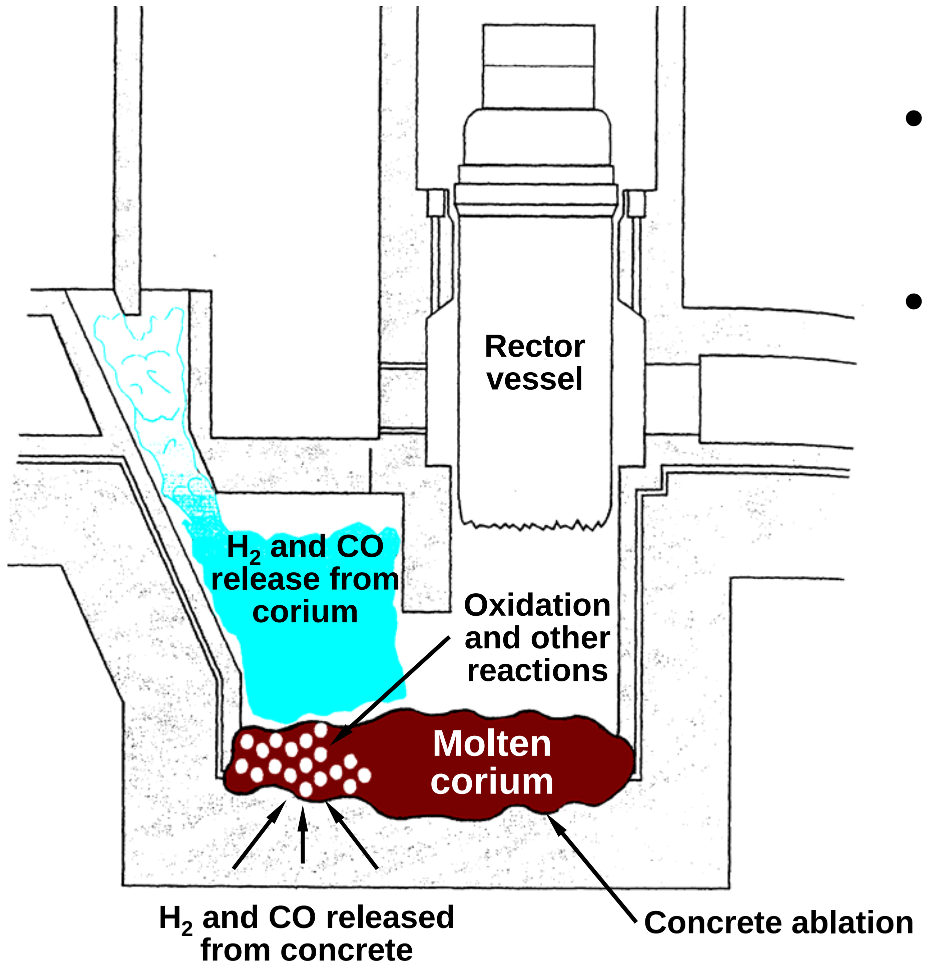
- Containment design
  - Subcompartment configuration
  - Cavity flow paths
  - Water availability (flooded height)
  - Containment fragility

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



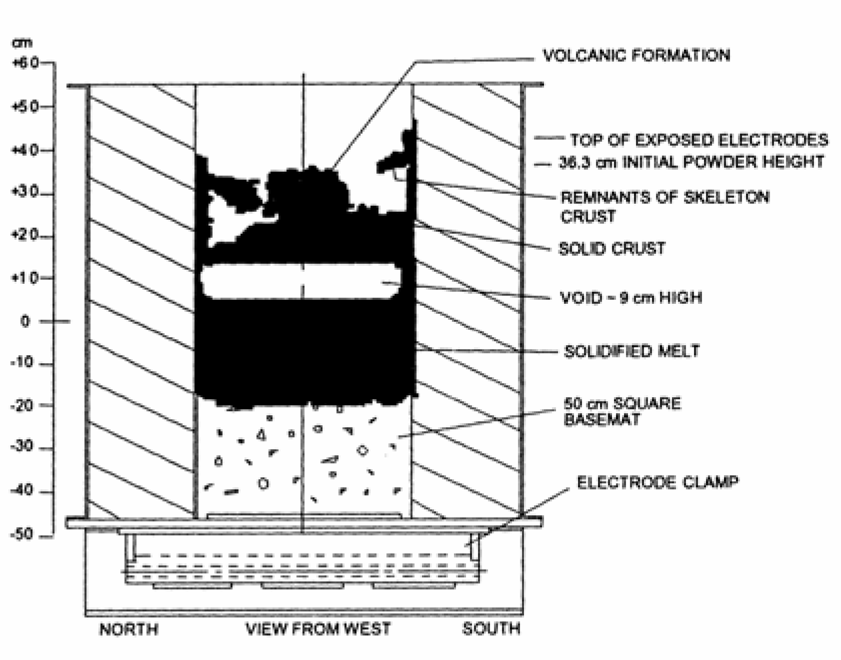
- Compartmentalization (CCFP < 0.001 for most W plants)
- Higher potential for induced RCS depressurization (lower likelihood for HPME)
- Realistic initial melt conditions based on SCDAP/RELAP5 calculations (smaller melt mass, less unoxidized metallics)

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 prototypic corium
  - 30 cm x 30 cm to 120 cm x 120 cm concrete basemat area
  - UO_2, ZrO_2, and Zr corium materials heated up to 2350 K
  - DEH 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.

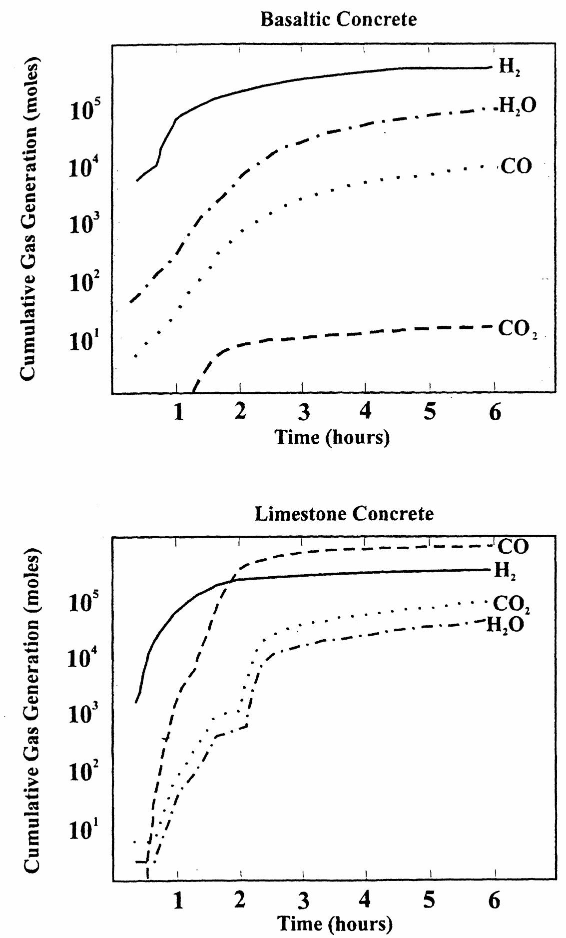
Several Factors Influence MCCI

- Design dependent
  - Type of concrete
  - Basemat thickness
  - Cavity size and geometry
- Sequence dependent
  - Melt mass released
  - Melt composition
  - Melt configuration (coolability)
  - Presence of water

Phenomena

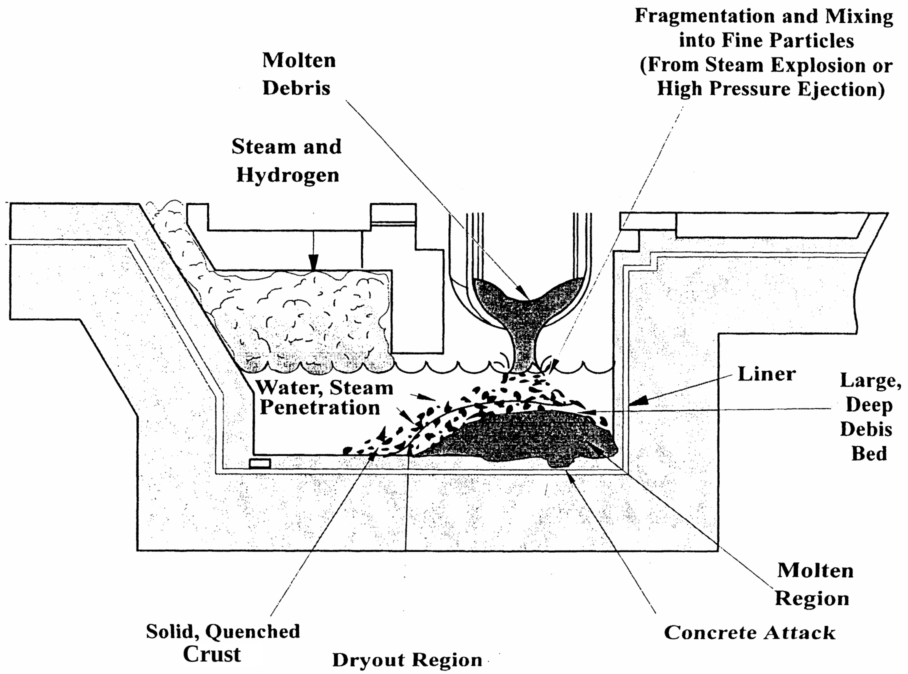
Concrete Composition Affects Gas Generation

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



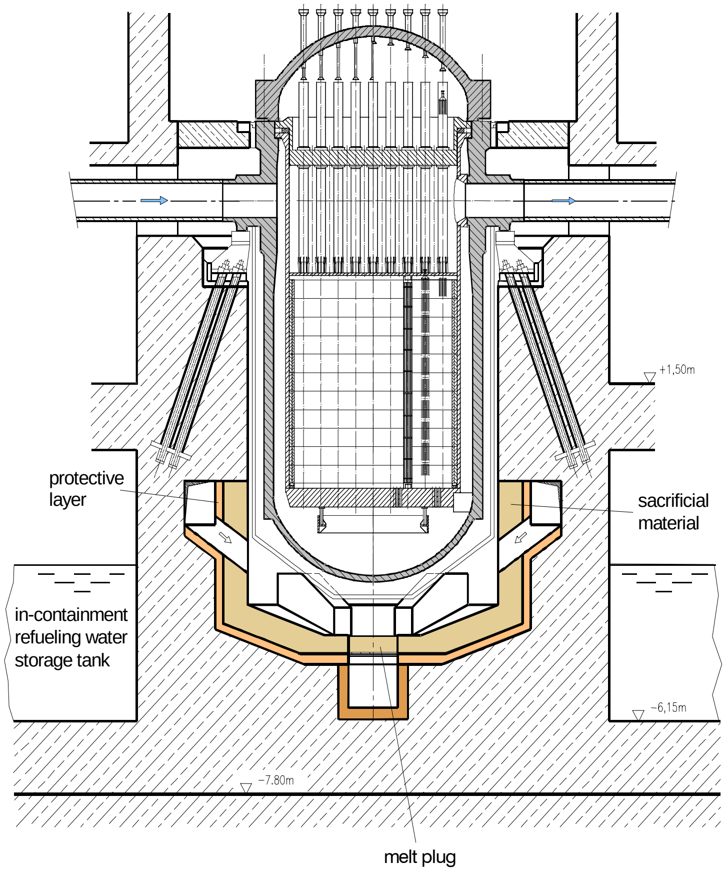
Limestone concrete ablates more rapidly and produces more combustible gases

Presence of Water Does Not Guarantee Coolability

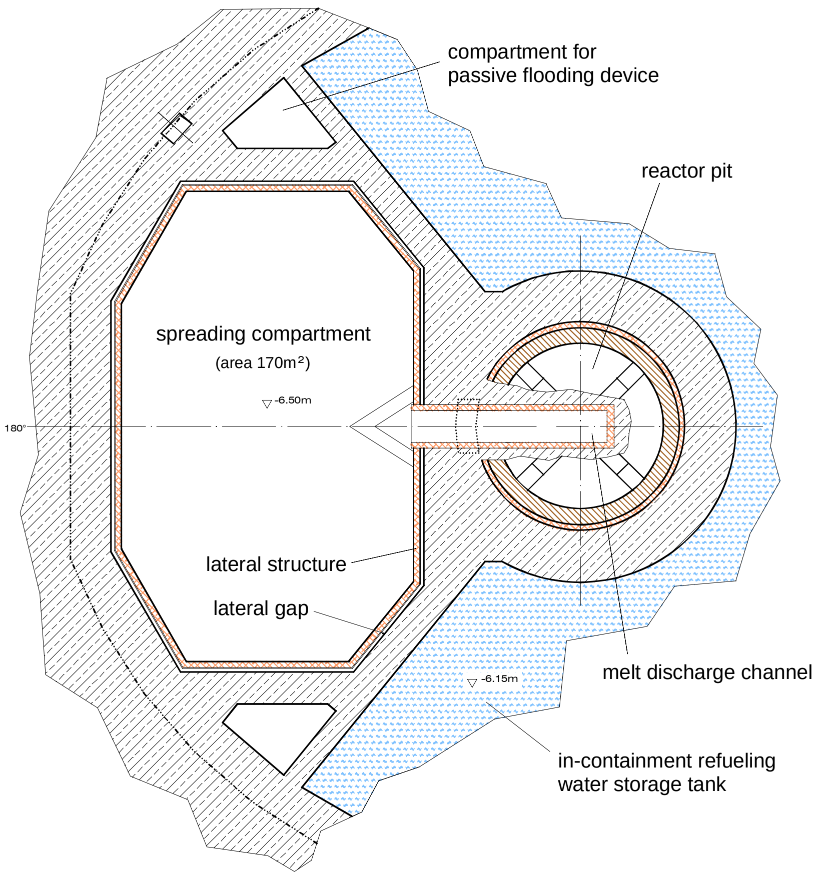


Water can cool released gases and retain some released fission products

EPR Relies on Large Spreading Area to Guarantee Coolability



Reactor Cavity



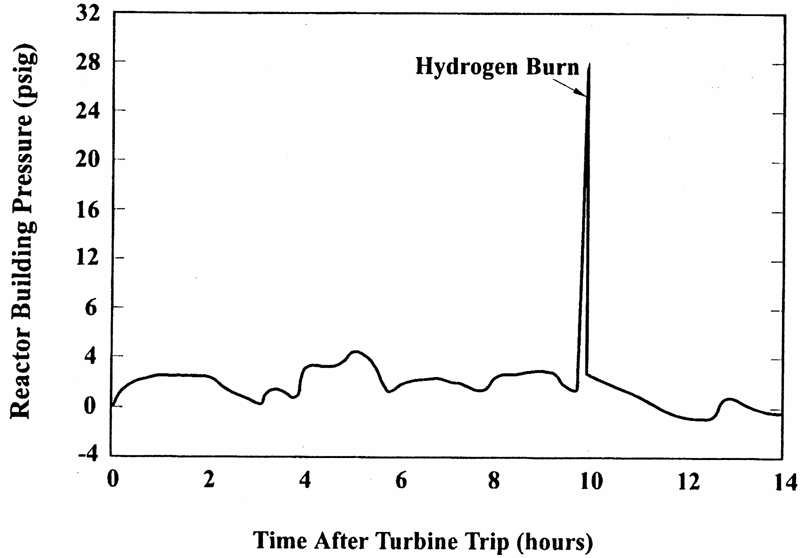
Spreading Area

Hydrogen Combustion Issues

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

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

Hydrogen ignition increased TMI-2 containment pressure by 28 psig.

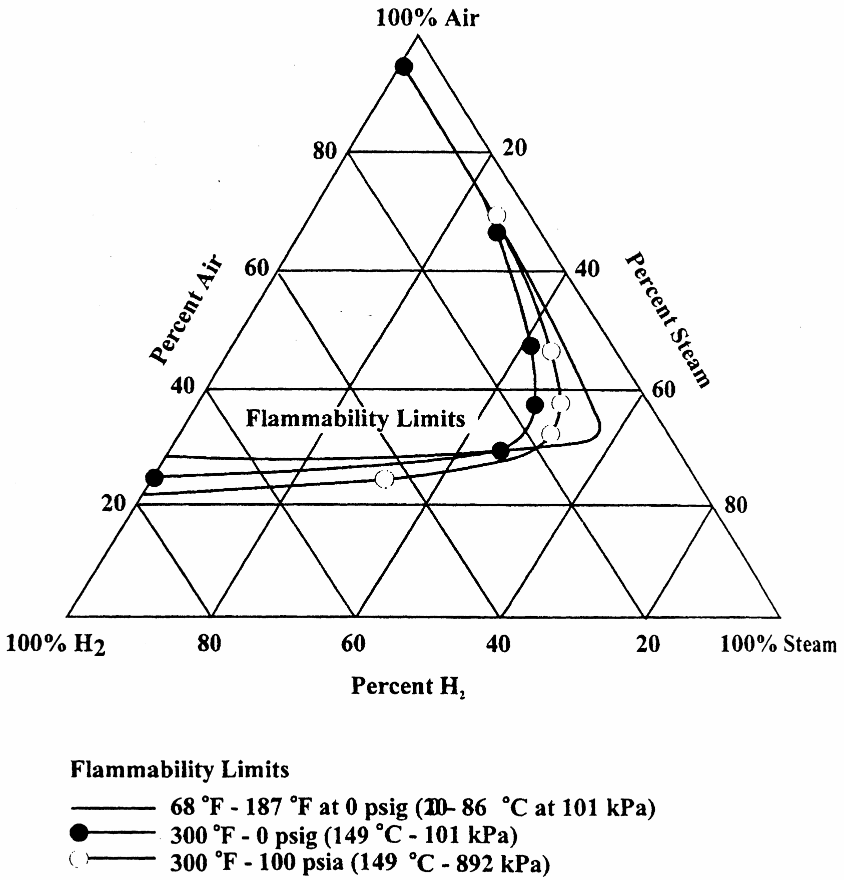


- During core heatup, between 270 to 370 kg hydrogen released through PORVs (~40% of zirconium oxidized)
- Pressure rise corresponds to complete combustion of approximately 8% hydrogen atmosphere
- Concerns exist about the integrity of containments with smaller net free volumes exposed to similar threats

Two Types of Combustion

- Deflagration waves
  - travel subsonically (< 35 m/s)
  - heat unburned gases to temperatures high enough for chemical reactions to occur
  - produce quasi-static containment loads
- Detonation waves
  - travel supersonically (at least 2200 m/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).

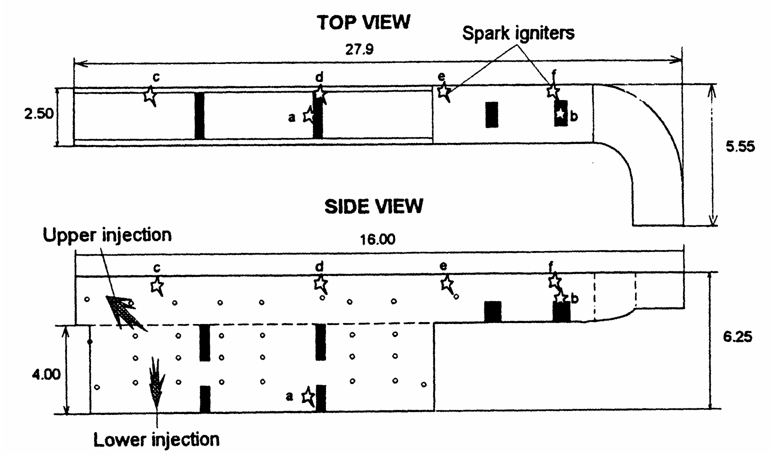
Shapiro and Moffette Diagram Depicts Hydrogen: Air: Steam Flammability Limits



Limits vary with:

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

RUT Experimental Data Provides Insights about Hydrogen Ignition



- Series of tests with dynamic hydrogen injection and spark ignition
  - Up to 480 m^3
  - 0.6 to 1 kg/s and 0.1 to 0.18 kg/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 can be used to optimize number and location of igniters in containments.

Localized Effects May Be Important

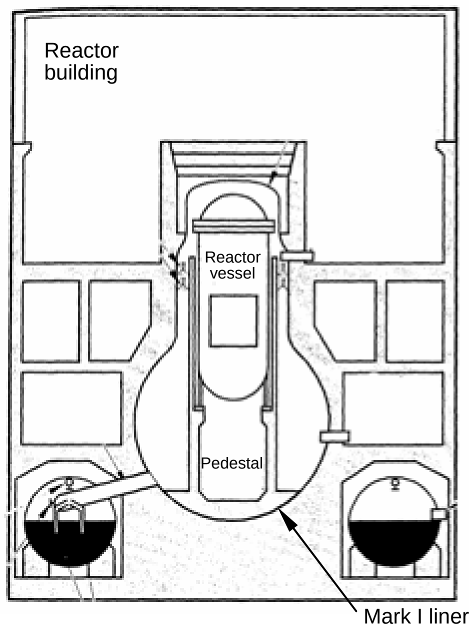
- Higher concentrations of hydrogen near release points, under ceilings or dome due to density stratification, or near steam removal locations, such as ice condensers, suppression pools, and fan coolers
- Equipment susceptible to high pressure or temperature
- Compartments with smaller volumes
- Structures or regions at higher temperature or with more ignition sources

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 (unless containment steam inerted)

Phenomena

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?

NRC-Sponsored Mark I Liner Failure Studies Focused on Limited Cases

RCS Pressure	Flooded Drywell Floor	Vessel Failure Mode	Analyzed in Mark I Liner Study	Estimate for Prompt Liner Failure
High	Either	Any	No	High
Low	No	Penetrations	Yes	High
		Global rupture	No	Medium to High
	Yes	Penetrations	Yes	Low
		Global rupture	No	Low to Medium

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

Several actions reduce contribution of drywell shell meltthrough to early containment failures.

- Improved success for vessel depressurization
  - Revised procedures
- Improved 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
- Improved containment venting prior to core damage
  - Direct path between wetwell to outside the containment building

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

DCH Resolution Methodology

Resolution Criterion:

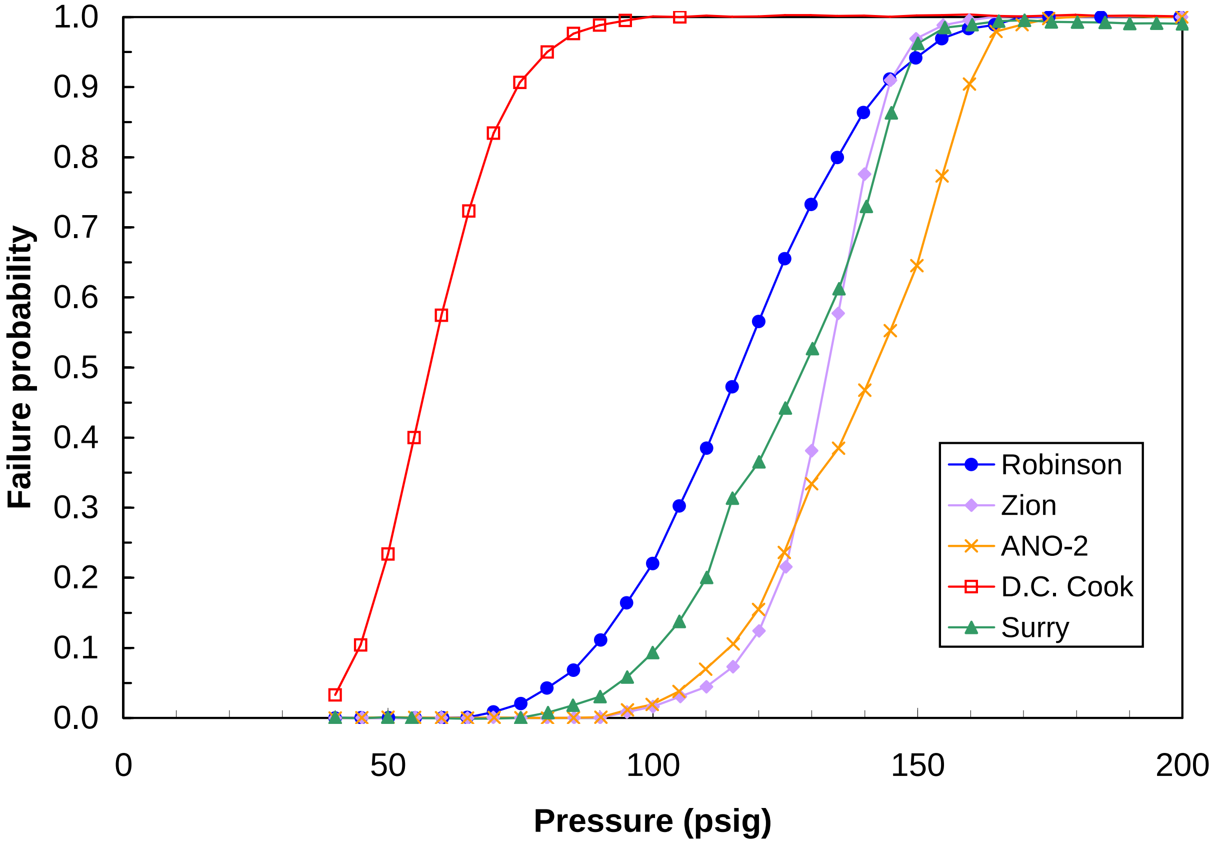
For events with core damage, threat of early containment failure due to DCH $\leq$ 0.1

Procedure:

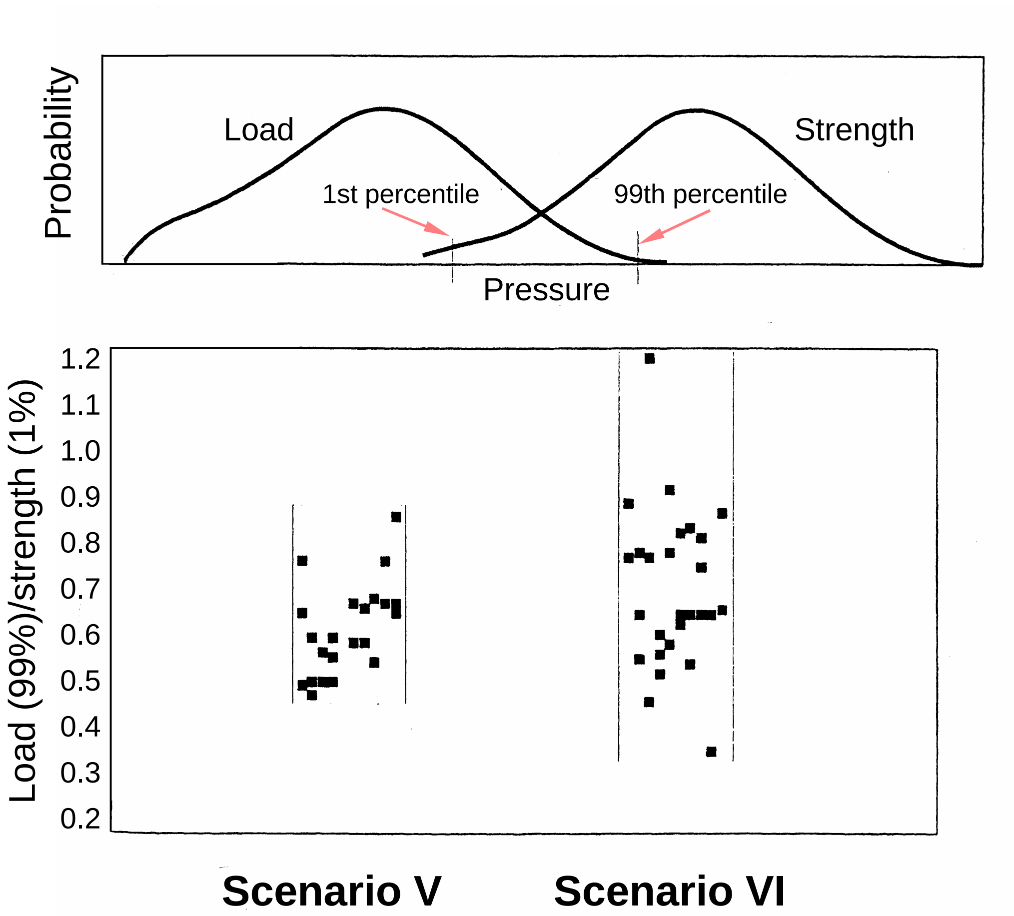
- Analyze several splinter scenarios to envelop conditions for release (melt mass, composition, vessel pressure, etc.)
- Predict containment pressurization pdf.
- Estimate CCFP using plant specific containment fragility curve (from IPEs).
- If CCFP > 0.01 (screening criterion), perform more detailed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

Case Study and Problem

IPE containment fragility curves assumed for DCH resolution study.



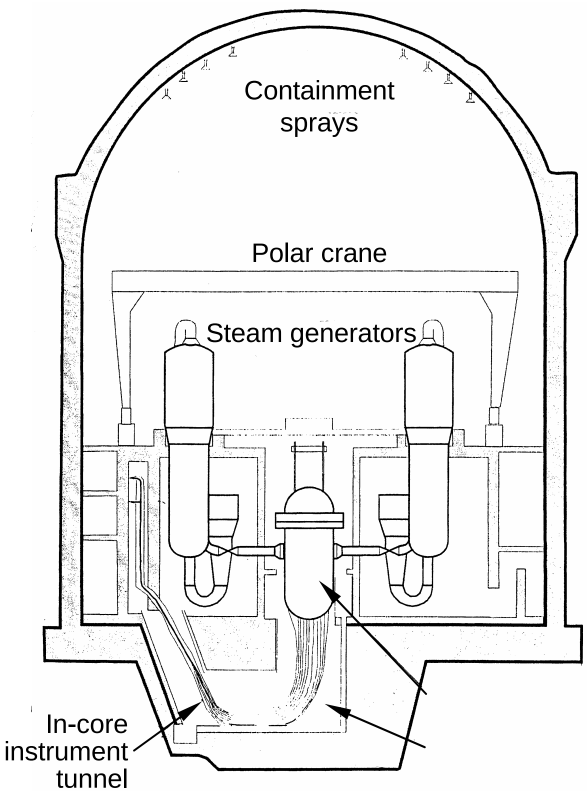
Mean CCFP < 0.01 for all Westinghouse Large Dry and Subatmospheric Containments



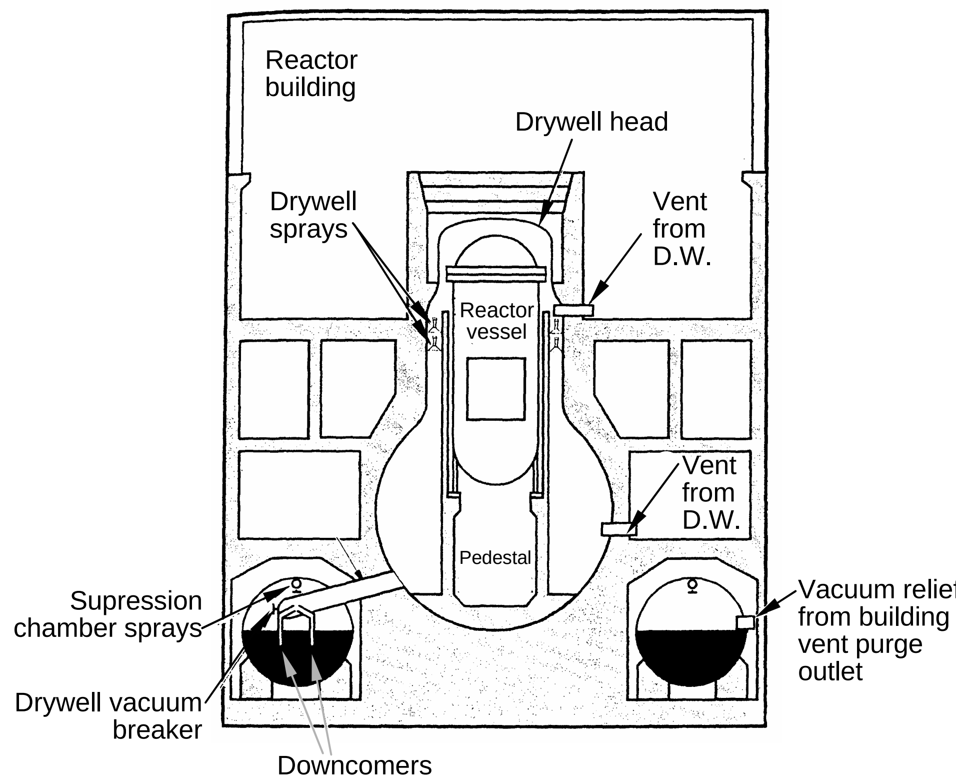
- No intersections of load distributions with fragility distributions for most plants (CCFP ~ 0).
- Finite, but negligible, intersection predicted for H.B. Robinson (broad containment fragility distribution and dome transport characteristics).

Case Study and Problem

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

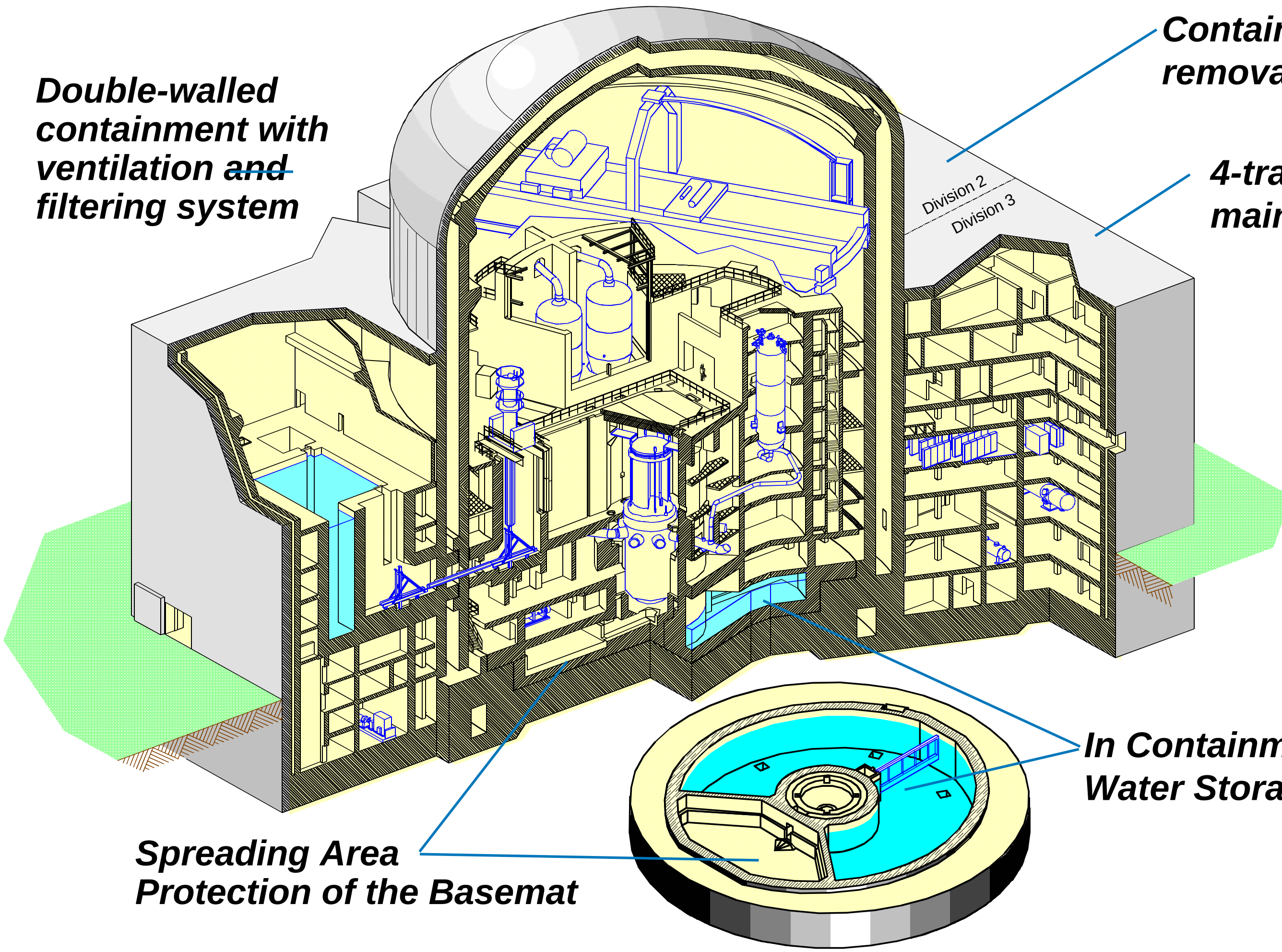


Large dry



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?

Research since NUREG-1150 Confirms Challenges Less Likely

General

- *General Individual Plant Examination Program: Perspectives on Reactor Safety and Plant Performance*, Final Report, NUREG-1560, December 1997.

In-Vessel and Ex-Vessel Steam Explosions

- S. Basu and T. Ginsberg, *A Reassessment of the Potential for an Alpha-Model Containment Failure and a Review of the Current Understanding of Broader Fuel-Coolant Interaction Issues*, NUREG-1524, 1995.
- S. Basu and T. P. Speis, “An Overview of Fuel-Coolant Interactions (FCI) Research at NRC,” *presented at the 23rd Water Reactor Safety Information Meeting*, October 23-25, 1995, Bethesda, Maryland.
- OECD Nuclear Energy Agency and Center for Risk Studies and Safety (UCSB), *Proceedings of the CSNI Specialists Meeting on Fuel-Coolant Interactions*, NUREG/CP-0127 (NEA/CSNI/R(93)8), January 1993.
- M. L. Corradini, "VAPOR Explosions: A Review of Experiments for Accident Analysis," *Nuclear Safety*, **32, No. 3**, July - September 1991.
- D. F. Fletcher, "Steam Explosion Triggering: A Review of Theoretical and Experimental Investigations," *Nuclear Engineering and Design*, **155**, 1995.
- H. H. Kim, et al., “A Study on Intermediate Scale steam Explosion Experiments with Zirconia and Corium Melts,” *Transactions of the International Congress on Advanced Nuclear Power Plants (ICAPP),* June 9-13, 2002, Hollywood FL, USA

Research since NUREG-1150 Confirms Challenges Less Likely (continued)

Direct Containment Heating (DCH)

- M. M. Pilch, et al., *The Probability of Containment Failure by Direct Containment Heating in Zion*, NUREG/CR-6075, Supplement 1, December 1994.
- M. M. Pilch, et al., *The Probability of Containment Failure by Direct Containment Heating in Surry*, NUREG/CR-6109, May 1995.
- M. Pilch, et al., *Resolution of the Direct Containment Heating Issue for all Westinghouse Plants with Large Dry Containments or Subatmospheric Containments*, NUREG/CR-6338, February 1996.
- M. Pilch, et al., *Resolution of the Direct Containment Heating Issue for Combustion Engineering Plants and Babcock & Wilcox Plants*, NUREG/CR-6475, November 1998.
- *Assessment of the DCH Issue for Plants with Ice Condenser Containments*, NUREG/CR-6427, April 2000.

Research since NUREG-1150 Confirms Challenges Less Likely (continued)

DCH (continued)

- T. K. Blanchat, et al., *Experiments to Investigate Direct Containment Heating Phenomena with Scaled Models of the Surry Nuclear Power Plant*, NUREG/CR-6152, June 1994.
- T. K. Blanchat, et al., *Experiments to Investigate Direct Containment Heating Phenomena with Scaled Models of the Zion Nuclear Power Plant in the Surtsey Test Facility*, NUREG/CR-6044, 1995.
- M. D. Allen, et al., *Experiments to Investigate the Effects of 1:10 Scale Zion Structures on Direct Containment Heating (DCH) in the Surtsey Test Facility: The IET-1 and IET-1R Tests*, SAND92-0255, 1992.
- D. C. Williams, "An Interpretation of the Results of Some Recent Direct Containment Heating (DCH) Experiments in the Surtsey Facility," *presented at NURETH-5*, September 21-24, 1992, Salt Lake City, Utah.

Research since NUREG-1150 Confirms Challenges Less Likely (continued)

Molten Core Concrete Interactions

- *Papers presented at the Second OECD (NEA) CSNI Specialists Meeting on Molten Core Debris-Concrete Interactions*, KfK 5108, NEA/CSNI/R(92)10, April 1-3, 1992.
- E. R. Copus and D. R. Bradley, *Interaction of Hot Solid Core Debris with Concrete*, NUREG/CR-4558, SAND85-1739, June 1986.
- R. K. Cole, et al., *CORCON-Mod2: A Computer Program for Analysis of Molten-Core Concrete Interactions*, NUREG/CR-3920, SAND84-1246, August 1984.
- M. T. Farmer, et al., “Results of MACE Core Coolability Experiments M0 and M1b,” *Proceedings of the 8th international Conference on Nuclear Engineering*, April 2-6, 2000, Baltimore, MD.

Research since NUREG-1150 Confirms Challenges Less Likely (continued)

Mark I Liner Failure

- T. G. Theofanous, et al; *The Probability of Liner Failure in a Mark-I Containment*, NUREG/CR-5423, August 1991.
- T. G. Theofanous, et al., *The Probability of Mark-I Containment Failure by Melt-Attack of the Liner*, NUREG/CR-6025, November 1993.

Hydrogen Combustion

- A. L. Camp, et al., *Light Water Reactor Hydrogen Manual*, NUREG/CR-2726, SAND92-1137, August 1983.
- *Technical Aspects of Hydrogen Control and Combustion in Severe Light Water Reactors Accidents*, National Academy Press, Washington, D.C., 1987.
- S. B. Dorofeev, et al., *Recent Experimental and Analytical Results on Hydrogen Combustion at RRC* "Kurchatov Institute," presented at the 23rd Water Reactor Safety Information Meeting, October 23-25, 1995, Bethesda, Maryland.

Accident Progression Analysis (P-300)

6. CET Development

May 10-12, 2005 - Rockville, MD

Session Objectives

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

Three Steps In Containment Performance Analysis

- **Assessment of the range of challenges to containment integrity (i.e., loads resulting from severe accidents)**
- **Characterization of the capacity of the containment (i.e., strength - often in terms of a fragility curve)**
- **Framework for combining the two above to estimate the conditional (on a given accident sequence) failure probability**
  - **also needs to accommodate uncertainty**

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

- Examples that involve considerable uncertainty include:
  - Hydrogen generation and combustion
  - Induced failure of RCS pressure boundary
  - Debris bed coolability and core-concrete interaction
  - Fuel-coolant interactions (steam explosions)
  - Melt/debris ejection following RV failure (DCH)
  - Shell melt-through failure in Mark-I containments
- Each phenomena depends on a number of accident progression characteristics

Performance Issues Depend on Containment Design

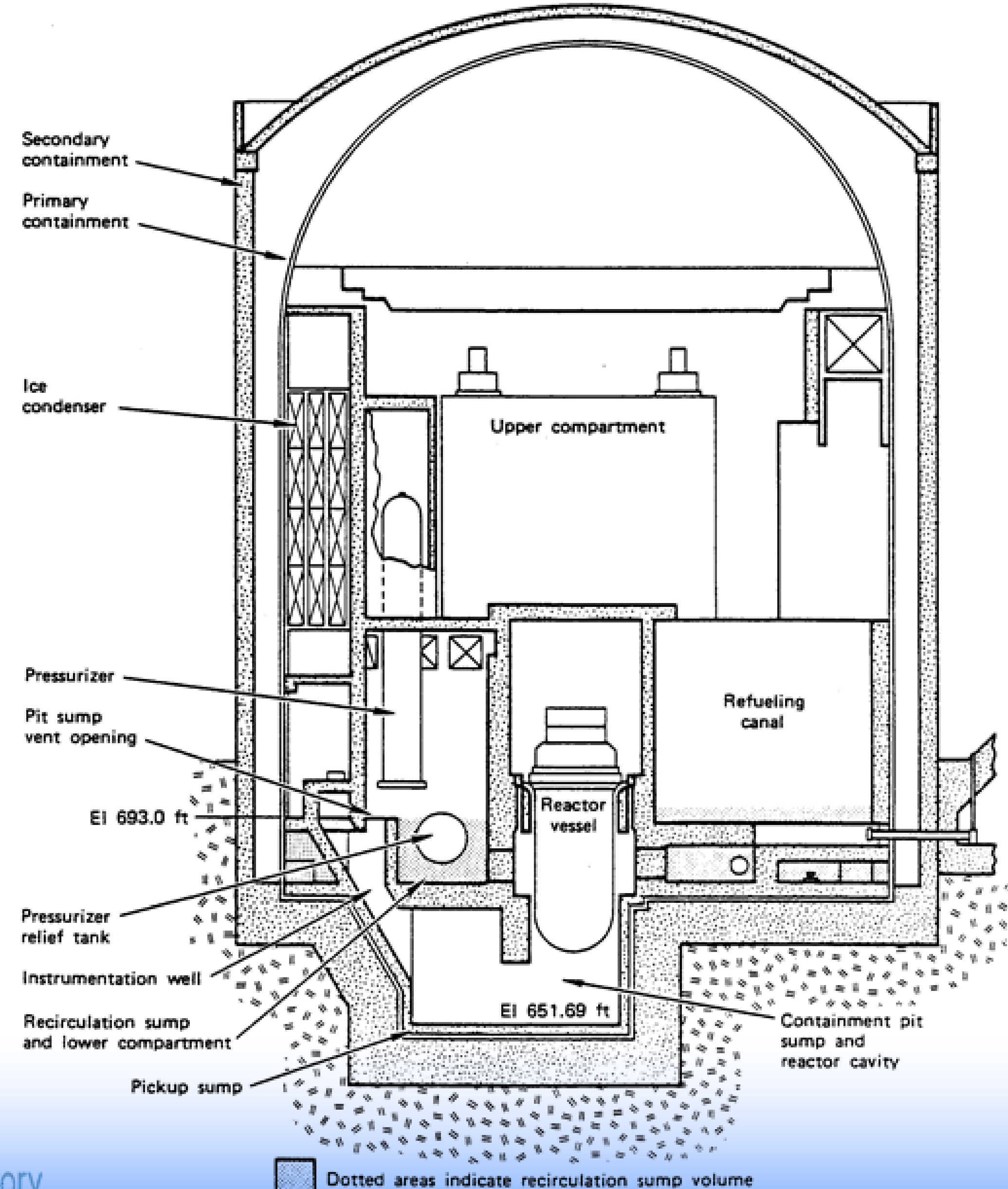
- Containments can be grouped into two general categories
  - Large Dry
    - Prestressed concrete (Palisades)
    - Free-standing steel (St. Lucie)
    - Subatmospheric, reinforced concrete (Surry)
  - Pressure Suppression
    - Ice Condenser (Sequoyah)
    - Mark I (Peach Bottom)
    - Mark II (Limerick)
    - Mark III (Grand Gulf)

Example Features of Large-Dry and Subatm. Containment Designs

- **Large containment volume results in relatively low containment failure probabilities**
  - **Early failure of CHR allows containment base pressure to rise such that the containment pressure at vessel failure (HPME) poses significant hazard**
  - **Some IPEs claim early containment failure is not a credible response to any core damage scenario**

Example Features of Ice Condenser Containment Designs

- **Glow plug igniters installed to burn off accumulated hydrogen (similar to BWR Mark-III design)**
  - **Therefore hydrogen combustion is a threat only during SBO (i.e., igniters not operable)**
- **Instrument cable tunnel and seal table room designs allow potential for HPME to result in core debris to directly impinge upon containment wall**
- **Typical of PWR analyses, containment bypass (ISLOCA, SGTR and I-SGTR) is an important contributor to early containment failure**



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Example Features of Mark-I Containment Designs

- Relatively small volume design is more susceptible to overpressure failures
- Suppression pool design and presence of reactor building (secondary containment) make containment failure location important to consequence analysis
- Containment atmosphere inerted (nitrogen) to prevent hydrogen combustion
- Containment venting possible (from torus) via hard-pipe flow path

Example Features of Mark-II Containment Designs

- **Similar to Mark-I design**
  - **Relatively high strength, but small volume**
  - **Containment atmosphere nitrogen inerted**
- **Different reactor cavity designs (different from Mark-I and various Mark-II plants differ from each other)**
- **Drain lines in drywell**
  - **Failure of drain lines fails containment**

Example Features of Mark-III Containment Designs

- **Significantly different from Mark-I and Mark-II designs**
  - **Larger volume, lower strength**
- **Four single-unit BWR/6 reactor sites**
  - **Clinton, Grand Gulf, Perry & River Bend**
  - **BWR/6 core contains more zirconium than any other U.S. reactor design**
    - **Potential for large amounts of hydrogen to be produced during core degradation**
      - **Glow plug igniters used for hydrogen mitigation (not inerted)**

Variations Exist Within Containment Designs

Many design features affect containment response

- **Free Volume**
- **Design pressure and ultimate pressure**
- **Type of concrete**
- **Presence of water in reactor cavity**
- **Basemat thickness**
- **Presence of missile shields (mitigate IVSE)**
- **Seismic rating**
- **Design of penetration seals**
- **Design leak rate**
- **Isolation features**
- **Atmospheric cooling provisions**
- **Ignition sources**
- **Presence of secondary containment**
- **Interfacing LOCA paths**

Example: Type of Concrete

- **Type of used for the containment (floor) affects the quantity of gas released in core concrete interaction (chemical reaction, which is affected by chemical composition of constituents.)**
  - **Limestone concrete releases significantly more gases than basalt concrete**
  - **Types of concrete include:**
    - **Limestone**
    - **Limestone-sand**
    - **Siliceous**
    - **Basalt**

Severe Accident Progression Model (Level-2 CET or APET)

- Purpose is to identify the various ways a possible core damage accident might progress
  - Challenges to plant mitigation systems
    - Failure probabilities
  - Challenges to core and reactor pressure vessel integrity
  - Response of containment structure
  - Timing and magnitude of release
    - Affect consequences to the public

Severe Accident Modeling Includes Both In-Vessel and Ex-Vessel

- Potential for recovery before RPV fails
  - "Core Damage" from Level-1 PRA not really core damage
  - Restore ac power (if SBO)
  - Restore core cooling (alternate injection)
- Characteristics of core melt and RPV failure major impact on containment loads

Containment Failure Categories

- 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
- Bypass Events
  - Vessel failure not required for release
    - Event V or Interfacing System LOCA (ISLOCA)
    - SGTRs

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)**

Containment Bypass Events

- Containment is bypassed (by definition) therefore, little analysis needed
- Generally only a small number of bypass scenarios are identified
  - ISLOCA
  - SGTR
- Need to determine:
  - leak rate (size of the bypass flow path)
  - whether or not flow path is submerged
    - Includes scrubbing by fire suppression sprays

Containment Bypass

- **Core damage scenario and associated release mechanism in which status of containment integrity is irrelevant**
  - **Failure event that opens primary coolant system directly to environment, bypassing containment**
  - **Typically includes:**
    - **interfacing system loss of coolant accident (ISLOCA)**
      - **also known as Event-V**
    - **steam generator tube rupture (SGTR)**
      - **Level-1 initiating event**
      - **Severe accident induced**
  - **Does NOT include containment isolation failure**
    - **which is an early containment failure mode**

Bypass Often Important to Risk

- Bypass sequences often NOT significant contributors to core damage frequency, but:
  - Unmitigated radioactive release produces greatest consequences
- ISLOCA frequency analyzed as part of Level-1
  - Sometimes Level-2 analysis considers details of ISLOCA release (i.e., could be scrubbed)
    - release point might be submerged
    - fire suppression spray operation

Two Types of SGTR Events

- Core damage SGTR sequence typically small contributor to overall CDF
  - Emergency operating procedures in place to isolate or recover SGTR
- Severe accident (temperature) induced SGTR analyzed as a possible containment failure mode
  - Requires dry secondary side of S/G and primary pressure at or above normal operating pressure
    - typically associated with station blackout sequences

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)

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Containment Design Details Also Affect Model Structure

- **Analysis of Ice Condenser design needs to address effects of ice**
- **Reactor cavity design affects likelihood of a flooded cavity**
  - **Does molten core fall into water or onto a dry floor?**
- **Is containment atmosphere nitrogen-inerted to prevent hydrogen combustion? Does is rely on igniters that need ac power?**

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 Available

- 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*

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

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

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 overwhelming
  - Large, integrated models more difficult to review

Typical Phenomena and Systems Considered in CETs for PWR

1 Occurrence of SGTR or ISLOCA

2 Status of Cont. Isolation

3 RCS press. at VB
   - PORV setpoint
   - High
   - Intermediate
   - Low

4 Mode of VB
   - No VB
   - Small VB area
   - Large VB area

5 Early CF Mode
   - No failure
   - Leak
   - Rupture
   - Catastrophic

6 Status of CHR

7 Presence of Water in RV Cavity

8 Occurrence of CCI

9 Late CF Mode
   - No failure
   - Leak
   - Rupture
   - Catastrophic

Typical Phenomena and Systems Considered in CETs for BWR (Part 1 of 2)

1 Status of Vessel Breach (VB)
- No VB
- VB at low press.
- VB at SRV setpoint

2 Level of Flow from RPV to Drywell (DW)
- None
- Partial
- Complete

3 Early status of DW Sprays

4 Early Containment Venting

5 Early Containment Failure
- No Early Failure
- Leak in Wetwell (WW)
- Leak in DW
- Leak in DW head
- Rupture in WW
- Rupture in DW
- Rupture in DW head

6 Late injection of water to RPV cavity

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Typical Phenomena and Systems Considered in CETs for BWR (Part 2 of 2)

7 CCI
- None
- CCI in flooded cavity
- CCI in dry cavity

8 Late status of DW sprays

9 Late Containment Venting

10 Late Containment Failure
- No Early Failure
- Leak in Wetwell (WW)
- Leak in DW
- Leak in DW head
- Rupture in WW
- Rupture in DW
- Rupture in DW head

11 Level of Reactor Building Bypass
- Nominal or small bypass
- Partial or complete

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 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

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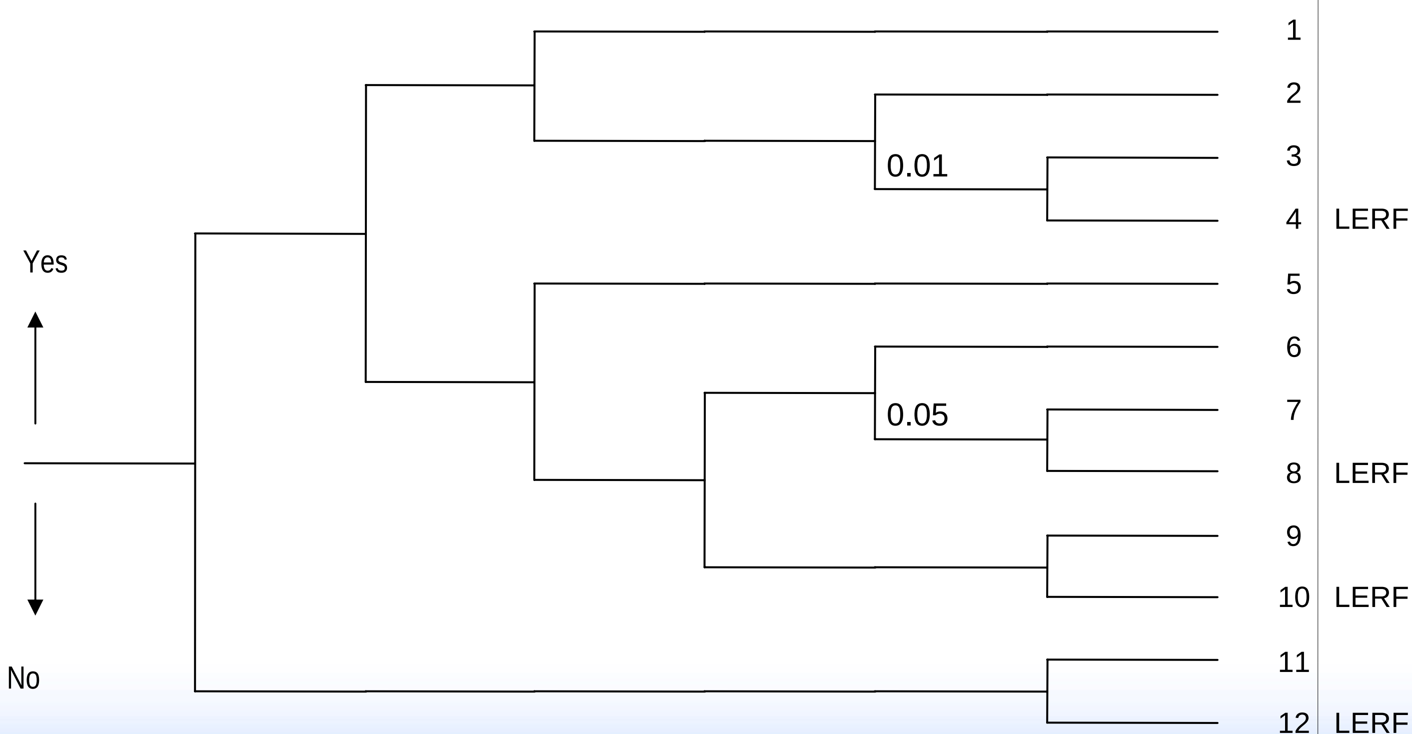
Core Damage	Containment Isolated and Not Bypassed	RCS Depress.	Core Damage Arrested without VB	No Induced Steam Generator Tube Rupture	No Containment Failure at VB	No Potential for Early Fatalities		
CD	NCI	HIPR	VB	I-SGTR	ECF	EF	#	ES



NCI - No Containment Isolation (or 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

- Pressure in reactor vessel at time of core damage
- 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 hot leg failure after core damage

VB - Vessel Breach

- Addresses 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
- Not probable 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

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 PRA

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)

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 Developed Using Variety of Sources

- Initial structure often based on a CET from a similar plant
  - Systematically identify design differences and similarities, and understand their implications
- Computer code runs
  - Heavily dependent on user developed input files
  - Extensive sensitivity analyses are a necessity to evaluate effects of various assumptions
- Experimental results support including or excluding various phenomena
- Engineering analyses to address specific issues

Most Level-2 Analyses Involve a Mix of Supporting Information

- **Plant-specific code calculation**
  - **MAAP, MELCOR, SCADAP/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	

Session Review

- Basic Steps in CET development
  - Loads, strength, framework
- Loads generated by
  - In-vessel and ex-vessel phenomena
- Containment Failure Modes
  - Early, Late, and Bypass
- Objective of CET
  - Source terms

7. Severe Accident Simulation Codes

- Introduction
- Methods
- Case Studies
- Summary and Discussion
- Study Questions
- References

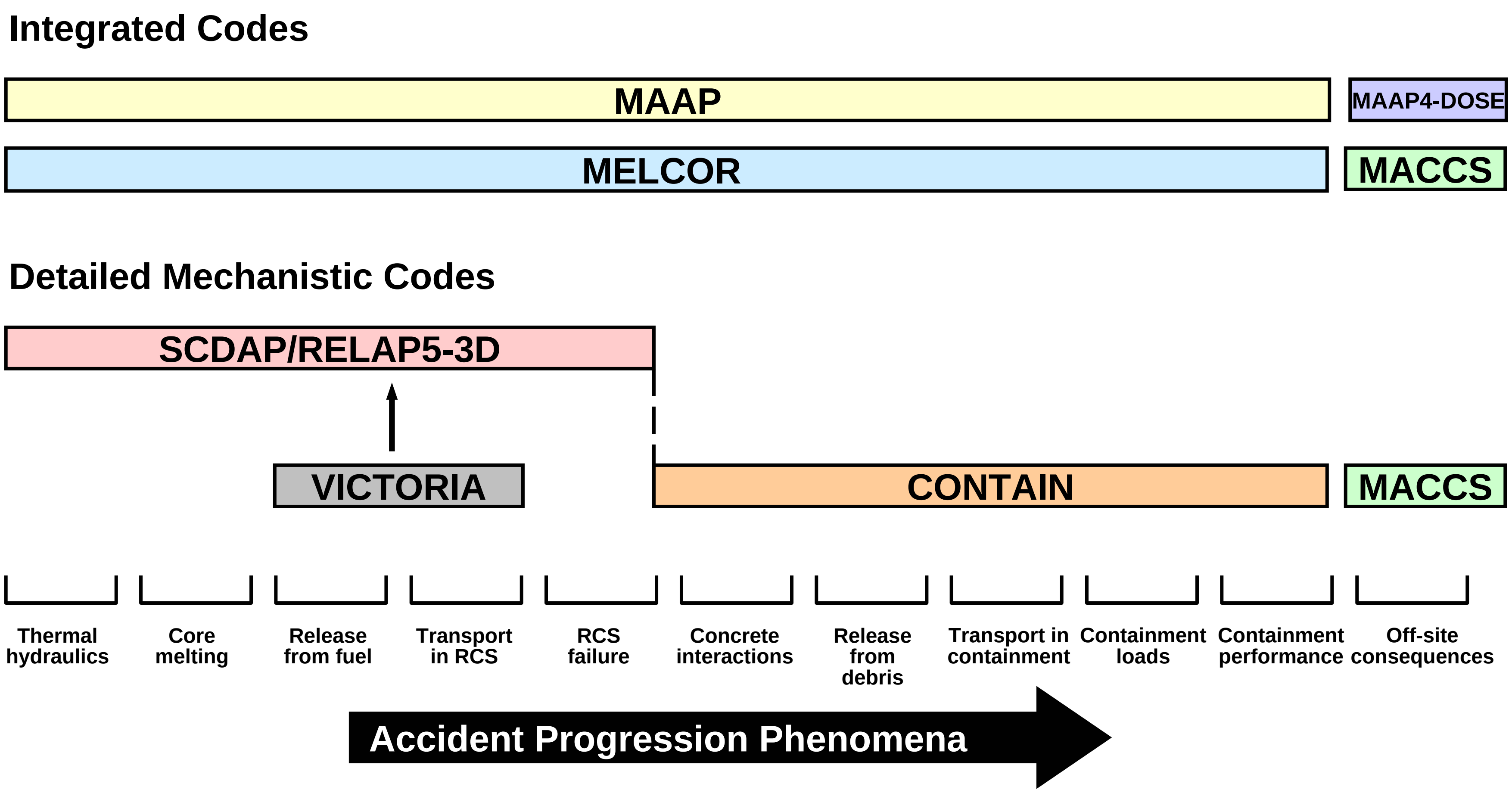
Objectives

- 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.

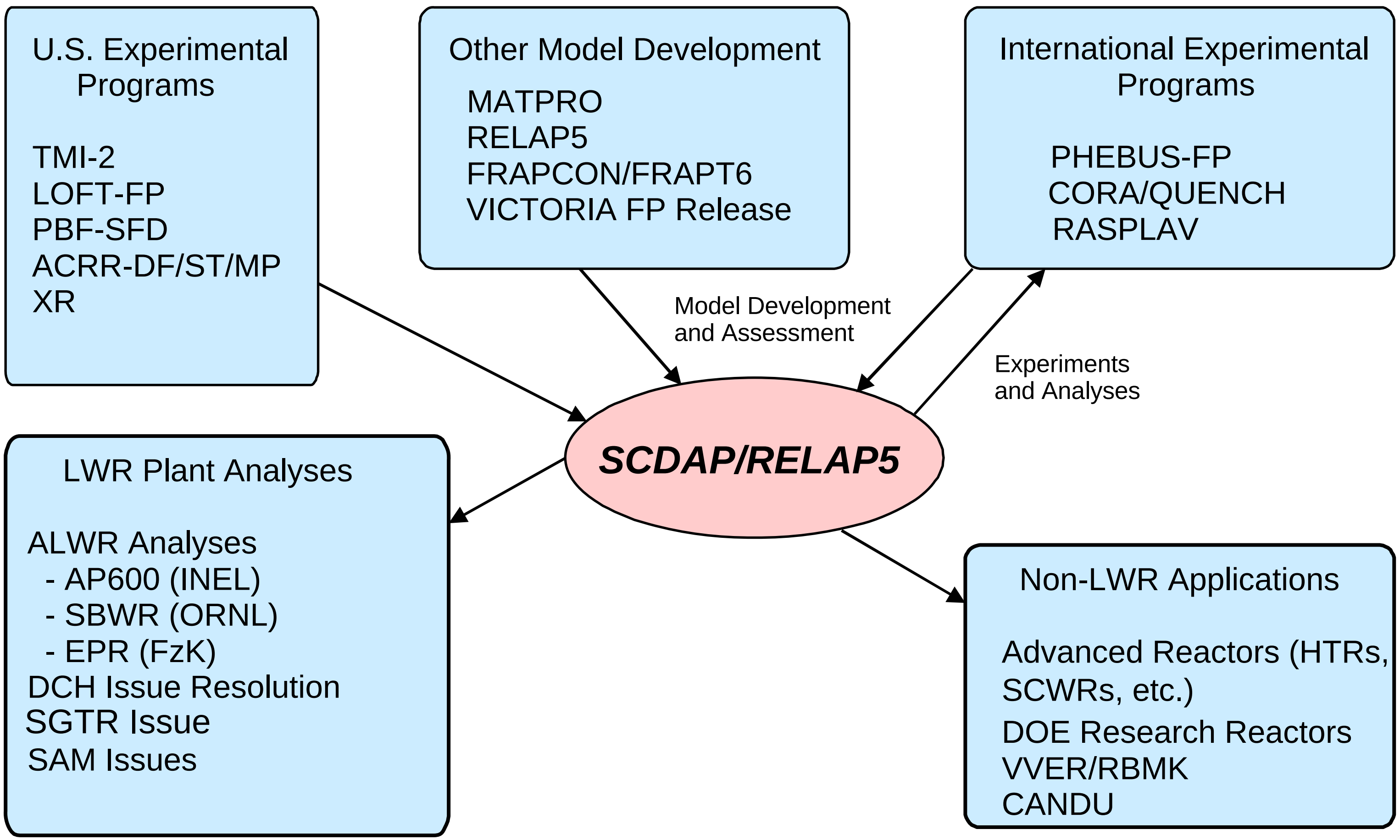
Code Design Philosophies Differ

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 dials Separate versions for each reactor type (BWR, PWR, etc.)
ICARE	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

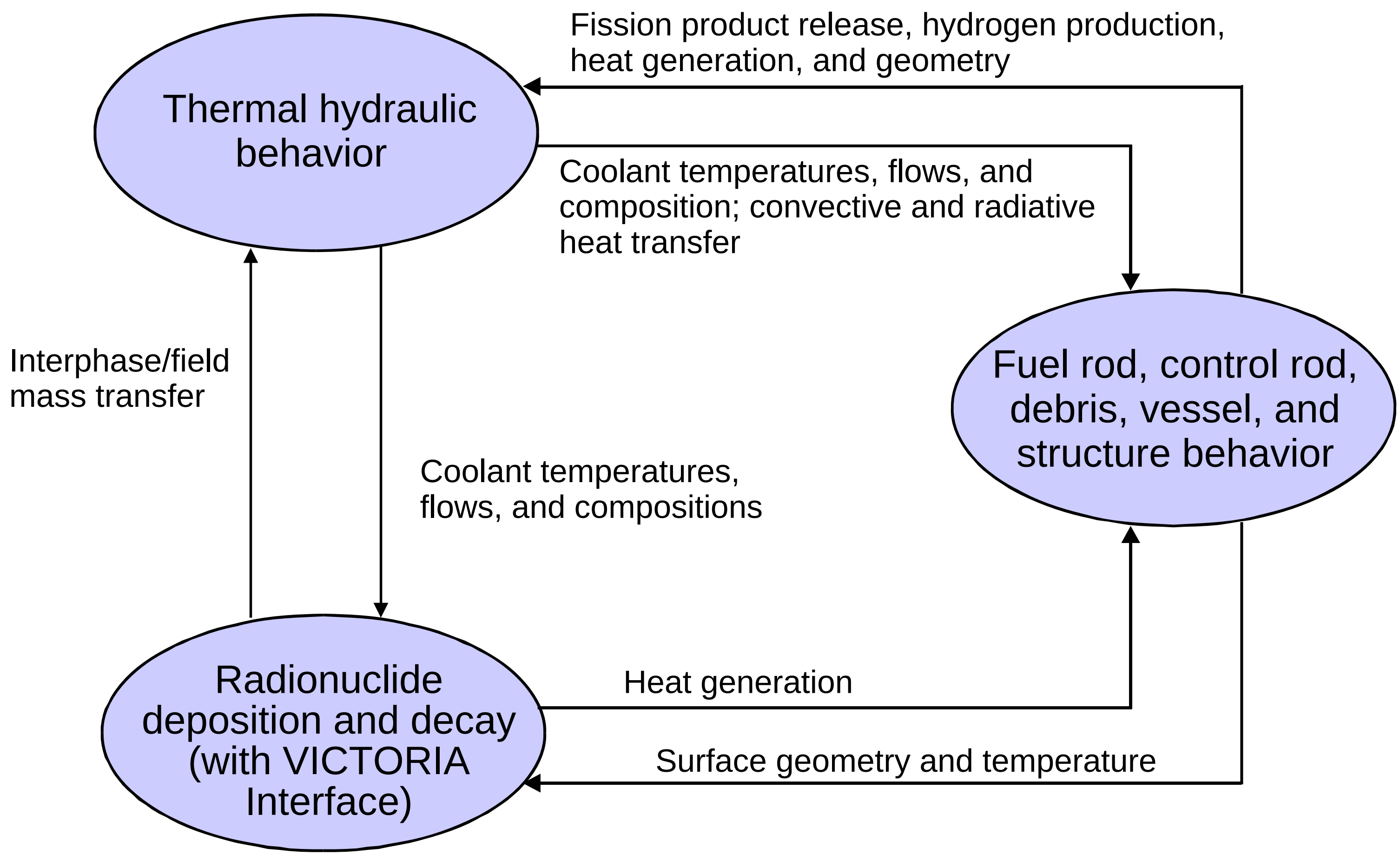
Approximate Accident Phenomena Covered by U.S. Severe Accident Computer Codes



SCDAP/RELAP5 Embodies Understanding of Severe Accident Processes



SCDAP/RELAP5 Models Wide Range of Accident Phenomena



SCDAP/RELAP5 Design Facilitates Assessments

- Building block approach for system thermal hydraulics
- Control system/trip logic
- Representative 2D fuel rods, control rods/blades, and structures for assessing early phases of melt progression
- Lumped parameter and 2D finite element models for simulating late phase behavior of debris and structures
  - In-core formation, growth, and collapse of molten pools, debris/melt/structural interactions treated with detailed, lumped parameter models
  - Relocation of molten core materials and upper plenum structures into lower head including interaction with (and degradation of) lower core support structures
  - Formation and growth of molten pool (natural circulation treated with effective conductivity approach in 2D FE model)

SCDAP/RELAP5 User Parameters Intentionally Minimized

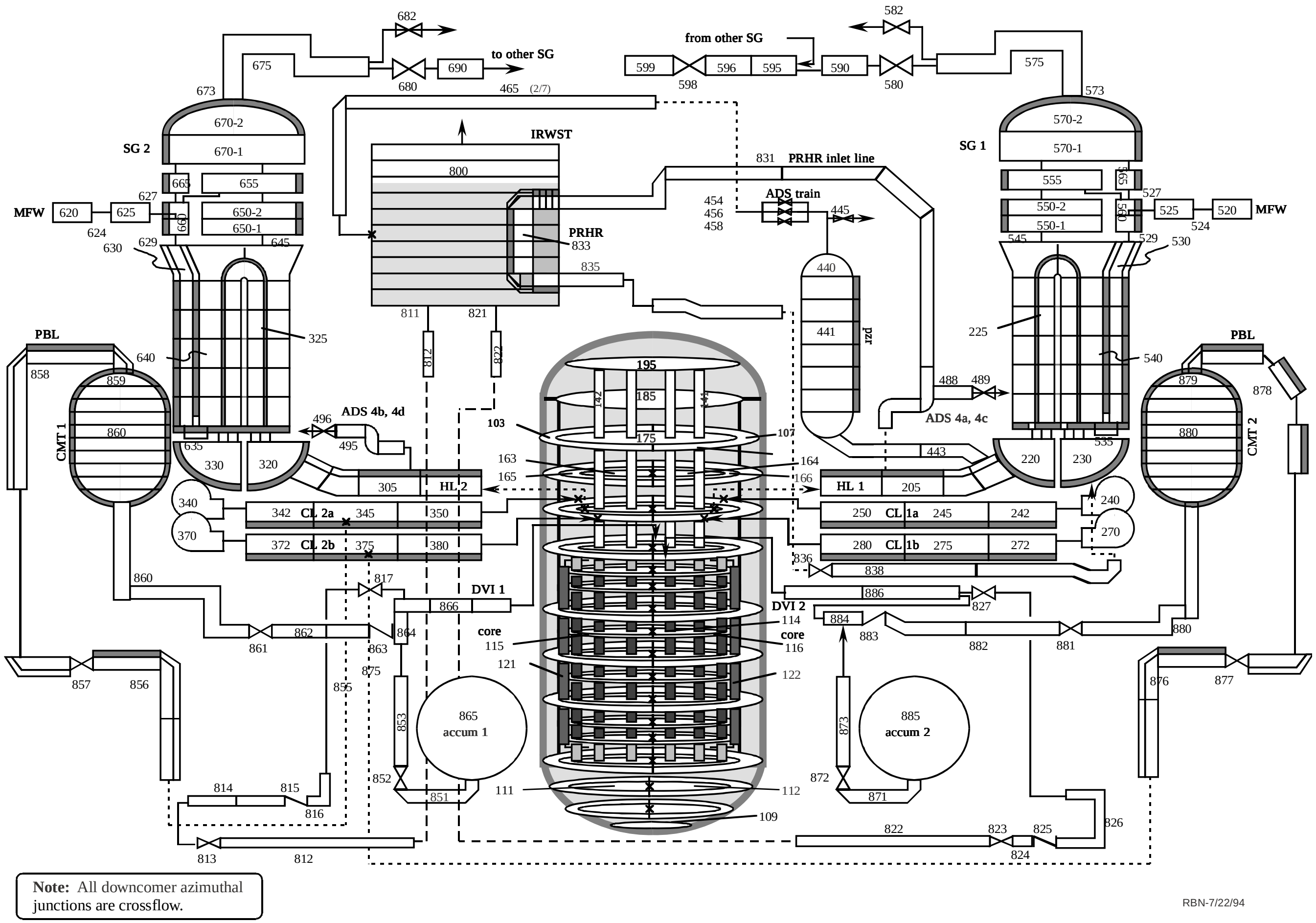
- TH nodalization and selection of representative core components
- TH models automatically select from 11 flow regimes or select alternative correlations or models
- Damage progression parameters limited to critical areas with significant modeling uncertainty:
  - Defaults provided for user applications
  - Defaults set to best estimate values obtained from code-to-data comparisons

SCDAP/RELAP5 Employs Two-Fluid, Non-Equilibrium Model

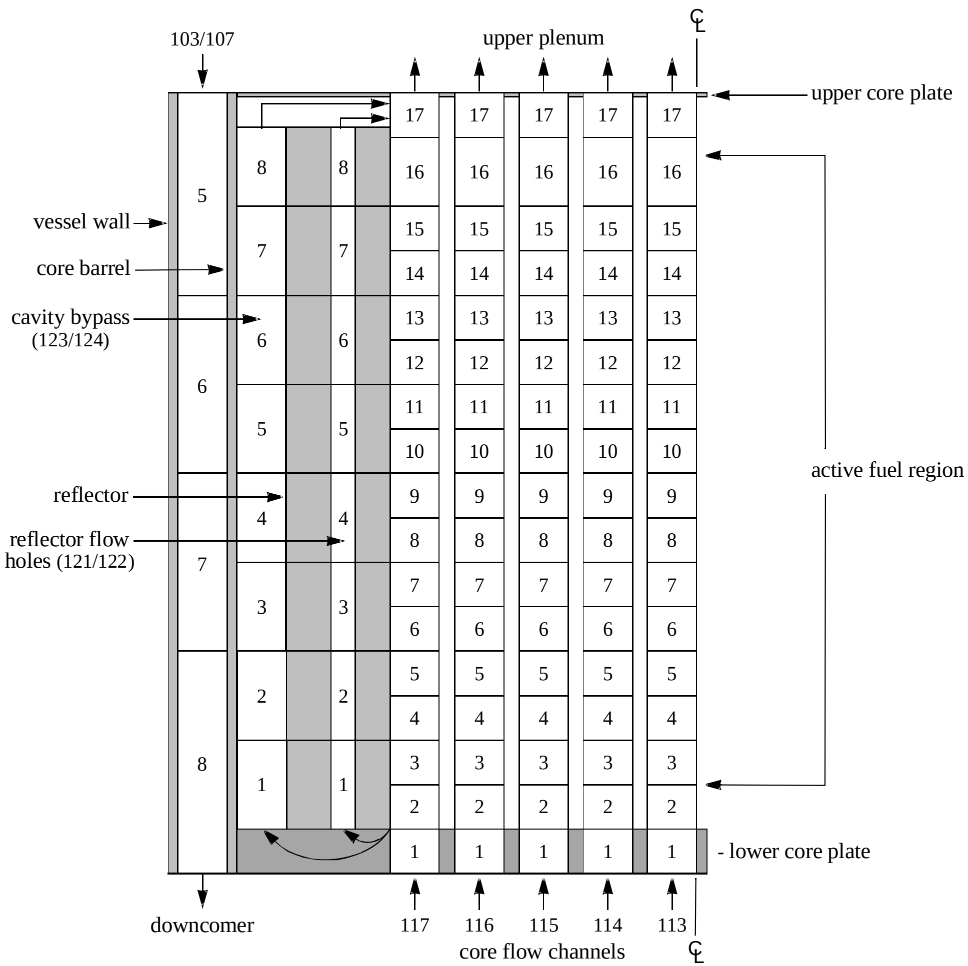
- 2D hydrodynamics typically used in the vessel to predict flow patterns associated with natural circulation and changes in geometry

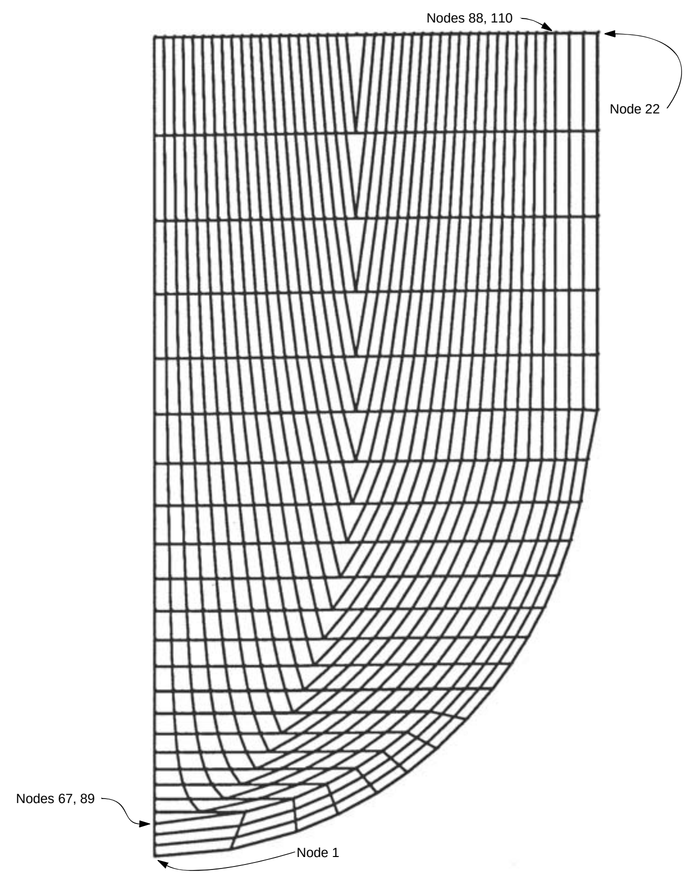
- Empirical models developed for hot leg natural circulation used in PWR calculations.

Representative SCDAP/RELAP5 Analysis Includes Several Models

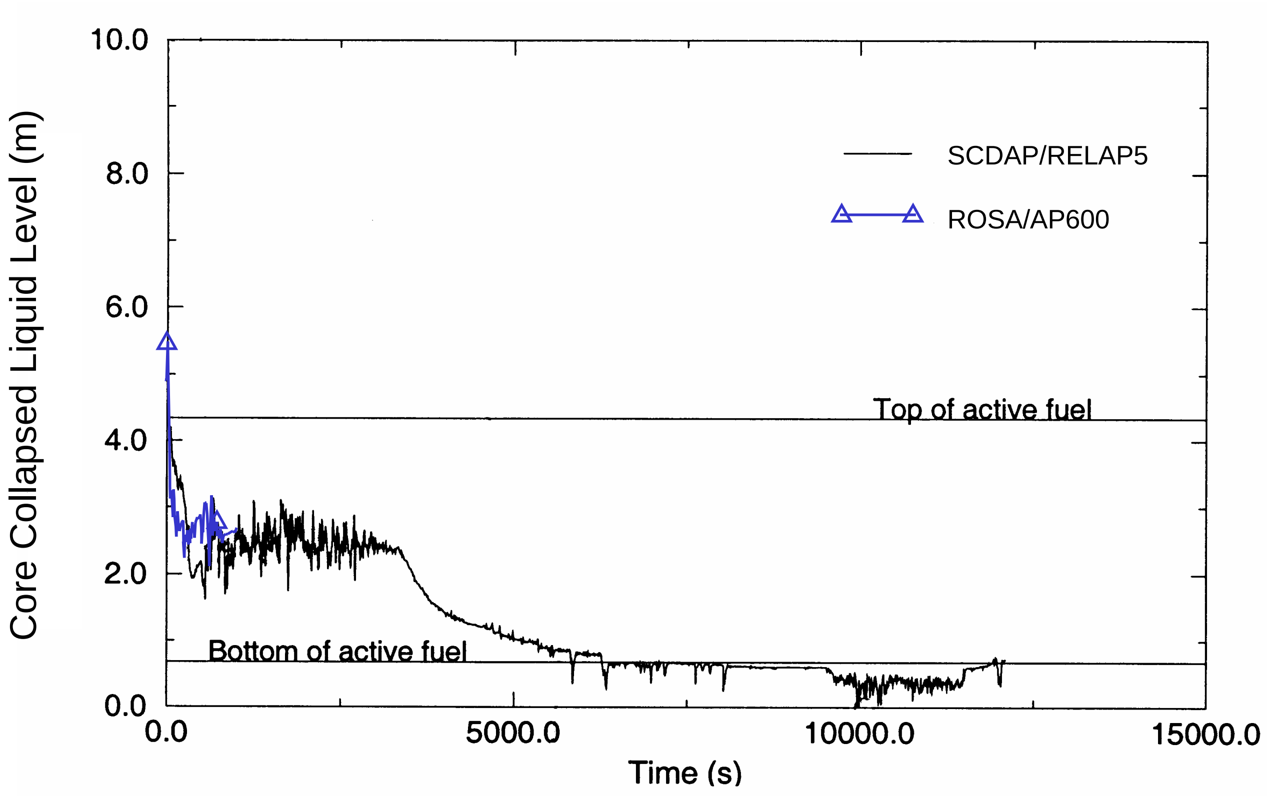


Representative SCDAP/RELAP5 Analysis Includes Several Models (continued)



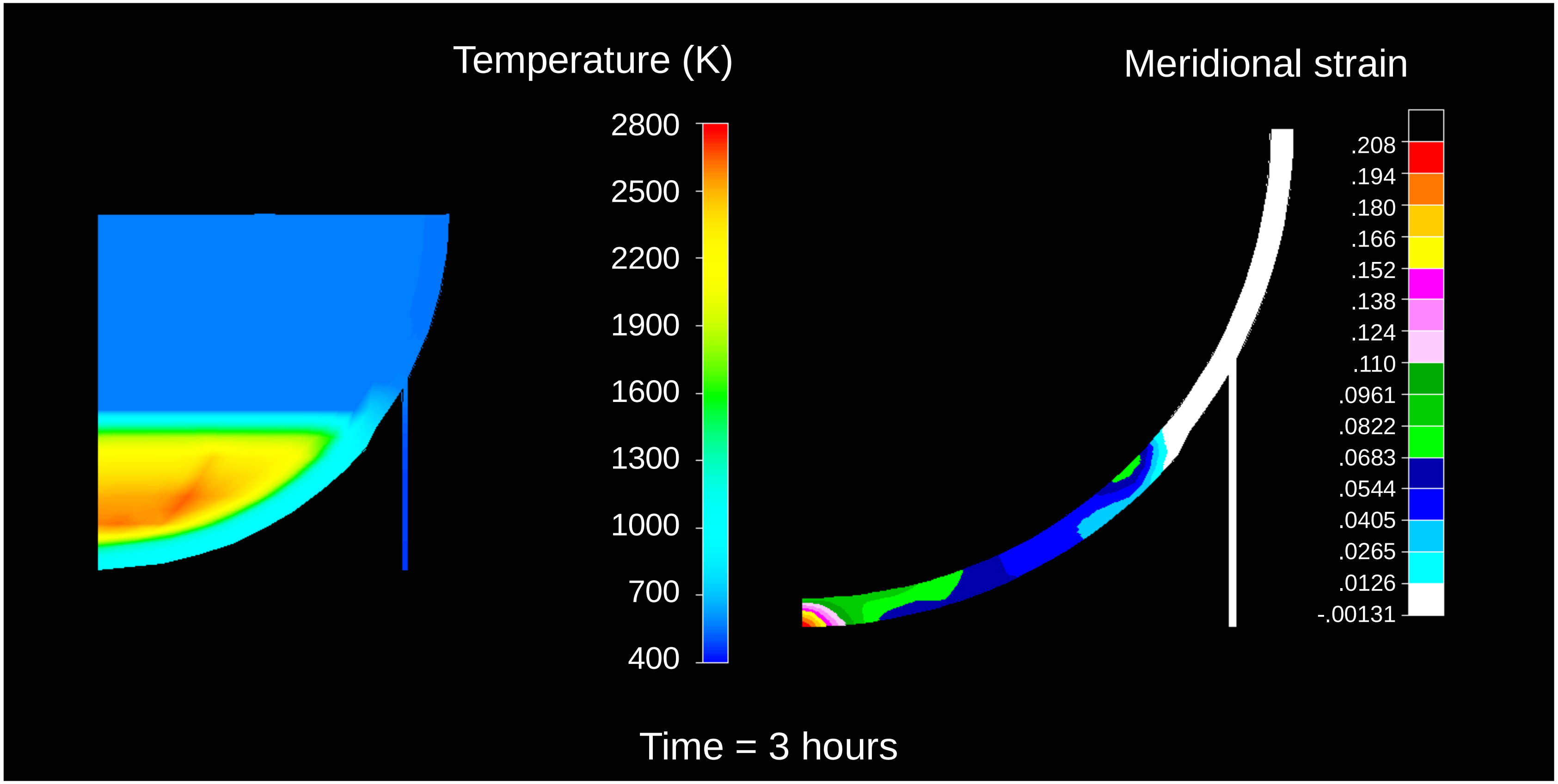


Code-to-data Comparisons Confirm SCDAP/RELAP5 Modeling Capabilities

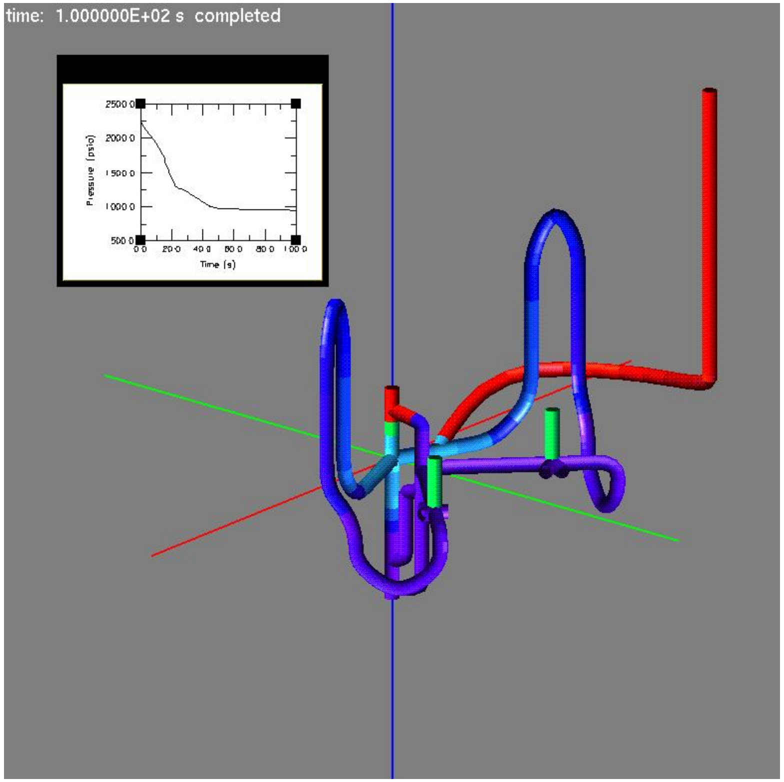


SCDAP/RELAP5 depressurization predictions consistent with ROSA/AP600 data for 3BE LOCA transient with vessel reflood

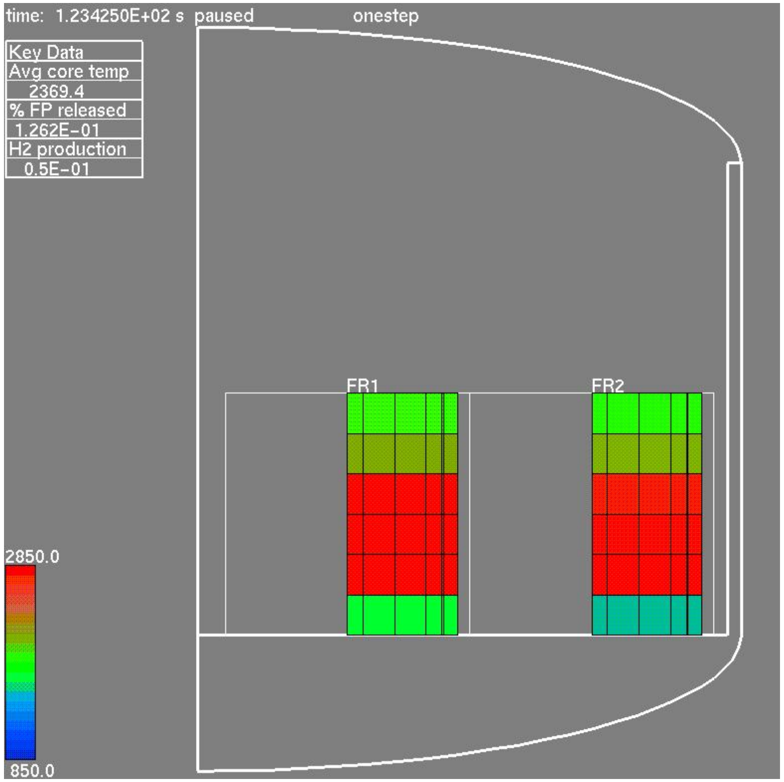
Coupled SCDAP/RELAP5 - ABAQUS - PATRAN Analysis Allows Detailed Structural Response



SCDAP/RELAP5 GUI Allows Users to Identify Input Errors and View Run-Time Results

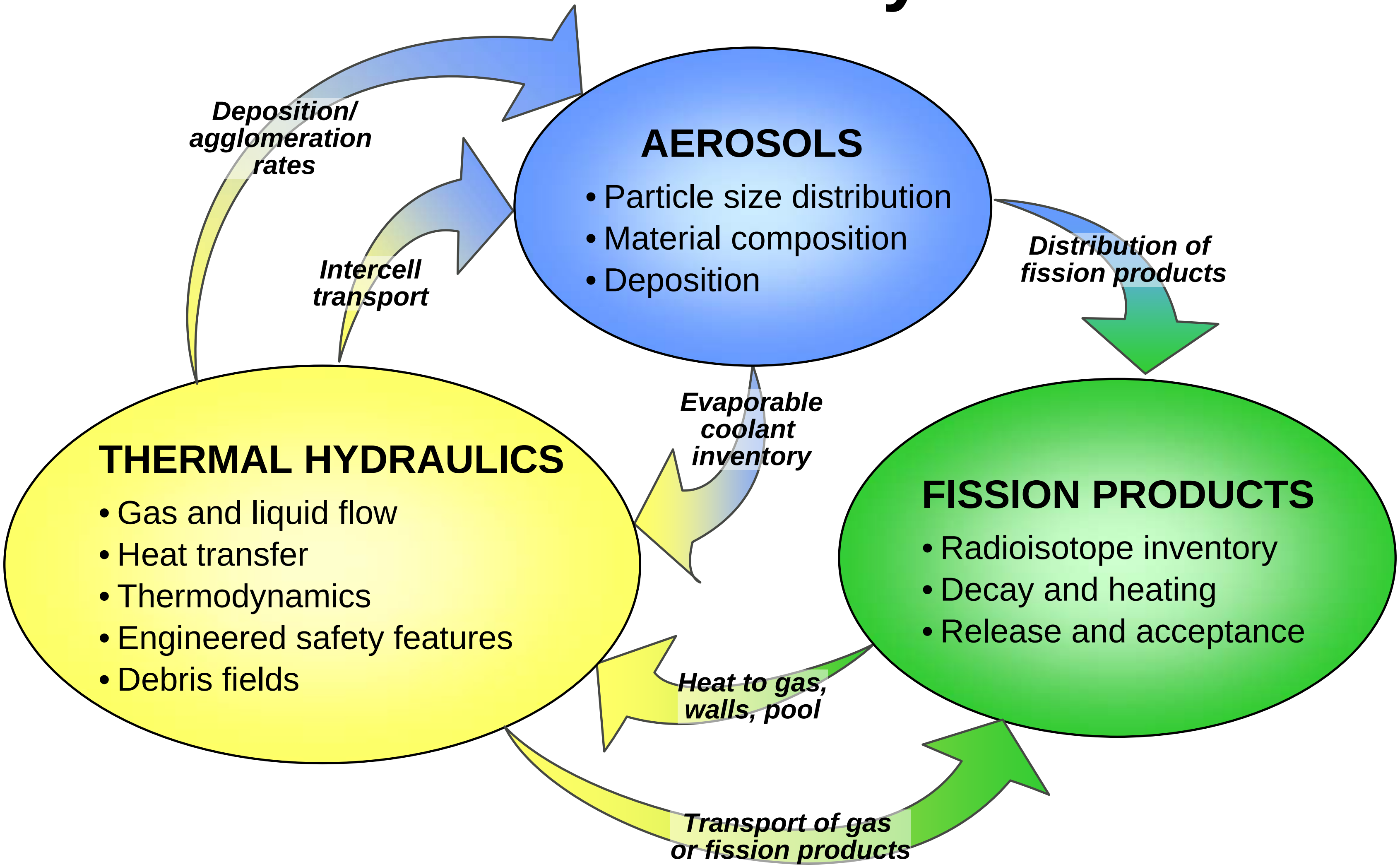


RELAP5

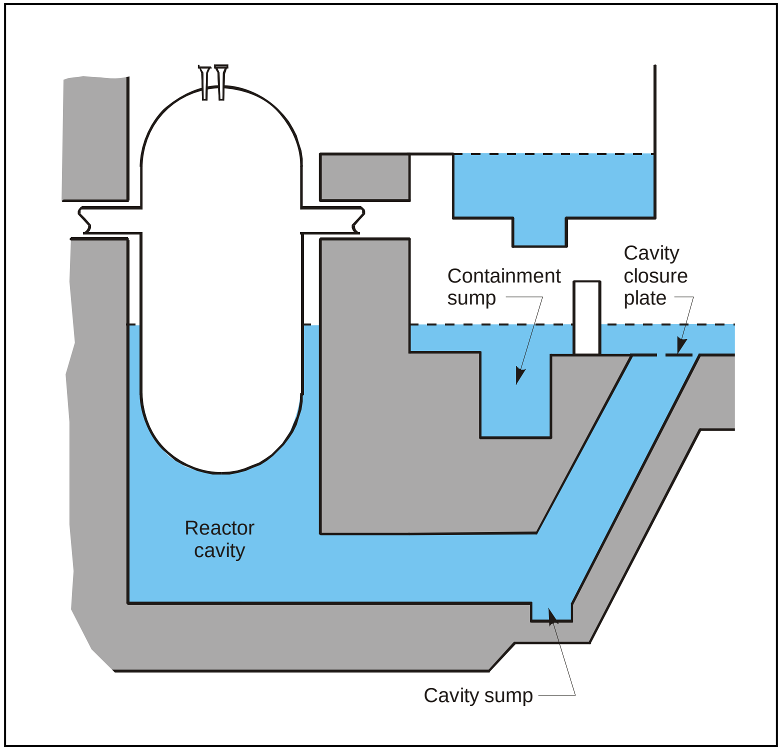


SCDAP

CONTAIN provides mechanistic containment analyses tool.



Linked mechanistic vessel/containment response analysis tool important for advanced reactors.



- Increased dependence on passive systems
  - ERVC
  - IRWST
  - PCCS
- Requires analyses with increased fidelity in heat and mass transfer between RCS and containment

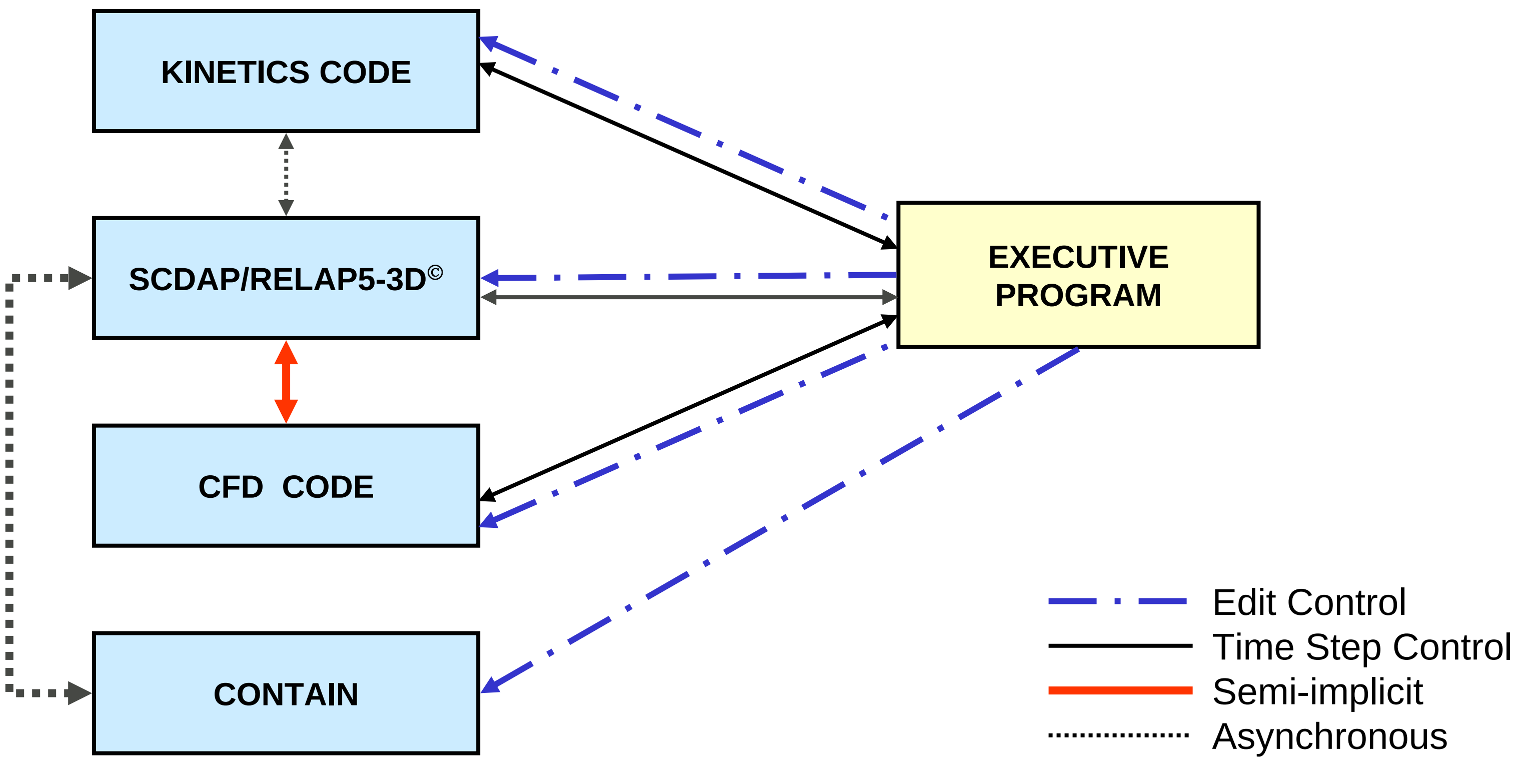
Initial SCDAP/RELAP5-CONTAIN PVM linkage completed by PSU.

- Used PVM software to link SCDAP/RELAP5/MOD3.0 and CONTAIN 1.1
- “Limited” linkage”:
  – Transfers break flowrates, SRV discharges, pool injection sources
  – Heat transfer from selected structures
  – Instantaneous flowrates, rather than integrated flowrates
  – Doesn’t transfer non-condensable gases (hydrogen), fission products, aerosols, or discharged corium
- Demonstrated capabilities by analyzing Brown’s Ferry ATWS
  – Linked code predicted more defensible vessel/containment response

New PVM linkage improves upon previous linkage efforts.

- SCDAP/RELAP5-3D© and CONTAIN 2.0 codes integrated using PVM software and Executive Program
- Applied insights gained from previous linkage efforts:
  - Relative time for calculations to advance (and relative timestep)
  - Form of variables for data transfer (enthalpy and mass flowrate)
  - Subroutines selected for extracting and receiving data
  - Review previous coding and improve, as needed
- Compared reactor plant analysis with and without integrated code

New PVM linkage improves upon previous linkage efforts. (continued)

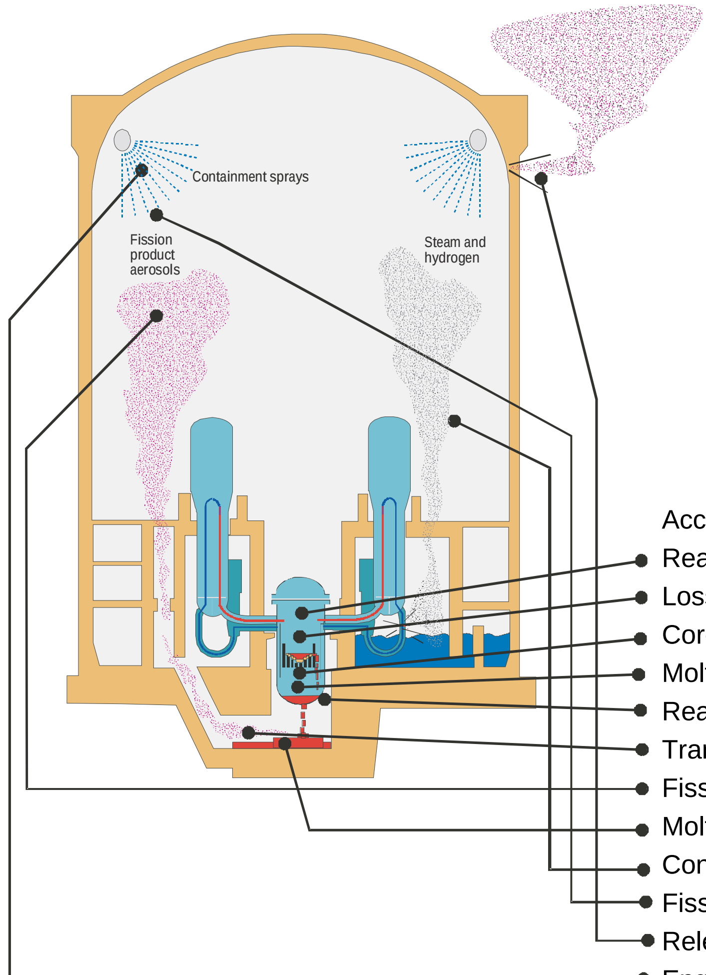


MELCOR Developed to Support Risk Assessment Studies

- Developed at Sandia National Laboratories for NRC in 1982 for full-plant assessments.
- Major concern was integration
  - Replace simple, special purpose codes (STCP)
  - Eliminate tedious hand-coupling between modules
  - Capture feedback effects
    - Coupling of temperatures, release rates, and decay heating
    - Track relocation of heat sources

Methods

MELCOR Provides Integrated Accident Analysis



Accident initiation

- Reactor coolant thermal hydraulics
- Loss of core coolant
- Core meltdown and fission product release
- Molten fuel/coolant interactions
- Reactor vessel failure
- Transport of fission products in RCS and containments
- Fission product aerosol dynamics
- Molten core/basemat interactions
- Containment Thermal Hydraulics
- Fission Product Removal Processes
- Release of Fission Products to Environment
- Engineered safety systems - sprays, fan coolers, iodine chemistry, etc.

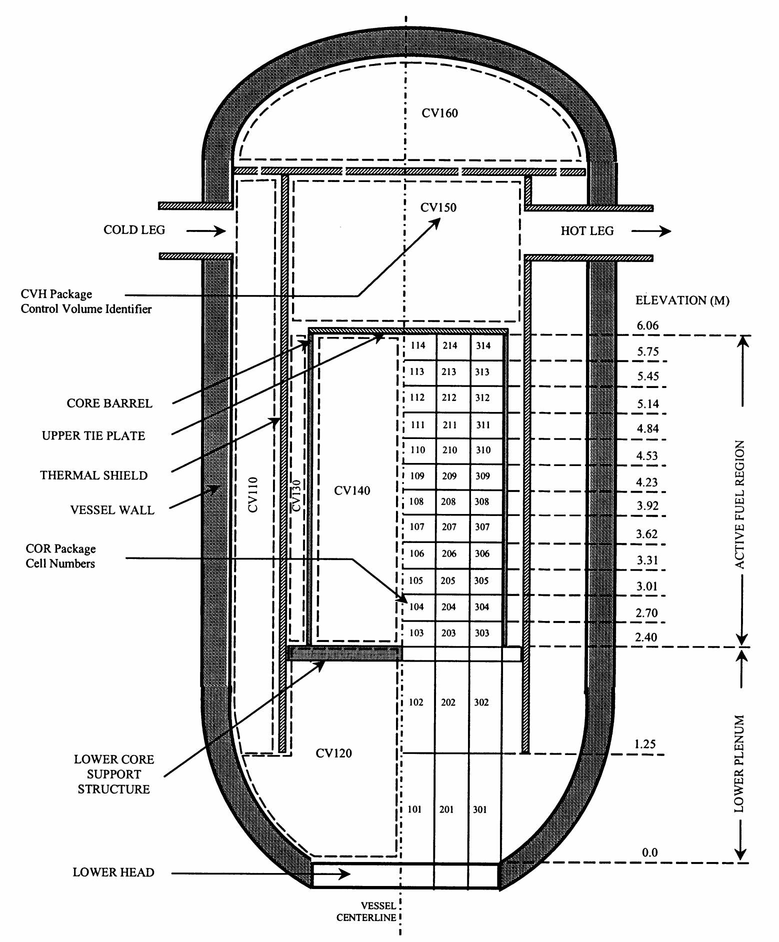
MELCOR Role Evolving

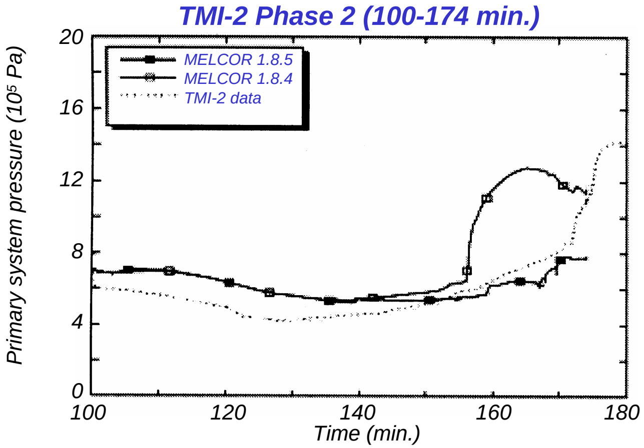
- Original role for PRAs required simpler, fast-running code
  - Uncertainties assessed through sensitivity studies
  - Substantial user flexibility allowed for parametric studies
- Recent role using more detailed models
  - NRC consolidating to one code
  - Assessments against more detailed codes used to determine required model complexity
  - More mechanistic models implemented as determined necessary

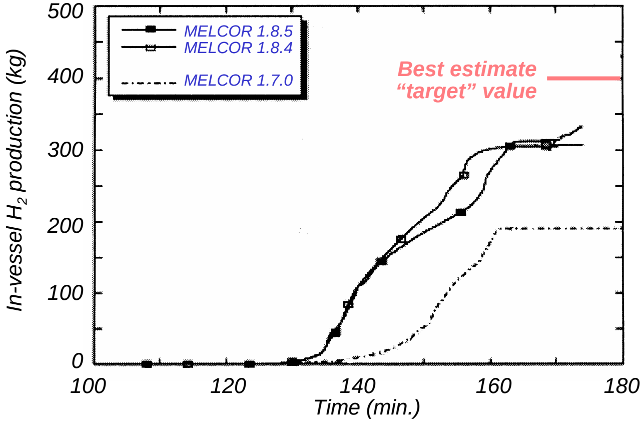
MELCOR Role Evolving (continued)

- Recent role using more flexible modeling geometry
  – Generic modeling without “built-in” nodalization
  – Control volume approach used to define plant system
  – Reactor-specific geometry imposed in modeling reactor core
  – BWR-specific geometry invoked in lower plenum modeling package
- Application NOT limited to LWR reactor accident analysis

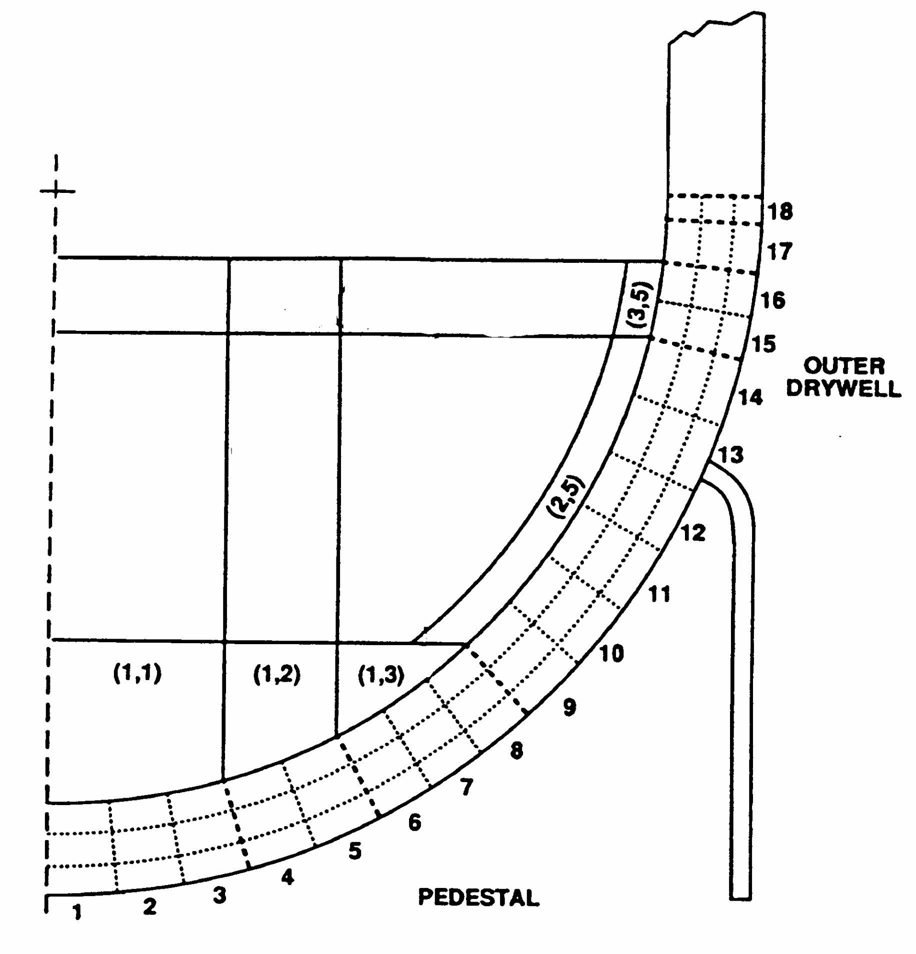
Recent MELCOR Release More Closely Represents TMI-2 Data







MELCOR BWR Lower Plenum Models Consider Lower Plenum Debris Bed



MELCOR Assessment with More Mechanistic Codes Underway

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

Many CONTAIN Models Embodied in MELCOR

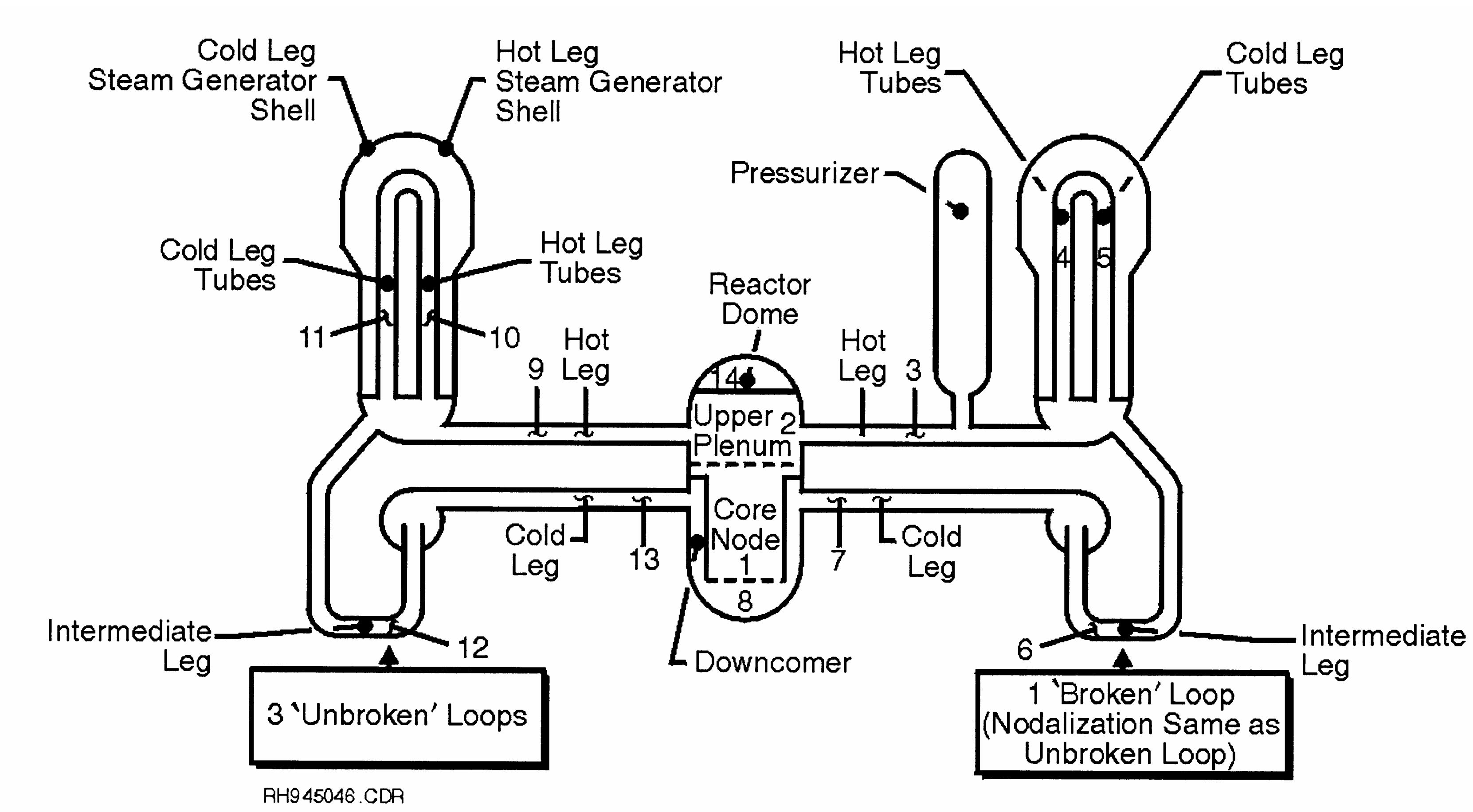
- CORCON-MOD3 CCI used in both codes
- HECTR 1.5 models used to predict combustion for both codes
- Models for aerosol transport similar
  – MELCOR includes model for fission product vapor transport
  – MELCOR neglects fission product transmutation
- Radiation from corium to containment structures
  – MELCOR neglects, but CONTAIN considers
- DCH material transport models differ
  – MELCOR requires user-specified transport input
- Flow solvers differ
  – MELCOR flow path approach requires judicious nodalization to prevent overmixing
  – CONTAIN advanced hybrid solver automatically mitigates overmixing and allows stratification
- Similar models for ESFs (sprays, fan coolers, etc.)
- Containment failure analysis approach similar

MAAP Designed for Full-Plant Calculations

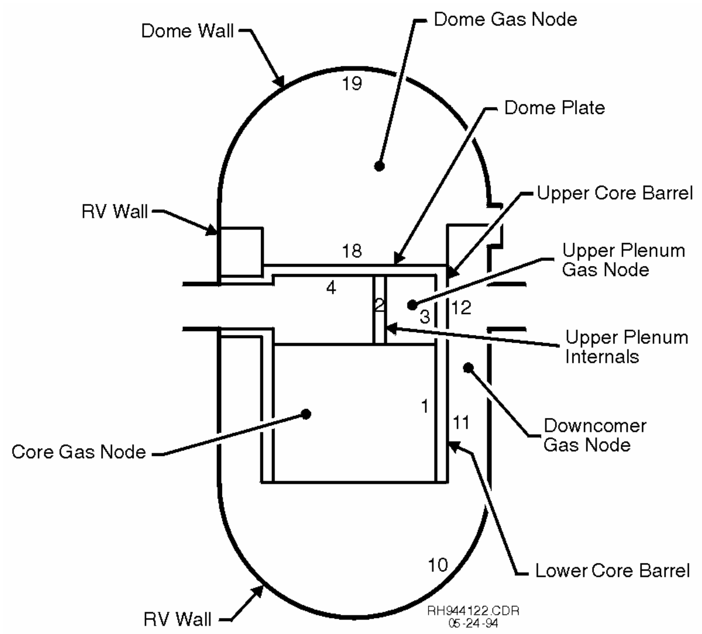
- Used by industry for risk assessments, IPEs, IPEEs, etc.
- Integrated RCS and containment analysis
- Extensive user dials for parametric analysis
- Control system/trip logic functions
- Lumped parameter models for global approximations
- Design specific versions (BWR, PWR, …) with relatively fixed thermal-hydraulic system representations
- Model validation against experimental data requires special models or versions.

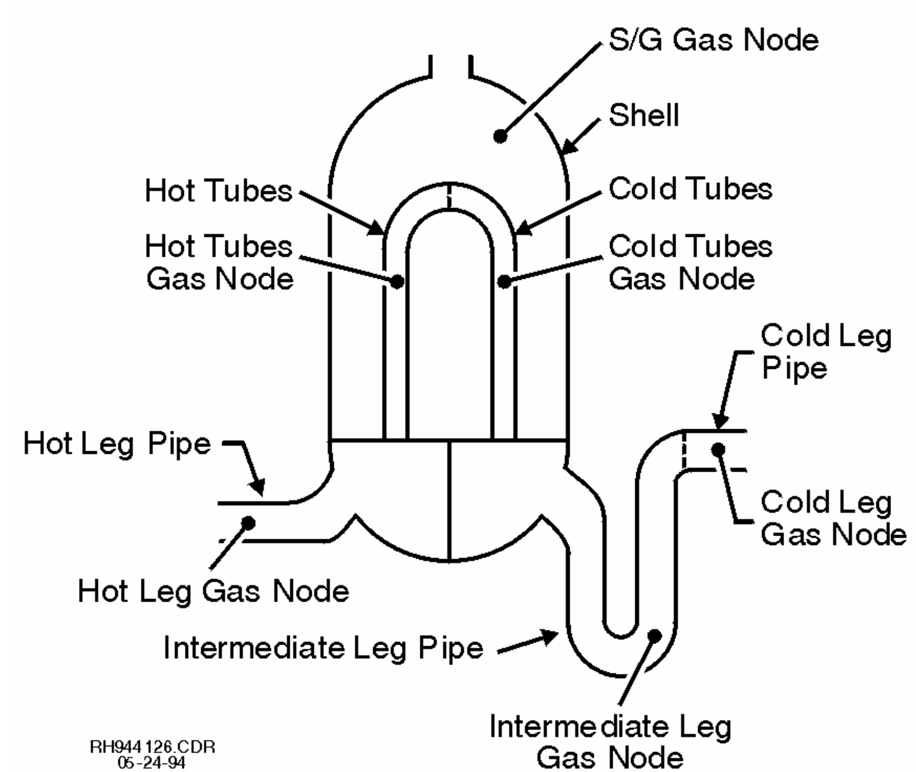
Methods

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

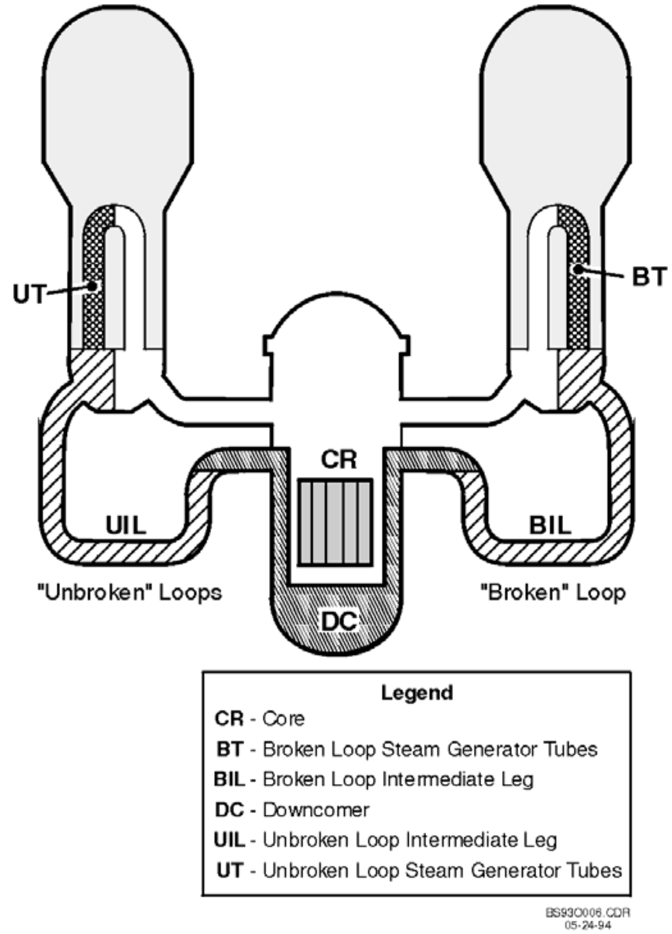


Representative MAAP PWR Analysis Considers Gas Nodes, Heat Structures, and Water Nodes (continued)

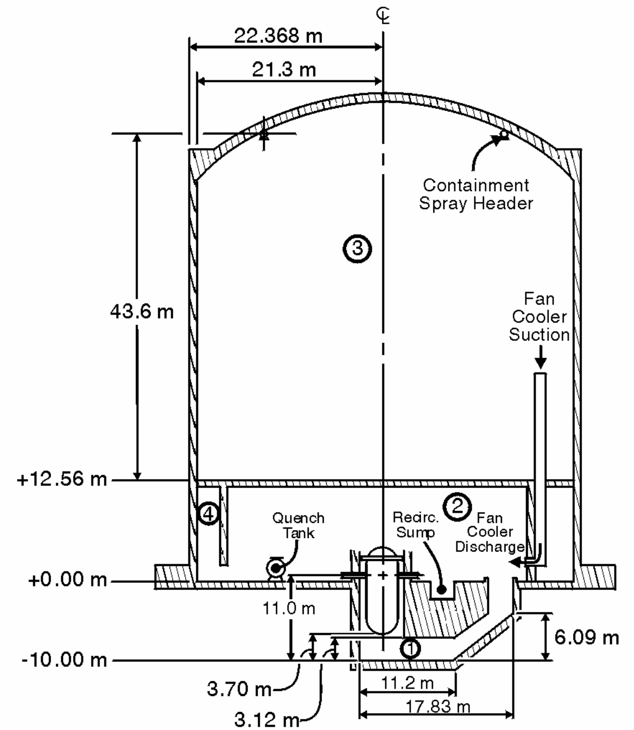




Representative MAAP PWR Analysis Considers Gas Nodes, Heat Structures, and Water Nodes (continued)



Representative MAAP PWR Containment Analysis Considers Interconnected Compartments and Flowpaths (continued)



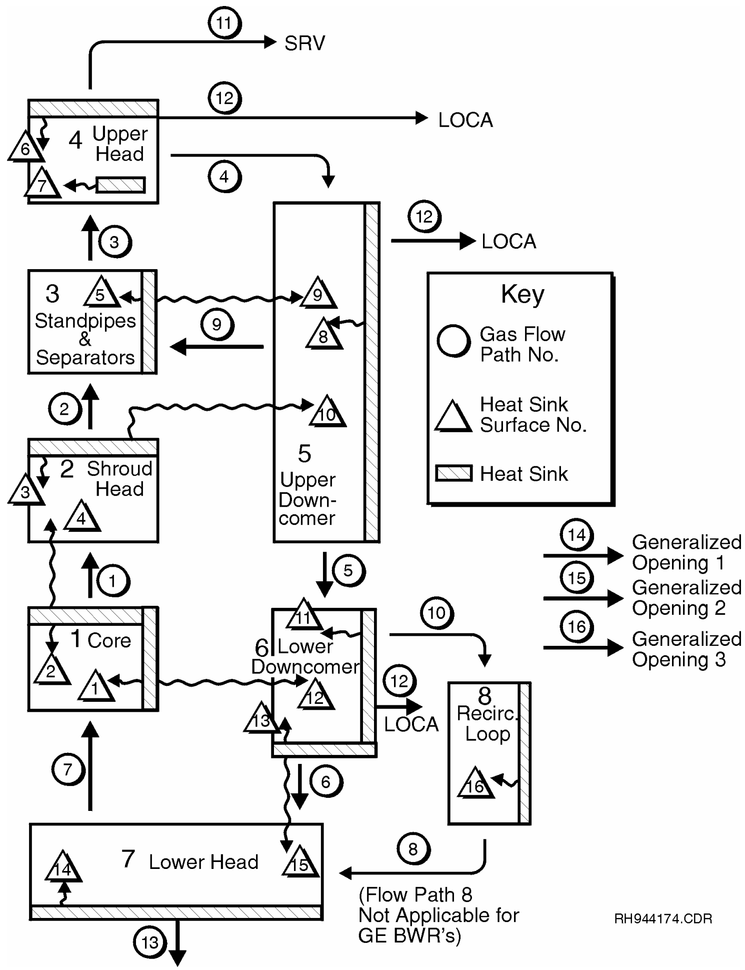
Compartment ID	Description
1	Cavity
2	Lower Compartment
3	Upper Compartment
4	Annular Compartment

TE935047.CDR

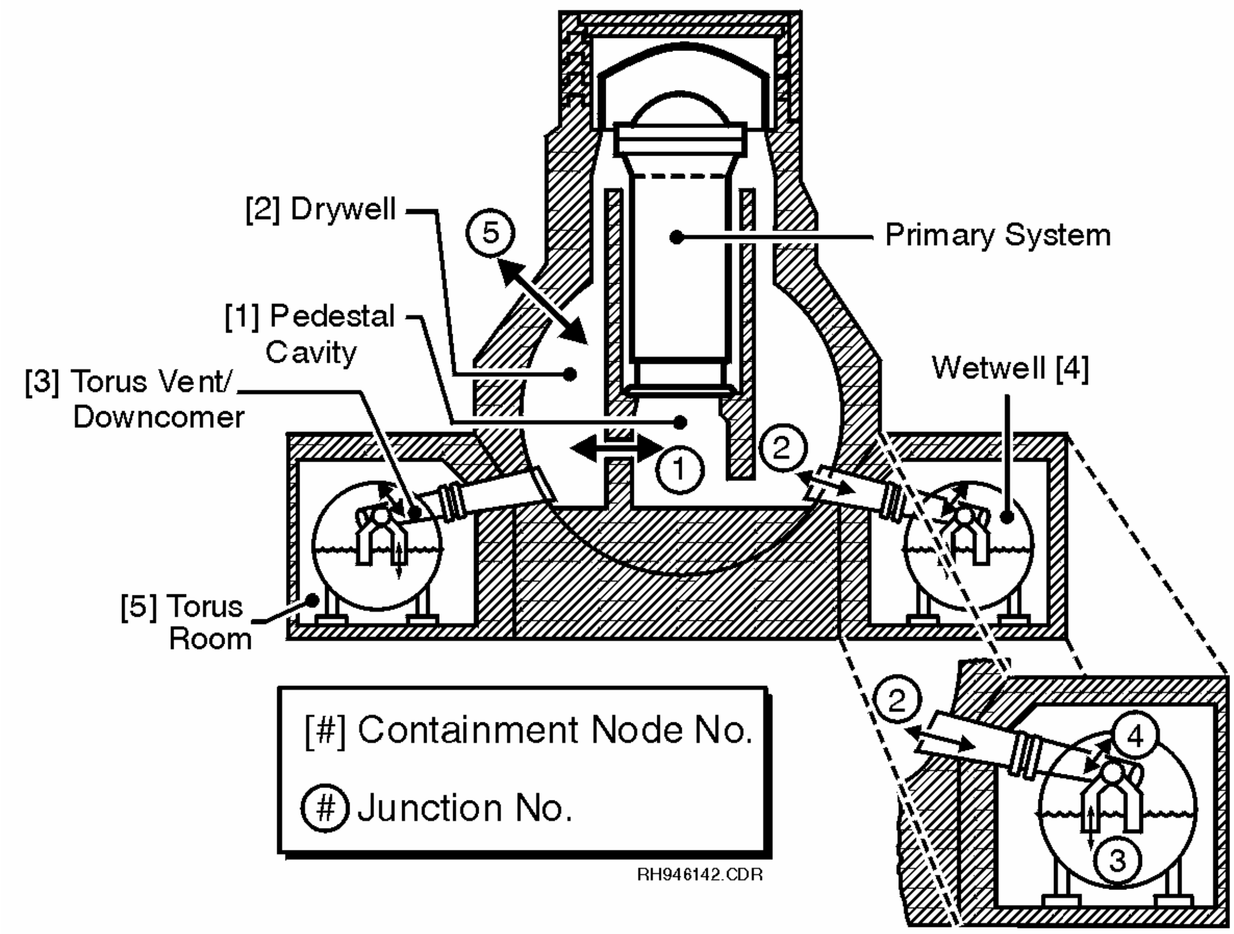
Zion-Like
MAAP4 Containment
Vertical Section

Methods

Specialized Components and Containment Designs Modeled in MAAP BWR



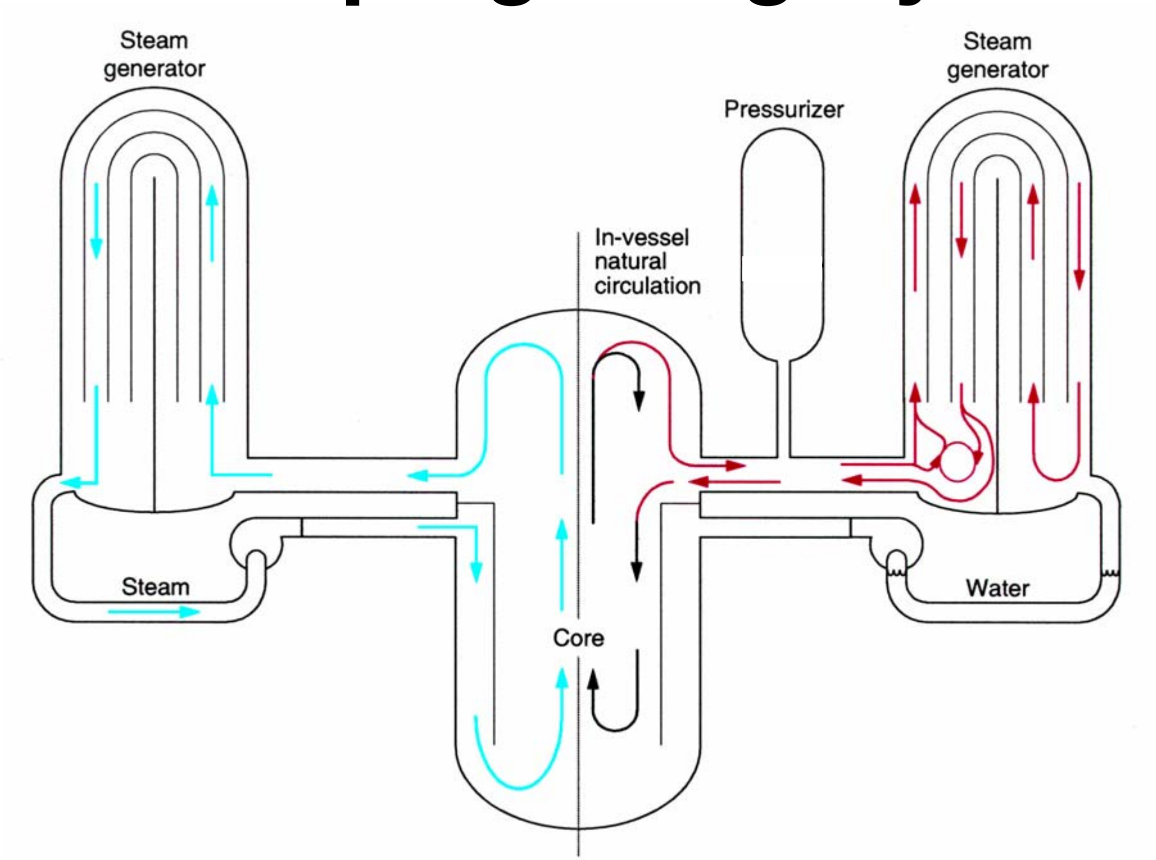
Specialized Components and Containment Designs Modeled in MAAP BWR (continued)



Case Study 1: SCDAP/RELAP5 Evaluation of Potential for SGTR

Case Study 1

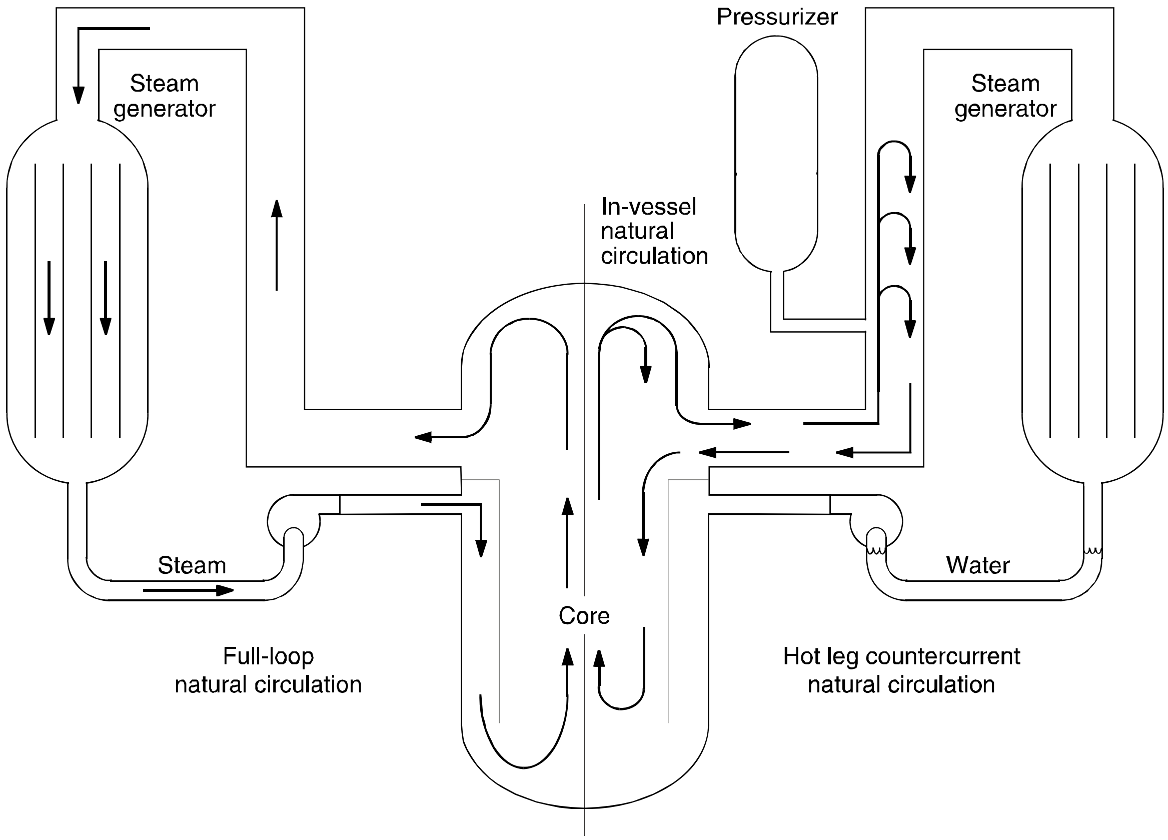
Natural Circulation Affects RCS Piping Integrity



PWRs with U-Tube SGs

Case Study 1

Natural Circulation Affects RCS Piping Integrity (continued)



PWRs with Once-Through SGs

Large Number of SCDAP/RELAP5 SBO Calculations Performed to Assess SGTR Potential

- Initiated by loss of off-site power followed by loss of all AC power and feedwater
- Turbine stop valves close, isolating SG secondaries
- SG pressures increase until relief valves cyclically open
- RCS pressures and temperatures increase until PORVs open cyclically
- If RCP seal LOCA occurs, RCS depressurizes.
- Without power recovery, RCS coolant heats to saturation and boiloff occurs

Large Number of SCDAP/RELAP5 SBO Calculations Performed to Assess SGTR Potential (continued)

- Wide spectrum of plants analyzed
  - Surry - 3 Loop W
  - Zion - 4 loop W
  - Calvert Cliffs - CE with pressurizer PORV
  - ANO-2 - CE without pressurizer PORV
  - Oconee - B&W, once-through SG
- All calculations assumed defect-free SG tubes. (code modified to consider tube flaw distributions for subsequent calculations)

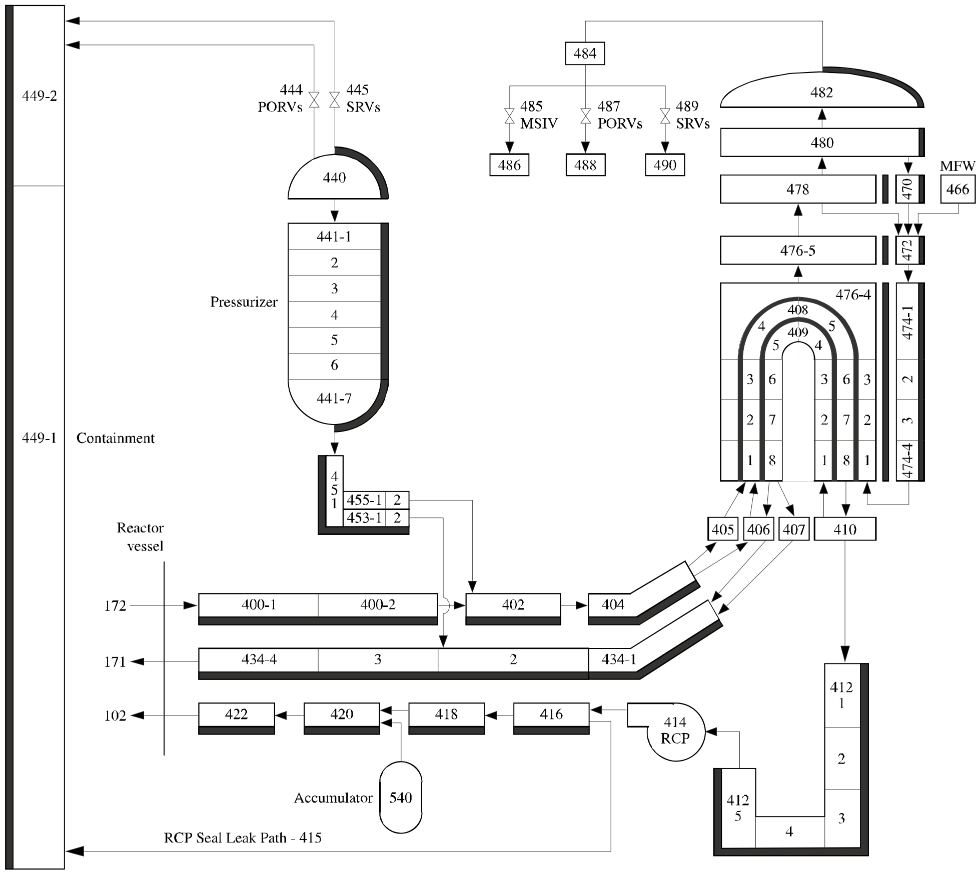
Case Study 1

Large Number of SCDAP/RELAP5 SBO Calculations Performed to Assess SGTR Potential (continued)

- SG U-Tube Split
- Mixing Fraction
- Recirculation Ratio
- RCS Pressure
- Secondary Pressure
- Upper Plenum Steel Relocation Mass
- Heat transfer modeling uncertainties
- Natural circulation modeling uncertainties
- Synergistic effects associated with natural circulation.
- Plant design
- Nodalization
- SG sludge accumulation

Case Study 1

S/R5 Counter Current Flow Model Benchmarked Against Westinghouse Data



Surry pressurizer loop nodalization for hot leg countercurrent natural circulation

Case Study 1

S/R5 RCS Failure Predictions Based on Larson-Miller Creep Rupture Theory

- Rupture time given by

$$t_r = 10^{\left[\frac{P}{T} + C_1\right]}$$

where $P = C_2 \log\sigma + C_3$

T = absolute temperature

C_1, C_2, C_3 = empirically derived constants

- Creep damage index given by

$$CDI(t + \Delta t) = CDI(t) + \frac{\Delta t}{t_r}$$

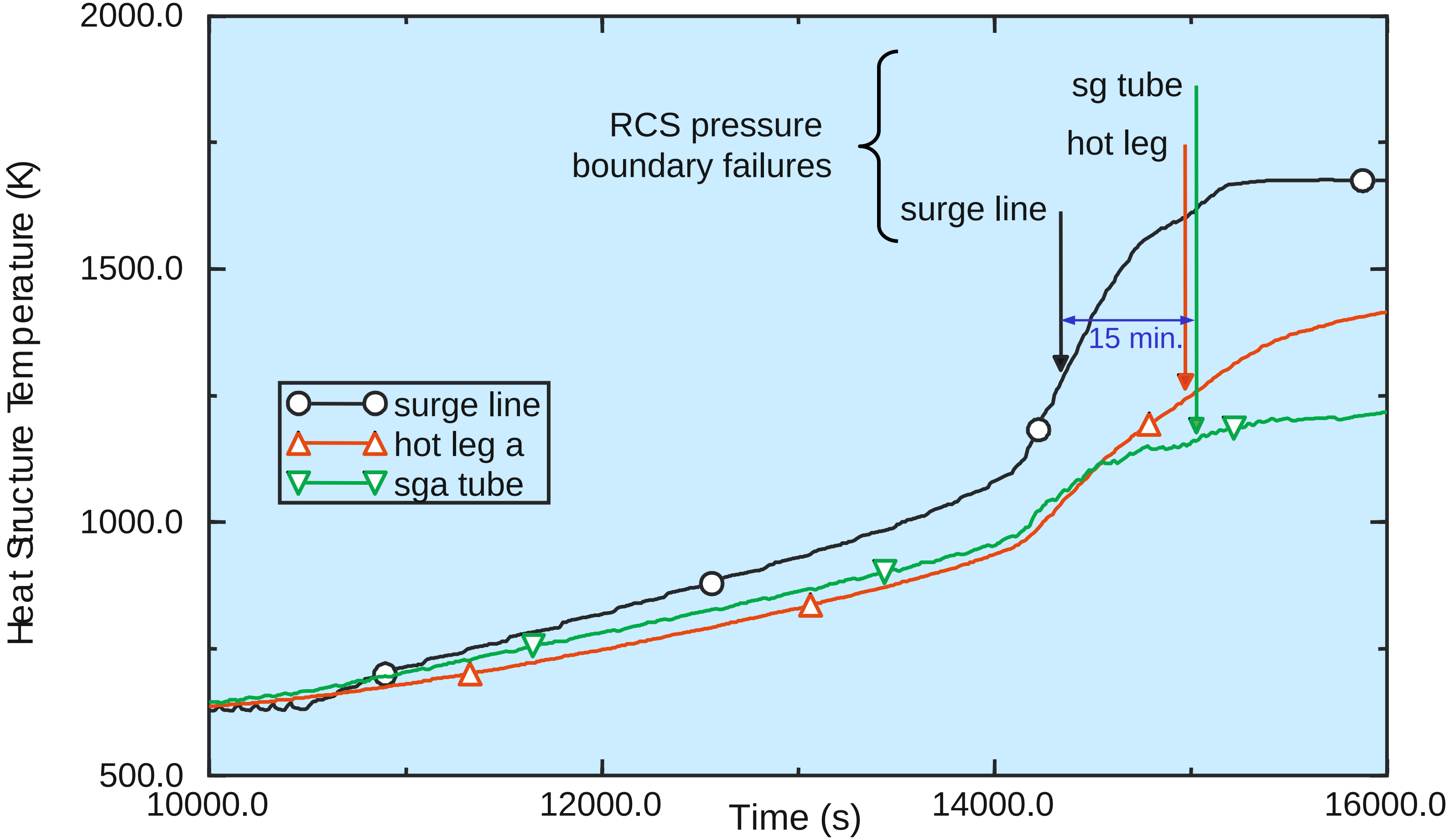
where t = current problem time

Δt = time step

S/R5 Results Provide Key Insights

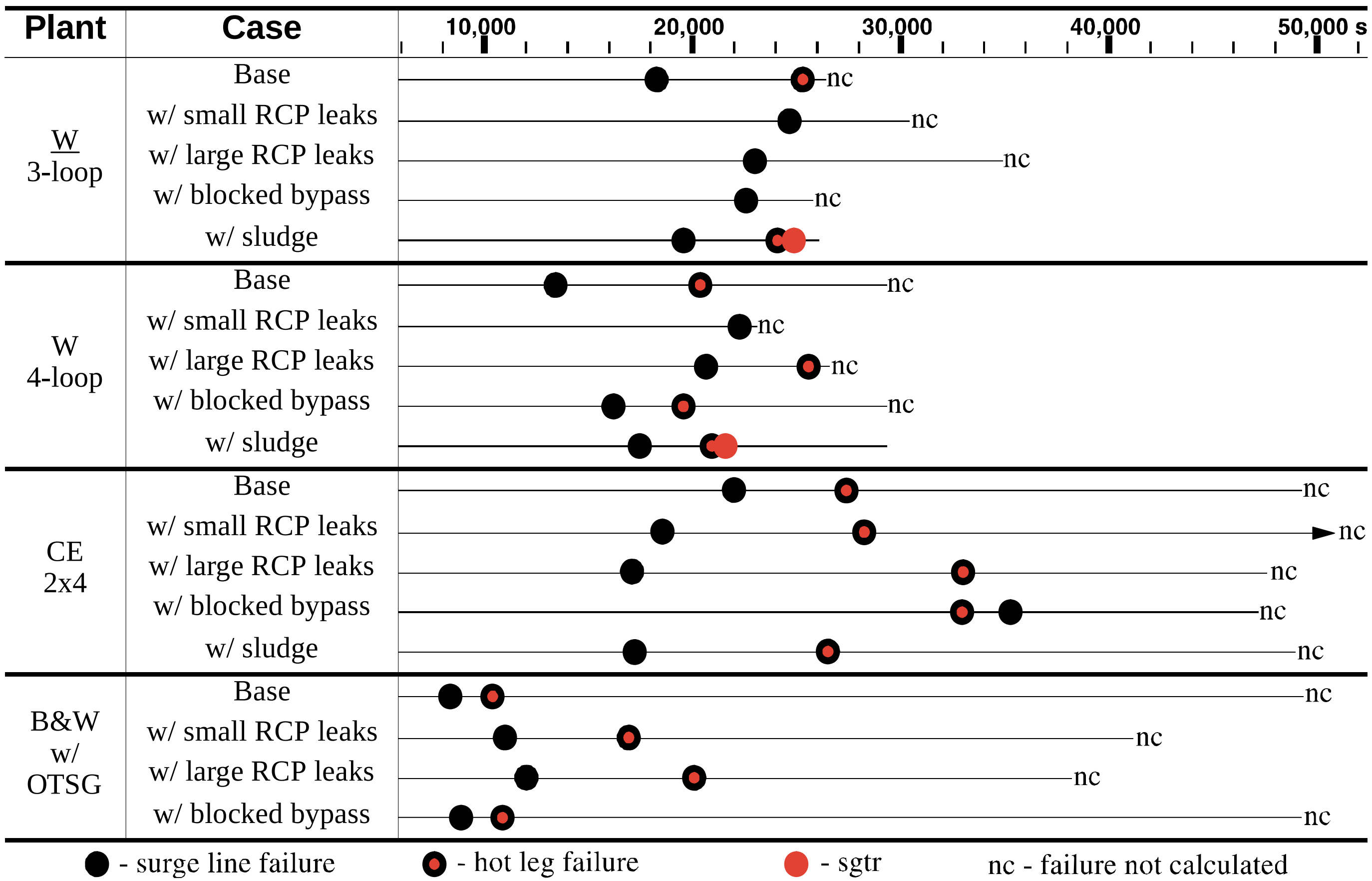
- Larson-Miller creep rupture model predicted surge line or hot leg failures prior to SG failures
- RCS depressurization and accumulator injection following surge line/hot leg failure preclude SGTR
- Cases with depressurized secondary side present more serious challenges to SG tubes

Margins Between Surge Line Failure and SGTR



Case Study 1

Margins Between Surge Line Failure and SGTR (continued)



Case Study 2: Comparison of Code Results for AP600 Analysis

Case Study 2

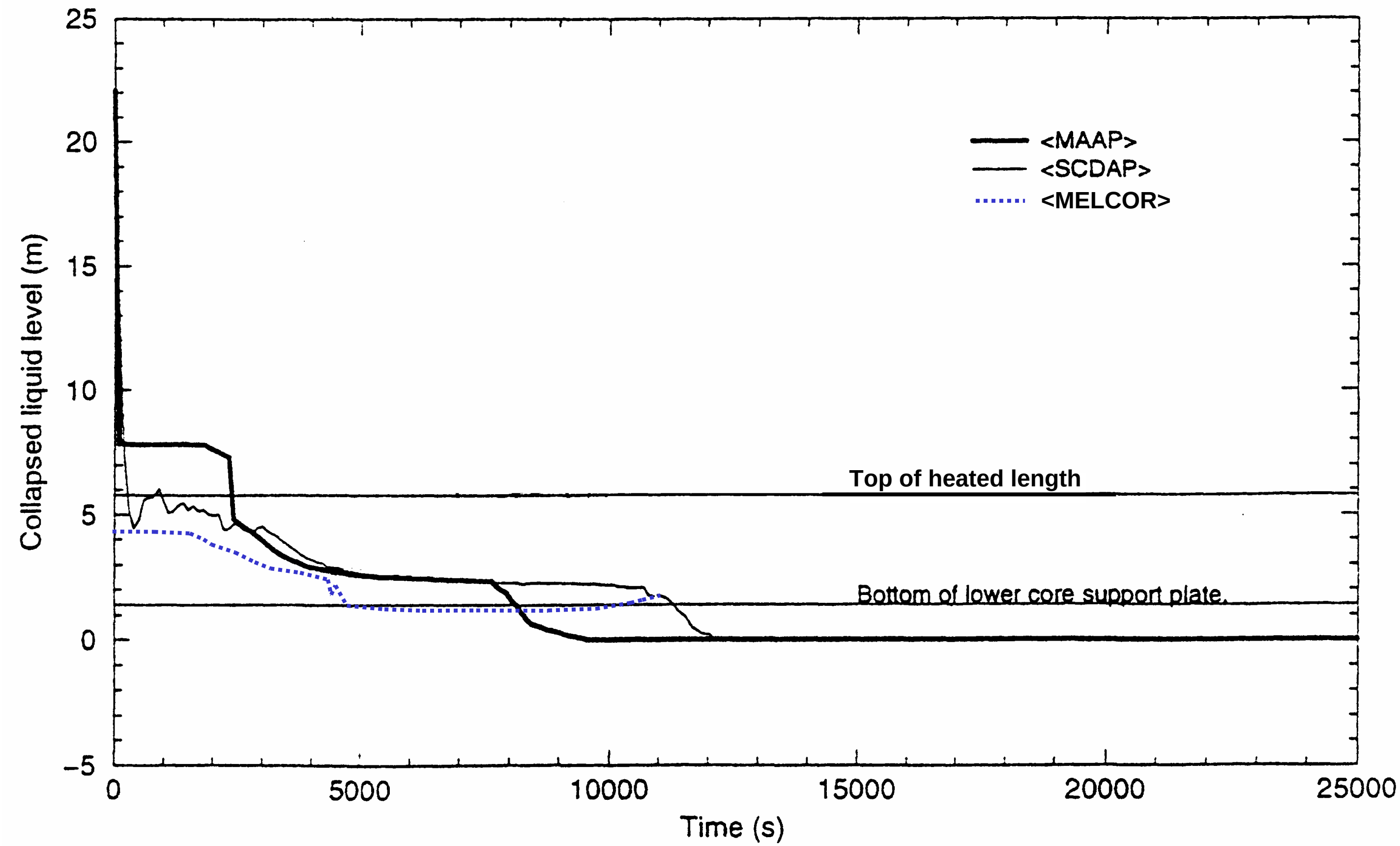
Code Models and Assumptions Impact 3BE AP600 Analysis Results

- 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 due to eutectic formation	At 3100 K (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, data needed to validate)	Enhanced cooling from water in user-specified gaps with user-specified heat transfer	No enhanced debris cooling (model developed, data needed to validate)

Case Study 2

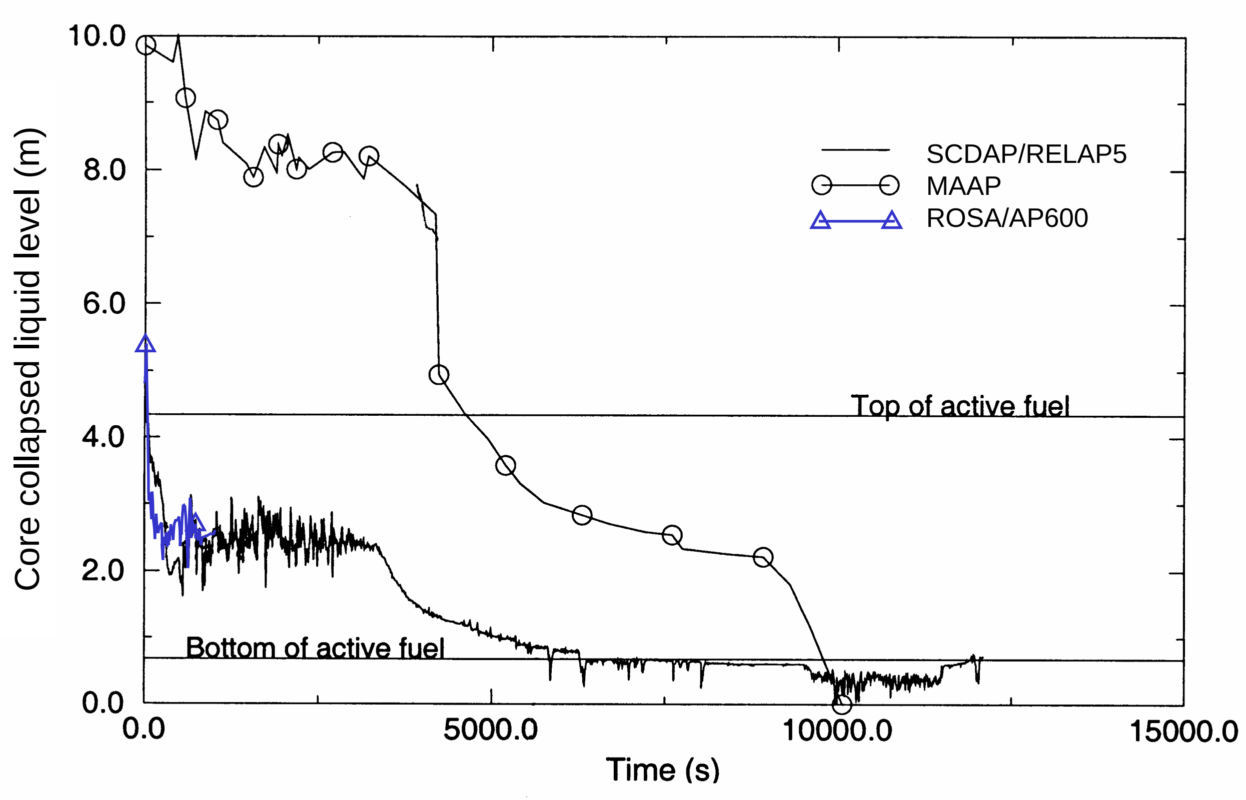
Code Models and Assumptions Impact 3BE AP600 Analysis Results (continued)



Unexplained additional coolant retained in RCS for MAAP calculation

Case Study 2

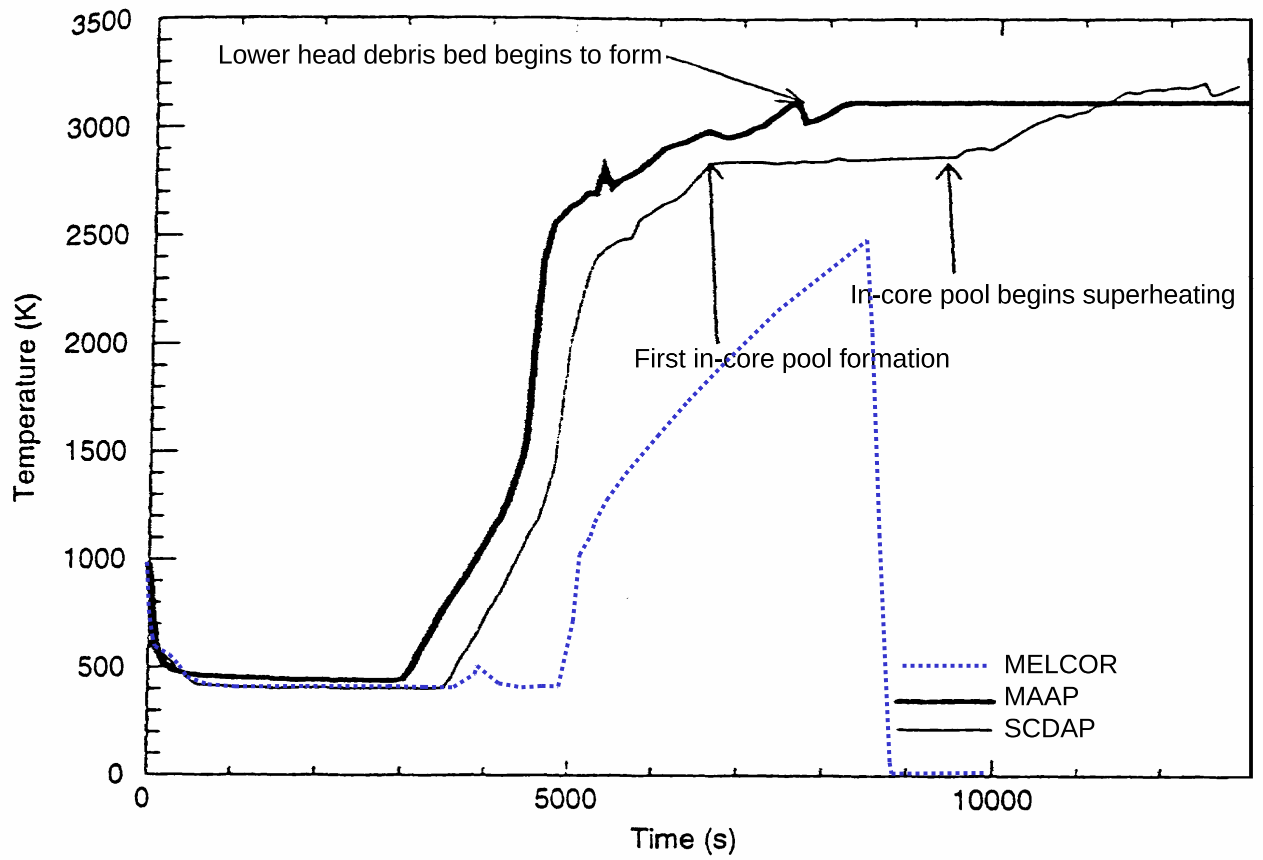
Code Models and Assumptions Impact 3BE AP600 Analysis Results (continued)



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

Case Study 2

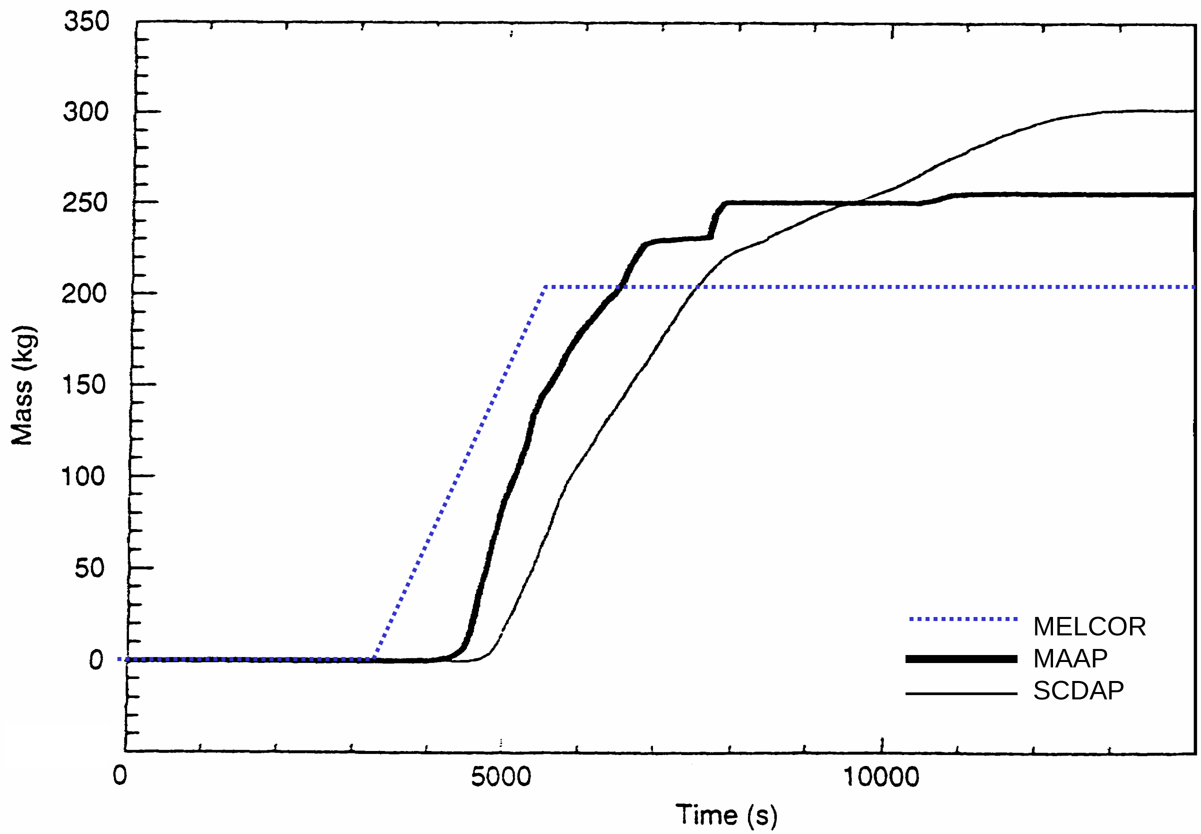
Code Models and Assumptions Impact 3BE AP600 Analysis Results (continued)



MELCOR shows delayed core heatup despite early core uncovery.

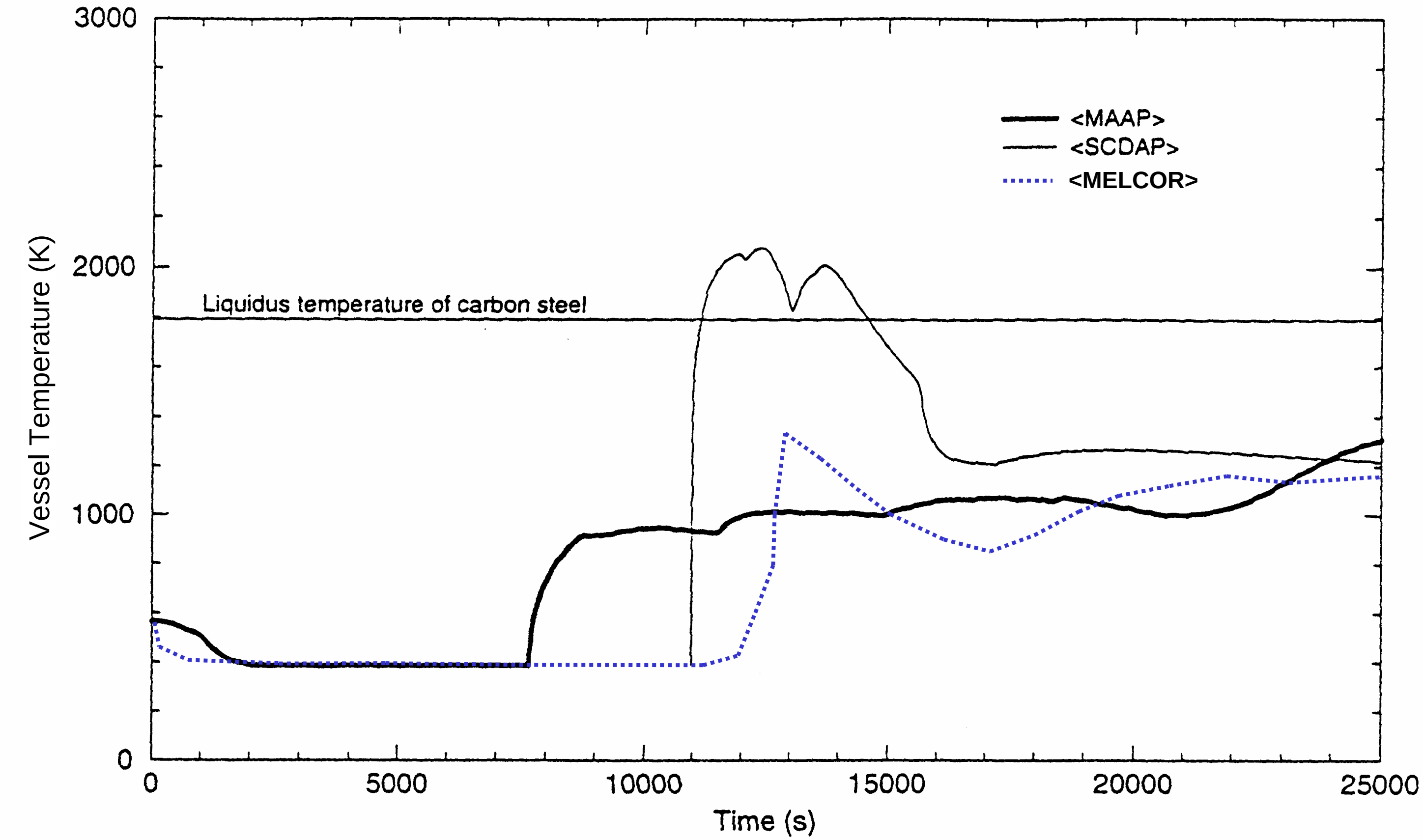
Case Study 2

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

Study Questions

- Name U.S.-developed codes used in severe accident analysis
- Draw a timeline of phenomena that must be considered in severe accident analysis and indicate what is modeled by each of these codes
- Discuss differences in code modeling approaches that may impact code predictions
- List some key questions to ask when reviewing an analysis

Code References

- M. L. Corradini, et al., *SCDAP/RELAP5 Independent Peer Review*, Los Alamos National Laboratory, LA-12381, January 1993.
- E. W. Coryell, et al., “The Development and Application of SCDAP-3D,” *Tenth International Conference on Nuclear Engineering*, April 14-18, 2002, Arlington, VA, USA
- L. J. Siefken, et al., *SCDAP/RELAP5/MOD3.3 Code Manuals*, Volumes 1-5, NUREG/CR-6150, INEL-96/0422, Rev. 2, January 2001, (http://www.nrc.gov/RES/SCDAP/nrc.html).
- L. J. Siefken, et al., *SCDAP/RELAP5-3D© Code Manuals*, Volumes 1-5, INEEL-EXT-01-00917, July 2001, (also see http:www.inel.gov/relap5 or www.inel.gov/relap/scdap).
- Papers presented at the RELAP5 International User Seminar, *Sun Valley, Idaho, September 5-7, 2001.*

Code References (continued)

- "MELCOR Project Status NRC Severe Accident Code Consolidation," *presented at USNRC CSARP-2001 Meeting,* May 7-9, 2001, Bethesda, Maryland.
- *MELCOR 1.8.5 User's Manual and MELCOR 1.8.5 Reference Manual,* NUREG/CR-6119, Volumes 1 - 3, (http: www. melcor.sandia.gov).
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- B. E. Boyack, et al., *MELCOR Peer Review*, Los Alamos National Laboratory, LA-12240, March 1992.
- R. O. Gaurtt, "MELCOR 1.8.5 Simulation of TMI-2 Phase 2 with an Enhanced 2-Dimensional In-Vessel Natural Circulating Model," *Tenth International Conference on Nuclear Engineering (ICONE 10),* April 14-18, Arlington, VA.

Code References (continued)

- *MAAP4 User's Manual,* Fauske and Associates, (http:www.maap4.com).
- *MAAP User's Group Meeting Presentations and Meeting Minutes*, October 7-9, 1998, FAI Technical Report TR-111240-V1, December 1998.
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- K. K. Murata, et al., *Code Manual for CONTAIN 2.0: A Computer Code for Nuclear Rector Containment Analysis*, NUREG/CR-6533, SAND97-1735, Sandia National Laboratories, December 1997.
- K. A. Smith*, Multi-Processor Based Simulation of Degraded Core and Containment Responses*, PhD Thesis, The Pennsylvania State University, December 1992.

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- F. Fichot, et al., *ICARE/CATHARE: A Computer Code for Analysis of Severe Accidents in LWRs. ICARE2 V3mod0: Description of Physical Models*, NT SEMAR 98/123, DRS/SEMAR/LECTA, 1998.
- H. Ujita, et al., "Development of Severe Accident Analysis Code SAMPSON in IMPACT Project," *J. Nucl. Sci. and Tech.,* **36** (11), 1999.
- K. Trambauer, "Coupling Methode of Thermal-hydraulic Models with Core Degradation Models in ATHLET- CD," *Sixth International Conference on Nuclear Engineering (ICONE6)*, San Diego, CA, May 1998.

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

Most Volatile Radionuclides Reside in Reactor Core

Location	Inventory, Ci	
	Noble Gases (Xe, Kr)	Iodine (I)
Core	4.0E+8	7.5E+8
- Gap between UO_2 fuel and Zr cladding	3.0E+7	1.4E+7
Spent fuel storage pool	1.0E+6	5.0E+5
Primary coolant[3]	1.0E+4	6.0E+2

[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	0.13
TMI-2 accident (March 1979)	2.50E+6	15
Chernobyl accident (April 1986)	1.90E+8	4.5E+7

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,

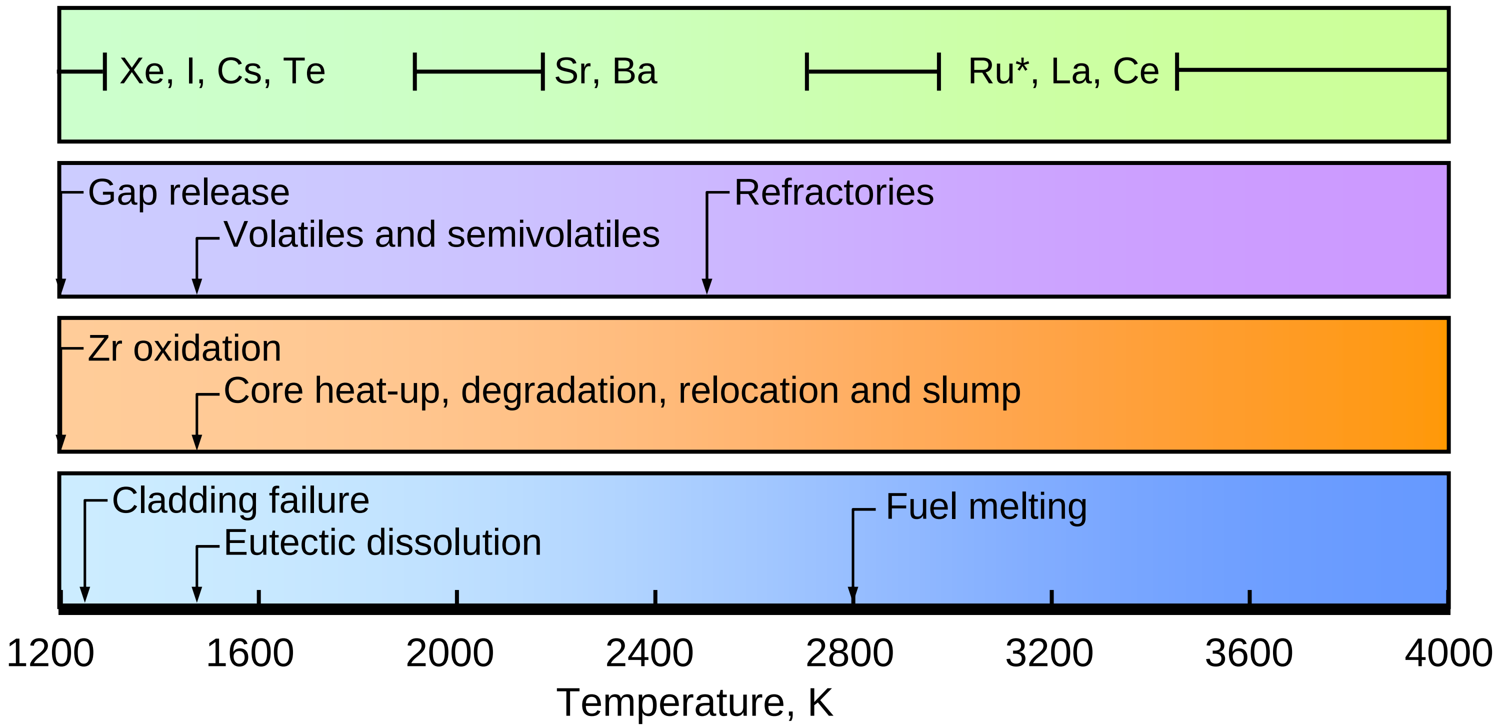
$Q_n(t)$ - activity source rate due to fuel release, MCCI, contribution entering from another volume, etc.

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
2	Halogens	Volatile	I-131, I-132, I-133, I-134, I-135	7.71E+08
3	Alkali Metals		Cs-134, Cs-136, Cs-137, Rb-86	2.18E+07
4	Tellurium		Sb-127, Sb-129, Te-127, Te-127m, Te-129, Te-129m, Te-131m, Te-132	2.13E+08
5	Strontium	Non-volatile	Sr-89, Sr-90, Sr-91, Sr-92	3.57E+08
6	Noble Metals		Co-58, Co-60, Mo-99, Rh-105, Ru-103, Ru-103, Ru-105, Tc-99m	5.94E+08
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
8	Corium (Cerium)		Ce-141, Ce-143, Ce-144, Np-239, Pu-238, Pu-239, Pu-240, Pu-241	2.15E+09
9	Barium		Ba-139, Ba-140,	3.38E+08

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

Group Release Tied to Fuel Temperature

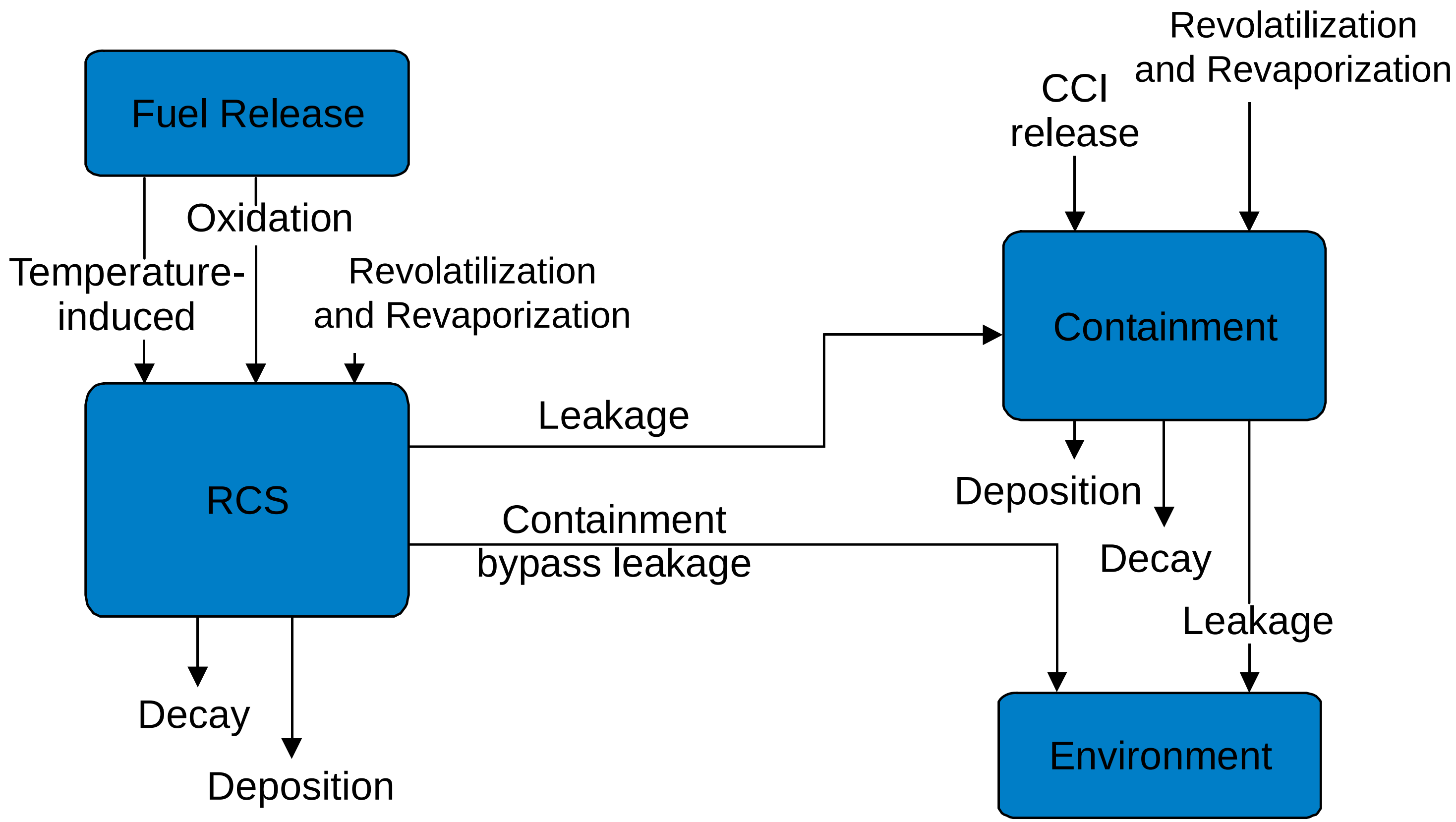


*In highly oxidizing environment, Ru is volatile

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	1.21E-01	
4	Tellurium	Te-132	3.21E+00	Sb-132
5	Strontium	Sr-90	1.06E+04	Rb-90
6	Noble Metals	Co-60	7.29E-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



Several Factors Affect Release and Transport

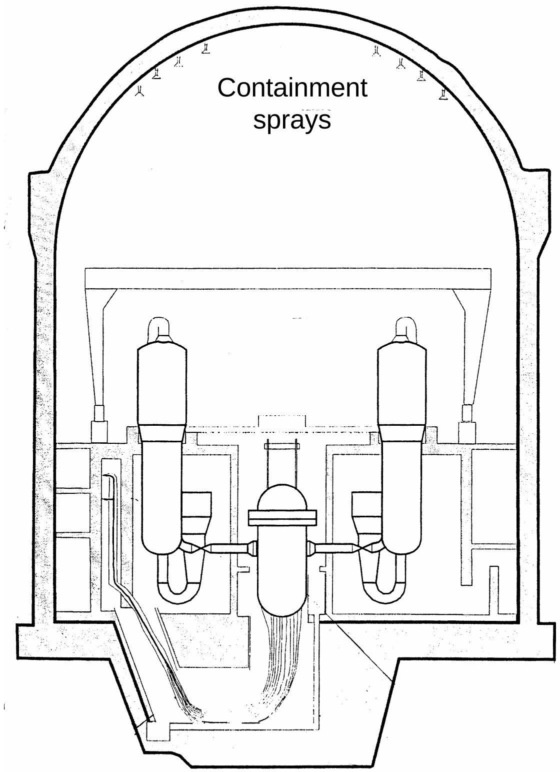
- Sequence dependent
  - Timing
  - Duration
  - Energy
  - Pressure
  - Chemical form
  - Physical form
  - Coolant chemistry
- Plant design dependent
  - Pathway (barriers, configuration, surface area, etc.)
  - Safety systems

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	2 to 4000
Overlying water layers	2 to 4000

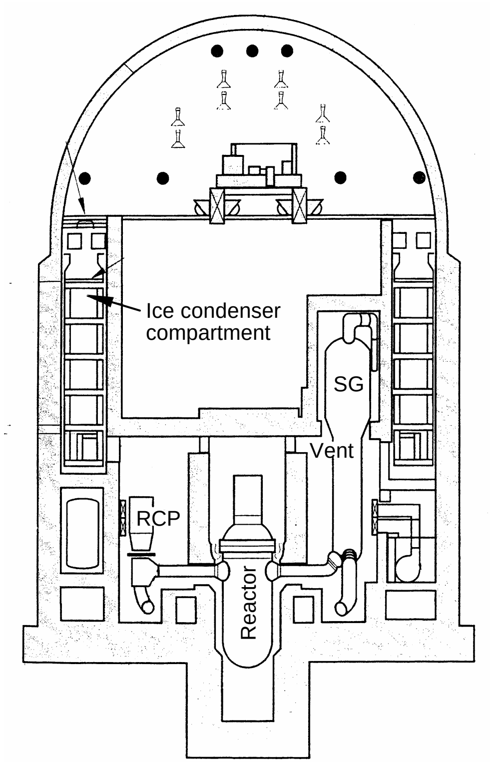
[1]Ratio of inlet to outlet concentrations.

Containment Sprays Rapidly Reduce Release



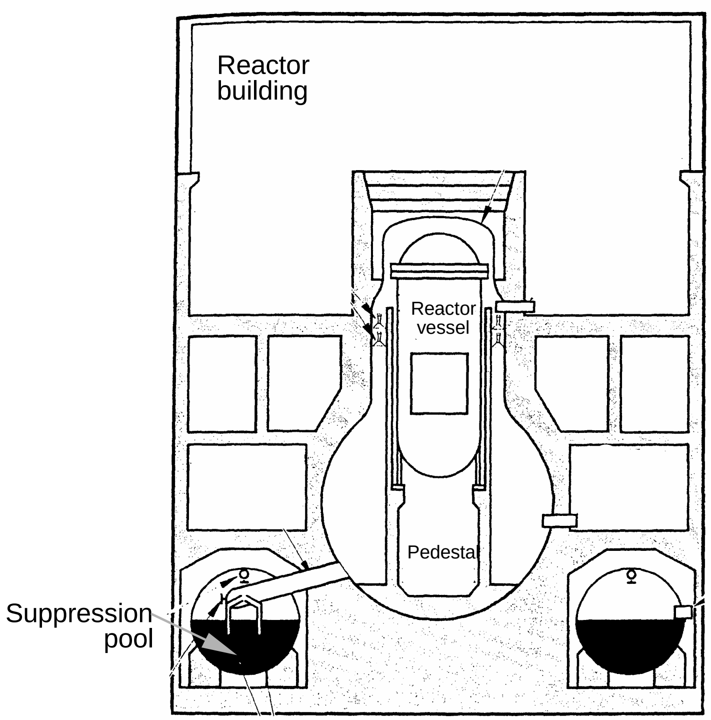
- Sprays reduce airborne concentration of aerosols and vapors in containment.
- Sprays may reduce airborne concentrations by order of magnitude in 15-20 minutes.

Ice Condensers Significantly Reduce Radioactive Release



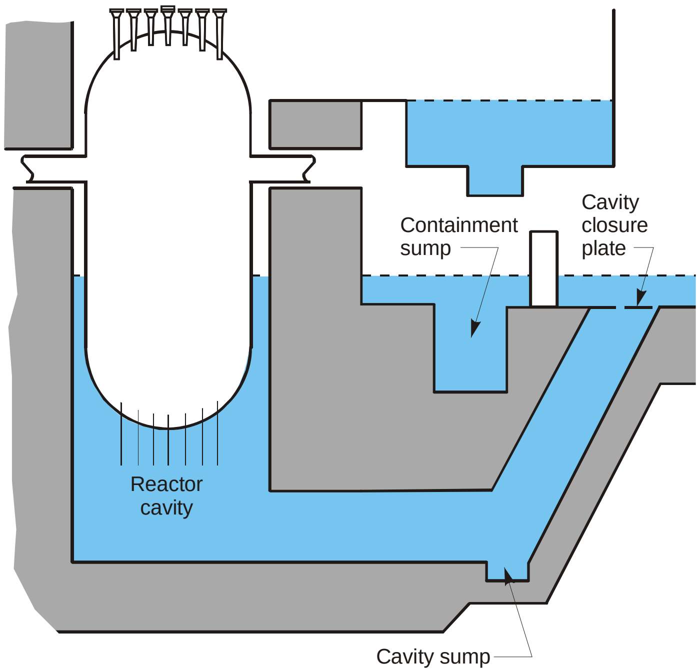
- Retain radioactive aerosols and vapors.
- Typical decontamination factors of 1 to 20 with a median of 3.
- Decontamination factor sensitive to steam and hydrogen fraction of gas that flows through them.

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.

Reactor Cavity Flooding Reduces Releases



Success of cavity flooding to mitigate MCCI releases requires:

- cavity geometry that promotes flooding
- sufficient water to flood cavity

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 or Alternate Source Term from NUREG-1465)

Source Terms Initially Based on TID-14844

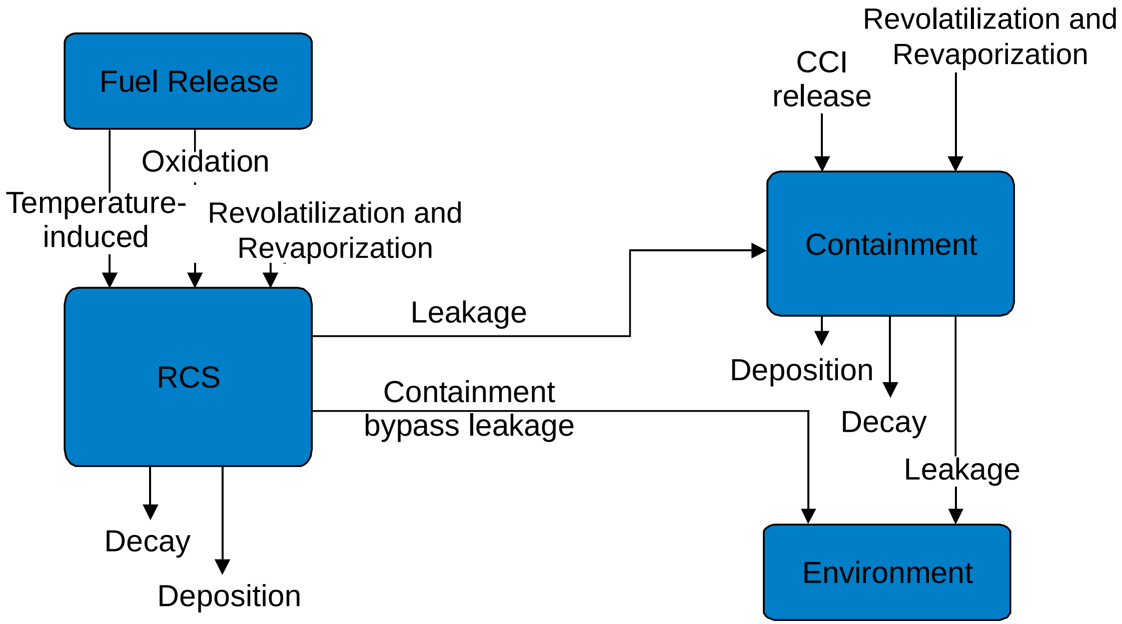
- Based on a postulated core melt accident and 1962 understanding of fission product behavior.
- As codified in Reg. Guides 1.3 and 1.4, assumed source term consists of an instantaneous release of:
  - 100% of core inventory of noble gases
  - 50% of core inventory of iodine
    - half assumed to subsequently 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

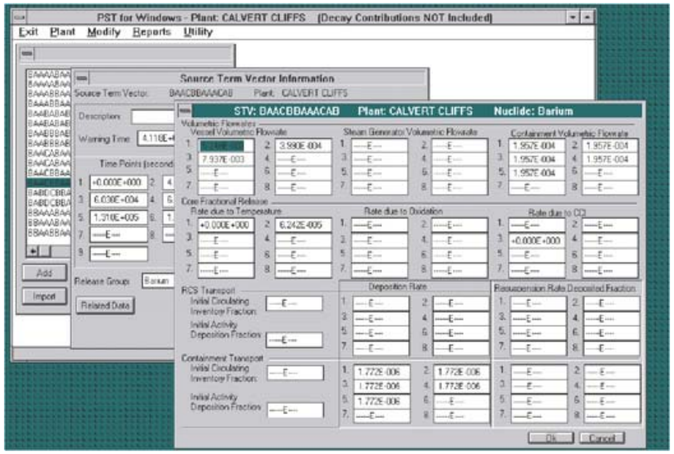
- Developed for five NUREG-1150 plants
- Doesn't consider knowledge gained from last decade of severe accident research.
- XSOR method decomposed source term into release fractions for various time periods and release barriers and quantified release fractions using expert opinion
  - Approach is time-consuming.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

ASP Program Source Terms Estimated with Parametric Source Term (PST)

- Models activity transport between volumes.
- User specifies connections between volumes
- General transport equation applied to each volume
- Considers physics and time-dependence
- Code results and expert review used to quantify lower level, PST input parameters.



ASP Source Terms Estimated with PST Code (continued)



- PST development emphasizes computational efficiency and user-friendliness
  - Easy to update source terms
  - Easy to apply to additional plants
  - Developed to provide point estimates and uncertainty distributions
  - Developed to read ASP Level 2 CET output and produce output that interfaces with NUREG-1150 codes.
  - Windows environment simplifies user-interface.
  - Being adapted for space reactor analyses

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 In-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).

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.

References for Additional Reading

- J.J. DiNunno, et.al, “Calculation of Distance Factors for Power and Test Reactors,” *Technical Information Document (TID)-14844*, U.S. Atomic Energy Commission, 1962.
- U.S. Nuclear Regulatory Commission, “Assumptions Used for Evaluating the Potential Consequences of a Loss of Coolant Accident for Boiling Water Reactors,” *Regulatory Guide 1.3, Rev. 2,* 1974.
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- U.S. Nuclear Regulatory Commission, *Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants*, NUREG-1150, December 1990.
- J. L. Rempe, M. Cebull, and B. G. Gilbert, *PST User’s Guide*, INEL/96-0308, September 1996.
- *Procedures for Conducting Probabilistic Safety Assessments of Nuclear Power Plants (Level 2)*, IAEA Safety Series No. 50-P-8, 1995.
- L. Soffer, et al., *Accident Source Terms for Light-Water Nuclear Power Plants, Final Report*, NUREG-1465, February 1995.
- H. P. Nourbakhsh, *Estimates of Radionuclide Release Characteristics into Containment under Severe Accidents*, NUREG-CR-5747, Nov. 1993.

References for Additional Reading (continued)

- Papers presented in the session, “Radiological Analysis Utilizing Revised Accident Source Terms,” at the American Nuclear Society Meeting, Washington, D.C., *Trans. Am. Nucl. Soc.,* 75, 1996, p. 308-315.
- Papers presented in the session, “Implementation of the New Source Term at Operating Plants,” at the American Nuclear Society Meeting, Albuquerque, NM, *Trans. Am. Nucl. Soc*. **77**, 1997 p. 304-308.
- L. J. Callan, EDO, “Results of the Revised (NUREG-1465) Source Term Rebaselining for Operating Reactors,” SECY-98-154, June 30, 1998.
- J. H. Schaperow and J. Y. Lee, “Implementation of the Revised Source Term at U.S. Operating Reactors,” presented at the U.S. WRSIM,October 27, 1999.

Accident Progression Analysis (P-300)

9. PRA Integration and Quantification

May 10-12, 2005 - Rockville, MD

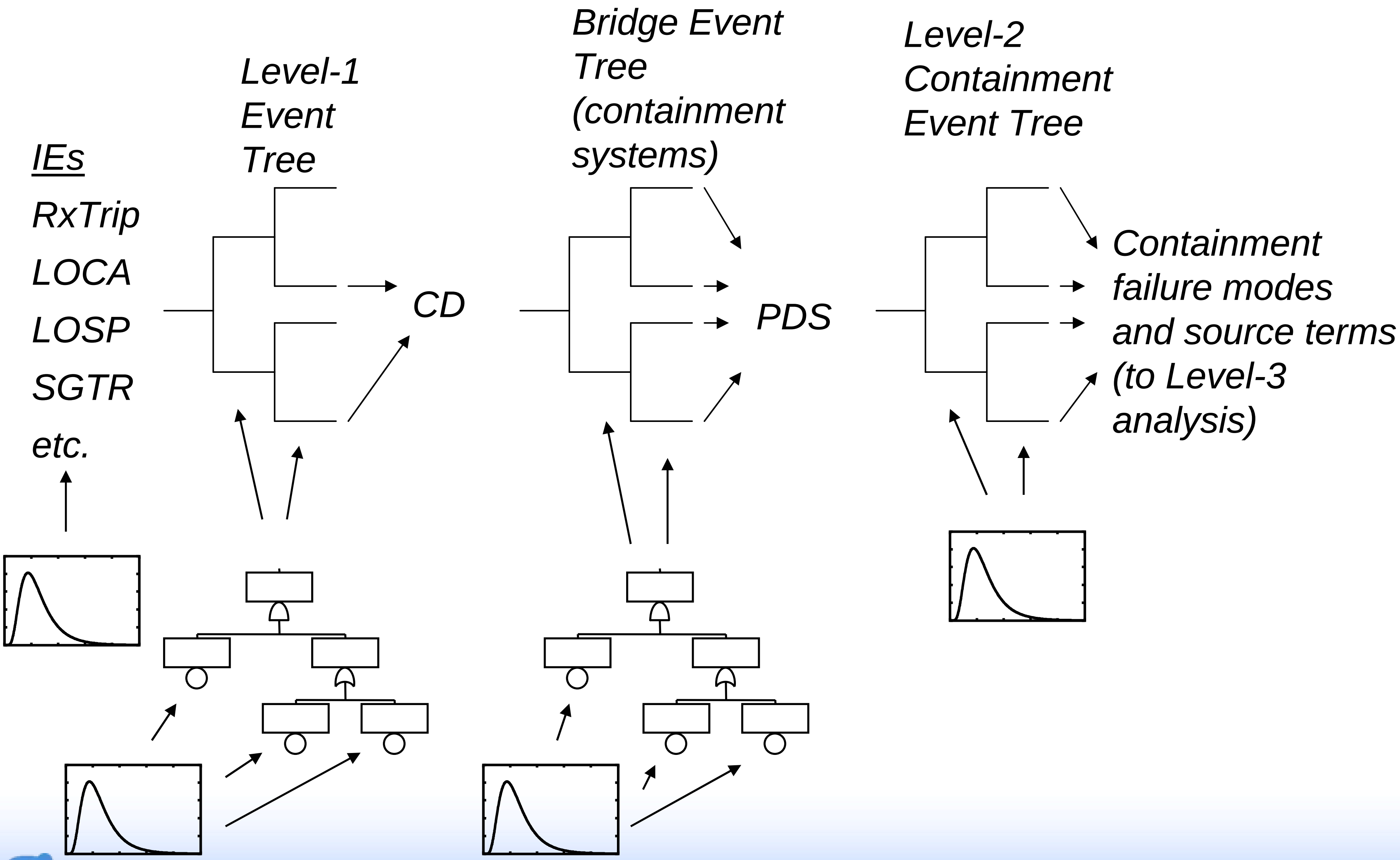
Session Objectives

- 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 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 and Level-2 Analyses Approach

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

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

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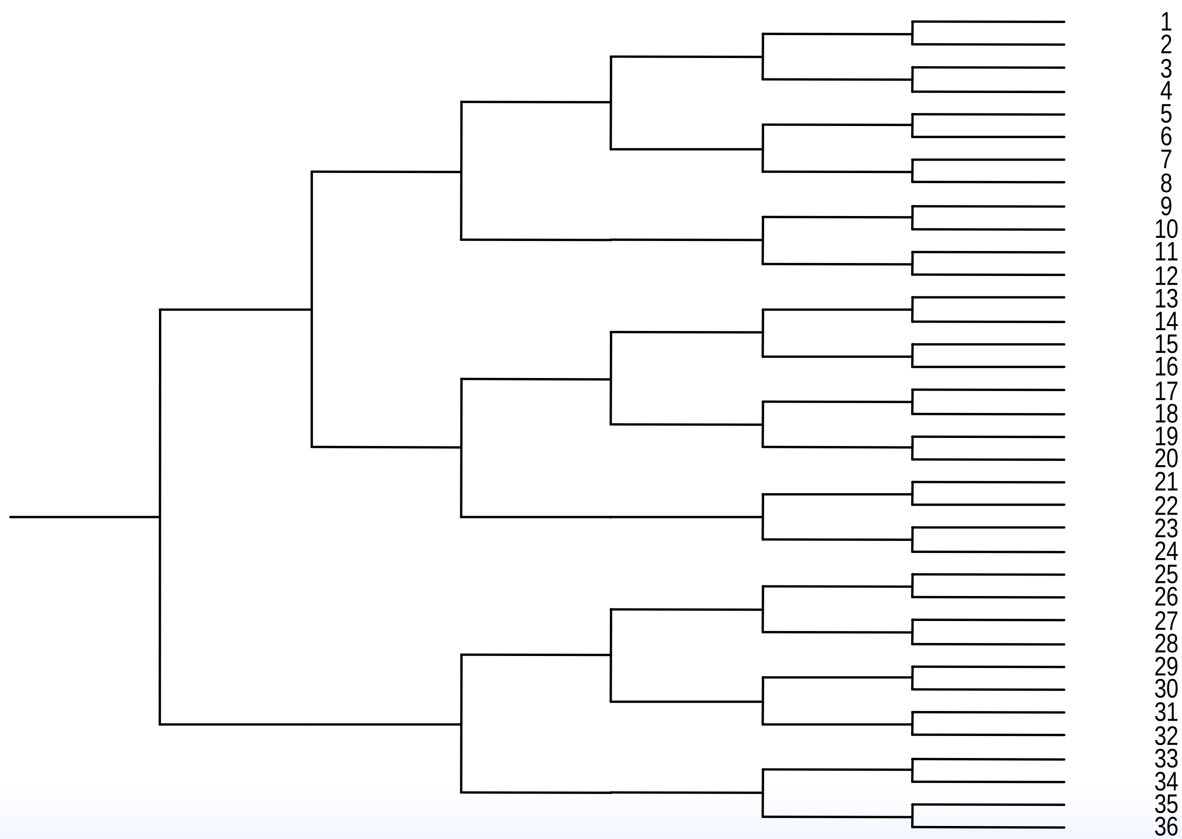
Example Bridge Event Tree

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



BET-01 - Bridge Event Tree

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Each CD Cutset Unique

- Each cutset represents a unique set of events (e.g., component failures, human actions) that is expected to lead to CD (i.e., UTAF)
- Individual cutsets 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 Cutset Assigned to PDS

- To accommodate different impacts on Level-2 analysis, each CD cutset explicitly mapped into a PDS (sometimes referred to as binning)
- Two approaches to binning Level-1 cutsets 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)**
    - **Cutset #1: Small LOCA with ECCS pump fails**
    - **Cutset #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*	*--*	*U*	*A*
CS#2	*S*	*U*	*U*	*U*	*A*	*--*	*U*	*U*

- *Frequency from cutsets #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 any and all relevant information needed to assess containment performance

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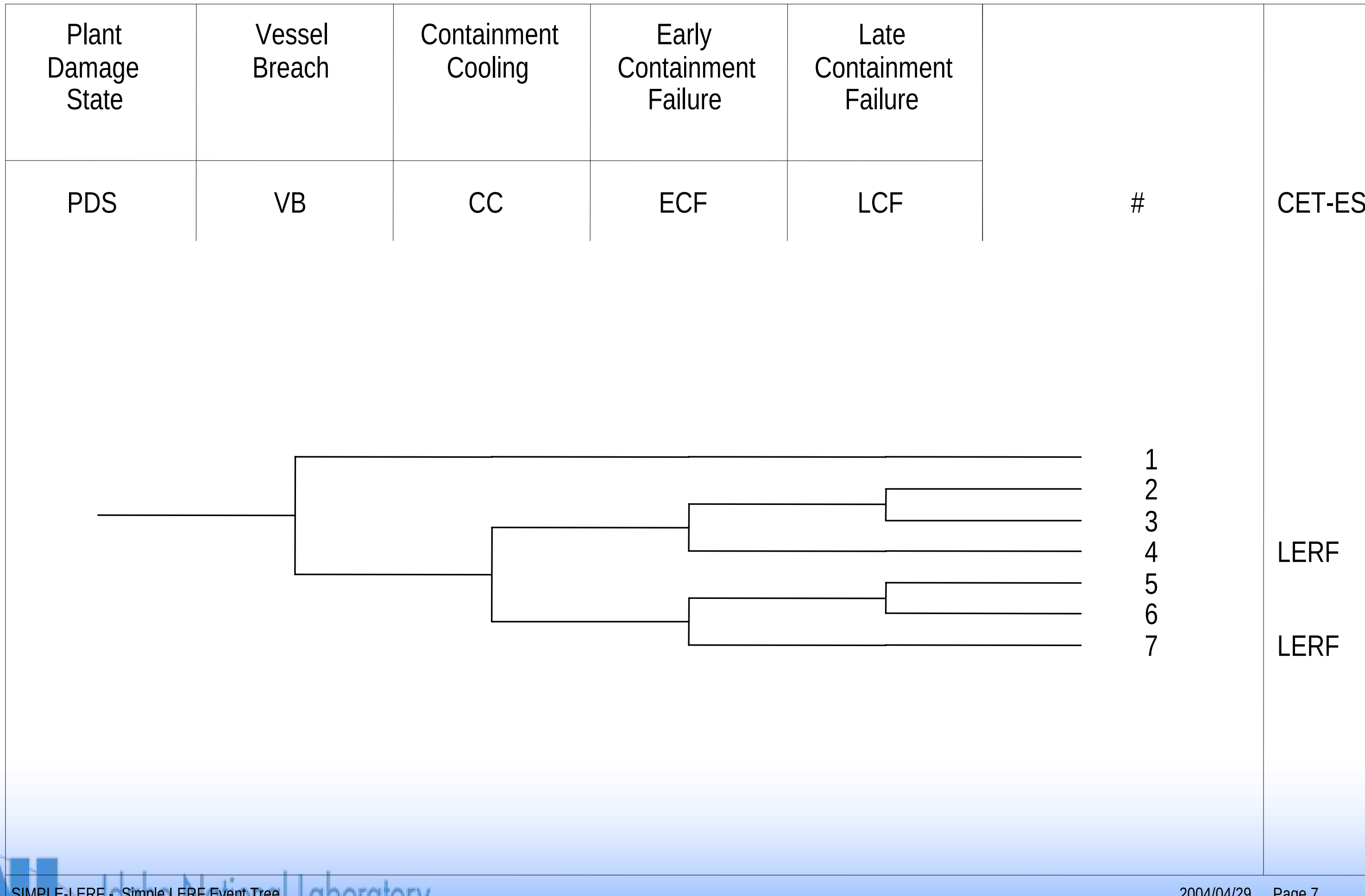
Accident Progression Quantified Different Ways

- 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 in form of “If-Then” statements
- Sometimes branch probability is a weighted average of different accident sequences
  - One way to account 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

- **What is 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 is assumed
  - Can also include variability in boundary conditions
- Epistemic - State of knowledge, subjective or confidence uncertainty
  - A produced by a lack of data
    - Classically represented by a statistical confidence interval
    - 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**

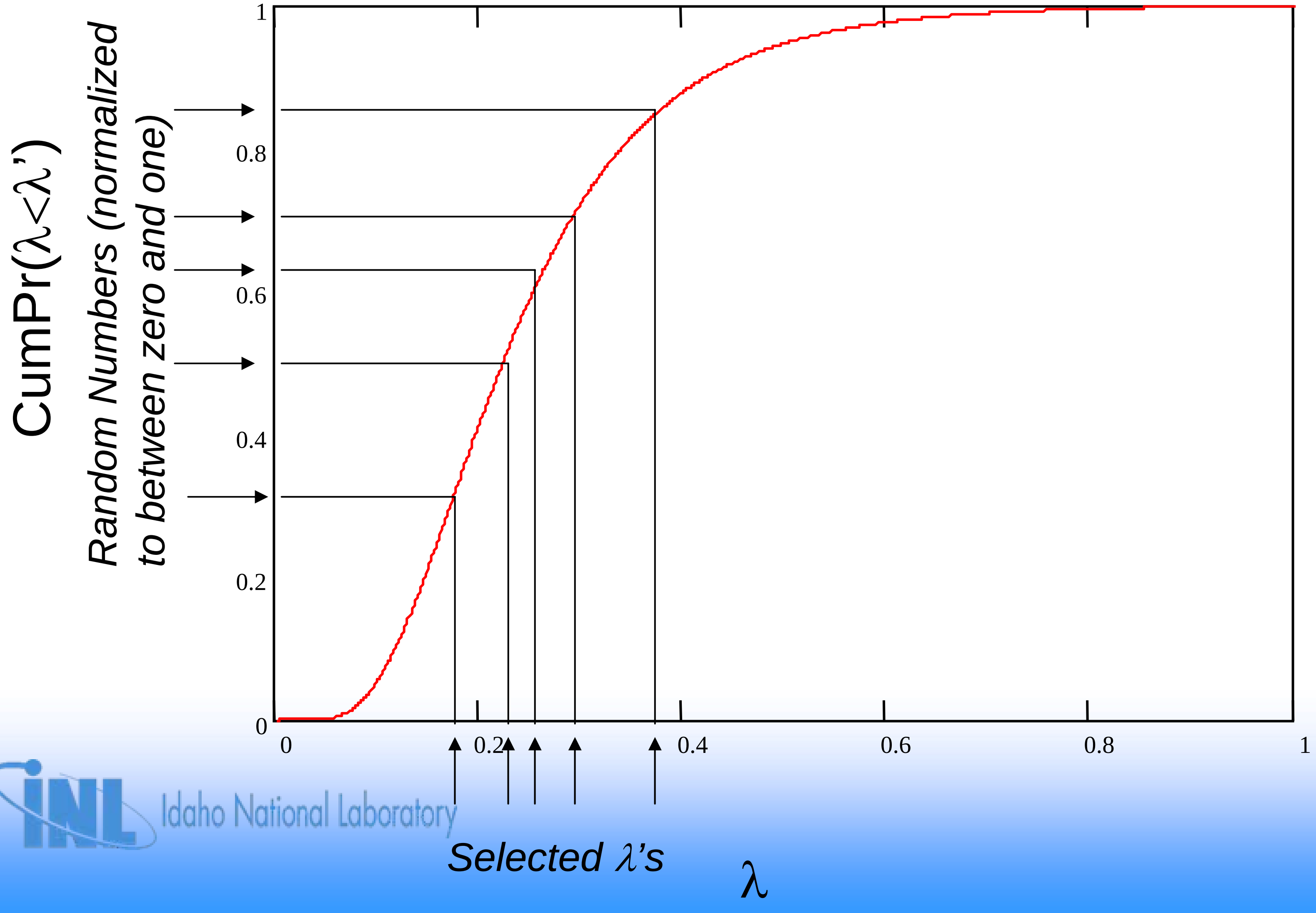
Uncertainty Needs Propagated Through Entire PRA

- **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**

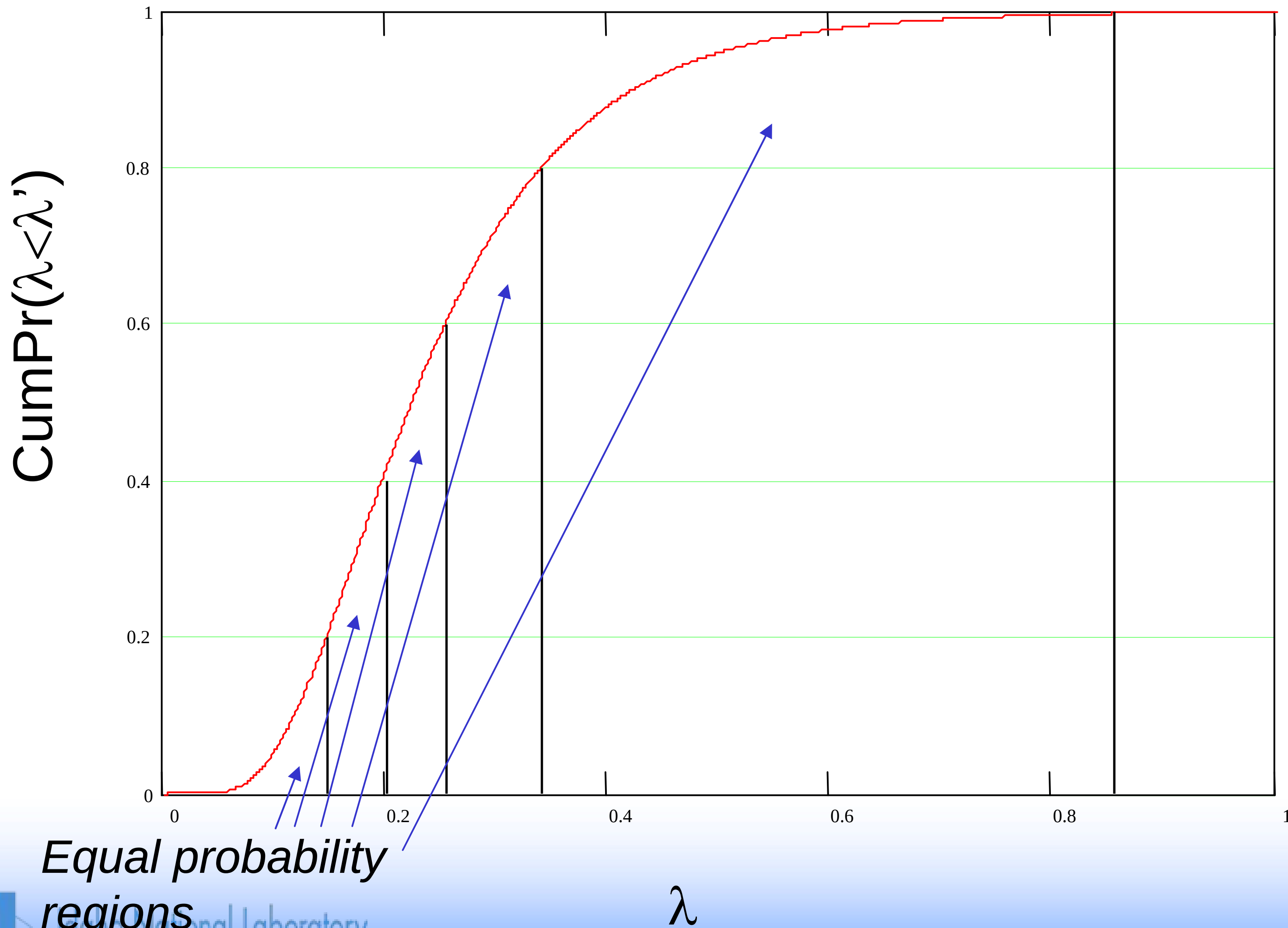
Simulation Techniques Used to Quantify Models

- 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

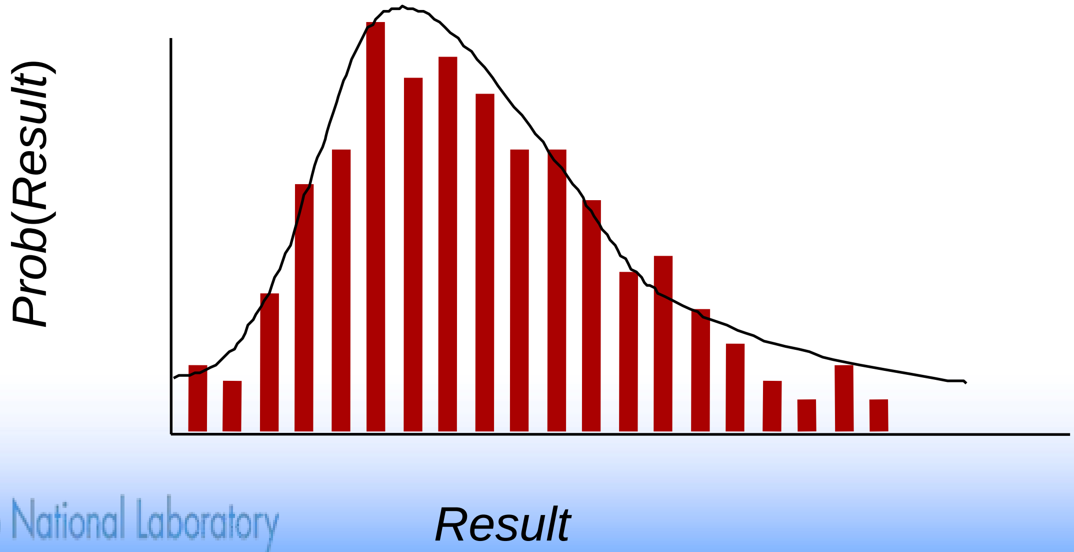


Latin Hypercube Sampling (one λ selected from each equal-probability area)



Propagation of Uncertainties

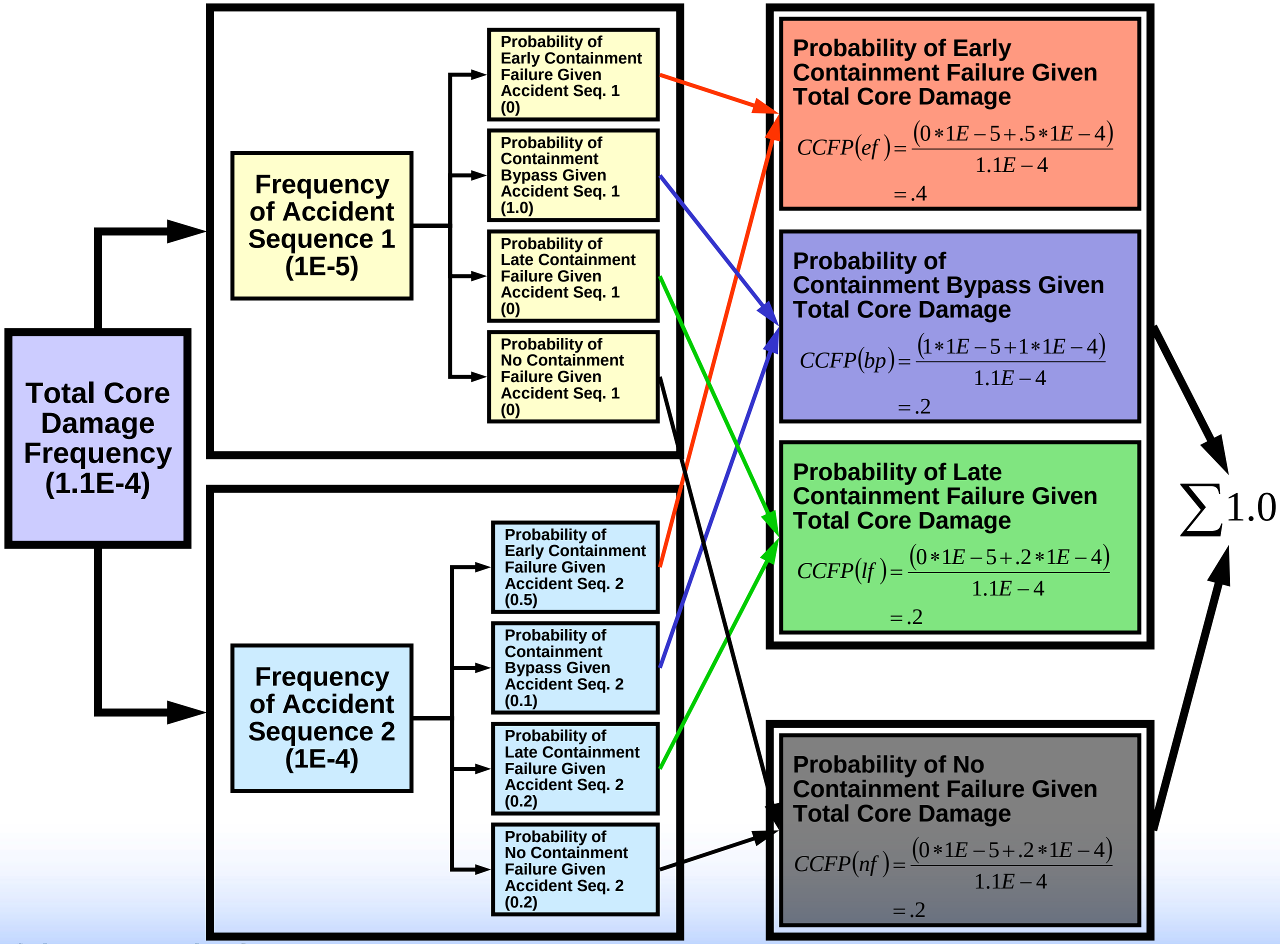
- **Simulation Process (either Monte Carlo or Latin Hypercube)**
  - **Generates frequency histogram for Result = f(X, Y) by sampling from distributions for X and Y re-calculating result for each of simulation samples**



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

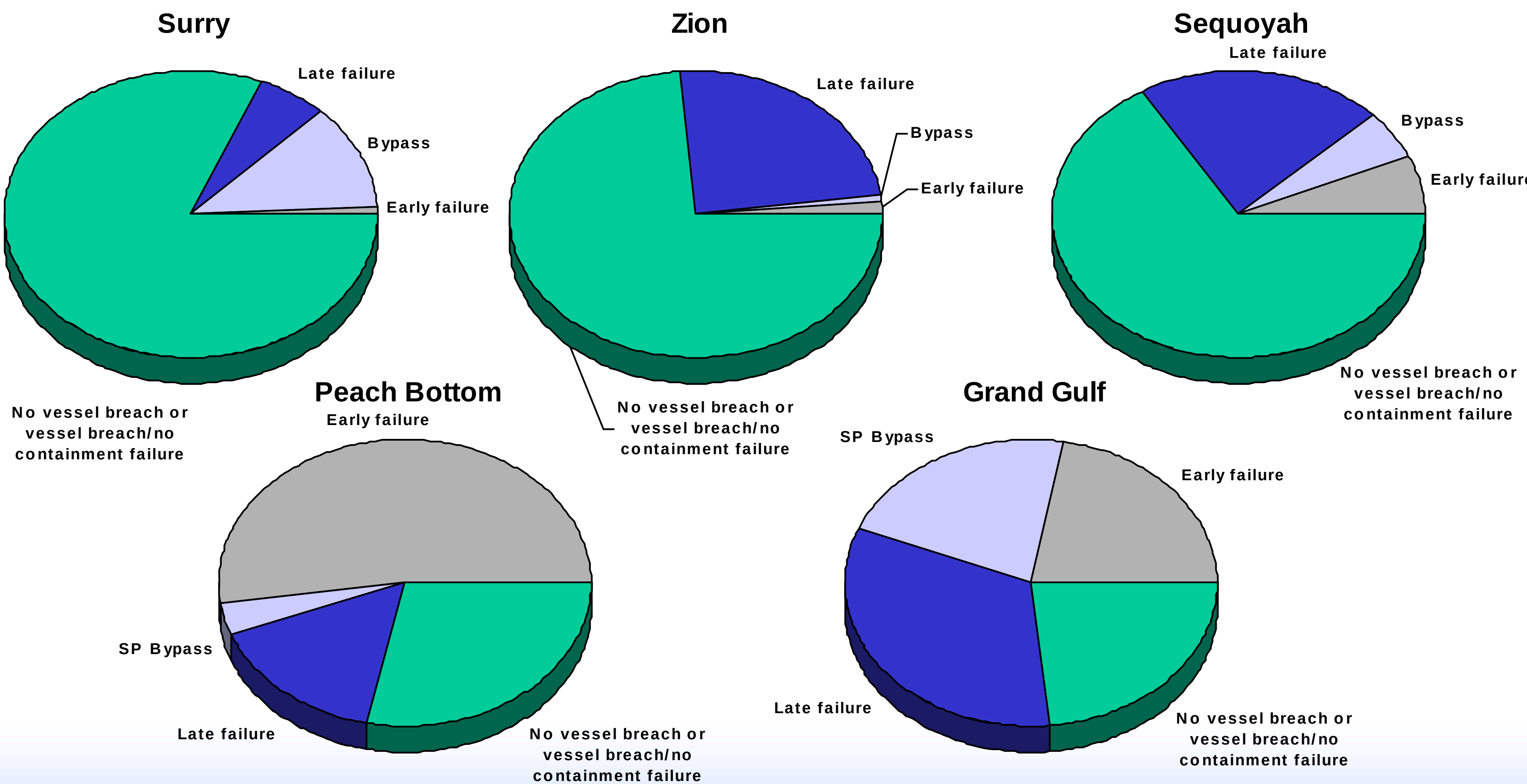
PLANT DAMAGE STATE
(Mean Core Damage Frequency)

ACCIDENT PROGRESSION BIN	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

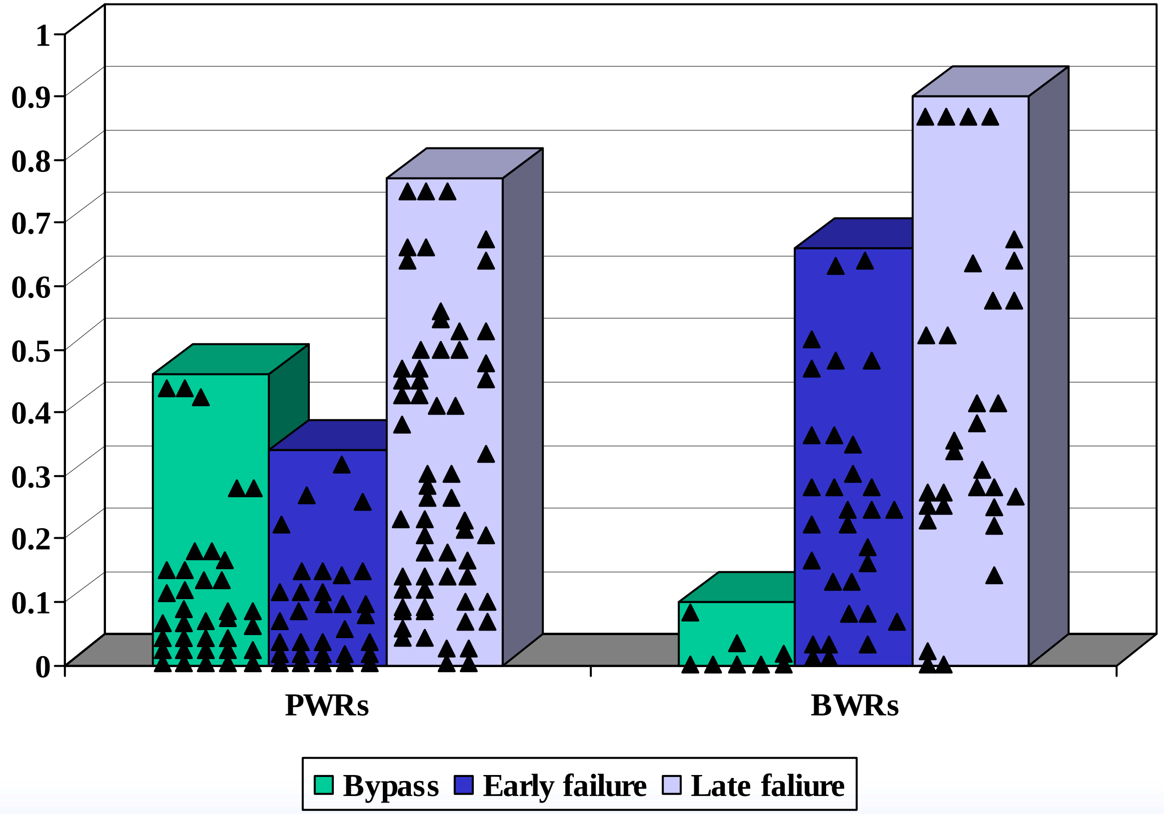
Sequoyah

NUREG-1150 Results Indicate BWR Early Containment Failures More Likely



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

More Recently Completed 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)

May 10-12, 2005 - Rockville, MD

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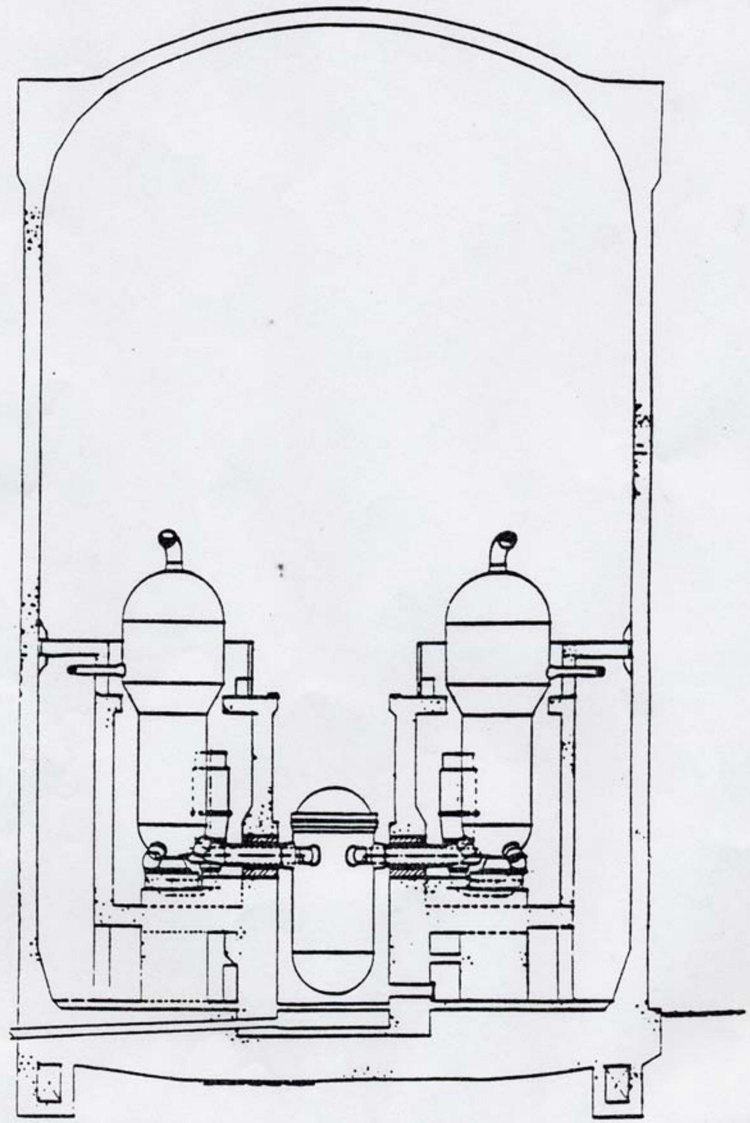
Example: Palisades IPE

- 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 283^{o}F
- Complete Level-2 PRA submitted as Individual Plant Examination (IPE) to NRC on January 29, 1993.

PALISADES NUCLEAR PLANT
GENERAL CONTAINMENT ARRANGEMENT



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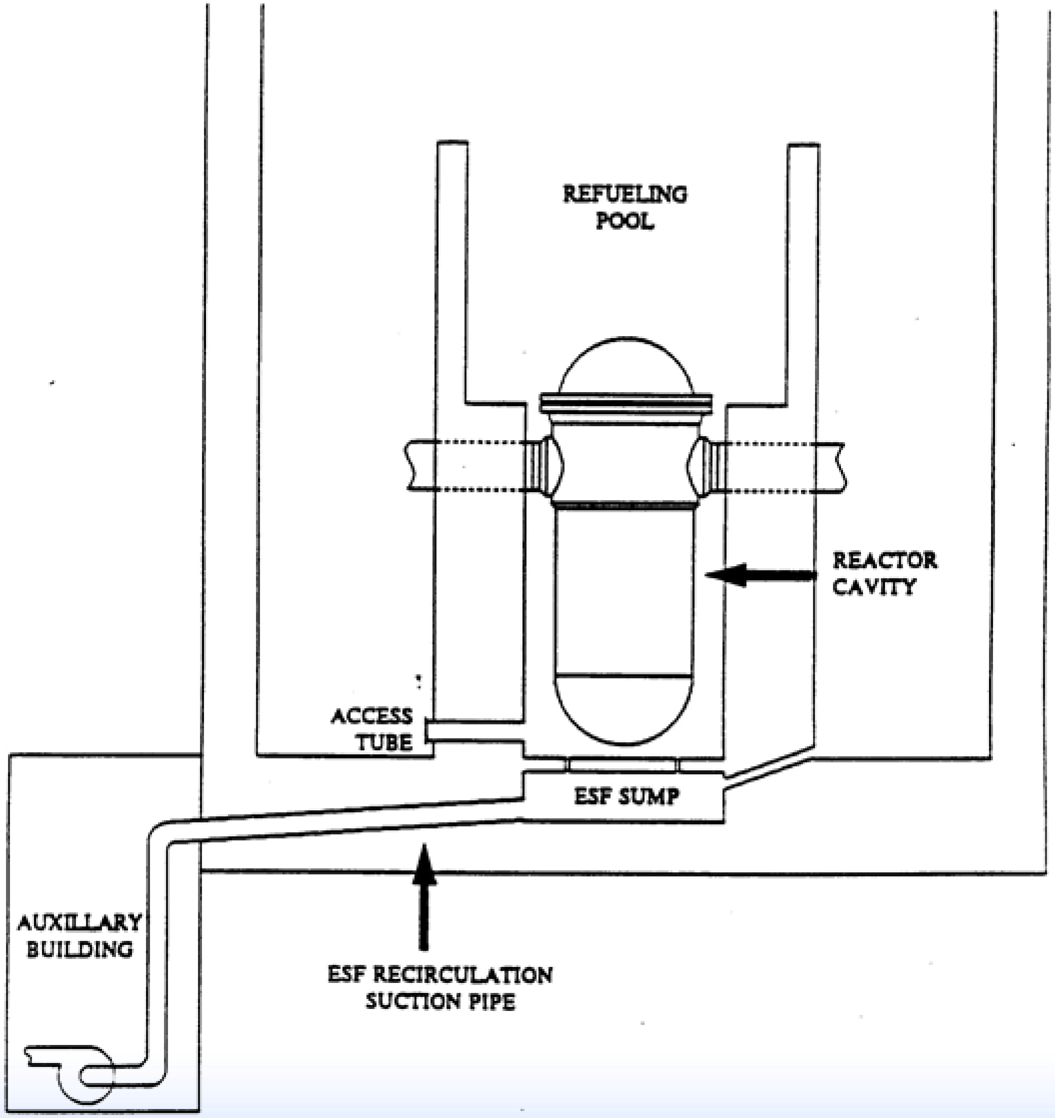

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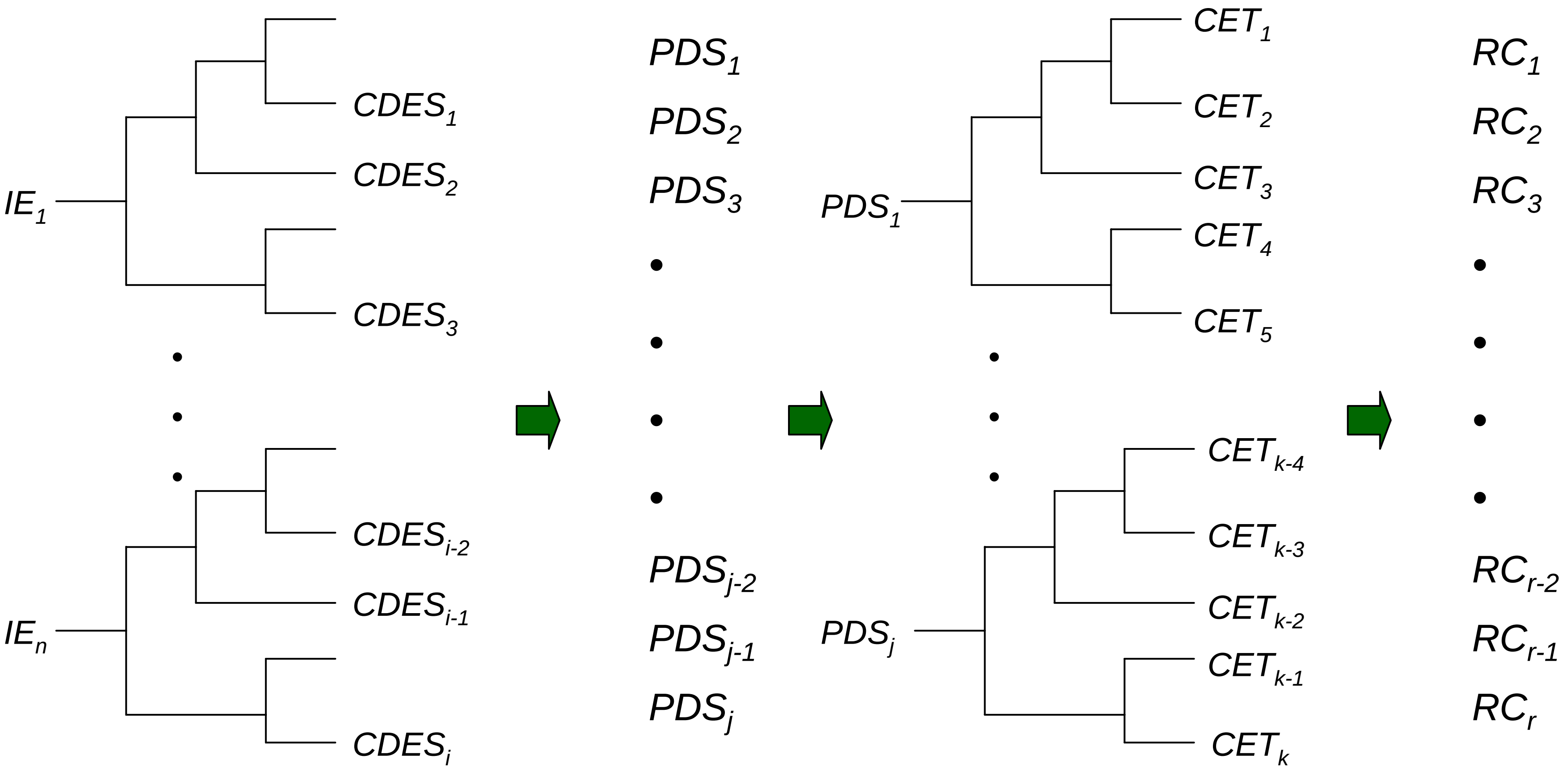
PALISADES REACTOR CAVITY-ENGINEERED SAFEGUARDS SUMP ARRANGEMENT

Palisades Level-2 PRA Analysis Process



$$\sum_{m=1}^{i} \Pr(CDES_m) = CDF \qquad \sum_{n=1}^{j} \Pr(PDS_n) = CDF \qquad \sum_{p=1}^{k} \Pr(CET_p) = CDF \qquad \sum_{q=1}^{r} \Pr(RC_q) = CDF$$

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 (Initiator)

ID	Description
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*

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Palisades IPE PDS Char. #5 (Containment Systems)

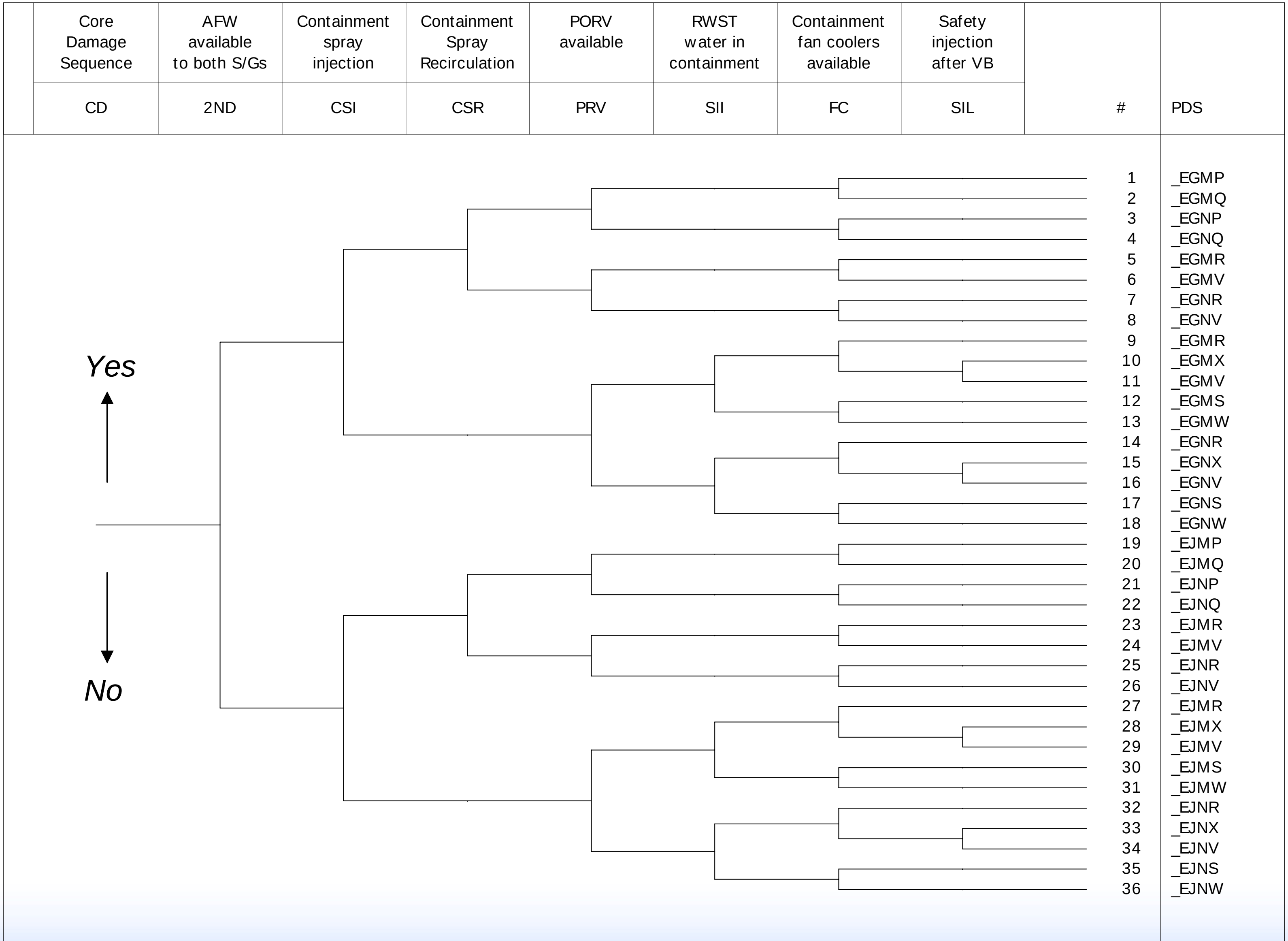
ID	*Description*
P	*Containment sprays and air coolers available*
Q	*Cont. sprays avail. and cont. air coolers NOT avail.*
R	*Only cont. air coolers avail., RWST contents in cont.*
S	*Only cont. air coolers avail., RWST contents NOT in cont.*
V	*No cont. systems avail., RWST contents in cont.*
W	*No cont. systems avail., RWST contents NOT in cont.*
X	*Late (post VB) operation of only HPSI/LPSI*

Palisades 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 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



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

CET Top Event Quantification Focus on Probability of Containment Failure

- **Need to know how strong the containment structure is**
- **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
  - $PrF(p) = 1- \prod_{i=1,n}[1-PrF_i(p)]$
  - 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

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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

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)

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

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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

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

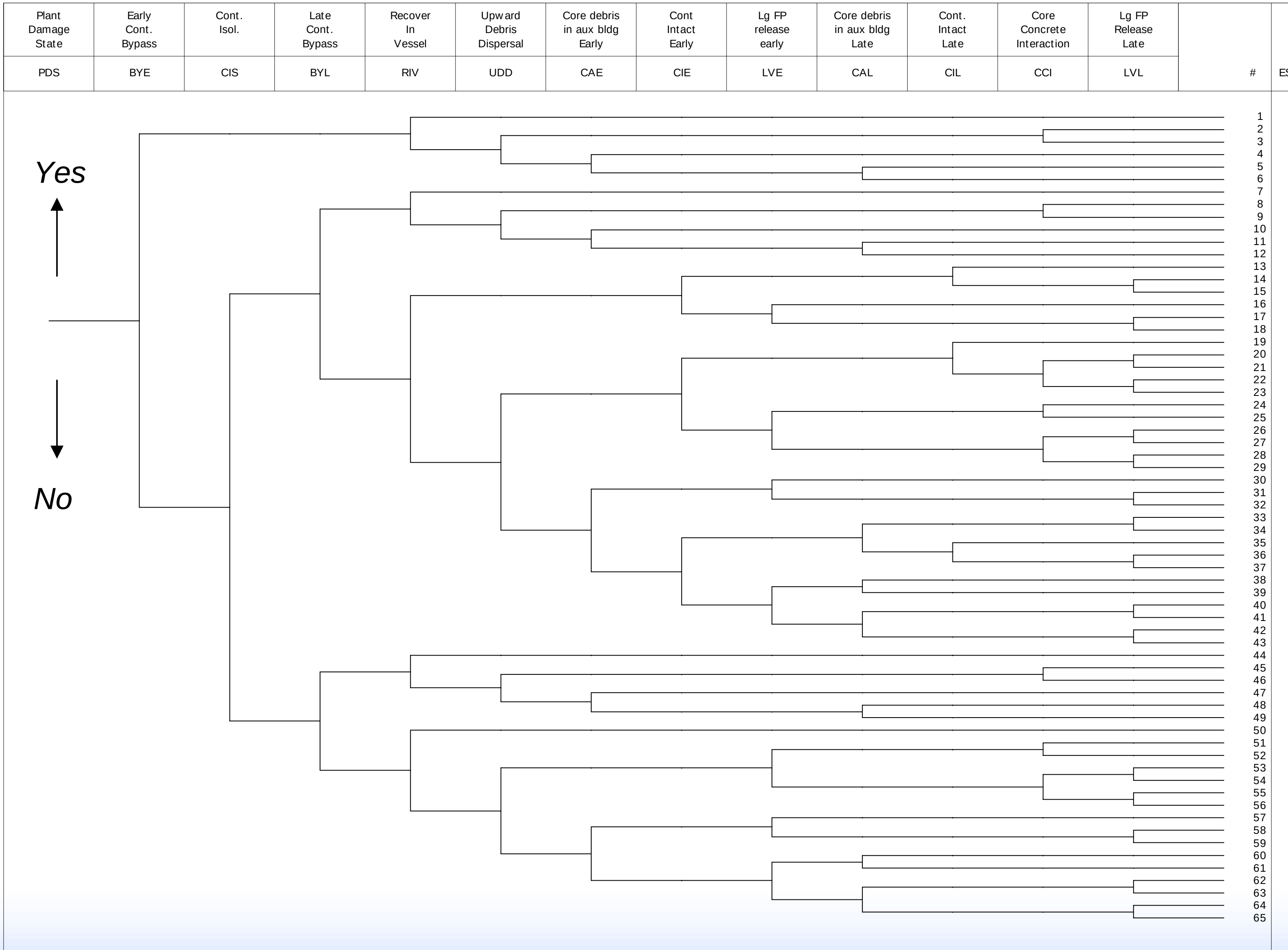
LVE - Large volatile fission product release early

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

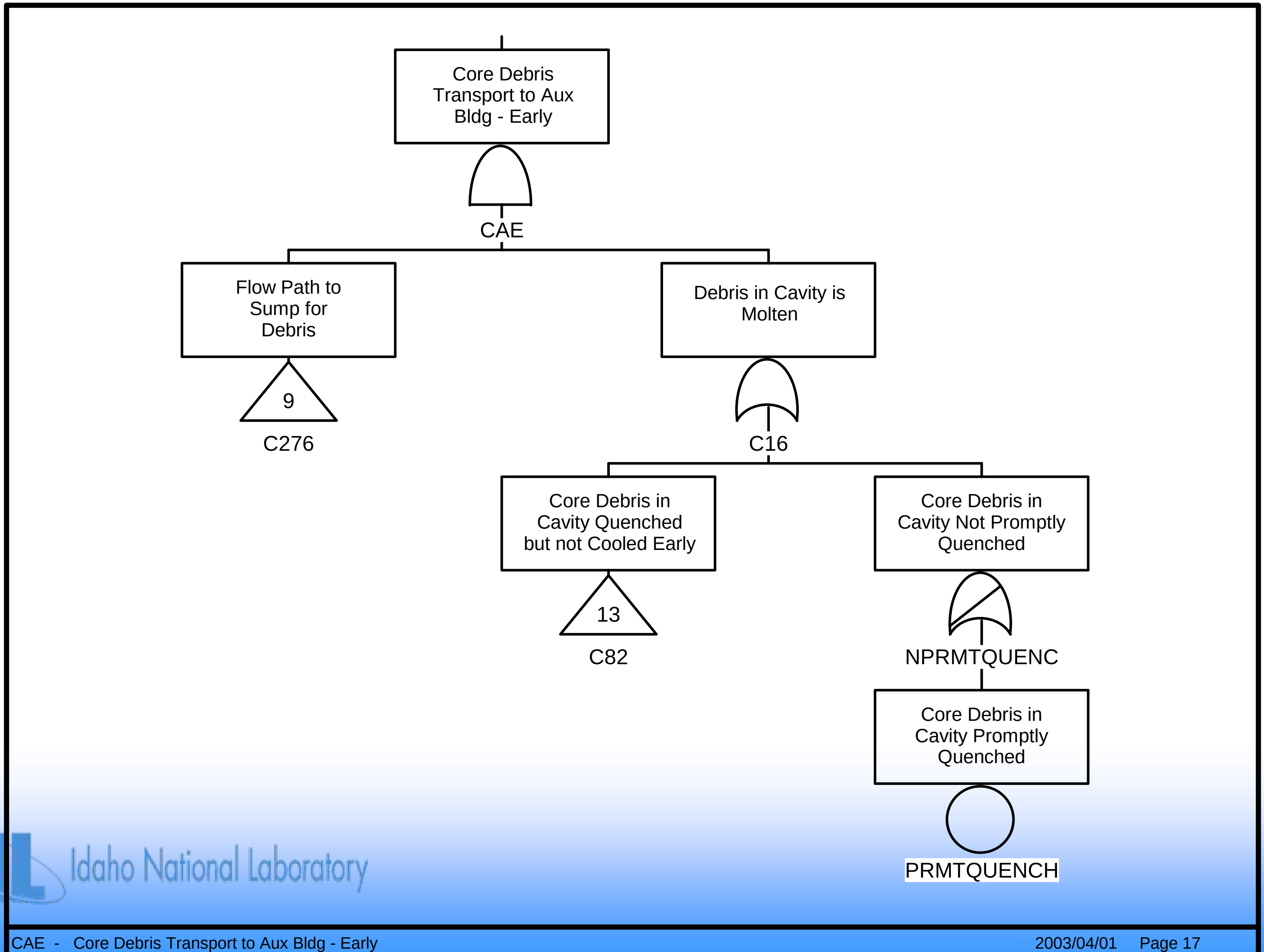


PALISADES-CET - Palisades IPE (Jan. 1993) Containment Event Tree 2001/06/14 Page 3

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 Fault Tree



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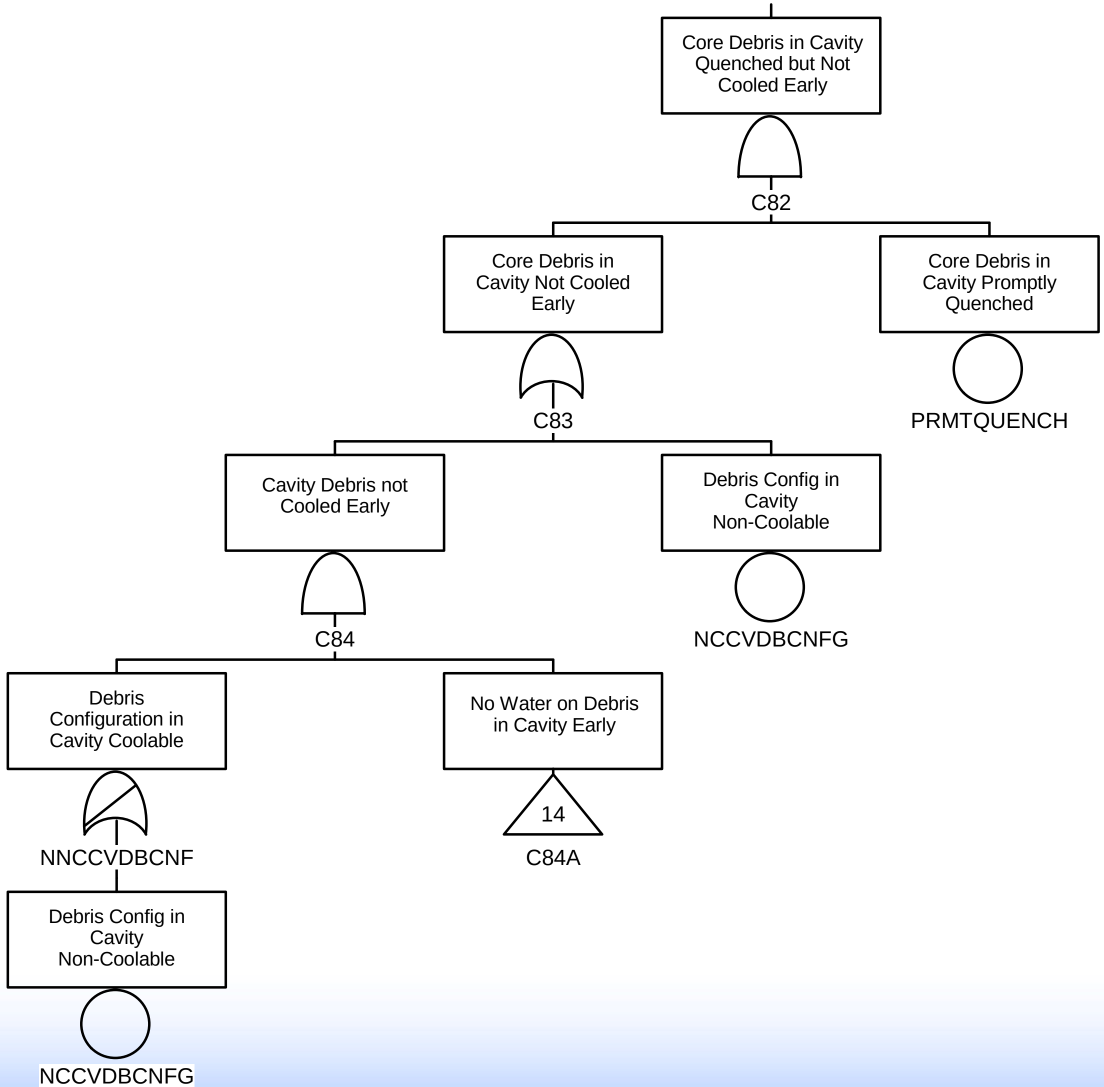
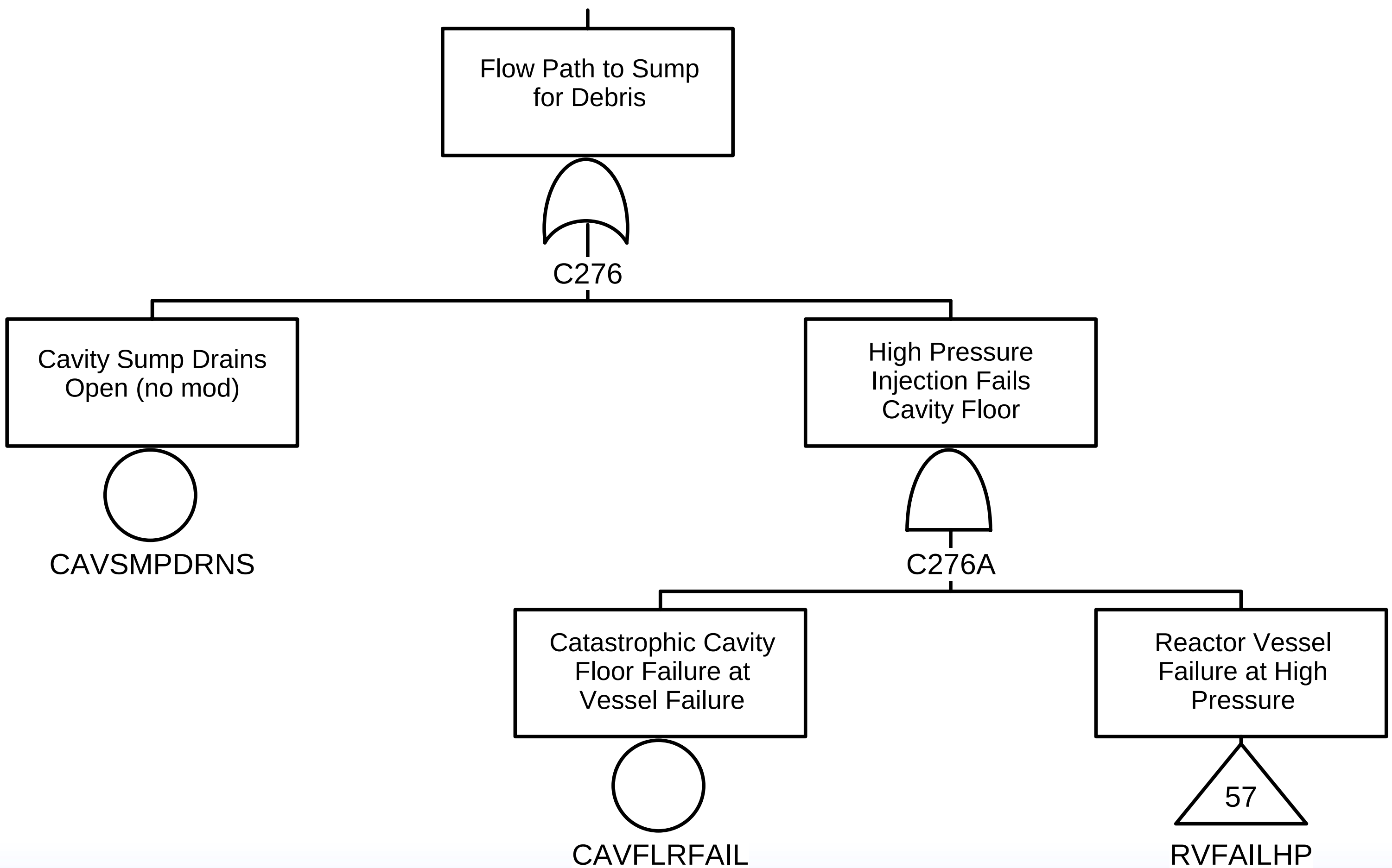

Core Debris in Cavity Quenched but Not Cooled Early
C82
Core Debris in Cavity Not Cooled Early
Core Debris in Cavity Promptly Quenched
C83
PRMTQUENCH
Cavity Debris not Cooled Early
Debris Config in Cavity Non-Coolable
C84
NCCVDBCNFG
Debris Configuration in Cavity Coolable
No Water on Debris in Cavity Early
NNCCVDBCNF
14
C84A
Debris Config in Cavity Non-Coolable
NCCVDBCNFG



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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)

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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		

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	2.36E-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	2.05E-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	1.34E-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%)
2	8.47E-7	DEJS(99%)
3	8.53E-7	DEJS(99%)
4	4.18E-7	CEJW(55%) DEJS(44%)
10*	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**	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

* *Containment bypass; ** Containment isolation*

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%)
2	8.47E-7	DEJS(99%)
3	8.53E-7	DEJS(99%)
4	4.18E-7	CEJW(55%) DEJS(44%)
10*	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**	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

*Containment bypass; ** Containment isolation*

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)

Review

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?