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Accident Progression Analysis (P-300)

Course Presented by

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April 19 - 21, 2016
NRC Professional Development Center
Rockville, MD
Course Objective

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

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

• WASH-1400, Reactor Safety Study: An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants, October 1975
  – Original Level-2 analysis.

  – 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-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.

  – Comprehensive analysis of all referenced plants, includes PWR containment design details extracted from IPEs, including fragility curves.

  – Detailed analysis of issue, benefited from a public workshop and an extensive peer review process.
Annotated Bibliography (cont.)

  - EPRI developed method for predicting containment failure mechanisms and leakage locations.
  - 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.

Acronyms

ACRS  Advisory Committee on Reactor Safeguards
ADS   Automatic Depressurization System
AFW   Auxiliary Feedwater System
AM    Accident Management
AP-600 Westinghouse Advanced PWR (600 MWe)
APB   Accident Progression Bin
APET  Accident Progression Event Tree
ASP   Accident Sequence Precursor
AST   Accident Source Term
ATWS  Anticipated Transient Without SCRAM
B&W  Babcock & Wilcox
BWR   Boiling Water Reactor
CCFP  Conditional (on core damage) Containment Failure Probability
CCI   Core Concrete Interaction
CD    Core Damage
CDF   Core Damage Frequency
CE    Combustion Engineering
CET   Containment Event Tree
CFD   Computational Fluid Dynamics
CFF   Containment Failure Frequency
CHF   Critical Heat Flux

CHR  Containment Heat Removal
CRD  Control Rod Drive
CS   Cut Set
CSR  Containment Spray Recirculation
CSS  Containment Spray System
DCH  Direct Containment Heating
DW   Drywell (BWR)
ECCS Emergency Core Cooling System
ECI  Emergency Coolant Injection
ECR  Emergency Coolant Recirculation
ERVVC  External Reactor Vessel Cooling
FAI  Fauske Associates, Incorporated
FCI  Fuel-Coolant Interaction
FEM  Finite Element Method
FIBS  Final Bounding State
H2   Hydrogen
HPIS  High Pressure Injection Systems
HPME  High Pressure Melt Ejection
IPE  Individual Plant Examination
ISLOCA Interfacing System Loss of Coolant Accident
IVR  In-Vessel Retention
<table>
<thead>
<tr>
<th>Acronym</th>
<th>Description</th>
</tr>
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<tbody>
<tr>
<td>JAERI</td>
<td>Japan Atomic Energy Research Institute</td>
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<tr>
<td>KAERI</td>
<td>Korea Atomic Energy Research Institute</td>
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<tr>
<td>LERF</td>
<td>Large Early Release Frequency</td>
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<tr>
<td>LHF</td>
<td>Lower Head Failure</td>
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<tr>
<td>LOCA</td>
<td>Loss of Coolant Accident</td>
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<tr>
<td>LPIS</td>
<td>Low Pressure Injection System</td>
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<td>LWR</td>
<td>Light Water Reactor</td>
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<tr>
<td>MAAP</td>
<td>Modular Accident Analysis Program</td>
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<td>MACCS</td>
<td>MELCOR Accident Consequence Code System</td>
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<td>MCCI</td>
<td>Molten Core Concrete Interaction</td>
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<td>MSSV</td>
<td>Main Steam Safety Valve</td>
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<tr>
<td>OECD</td>
<td>Organization for Economic Cooperation and Development</td>
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<tr>
<td>OTSG</td>
<td>Once-Through Steam Generator</td>
</tr>
<tr>
<td>PCS</td>
<td>Power Conversion System</td>
</tr>
<tr>
<td>PDF</td>
<td>Probability Density Function</td>
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<tr>
<td>PDS</td>
<td>Plant Damage State</td>
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<tr>
<td>PORV</td>
<td>Power (or Pilot) Operated Relief Valves</td>
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<td>PST</td>
<td>Parametric Source Term</td>
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<tr>
<td>PWR</td>
<td>Pressurized Water Reactor</td>
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<tr>
<td>QHO</td>
<td>Quantitative Health Objective</td>
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<td>RCP</td>
<td>Reactor Coolant Pump</td>
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<tr>
<td>RCS</td>
<td>Reactor Coolant system</td>
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<td>ROAAM</td>
<td>Risk Oriented Accident Analysis Methodology</td>
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<td>RPS</td>
<td>Reactor Protection System</td>
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<td>RPV</td>
<td>Reactor Pressure Vessel</td>
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<td>Severe Accident Management Guidelines</td>
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<td>Steam Generator</td>
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<td>SGTR</td>
<td>Steam Generator Tube Rupture</td>
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<tr>
<td>SNL</td>
<td>Sandia National Laboratory</td>
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<tr>
<td>SRV</td>
<td>Safety Relief Valve</td>
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<tr>
<td>TAF</td>
<td>Top of Active Fuel (in reactor core)</td>
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<tr>
<td>TEDE</td>
<td>Total Effective Dose Equivalent</td>
</tr>
<tr>
<td>TMI-2</td>
<td>Three Mile Island Unit 2</td>
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<tr>
<td>UCSB</td>
<td>University of Santa Barbara</td>
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<tr>
<td>UHI</td>
<td>Upper Head Injection</td>
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<tr>
<td>VB</td>
<td>(Reactor Pressure) Vessel Breach</td>
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<tr>
<td>WW</td>
<td>Wetwell (BWR)</td>
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<tr>
<td>Pressure</td>
<td>psi</td>
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1. Risk-Informed Regulatory Background and Review of PRA Basic Concepts
Session Objectives

• To understand the motivation for Level-2 PRA
  – NRC regulatory philosophy
    • PRA Policy Statement
    • Reactor Safety Goal Policy Statement
    • Regulatory Guide 1.174

• To understand some of the basic PRA concepts
  – Risk
    – Large Early Release Frequency (LERF)
PRA Policy Statement

The Nuclear Regulatory Commission's (NRC's) policy for implementing risk-informed regulation was expressed in the 1995 policy statement on the use of probabilistic risk assessment (PRA) methods in nuclear regulatory activities. The policy statement states:

- The use of PRA technology should be increased in all regulatory matters to the extent supported by the state-of-the-art in PRA methods and data and in a manner that complements the NRC's deterministic approach and supports the NRC's traditional defense-in-depth philosophy.

- PRA and associated analyses (e.g., sensitivity studies, uncertainty analyses, and importance measures) should be used in regulatory matters, where practical within the bounds of the state-of-the-art, to reduce unnecessary conservatism associated with current regulatory requirements, regulatory guides, license commitments, and staff practices. Where appropriate, PRA should be used to support the proposal of additional regulatory requirements in accordance with 10 CFR 50.109 (Backfit Rule). Appropriate procedures for including PRA in the process for changing regulatory requirements should be developed and followed. It is, of course, understood that the intent of this policy is that existing rules and regulations shall be complied with unless these rules and regulations are revised.
PRA Policy Statement (Continued)

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

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

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

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

• An Approach for Using Probabilistic Risk Assessment In Risk-Informed Decisions On Plant-Specific Changes to the Licensing Basis

• Defines the five principles of risk-informed integrated decision-making
  – #4. Proposed increases in CDF or risk are small and consistent with Commission’s Safety Goal Policy Statement
    • Use of CDF and LERF as bases for PRA acceptance guidelines is an acceptable approach to addressing Principle 4.
Large Early Release Frequency (LERF)

• In the context of Reg Guide 1.174, LERF is used as a surrogate for the early fatality QHO

• Defined as: the frequency of those accidents leading to significant, unmitigated releases from containment in a time frame prior to effective evacuation of the close-in population such that there is a potential for early health effects
  – No quantitative definition (w.r.t. timing or magnitude)
  – By definition, late releases would result in no early fatalities
RG-1.174 Acceptance Guidelines for Core Damage Frequency

- **Region I**
  - No Changes Allowed

- **Region II**
  - Small Changes
  - Track Cumulative Impacts

- **Region III**
  - Very Small Changes
  - More Flexibility with Respect to Baseline CDF
  - Track Cumulative Impacts
RG-1.174 Acceptance Guidelines for Large Early Release Frequency

- **Region I**
  - No Changes Allowed

- **Region II**
  - Small Changes
  - Track Cumulative Impacts

- **Region III**
  - Very Small Changes
  - More Flexibility with Respect to Baseline LERF
  - Track Cumulative Impacts
Common PRA Terms

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

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

• Vector Definition
  – Risk Triplet: \( \text{Risk} = \{S_i, F_i, C_i\} \),
    • where: \( S_i \) = Accident sequence \( i \),
      \( F_i \) = Frequency of sequence \( i \),
      \( C_i \) = Consequence of sequence \( i \).

• Scalar Definition
  – Risk = \( \sum_{i=1}^{n} F_i \times C_i \)
  – Sometimes called aggregated risk
Sequence Frequency Quantified by Combining Challenges and Failures

- Initiating events (IE) challenge plant systems to respond 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)
  \[ \text{CDF} = \lambda \varphi \]
  where: \( \lambda \) = Initiating event frequency
  \( \varphi \) = Failure probability of safety barriers (systems)
PRAs Characterized as Level-1, Level-2 or Level-3

• Level 1: Core damage risk
  – Quantifies the frequency of accidents that result in core damage

• Level 2: Radioactive material release risk
  – Core damage frequency combined with the conditional probability the containment structure fails to prevent the release

• Level 3: Health consequence risk
  – Combines radioactive material release frequency with the health consequences associated with each release
Full Scope PRA Process/Structure

**LEVEL 1**
- **SYSTEMS ANALYSIS**
  - Plant System Models, and Equipment and Operator Failure Data

**LEVEL 2**
- **ACCIDENT PROGRESSION ANALYSIS**
  - Models Progression of Severe Accident (APET or CET)
- **SOURCE TERM ANALYSIS**
  - Parametric Information About Fission Product Transport and Removal

**LEVEL 3**
- **CONSEQUENCE ANALYSIS**
  - Demographic and Meteorological Data, and Radiological Consequences (Health Effects and Costs)
- **RISK INTEGRATION**
  - Combines core damage accident sequence frequency with the consequences associated with that particular accident sequence

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

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

**Risk (frequency of public consequences) - e.g., fatalities/year, cost-of-accidents/year**
Uncertainty is a Vital and Integral Component in Any PRA

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

- Why is Level-2 PRA important?
- What are some basic PRA concepts?
Accident Progression Analysis (P-300)

2. Overview of PRA
Session Objectives

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

IEs
RxTrip
LOCA
LOSP
SGTR
etc.

Level-1 Event Tree

Bridge Event Tree (containment systems)

Level-2 Containment Event Tree

Level-3 Consequence Analysis

Consequence Code Calculations (MACCS)

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.

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

IEs
RxTrip
LOCA
LOSP
SGTR
etc.

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

ok

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

Total \( CDF = \sum_{i=1}^{n} CDF_i \)
Purpose of Level-2 PRA Analysis

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

• Characterize challenges to containment resulting from various core damage sequences
  – e.g., core degradation produces H2, which can burn
• Estimate strength of containment
• Identify probable containment failure mode (e.g., failure due to hydrogen detonation or steam explosion, melt through, leakage)
• Describe radioactive source term released into the environment
  – Including the energy associated with containment failure and radioactive material release
Level-2 PRA (Containment Event Tree)

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

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

Core Damage (Plant Damage State)

CD

no CF

no CF

CF\(_1\)

CF\(_n\)

Total CF = \(\sum_{i=1}^{n} CF_i\)

Note that this example focuses on Containment Failure (CF), some Level-2 analyses estimate releases (i.e. source terms) or Large Early Release Frequency (LERF)
Purpose of Level-3 PRA Analysis

• Estimate the public consequences (mostly health) of a severe accident
  – Person-rem (individual and population), early fatalities, latent cancers, financial cost, etc.

• Site-specific calculation
  – Considers local demographics, weather, emergency plan
Level 3 PRA Analysis Approach

• Source term information from Level-2 analysis result used as input to Level-3 consequence analysis

• Source term information includes:
  – radionuclide composition, energy associated with release, timing and duration of release, etc.

• Source term transport and offsite consequences (both health and economic) modeled using consequence code
  – MACCS2 (1998)
  – MACCS (1987 - NUREG-1150)
  – CRAC2 (1982)
  – CRAC (1975 - WASH-1400)
Level-3 Analysis Combines Source Term Frequencies and Consequences

Source Terms (for each STG) Demographics Weather data

MACCS Code

Public Consequences for each Source Term Group

Risk Integration

Frequency of each Source Term Group (from Level-2)

Public Risk (both health and financial)
Level 1/2/3 PRA Integration Issues

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

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

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

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

- Additional system models and analyses needed before containment analysis can be performed
  - “Core Damage” result from Level-1 is not adequate for starting containment analysis
  - Some containment systems not relevant to CDF are important for containment response
  - Containment system models need to be integrated with Level 1 system analysis (i.e., need to account for dependencies)
  - Bridge Event Tree (BET) used to model additional systems/phenomena, linked to Level 1 event trees
Plant Damage States (PDS) Framework Used As Input to Level-2 (from Level-1)

• Output (end states) of BET defined in terms of specific details about CD accident sequence

• Method utilizes a vector identifier
  – Each character position of the vector identifies the status of a particular system or event
    • e.g., ACCBABDC
  – Vector is “read” by the Level 2 analysis
Expanded Systems Analysis Needed to Support Level-2 Model

Level-1 Event Tree

IEs
RxTrip
LOCA
LOSP
SGTR
etc.

IE
ECI
ECR

ok

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

CD₁
CSS
CSR

PDS₁
PDS₂
PDS₃
...
PDSₙ

April 2016 Accident Progression Analysis (P-300)
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

<table>
<thead>
<tr>
<th>Character</th>
<th>PWR</th>
<th>BWR</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>Status of RCS at onset of core damage</td>
<td>Status of RPS</td>
</tr>
<tr>
<td>2</td>
<td>Status of ECCS</td>
<td>Status of electric power</td>
</tr>
<tr>
<td>3</td>
<td>Status of containment heat removal</td>
<td>RPV integrity</td>
</tr>
<tr>
<td>4</td>
<td>Status of electric power</td>
<td>RPV pressure</td>
</tr>
<tr>
<td>5</td>
<td>Status of contents of RWST</td>
<td>Status of HPI</td>
</tr>
<tr>
<td>6</td>
<td>Status of heat removal from S/Gs</td>
<td>Status of LPI</td>
</tr>
<tr>
<td>7</td>
<td>Status of cooling for RCP seals</td>
<td>Status of containment heat removal</td>
</tr>
<tr>
<td>8</td>
<td>Status of containment fan coolers</td>
<td>Status of containment venting</td>
</tr>
<tr>
<td>9</td>
<td></td>
<td>Level of pre-existing leakage from containment</td>
</tr>
<tr>
<td>10</td>
<td></td>
<td>Time to core damage</td>
</tr>
</tbody>
</table>
### Example PDS Scheme - Grand Gulf (NUREG-1150)

<table>
<thead>
<tr>
<th>Character #</th>
<th>Description</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>Initiating event</td>
</tr>
<tr>
<td>2</td>
<td>Reactor vessel pressure</td>
</tr>
<tr>
<td>3</td>
<td>Status of both high and low pressure injection</td>
</tr>
<tr>
<td>4</td>
<td>Status of containment spray and suppression pool cooling</td>
</tr>
<tr>
<td>5</td>
<td>Status of containment and containment systems as start of core damage</td>
</tr>
<tr>
<td>6</td>
<td>Time of core damage (early or late)</td>
</tr>
</tbody>
</table>
# PDS Scheme from NUREG-1150 (Grand Gulf)

<table>
<thead>
<tr>
<th>#</th>
<th>ID</th>
<th>Description</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>B1</td>
<td>Station blackout (SBO) transient has occurred. Offsite power is not recoverable because there is no emergency DC power.</td>
</tr>
<tr>
<td></td>
<td>B2</td>
<td>SBO transient has occurred. Offsite power is recoverable.</td>
</tr>
<tr>
<td></td>
<td>T2</td>
<td>Loss of PCS transient has occurred. Offsite or onsite power is available.</td>
</tr>
<tr>
<td></td>
<td>TC</td>
<td>ATWS has occurred. Offsite or onsite power is available.</td>
</tr>
<tr>
<td>2</td>
<td>P1</td>
<td>The reactor vessel (RV) is at high pressure (HP) at the onset of core damage (CD) and depressurization is not possible.</td>
</tr>
<tr>
<td></td>
<td>P2</td>
<td>The RV is at HP at the onset of CD because the operator failed to depressurize; depressurization is possible.</td>
</tr>
<tr>
<td></td>
<td>P3</td>
<td>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.</td>
</tr>
<tr>
<td></td>
<td>P4</td>
<td>The RV is at low pressure (LP)</td>
</tr>
</tbody>
</table>
# PDS Scheme from NUREG-1150 (Grand Gulf) - cont.

<table>
<thead>
<tr>
<th>#</th>
<th>ID</th>
<th>Description</th>
</tr>
</thead>
<tbody>
<tr>
<td>3</td>
<td>I1</td>
<td>Injection to the RV is not available after the onset of CD.</td>
</tr>
<tr>
<td></td>
<td>I2</td>
<td>Injection with the Firewater system is available before and after the onset of CD.</td>
</tr>
<tr>
<td></td>
<td>I3</td>
<td>Injection with the Condensate system is recoverable with the restoration of offsite power.</td>
</tr>
<tr>
<td></td>
<td>I4</td>
<td>Injection with the LP systems [core spray (LPCS) and coolant injection (LPCI)] is recoverable with the restoration of offsite power (or RV depressurization).</td>
</tr>
<tr>
<td></td>
<td>I5</td>
<td>Injection with both the HP and LP systems is recoverable with the restoration of offsite power.</td>
</tr>
<tr>
<td></td>
<td>I6</td>
<td>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).</td>
</tr>
<tr>
<td>4</td>
<td>H1</td>
<td>Containment Spray (CS) is not available at the onset of CD, neither is it recoverable.</td>
</tr>
<tr>
<td></td>
<td>H2</td>
<td>At least one train of CS is recoverable with the restoration of offsite power.</td>
</tr>
<tr>
<td></td>
<td>H3</td>
<td>At least one train of CS is available at the onset of CD.</td>
</tr>
<tr>
<td>5</td>
<td>M1</td>
<td>Miscellaneous systems (Venting, SBGT, Cl, H2I) are not available at the onset of CD.</td>
</tr>
<tr>
<td></td>
<td>M2</td>
<td>Miscellaneous systems (Venting, SBGT, Cl, H2I) are recoverable with the restoration of offsite power.</td>
</tr>
<tr>
<td></td>
<td>M3</td>
<td>Miscellaneous systems (Venting, SBGT, Cl, H2I) are available at the onset of CD.</td>
</tr>
<tr>
<td>6</td>
<td>ST</td>
<td>CD occurs in the short term (at ~1 hour).</td>
</tr>
<tr>
<td></td>
<td>LT</td>
<td>CD occurs in the long term (at &gt;12 hours).</td>
</tr>
</tbody>
</table>
## List of PDS from NUREG-1150 (Grand Gulf)

<table>
<thead>
<tr>
<th>PDS</th>
<th>PDS Character Vector</th>
<th>Accident Sequence</th>
</tr>
</thead>
<tbody>
<tr>
<td>PDS-1</td>
<td>B2-P3-I5-H2-M2-ST</td>
<td>T1B-16</td>
</tr>
<tr>
<td></td>
<td></td>
<td>T1B-17</td>
</tr>
<tr>
<td></td>
<td></td>
<td>T1B-21</td>
</tr>
<tr>
<td>PDS-2</td>
<td>B2-P3-I5-H1-M2-ST</td>
<td>T1B-16</td>
</tr>
<tr>
<td></td>
<td></td>
<td>T1B-17</td>
</tr>
<tr>
<td></td>
<td></td>
<td>T1B-21</td>
</tr>
<tr>
<td>PDS-3</td>
<td>B2-P3-I3-H1-M2-ST</td>
<td>T1B-16</td>
</tr>
<tr>
<td></td>
<td></td>
<td>T1B-17</td>
</tr>
<tr>
<td></td>
<td></td>
<td>T1B-21</td>
</tr>
<tr>
<td>PDS-4</td>
<td>B2-P4-I5-H2-M2-LT</td>
<td>T1B-14</td>
</tr>
<tr>
<td>PDS-5</td>
<td>B2-P4-I5-H1-M2-LT</td>
<td>T1B-14</td>
</tr>
<tr>
<td>PDS-6</td>
<td>B2-P4-I2-H1-M2-LT</td>
<td>T1B-14</td>
</tr>
<tr>
<td>PDS-7</td>
<td>B1-P1-I1-H1-M1-ST</td>
<td>T1B-16</td>
</tr>
<tr>
<td></td>
<td></td>
<td>T1B-17</td>
</tr>
<tr>
<td></td>
<td></td>
<td>T1B-21</td>
</tr>
<tr>
<td>PDS-8</td>
<td>B1-P1-I1-H1-M1-LT</td>
<td>T1B-13</td>
</tr>
<tr>
<td>PDS-9</td>
<td>TC-P2-I6-H3-M3-ST</td>
<td>TC-74</td>
</tr>
<tr>
<td>PDS-10</td>
<td>TC-P2-I4-H3-M3-LT</td>
<td>TC-74</td>
</tr>
<tr>
<td>PDS-11</td>
<td>T2-P2-I5-H3-M3-ST</td>
<td>T2-56</td>
</tr>
<tr>
<td>PDS-12</td>
<td>T2-P2-I5-H3-M3-LT</td>
<td>T2-56</td>
</tr>
</tbody>
</table>
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
<table>
<thead>
<tr>
<th>Core Damage Sequence</th>
<th>Containment Rupture due to a Reactor Vessel Steam Explosion</th>
<th>Containment Leakage</th>
<th>Containment Rupture due to Hydrogen Burning</th>
<th>Containment Rupture by Overpressurization</th>
<th>Containment Rupture by Meltdown</th>
<th>#</th>
<th>CF-Mode</th>
</tr>
</thead>
<tbody>
<tr>
<td>CD</td>
<td>CRVSE</td>
<td>CL</td>
<td>CR-B</td>
<td>CR-OP</td>
<td>CR-MT</td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

No Containment Failure

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

No Containment Failure

Containment Failure
Zion APET from NUREG-1150

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

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

• Containment failure details
  – Size of containment failure
  – Timing of failure
  – Energy associated with failure

• In-containment transport of radioactive material also modeled in CET/APET
  – Quality and quantity of radioactive material escaping containment
Level-3 Analysis Estimates Health Consequences for Each Release Event

- Output of Level-2 analysis (i.e., details of the radioactive material source term release) provide one input to the Level-3 analysis
- Each source term combined with site-specific information on demographics, weather, emergency planning, etc. to calculate health and economic consequences to the surrounding population
- MACCS code used to perform consequence calculations
MACCS2 Code Features

• Atmospheric transport and deposition under time-variant meteorology
• Short- and long-term mitigative actions and exposure pathways
  – evacuation, sheltering and relocation of people
  – interdiction of milk and crops
  – decontamination or interdiction of land and buildings
• Deterministic and stochastic health effects, and economic costs
  – Includes Direct (cloudshine, inhalation, groundshine, and skin deposition) and indirect (ingestion) radiation dose pathway
MACCS2 Available Since 1998

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

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

• PRA structure and outputs
  – Level-1 PRA
  – Level-2 PRA
  – Level-3 PRA
Accident Progression Analysis (P-300)

3. LWR Containment Designs
Session Objectives

• To understand the various LWR containment designs
  – Features important to severe accident response
Seven Major Types of LWR Containment Designs

- Boiling Water Reactors (BWRs)
  - Mark I (e.g., Peach Bottom 2 & 3, Cooper and Fukushima Daiichi 1-5)
  - Mark II (e.g., Limerick 1 & 2, Columbia)
  - Mark III (e.g., Clinton, Grand Gulf)

- Pressurized Water Reactors (PWRs)
  - Large Dry (e.g., ANO 1 & 2, Indian Point 2 & 3)
  - Subatmospheric (e.g., Surry 1 & 2, Millstone 3)
    - Subatmospheric usually grouped with Large Dry
  - Ice Condensers (e.g., Sequoyah 1 & 2, D. C. Cook 1 & 2)
  - AP1000 (e.g. Vogtle 3 & 4)

- Design variations within each group
## Significantly Larger Number of Dry Containments

<table>
<thead>
<tr>
<th>Containment Type</th>
<th>Number</th>
</tr>
</thead>
<tbody>
<tr>
<td>Large dry</td>
<td>58</td>
</tr>
<tr>
<td>- ANO 1 &amp; 2, Indian Point 2 &amp; 3</td>
<td></td>
</tr>
<tr>
<td>Subatmospheric</td>
<td>7</td>
</tr>
<tr>
<td>- Surry 1 &amp; 2, Millstone 3</td>
<td></td>
</tr>
<tr>
<td>Ice Condenser</td>
<td>9</td>
</tr>
<tr>
<td>- Sequoyah 1 &amp; 2, D.C. Cook 1 &amp; 2</td>
<td></td>
</tr>
<tr>
<td>Mark I</td>
<td>24</td>
</tr>
<tr>
<td>- Peach Bottom 2 &amp; 3, Cooper</td>
<td></td>
</tr>
<tr>
<td>Mark II</td>
<td>8</td>
</tr>
<tr>
<td>- Limerick 1 &amp; 2, Columbia</td>
<td></td>
</tr>
<tr>
<td>Mark III</td>
<td>4</td>
</tr>
<tr>
<td>- Clinton, Grand Gulf</td>
<td></td>
</tr>
</tbody>
</table>
Containment Free Volumes and Design Pressures Differ

### Containment design pressure (psig)

<table>
<thead>
<tr>
<th>System</th>
<th>Net Free Volume x 10^6 (ft^3)</th>
<th>Design Pressure (psig)</th>
</tr>
</thead>
<tbody>
<tr>
<td>BWR Mark I</td>
<td>0.3 x 10^6 ft^3</td>
<td>62 psig</td>
</tr>
<tr>
<td>BWR Mark II</td>
<td>0.4 x 10^6 ft^3</td>
<td>45 psig</td>
</tr>
<tr>
<td>PWR Ice Condenser</td>
<td>1.65 x 10^6 ft^3</td>
<td>12 psig</td>
</tr>
<tr>
<td>BWR Mark III</td>
<td>1.5 x 10^6 ft^3</td>
<td>15 psig</td>
</tr>
<tr>
<td>PWR Sub-Atmospheric</td>
<td>1.85 x 10^6 ft^3</td>
<td>45 psig</td>
</tr>
<tr>
<td>PWR Large Dry</td>
<td>2.1 x 10^6 ft^3</td>
<td>60 psig</td>
</tr>
</tbody>
</table>
BWR Containment Designs Differ

Mark I

Mark II

Mark III

Hydrogen igniter

Containment

Drywell

Reactor shield wall

Weir wall

Suppression pool

Horizontal vents

Upper pool

Containment sprays

Reactor

Drywell

Vacuum breaker

Downcomer

Wetwell sprays

Vacuum relief from building vent purge outlet

Suppression pool purge exhaust line

Drywell sprays

Drywell purge exhaust line

Suppression chamber sprays

Drywell vacuum breaker

Downcomers

Vent from D.W.

Vent from D.W.

Pedestal

Reactor vessel

Drywell head

Reactor building

Reactor building

Suppressor pool purge exhaust line

Accident Progression
Mark I Design Used in Older BWRs

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

Spent fuel pool

Reactor service floor

Concrete reactor building

Reactor pressure vessel

Primary containment drywell

Suppression pond wetwell
Mark I Containment Heat Removal Relies Primarily on Suppression Pool Water
Mark II Design More Unified than Mark I Design

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

LaSalle Units 1 & 2

Columbia (WNP-2)

Limerick 1 & 2
Mark II Design More Unified than Mark I Design (continued)

Susquehanna Units 1 & 2

Nine Mile Point 2
Containment Heat Removal for Mark II Containment
Mark III Dramatically Differs from Mark I and II Designs

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

Free standing steel structure

- Reactor
- Reactor shield wall
- Weir wall
- Suppression pool
- Horizontal vents

- Hydrogen igniter
- Containment sprays

Reinforced concrete

- Primary containment
- 0.25-in steel liner
- 2.5-ft concrete (Detail drawing not to scale)
- Not: Upper containment cavity and lower wellwell communicate with each other

- Upper containment pool
- Vessel annulus
- Shield wall
- Weir wall
- Horizontal vents
- Containment basement

April 2016 Accident Progression Analysis (P-300)
Mark III Containment Heat Removal Accomplished via Sprays and Suppression Pool
PWR Containment Designs Differ

Large dry

Subatmospheric

Ice condenser
Diverse Types of Large Dry Containments

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

Large dry reinforced concrete
e.g., Diablo Canyon
(Most subatmospheric designs are of this type)
Diverse Types of Large Dry Containments (continued)

Large Dry Pre-stressed (or Post-tensioned) Concrete

*e.g., Palisades*

*(This is the most common containment design)*
Diverse Types of Large Dry Containments (continued)

Large dry steel containment with reinforced concrete secondary containment e.g., Davis Besse
Containment
Heat Removal for Large Dry Containment Design Uses Sprays and Fans Coolers
Ice Condenser Containments

- Three volumes: lower compartment, upper compartment, ice condenser
  - Ice condenser connects lower compartment containing RPV and RCS to upper compartment
  - Ice condenser holds approximately 2,300,000 lb. of borated ice in perforated metal baskets
- Relies on igniters for hydrogen control
- Most have cylindrical steel containment surrounded by concrete secondary containment
  - D. C. Cook: concrete containment with steel liner
Ice Condenser Containments (continued)
CHR for IC Design Uses Sprays and Ice Condensers
AP1000 Containment Design

Natural convection air discharge

Gravity drain water tank

Water film evaporation

Outside cooling air intake

Steel Containment Vessel

Internal condensation and natural recirculation

Air Baffle

Automatic Depressurization System

Refueling Water Storage Tank Gravity Feed

2 Core Makeup Tanks, Driven By Cold Leg Conditions

2 Accumulator Tanks, Driven By Gas Pressure

April 2016
AP1000 Containment Utilizes In-Vessel Core Damage Retention

Flow of water/steam past RV removes heat and prevents RV failure.
AP1000 Passive Containment Cooling System
Water drains by gravity to enhance cooling with evaporation.
Severe Accidents Pose Several Challenges to Containment Integrity

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

(Note: These will be discussed in detail in Chapter 5)
Containment Failure Pressures Significantly Higher than Design Pressures

![Graph showing pressures comparison between ultimate and design]

- Large Dry - Zion: Ultimate 150 psig, Design 49 psig
- Subatmospheric - Surry: Ultimate 126 psig, Design 45 psig
- Ice Condenser - Sequoyah: Ultimate 65 psig, Design 10.8 psig
- Mark I - Peach Bottom: Ultimate 148 psig, Design 56 psig
- Mark II - LaSalle: Ultimate 191 psig, Design 45 psig
- Mark III - Grand Gulf: Ultimate 55 psig, Design 15 psig
Conditional Probability for Containment Failure for Each Sequence Calculated Probabilistically

Conditional containment failure probability (CCFP) =

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

\[ P_c = \text{Peak containment pressure} \]

\[ P_f = \text{Containment failure pressure} \]

Overlap of two curves represents the probability of containment failing
**Containment Structural Response and Failure Characterization**

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

- Membrane failure in the hoop direction in the cylinder or dome
- Membrane failure in the meridional 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:
  \[ P = P' \times \varepsilon_R \times \varepsilon_U \]. Where: $P'$ = median fragility, and

- $\varepsilon_R$ and $\varepsilon_U$ are random variable with unit medians that represent the inherent randomness (variability or aleatory uncertainty) and uncertainty (epistemic uncertainty) in the estimate of $P'$.

- $\varepsilon_R$ and $\varepsilon_U$ are assumed to lognormally distributed with logarithmic standard deviations of $\beta_R$ and $\beta_U$, respectively.
Containment Fragility Curves at Different Confidence Levels

Cumulative Probability of Containment Failure

$P$ (Pressure)

$P'$ (Median Failure Pressure or Fragility)

95% Confidence Level Curve

50% Confidence Level Curve

5% Confidence Level Curve
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:
  \( PrF(p) = 1 - \prod_{i=1}^{n}[1 - PrF_i(p)] \)
  where:
  \( PrF_i(p) = \) probability of failure mode \( i \) at pressure \( p \)
  \( n = \) total number of failure modes

• Note that this formulation assumes independence among the different failure modes
  – Assumption of independence in this case, is conservative
Containment Fragility and Severe Accident Loads are Integrated in CET

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

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

• What are the major containment designs?
• What are some of the characteristic features of each?
4. Phenomena Affecting Vessel Integrity

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

- Define reactor fuels effects on PWR, and BWR Reactors – and RIA and LOCA Behavior
- Identify various vessel failure modes and understand their likelihood in various reactor designs and accident scenarios.
- Describe possible end states for debris that relocates to vessel lower head.
- Discuss various mechanisms or actions that may prevent vessel failure.
<table>
<thead>
<tr>
<th></th>
<th>BWR</th>
<th>PWR</th>
</tr>
</thead>
<tbody>
<tr>
<td>Lattice</td>
<td>10x10</td>
<td>14x14 – 18x18</td>
</tr>
<tr>
<td>Lattice size</td>
<td>~5.3”</td>
<td>~9”</td>
</tr>
<tr>
<td>Height</td>
<td>120”-150”</td>
<td>144”-168”</td>
</tr>
<tr>
<td>Fuel</td>
<td>UO₂/MOX</td>
<td>UO₂/MOX</td>
</tr>
<tr>
<td>Fuel rods</td>
<td>~92</td>
<td>176-300</td>
</tr>
<tr>
<td>Part length rods</td>
<td>~14</td>
<td>0</td>
</tr>
<tr>
<td>Non-fueled rods</td>
<td>~2</td>
<td>20-25</td>
</tr>
<tr>
<td>Control</td>
<td>Ext. control rod</td>
<td>Int. control cluster</td>
</tr>
<tr>
<td>Cladding</td>
<td>Zr2</td>
<td>Zr4/Zirlo/M5</td>
</tr>
<tr>
<td></td>
<td>for PCI, nodular corrosion</td>
<td>for uniform corrosion &amp; hydrogen</td>
</tr>
<tr>
<td>Channels</td>
<td>Yes</td>
<td>No</td>
</tr>
<tr>
<td>Fuel mass</td>
<td>~180 kgU</td>
<td>~600 kgU</td>
</tr>
</tbody>
</table>

[Crawford, 2009]
Principal internal, fuel rod processes and their primary interactions
(Key design criteria are enclosed with a box and shown in red)

Fuel Designs Limited Heat Generation Rate and Total Exposure

Constraints based on MATPRO/FRAPCON; [Hagman 1993- Rudling and Patterson 2012]
UO₂ has ideal Fuel Properties Although New Accident Tolerant Fuels are Under Development

<table>
<thead>
<tr>
<th>Property</th>
<th>Uranium</th>
<th>UO₂</th>
<th>UC</th>
<th>UN</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>A. Chemical</strong></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Free energy of formation at 1000 °K (Kcal/mole)</td>
<td>-</td>
<td>-218.2</td>
<td>-25.2</td>
<td>-47</td>
</tr>
<tr>
<td>Corrosion resistance in water</td>
<td>Very poor</td>
<td>Excellent</td>
<td>Very poor</td>
<td>Poor</td>
</tr>
<tr>
<td>Compatibility with clad materials</td>
<td>Reacts with normal clad</td>
<td>Excellent</td>
<td>Variable</td>
<td>Variable</td>
</tr>
<tr>
<td>Thermal stability</td>
<td>Phase change at 665 and 770 °C</td>
<td>Good</td>
<td>Good in reducing atmosphere</td>
<td>Good, decomposes at 2600 °C</td>
</tr>
<tr>
<td><strong>B. Physical</strong></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Uranium (metal) density (g/cm³)</td>
<td>19.04</td>
<td>9.65</td>
<td>12.97</td>
<td>13.52</td>
</tr>
<tr>
<td>Theoretical Density (T.D.) (g/cm³)</td>
<td>10.96</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Melting point (°C)</td>
<td>1132</td>
<td>2865</td>
<td>2850</td>
<td>2850</td>
</tr>
<tr>
<td>Thermal conductivity W/cm/K</td>
<td>0.28 at 430 °C</td>
<td>0.03 at 1000 °C</td>
<td>0.25 at 100-700 °C</td>
<td>0.2 at 750 °C</td>
</tr>
</tbody>
</table>

after Garzarolli in [Rudling, et al. 2007]

4/19/2016

Accident Progression Analysis (P-300)
Primary Fission Sources are Uranium and activation Produce Plutonium

<table>
<thead>
<tr>
<th></th>
<th>Elastic Scattering $\sigma(\text{nn})$</th>
<th>Inelastic Scattering $\sigma(\text{nn}')$</th>
<th>Radiative Capture $\sigma(\text{n}\gamma)$</th>
<th>Fission $\sigma(\text{nf})$</th>
<th>Average neutron yield (ν-bar)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Fissile materials</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>$^{235}\text{U}$</td>
<td>15.98</td>
<td></td>
<td>86.70</td>
<td>504.81</td>
<td>2.433</td>
</tr>
<tr>
<td>$^{239}\text{Pu}$</td>
<td>7.90</td>
<td></td>
<td>274.32</td>
<td>699.34</td>
<td>2.882</td>
</tr>
<tr>
<td>$^{241}\text{Pu}$</td>
<td>12.19</td>
<td></td>
<td>334.11</td>
<td>936.65</td>
<td>2.946</td>
</tr>
<tr>
<td>Fertile materials</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>$^{238}\text{U}$</td>
<td></td>
<td></td>
<td>2.41</td>
<td>1.05E-05</td>
<td>2.489</td>
</tr>
<tr>
<td>$^{240}\text{Pu}$</td>
<td>1.39</td>
<td></td>
<td>262.65</td>
<td>6.13E-02</td>
<td>2.784</td>
</tr>
<tr>
<td>Fissile materials</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>$^{235}\text{U}$</td>
<td>152.82</td>
<td></td>
<td>131.97</td>
<td>271.53</td>
<td>2.438</td>
</tr>
<tr>
<td>$^{239}\text{Pu}$</td>
<td>155.87</td>
<td></td>
<td>184.06</td>
<td>289.36</td>
<td>2.876</td>
</tr>
<tr>
<td>$^{241}\text{Pu}$</td>
<td>148.68</td>
<td></td>
<td>169.13</td>
<td>570.66</td>
<td>2.933</td>
</tr>
<tr>
<td>Fertile materials</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>$^{238}\text{U}$</td>
<td>319.06</td>
<td></td>
<td>277.70</td>
<td>2.16E-03</td>
<td>2.490</td>
</tr>
<tr>
<td>$^{240}\text{Pu}$</td>
<td>913.76</td>
<td></td>
<td>8448.70</td>
<td>3.74</td>
<td>2.785</td>
</tr>
</tbody>
</table>


4/19/2016

Accident Progression Analysis (P-300)
Plutonium is Significant Contributor to Fission by end of Life

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

[Lundberg 2010, Rudling and Patterson 2012]
Plutonium Content Varies With Void Fraction

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

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

RIA and LOCA – Severe Accident

• During a Reactivity Initiated Accident and a Loss of Coolant Accident no fuel dispersal is allowed
• RIA -> Severe Accident
  – Chernobyl
• LOCA -> Severe Accident
  – TMI
  – Fukushima
# LWR Design Affects Severe Accident Response

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

4/19/2016
Reactor Kinetics

- About 99.4% of all neutrons are born directly in fission (*prompt neutrons*), with a very short lifetime. However, approximately 0.64% of the neutrons are delayed (*delayed neutrons*) – fraction of delayed neutrons $= \beta$

- In a system reached criticality, just as many neutrons are produced by fission as are lost by absorption and leakage from the reactor in a given time
  
  $- k_{\text{eff}} = \frac{\text{Rate of neutron production}}{\text{(Rate of neutron absorption + Rate of neutron leakage)}}=1$

- The fractional departure of a system from criticality is often expressed by the reactivity, $\rho$, and defined by:
  
  $\rho \equiv \frac{k_{\text{eff}} - 1}{k_{\text{eff}}}$

[Rudling and Patterson 2012]
Reactor Kinetics

• The conditions for a reactor power transient like RIA to be of a concern are:
  – It must be very fast.
  – The reactivity added must be larger than 0.006.

• However, water reactors are designed so that a power increase will generate negative *reactivity* feedback
  – where a fuel temperature increase gives a fast negative feedback (Doppler effect).
  – an increase in the moderator temperature and steam (void) fraction, gives a slower negative feedback.
  – A slow reactivity increase may not cause any harm even if it is larger than 0.006 because of the negative feedback mechanisms.

[Rudling and Patterson 2012]
RIA Kinetics

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

[Rudling and Patterson 2012]
RIA Pulse

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

Rudling and Patterson 2012

Measured/calculated
- Calculate adiabatic energy deposition
- Delayed neutron fission
- Heat losses

About 10% uncertainty in reported values
Pulse Characteristics

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

Average Power (kW/m) vs. Time (ms)

- 80 kW/M
- 40 ms
- 9.5 ms
- 150 ms

[Rudling and Patterson 2012] [Montgomery et al, 2003]
The graph shows the temperature profile over the radial position of a fuel pellet, with key points marked:

- **Max. temperature, 87 msec**
- **End-of-pulse, 160 msec**
- **Failure time, 74.5 msec**
- **Fuel clad interface**
- **Clad**

The graph is labeled "Accident Progression Analysis (P-300)" and cites sources:

- [Rudling and Patterson, 2012]
- [Montgomery & Rashid, 1997]
Fuel temperature
Radius
Narrow pulse (<10 ms)
Fuel dispersal

No fuel dispersal

Width pulse (>20 ms)
Detail of fuel dispersal mechanism
Rapid expansion of grain-boundary gas

Pellet expansion and PCMI loading

Fuel rod
Gas ejection
Fuel fragments
Fuel/coolant thermal interaction
Rapid expansion of steam bubbles

[Rudling and Patterson 2012]
RIA Effect on Fuel

[Rudling and Patterson, 2012]

[Le Saux et al, 2007]
Clad failure mechanism

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

[Rudling and Patterson 2012]
• Fuel Failure Modes
  – Low Burnups
    • Post-DNB failures
  – High Burnups
    • PCMI
    • Creep Rupture
• Fuel Dispersal

\[ \Delta H \]

Post-DNB failure

Pellet-clad gap

Clad ductility

30-40 GWd/T

Burnup

PCMI failure

ANT International, 2011

[Rudling and Patterson 2012]
PCMI Failure

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

Hydride blister

[Rudling and Patterson
BWR and PWR RIA Tests

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

Solid Symbols - Failure
- CDC-SPERT
- NSRR
- CABRI
- PBF
- Hydride blister/rim

Radial Average Peak Fuel Enthalpy (cal/gm)

US Core Coolability Limit

US Fuel Failure Threshold

Test Rod Burnup (MWd/tU)

[Rudling et al, 2004/05].

[Rudling and Patterson 2012]
Ballooning and Creep Burst Failure

Post-DBN Failure Risk

- Strong Pomi
- Impaired axial gas circulation
- $\Delta P \uparrow$
- Localized spallation
- Water overheating
- DNB
- Clad $T \uparrow$
- Mech. properties $\downarrow$
- Clad ballooning
- Localized $\sigma$ and $\varepsilon$
- Cladding failure $\uparrow$
- Fuel dispersion $\downarrow$

[Rudling and Patterson 2012]

[Waeckel, 1997]
Ballooning and Creep Burst Failure in VVER

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

0 degrees

90 degrees

[Rudling and Patterson
2012]

[Yegorova et al, 2006]
RIA Effect on Fuel

[Image of a flowchart illustrating the progression of events in a nuclear accident, including reactivity insertion, fuel temperature increase, cladding failure, and fuel dispersal.]

[Le Saux et al, 2007]
Introduction to LOCA

Cold-leg break in PWR – blowdown phase

Initial flow direction

[Rudling and Patterson 2012]
Representative Shutdown Response to a LOCA

Meneley and Muzumdar
Introduction to LOCA

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

[Figure showing the progression of a LOCA accident, with stages such as Clad, Oxidation, Ballooning, Burst, Coolant blockage, and Quenching.]

Temperatures °C

Time (seconds)

[Rudling and Patterson 2012]
LOCA, Decay Heat

• The removal of the decay heat is a significant reactor safety concern, especially shortly after normal shutdown or following a loss-of-coolant accident.

• Failure to remove decay heat may cause the reactor core temperature to rise to dangerous levels and has caused nuclear accidents, including the nuclear accidents at Three Mile Island and Fukushima I.

• The heat removal is usually achieved through several redundant and diverse systems, from which heat is removed via heat exchangers.

[Rudling and Patterson 2012]
Decay Heat

[Graph showing decay heat over time after shutdown, with two lines labeled "Retran" and "Todreas.

From Wikipedia, the free encyclopedia

[Rudling and Patterson 2012]
Fuel Temperature Profile

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

Initial temperature profile
Fuel temperature profile
Compression
Expansion
Pellet

Temperature inside the pellet
1500°C
1000°C
300°C

[Rudling and Patterson 2012]

[Maillat et al, 2003]
Summary

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

[Rudling and Patterson 2012]
Fuel Performance During LOCA

- During base irradiation and during the LOCA event several changes of the UO\textsubscript{2} pellet and within the fuel rod can occur which are significant for LOCA performance. These are:
  - FGR from the pellet during base irradiation increases the inner pressure, which affects the ballooning behaviour of the cladding and the probability and time of a burst during LOCA.
  - Degradation of the thermal conductivity of the UO\textsubscript{2} pellet resulting in increased fuel temperature (at constant rod power) which in turn will increase the FGR.
  - A high burnup rim zone is formed at the pellet periphery during the base irradiation at high burnups. This high burnup rim zone has a high inventory of fission gases, contained mainly in large overpressurized pores. During a LOCA the outer rim experiences a temperature increase, e.g. from 400 to 1100 °C, which may lead to a pronounced transient fission gas release TFGR during the LOCA.
  - The rod internal pressure at burst (the FGR-prior to the LOCA- and TFGR-during the LOCA-) constitutes the parts of the source term
  - Pellet-Clad Bonding

[Rudling and Patterson 2012]
Ballooning and Burst

• The basic parameters controlling fuel clad deformation and ballooning are:
  – Stress,
  – temperature and
  – creep strength, which is affected by oxidation, grain size and anisotropy.

• Burst leads to release of noble gases, iodine, caesium and other species released by the fuel (source term)

• Burst is facilitated by Hydrogen

[Rudling and Patterson 2012]
Effect of Decay Heat

---

- Cladding temperature at hot spot [°C]
- System pressure [bar]
- Pressure difference across cladding

Range of plastic deformation

- Blow down
- Refill
- Reflood

- Internal rod pressure: 70 bar
- Normal rating $F_q = 1.2$
- High rating $F_q = 2.5$

---

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

[Rudling and Patterson 2012]
Zr-O phase diagram

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

[Wright et al., 1986]

[Abriata et al., 1986]

[Rudling and Patterson, 2012]
In-vessel Severe Accident Progression

- Thermal-hydraulic and fuel rod degradation
- Hydrogen generation
- Degradation of core structure
- In-vessel fuel-coolant interaction
- Oxide/metal separation
- In-vessel debris formation
- RPV failure w/ or w/o high pressure melt ejection
Design of Fukushima-Daiichi-1 Provides Primary Containment Around Vessel

- Reactor service floor (steel construction)
- Concrete reactor building (secondary containment)
- Reactor pressure vessel
- Primary containment
- Pressure suppression pool (wetwell)

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

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

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

Fission product aerosols

Steam and hydrogen

Accident initiation
Reactor coolant thermal hydraulics
Loss of core coolant
Core heatup and uncovery
Zr oxidation/hydrogen production
Core degradation and relocation
Fission product release
Molten fuel/coolant interactions
Transport of fission products in RCS
Reactor vessel failure
BWR Vessels Also Penetrated by CRD Assemblies and Drain Line

Typical GE CRD Assembly Penetration

- Stub tube weld
- 0.23-cm (0.09 in.) annular flow gap
- SS 166 Inconel
- SS index tubes
- 0.23-cm (0.09 in.) annular flow gap
- Cladding
- Vessel wall
- Thermal sleeves consisting of 3 concentric 304 SS sleeves
- 15.24 cm (6.00 in.) dia vessel bore
- 304 SS tube

Typical GE Drain Line Nozzle Penetration

- 7.60 cm (2.99 in.)
- 6.40 cm (2.52 in.)
- 5.00 cm (1.97 in.)
- SA533B1 vessel
- Weld buildup
- SA105 II
- Through-butt weld
- SA106 B

4/19/2016
Insulation, Supports, and Cavities for Lower Heads Differ

(a) W
(b) B&W
(c) CE

Containment sump
Cavity closure plate
Reactor cavity
Cavity sump

Reactor vessel
Vessel support skirt

Carbon steel liner
Insulation

Normal drainage
Injection pump suction

4/19/2016
In-vessel Steam Explosion Issues

A. Initial separated configuration
B. Relocation and instantaneous mixing
C. Sustained energy transfer and slug acceleration
D. Slug impact

- Will in-vessel fuel/water interactions cause energetic reactions?
- Are such reactions sufficient to accelerate a slug that fails the vessel and/or create a missile that causes early containment failure?
Additional Data obtained since NUREG-1150 Evaluations

• Issues so controversial at time NUREG-1150 completed, expert panel refused to address.

• SNL staff internally developed distribution based on opinions expressed by Steam Explosion Review Group (SERG) in NUREG-1116.

• More recent experimental results indicate:
  – At low pressure [< 0.1 MPa (14.7 psi)], limited fuel mass expected to participate in energetic FCI
  – At higher pressures [> 1 MPa (147 psi)], explosion difficult to trigger

• All eleven SERG-2 experts estimated low probabilities for energetic in-vessel steam explosion
In-vessel Core Debris Coolability

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

• Event progression through RPV failure represents the largest source of uncertainty for SA mitigation
Large Uncertainties Associated with Early Methods for Quantifying Vessel Lower Head Failure Potential

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

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

- Identified and developed models for each failure mechanism
- Obtained high temperature creep and tensile data for vessel and penetration materials
- Applied methods to obtain insights for range of accident conditions and reactor designs
High-temperature Tensile and Creep Data Obtained for Vessel and Penetration Materials

- Data for penetration materials (SS304, Inconel 600 and SA105/106) also available

- For SA533B1:
  - Significant reduction in SA533B1 yield strength at temperatures above 1000 K
  - Stress versus time to rupture only moderately sensitive to phase transformation
  - Higher temperature thermal diffusivity and thermal expansion data smaller than extrapolated published values

- Calculations needed to assess impact of new data!

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Vessel Failure

Summary

• Research results suggest energetic in-vessel steam explosions not important from risk perspective

• Recent assessments and experiments provide key insights about potential for other failure modes:
  – Importance of RCS pressure and relocated debris mass, composition, decay heat distribution and melt fraction, and vessel material and fabrication
  – Experimental data and analyses suggest localized and global vessel failures more likely than penetration failures at high pressures
Debris Heat Loads

Debris Heat Loads Impact Quantification of Several Events

- Debris heat loads impact mode and timing of vessel failure and potential for containment failure.
- Information needed to address key questions:
  - What type of debris endstates may occur?
  - How does debris endstate affect vessel heat loads?
  - What phenomena affect debris coolability?
Debris Heat Load Considered by NUREG-1150
Experts Evaluating Vessel Failure Mode

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

**Vessel Failure Phenomena**

<table>
<thead>
<tr>
<th>Molten pool</th>
<th>Fragmented rubble</th>
<th>Molten pool beneath fragmented rubble</th>
</tr>
</thead>
</table>

**Homogeneous**

- Convection and radiation heat transfer to coolant
- Radiation and convection heat transfer to surroundings
- Vessel
- Crust
- Debris-to-vessel contact resistance

**Stratified**

- Molten metal (includes dissolved uranium in unoxidized zircaloy)
- Molten metal (includes some lower plenum structures)
- Molten ceramic pool

4/19/2016
Debris Heat Loads

Enhanced Cooling Possible As Relocated Core Material Solidifies

Intermittent debris-to-vessel gap

Enhanced upper surface corium surface area

Interconnected corium cracks
Debris Heat Loads

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

<table>
<thead>
<tr>
<th>Program</th>
<th>Insight</th>
<th>Materials</th>
<th>Pressure</th>
</tr>
</thead>
<tbody>
<tr>
<td>RRC/OECD RASPLAV</td>
<td>Natural convection heat fluxes, corium stratification</td>
<td>UO$_2$, ZrO$_2$, Zr, C, FeO, LaO</td>
<td>Low</td>
</tr>
<tr>
<td></td>
<td></td>
<td>W/Ta protected graphite in slice geometry</td>
<td>(0.1 MPa / 14.7 psi)</td>
</tr>
<tr>
<td>JRC/ISPRA FARO</td>
<td>Melt/water interactions, debris cooling, morphology, interactions with structures</td>
<td>UO$_2$, ZrO$_2$, Zr,</td>
<td>High</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Flat plate</td>
<td>(0.5 to 5 MPa / 72.5-725 psi)</td>
</tr>
<tr>
<td>OECD TMI-2 Vessel Investigation Program</td>
<td>Debris cooling, morphology, and interactions with structures</td>
<td>UO$_2$, ZrO$_2$, FeO$_2$, Ag, SS-304</td>
<td>High</td>
</tr>
<tr>
<td></td>
<td></td>
<td>SS-lined carbon steel vessel with penetrations</td>
<td>(3-15 MPa / 435-2175 psi)</td>
</tr>
</tbody>
</table>
Debris Heat Loads

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

- Furrows observed in relocated debris
- Intermittent contact between relocated debris and test plate
RASPLAV provides insights about stratification in relocated molten corium materials

Stratification dependent on presence of carbon and fraction of unoxidized zirconium (AW-200-2 used C-22 with 81.8 wt% UO₂, 5.0 wt% ZrO₂, 13.2 wt% Zr, and 0.3 wt% C)
Summary

- Experimental data suggest range of debris endstates possible
  - Data insufficient to select one bounding configuration
  - Data suggest melt progression scenario dependent
  - Additional research needed to assess potential for various configurations to occur and heat transfer conditions associated with various configurations

- Experimental data provide insights related to heat transfer from various configurations
  - Gaps, cracks, and increased upper surface area enhance ceramic melt coolability
Failure Mitigation Measures

Several mechanisms available to reduce potential for vessel failure

• External Reactor Vessel Cooling (ERVC)
  – Enhanced vessel/insulation arrangement
  – Enhanced vessel coatings

• RCS depressurization
  – Intentional
  – Unintentional
Requirements for Successful External Reactor Vessel Cooling

- Water must quickly cover lower vessel external surfaces
  - Flooding must occur prior to melt relocation
  - Sufficient coolant ingress and steam egress
  - Insulation must be designed to withstand forces associated with ERVC

- Heat flux to vessel must be less than heat removed from the vessel
  - Often translated to vessel heat flux must be less than Critical Heat Flux (CHF) for nucleate boiling on vessel outer surface
  - CHF dependent on angle, surface treatment, geometry (penetrations, junctions, insulation) and water height
Failure Mitigation Measures

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

• In many Individual Plant Examinations (IPEs), cavity flooding assumed to preclude vessel failure and reduce event consequences
  – Westinghouse vessels (Zion, Byron, etc.) penetrated by instrumentation tubes that travel through reactor cavity
  – CE vessels (Palisades, etc.) without lower head instrumentation tubes

• All four generic vendor Severe Accident Management Guidelines (SAMGs) invoke ERVC, although extent of reliance varies in plant-specific SAMGs

• Finnish safety authorities approved ERVC as an Accident Management strategy for Loviisa plant (modified to enhance ERVC)

• Proposed for many advanced reactor designs, such as Westinghouse AP600, AP1000, and the Korean APR1400.
Various Approaches used to Investigate ERVC

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

4/19/2016

Accident Progression Analysis (P-300)
Mitigating High Pressure Scenarios

- Progression of core damage under high pressure presents unique challenges
  - Steam generator tube rupture (bypass risk)
  - Hot leg/surgeline failure
  - Safety valve failure to close
  - RCP seal leakage
  - High pressure melt ejection
  - Direct containment heating

- Mechanisms to prevent by design
  - Primary depressurization system
  - Lower core power density
  - Minimize head penetrations
  - Minimize pathways to upper containment
SCDAP/RELAP5 calculations suggest induced RCS piping failure prior to significant core relocation.

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

• External Vessel Reactor Cooling (ERVC) may prevent vessel failure
  – Plant-specific evaluations needed to assure timing of flooding, sufficient water ingress, and steam egress.
  – Methods available to enhance ERVC.

• Several RCS depressurization mechanisms offer potential for accident mitigation:
  – RCP seal leakage
  – Induced RCS piping failures
  – Safety valve failing open
  – Intentional PORV depressurization.
Case Study: AP1000

Westinghouse Advanced PWR 1000 MWe (AP1000) focused on simplicity
- Heavily reliant on passive, rather than active, safety systems
- Reduced outages and maintenance
The AP1000™ is designed to mitigate a postulated severe accident such as core melt. In this event the AP1000 operator can flood the reactor cavity space immediately surrounding the reactor vessel with water to submerge the reactor vessel. The cooling is sufficient to prevent molten core debris in the lower head from melting the steel vessel wall and spilling into the containment. These water storage tanks hold enough water to cool the containment vessel for seventy two hours.
AP1000 Relies on Design Simplicity and Passive Cooling

- 50% Fewer Valves
- 35% Fewer Safety Grade Pumps
- 80% Less Pipe
- 45% Less Seismic Building Volume
- 85% Less Cable

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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.
References

In-Vessel Steam Explosion


References (continued)

Vessel Failure

References (continued)

External Reactor Vessel Cooling


References (continued)

Debris Endstate

- Papers presented at the OECD/NEA/CSNI Workshop on In-vessel Retention and Coolability, Garching, Germany, March 3-6, 1998.

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


References (continued)

RCS Depressurization – RCP Seal Leakage


RCS Depressurization - Induced depressurization


TMI Damage Implications for Fukushima and Status

Douglas Akers
Nuclear Physics
Idaho National Laboratory

4/19/2016
Overview

• TMI reactor accident core damage progression
• TMI reactor core damage and fuel relocation
  – Core material melt behavior
  – Relocation of fission products and core materials
• Damage to the lower head of the TMI reactor vessel
• RPV design differences between PWRs and BWRs
• Implications of the TMI accident for the Fukushima recovery
• Fukushima status
• Fukushima path forward and schedule
• Nuclear material accountability issues and TMI approach
TMI Core Damage Occurred Within 224 Minutes Including Core Relocation to RPV Lower Head
Melt Behavior Defines Core Damage Progression

Temperature (°C)

3000
2850
2690
≈2600
≈2400
2380
≈2060
1975
≈1900
1760
≈1450
≈1300
1200
1150
≈940
≈800

- Melting of UO₂
- Melting of ZrO₂
- Formation of a ceramic U-Zr-O melt
- Formation of α-Zr(0)-UO₂ and U-UO₂ monotectics
- Melting of B₄C
- Melting of Al₂O₃ (burnable poison rod: Al₂O₃ + B₄C)
- Melting of oxygen-stabilized α-Zr(0)
- Al₂O₃-UO₂ and Al₂O₃-ZrO₂ eutectics
- Melting of as-received Zircaloy-4
- Start of UO₂ dissolution by molten Zircaloy → formation of metallic (U,Zr,O) melt
- Melting of stainless steel and Inconel
- Fe-Zr, Al(Al₂O₃)-Zr eutectics
- Ni-Zr eutectic, Ag-Zr reactions
- B₄C-Fe eutectics
- Start of rapid Zircaloy oxidation by H₂O → uncontrolled temperature escalation
- Formation of first Fe-Zr and Ni-Zr eutectics
- Melting of Ag-In-Cd alloy
Extensive Sampling and Coring Used to define Core damage and Materials/Fission Product Behavior

- Core Examinations included
  - Control rod leadscrews
  - Upper RPV core debris
    - Core bores
    - lower RPV debris
  - Fuel - \((\text{UO}_2)\) and Zirconium Relocation
    - Upper core debris – 26,000 kg
    - Center core melt – 21000 kg
      - Lower crust – 5400 kg
    - Partial fuel assemblies - 40000 kg
    - Core support assembly – 8900 kg
    - RPV lower head -19000 kg
Limited Damage to Reactor Pressure Vessel Above Fuel Assemblies

Damage to Grid Above Fuel

Damage to Fuel Assembly End Fitting
(U,Zr)O$_2$ Previously Melted Reactor Fuel - Highly Inert and Depleted of Volatile Fission Products
(U,Zr)O$_2$ Melted Reactor Fuel - Highly Inert and Similar Composition in Central Core and Lower Head

- Composition is primarily (U,ZrO$_2$) nominally - 69%U, 26%Zr and 4.6% O
- Density of debris ranges from 7-9 g/cm$^3$
- Nominally <1% of volatiles inventory (e.g., $^{134,137}$Cs, radioiodine and noble gases)
- Dissolution of bulk specimens (>50 g) in some cases required multiple step processes and required days to complete dissolution
TMI Core Boring System (Modified Drilling System) to Break Up Debris

Core bore head
Specialized Gamma Spectrometry System used to Reconstruct Fuel and Fission Product Relocation
Fission Product Release (e.g., $^{137}$Cs) from Prior molten Reactor Fuel Evaluated using Gamma Tomography System
Relatively Intact Fuel In Metal Layer Below Central Core
19000 kg of Fuel Melt On TMI RPV Lower Head
Burned off Incore Instrument
Penetrations on Head Indicate Protection by Soldified Melt
TMI -2 Incore Nozzle Protected by Solidified Fuel Melt
Varying damage to Lower Head RPV Nozzles
Lower Head Boat Samples Indicate Max Temperature $< 1100^\circ\text{C}$
BWR RPV Assembly More Massive Than PWR

Figure 4 Reactor Assembly

PWR Lower RPV Head

BWR Lower RPV Head

4/19/2016

Accident Progression Analysis (P-300)
Significant Relocation of Volatile Fission Products From the Reactor Vessel to the Containment with little Release to the Environment

• Significant release of all highly volatility fission products where melting occurred - Approximately 50% of noble gases, iodines and cesium radionuclides
• Medium volatility radionuclides $^{125}\text{Sb}$ and $^{104}\text{Ru}$ accumulated in metal layer below mid core location
• Low volatility $^{144}\text{Ce}$, $^{154}\text{Eu}$ and $^{155}\text{Eu}$ fully retained in fuel
• Tc-99 releases also expected due to volatility and long half-life
**137Cs Retained in RB water (47%) and noble gases in containment (54%)**

<table>
<thead>
<tr>
<th>Repository</th>
<th>Core Inventory (%)</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>129I</td>
</tr>
<tr>
<td>RCS coolant</td>
<td>1.20</td>
</tr>
<tr>
<td>RCS surfaces</td>
<td>0.91</td>
</tr>
<tr>
<td>RCDT</td>
<td>0.01</td>
</tr>
<tr>
<td>RB structural surfaces</td>
<td>0.07</td>
</tr>
<tr>
<td>RB air cooling assembly surfaces</td>
<td>0.23</td>
</tr>
<tr>
<td>RB basement water</td>
<td>14.10</td>
</tr>
<tr>
<td>RB basement sediment</td>
<td>7.91 to 100.00</td>
</tr>
<tr>
<td>Auxiliary building media</td>
<td>6.11(^a)</td>
</tr>
<tr>
<td><strong>Total</strong></td>
<td>30.54 to 100.00</td>
</tr>
</tbody>
</table>

\(^a\)Not detected.
Post Accident Fuel Distribution Outside the Reactor Core

- Auxiliary and fuel handling buildings <17 kg
- Reactor building outside the RCS - <75 kg
- RCS outside the RV - <133 kg
  - Primarily steam generator tube sheet
- Reactor vessel following defueling <900 kg
- Several techniques used for post accident defueling assessment
  - Visual examination estimate 630 kg
  - Passive neutron measurement – 1332 kg
Fukushima Dahiichi NPP after Tsunami and explosions
Recent View of Fukushima Daiichi (Units 1 to 4)

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

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# Unit Specifications

<table>
<thead>
<tr>
<th>Unit</th>
<th>Output (MW)</th>
<th>Start of Operation</th>
<th>Reactor Type</th>
<th>Containment Model</th>
<th>General Contractor</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>460</td>
<td>3/26/1971</td>
<td>BWR-3</td>
<td>Mark I</td>
<td>GE</td>
</tr>
<tr>
<td>2</td>
<td>784</td>
<td>7/18/1974</td>
<td>BWR-4</td>
<td>Mark I</td>
<td>GE &amp; Toshiba</td>
</tr>
<tr>
<td>3</td>
<td>784</td>
<td>3/27/1976</td>
<td>BWR-4</td>
<td>Mark I</td>
<td>Toshiba</td>
</tr>
<tr>
<td>4</td>
<td>784</td>
<td>10/12/1978</td>
<td>BWR-4</td>
<td>Mark I</td>
<td>Hitachi</td>
</tr>
<tr>
<td>5</td>
<td>784</td>
<td>4/18/1978</td>
<td>BWR-4</td>
<td>Mark I</td>
<td>Toshiba</td>
</tr>
<tr>
<td>6</td>
<td>1100</td>
<td>10/24/1979</td>
<td>BWR-5</td>
<td>Mark II</td>
<td>GE &amp; Toshiba</td>
</tr>
</tbody>
</table>
JAEA Activities for Environmental Restoration

Decontamination technology development

- Challenge for development of macromolecule cesium collection material, soil exfoliation technology with solidification agent, etc.
- Demonstration of areal decontamination at the model areas, totaling 221 ha in size with various components and various dose rate levels from 5 to over 100 mSv/y, and decontamination technologies were carried out to provide valuable data for full scale decontamination work in the future.

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

<table>
<thead>
<tr>
<th>Location</th>
<th>Before decontamination (μSv/h)</th>
<th>After decontamination (μSv/h)</th>
<th>Reduction rate</th>
</tr>
</thead>
<tbody>
<tr>
<td>Forest</td>
<td>136.8</td>
<td>63.1</td>
<td>54%</td>
</tr>
<tr>
<td>Farmland</td>
<td>62.4</td>
<td>12.4</td>
<td>80%</td>
</tr>
<tr>
<td>Housing site</td>
<td>55.3</td>
<td>14.5</td>
<td>74%</td>
</tr>
<tr>
<td>Road</td>
<td>55.2</td>
<td>17.3</td>
<td>69%</td>
</tr>
<tr>
<td>Road (unpaved)</td>
<td>112.5</td>
<td>76.4</td>
<td>32%</td>
</tr>
</tbody>
</table>

Communications and instructions

- JAEA staff members talk face-to-face with parents and teachers, answering their questions on radiation and its health effects. 177 sessions have been held since July 2011, and a total of about 12,900 people joined.
- WBC measurement for Fukushima pref. residents (A total of 18,600 persons have been measured and those who were estimated as more than 1 mSv (a maximum of 2.8 mSv) are 0.07% of the whole.)
“Circulated cooling water injection” has been established to reuse the contaminated water in the buildings (accumulated water) for injection into the reactors (since 2011/6/27.)
Photos inside of PCV, Unit2 on Jan. 20th

Inside wall of PCV

Grating (OP. 9500)

Thermo couple

Inside wall of PCV

Grating (OP. 9500)

Shooting to this direction

4/19/2016 Accident Progression Analysis (P-360)
Inside inspection of damaged PCV, Unit1 on Sep. 27th

Bottom surface (dry well)

Surface on the deflector

CCD camera inspection

4/19/2016

[Diagram showing various views of the PCV with measurements and labels for grating and deflector]
Muon Model of Unit 1 Vessel and Core

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

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

Density-length is the multiplying structure density and length along with incidence path from the detector, which represents the extent of muon attenuation
Muon Imaging of Unit 1 Reactor – 26 days

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

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

Figure 2. Measured image from the detector 1 (North west side)
Defueling Floor Radiation Fields -100 R/hr Maximum (1 mSv = 100 mR)
Summary

• TMI data provides a basis for understanding the Fukushima reactor accident

• Improvements in nuclear technology provided methods for characterizing the reactor accident

• Significant retention of core debris in the control rod assemblies is likely

• Direct measurement of relocated fuel material is possible
5. Phenomena Affecting Containment Integrity

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

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

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

- Pre-existing leaks
- Overpressure
- Dynamic pressure (shock wave)
- Internal missiles
- External missiles
- Basemat meltthrough
- Bypass
- Isolation failures
Challenges Dominate at Different Time Periods

<table>
<thead>
<tr>
<th>Time Regime</th>
<th>Challenge</th>
</tr>
</thead>
<tbody>
<tr>
<td>Early</td>
<td>Start of accident: pre-existing leak, isolation failure, bypass</td>
</tr>
<tr>
<td></td>
<td>At or soon after vessel breach: RCS blowdown, hydrogen combustion, bypass, steam explosion, liner meltthrough</td>
</tr>
<tr>
<td>Late (&gt; 2 hours after vessel breach)</td>
<td>containment heat removal system failure, hydrogen combustion, non-condensable gas generation, basemat meltthrough</td>
</tr>
</tbody>
</table>
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?
Failure Analyses

NUREG-1150 Results Indicate BWR Early Containment Failures More Likely

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

4/19/2016 Accident Progression Analysis (P-300)
Failure Analyses

Individual Plant Examinations (IPEs) Suggest Late Failures Dominate

- PWR containments less likely to experience early failures than smaller BWR containments
- Bypass probabilities higher in PWRs due to higher operating pressures and use of steam generators
- Result variability due to differing containment features and analysis
Key Phenomena Challenging Containment Integrity

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

- Will in-vessel fuel/water interactions cause rapid energetic reactions?
- Are such reactions sufficient to accelerate a slug that fails vessel upper head and/or creates a missile that causes early ($\alpha$) containment failure?
Ex-Vessel Steam Explosion Issues

- Is sufficient water present in the reactor cavity or pedestal region for an energetic ex-vessel fuel/water reaction?
- Are such reactions sufficient to lead to containment failure?
NUREG-1150 Addresses SEs using Sensitivity Studies

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

<table>
<thead>
<tr>
<th>Facility/Location</th>
<th>Phenomena Investigated</th>
<th>Test Section Diameter</th>
<th>Melt Jet Diameter (mm)</th>
<th>Water Depth</th>
<th>System Pressure</th>
<th>Melt Composition and Mass</th>
</tr>
</thead>
<tbody>
<tr>
<td>FARO/ISPRA</td>
<td>Integral tests investigating premixing, quenching, propagation, and FCI energetics</td>
<td>700 mm (27.6 in.)</td>
<td>100 mm (4.0 in.)</td>
<td>0.1-5.0 m (0.3 -1.4 ft)</td>
<td>0.1 – 5.0 MPa (15 -730 psi)</td>
<td>UO$_2$-ZrO$_2$ (w/ and w/o Zr &amp; SS) 18 - 250 kg (40 - 550 lb)</td>
</tr>
<tr>
<td>KROTOS/ISPRA</td>
<td>Smaller scale tests investigating premixing, quenching, propagation, and FCI energetics</td>
<td>95-200 mm (3.7-7.9 in.)</td>
<td>30-50 mm (1.2-2.0 in)</td>
<td>1.0 m (3.3 ft)</td>
<td>0.1 - 1.0 MPa (15-150 psi)</td>
<td>UO$_2$-ZrO$_2$ Al$_2$O$_3$ 1.4 - 6.0 kg (3.1 - 13 lb)</td>
</tr>
<tr>
<td>TROI/KAERI</td>
<td>Integral tests investigating premixing, quenching, propagation, and FCI energetics</td>
<td>600 mm (24.0 in.)</td>
<td>~38 to 50 mm (~1.5 – 2.0 in.)</td>
<td>0.67 m (2.2 ft)</td>
<td>0.1 to 2.0 MPa (15.0 -290 psi)</td>
<td>ZrO$_2$ and UO$_2$-ZrO$_2$ 5 to 14 kg (11- 30 lb)</td>
</tr>
</tbody>
</table>
Experimental Data Provides Key Insights about Steam Explosions (continued)

Phenomena

FARO

KRYTOS

TROI

4/19/2016 Accident Progression Analysis (P-300)
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 (0.13 to 0.19 in.).
- No energetic steam explosions observed in tests simulating in-vessel conditions.
Key Parameters for Evaluating Ex-Vessel Steam Explosion Potential

- **Sequence**
  - Melt composition (amount of unoxidized metals)
  - Melt mass and energy
  - Melt pour area, rate, and geometry
  - Water availability

- **Containment design**
  - Cavity or pedestal geometry
  - Potential for shock wave transmission
  - Water availability
Recent Findings Suggest Lower Probability for Steam Explosions

• Experimental results indicate:
  – At low pressure (0.1 MPa/15 psia), limited fuel mass participates
  – At higher pressures (>1 MPa/150 psia), difficult to trigger
  – Debris composition affects ability to trigger spontaneous SE

• All eleven SERG-2 experts estimated low probabilities for $\alpha$-mode failure
  – Low conversion energy
  – Lower explosivity of corium
  – Intervening structures

• Nine of eleven SERG-2 experts declared issue of $\alpha$-mode failure induced by steam explosion resolved from risk perspective

• OECD-sponsored SERENA program designed to compare various SE models with data from FARO, KROTOS, and TROI.
Direct Containment Heating (DCH) Issues

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

Phenomena
Unique Experimental Facilities Provide Insights About Potential for DCH

Facility capabilities allowed measurement of:

- Pressure load
- Hydrogen distribution and combustion
- Containment compartment geometry effect
- Post-test debris distribution
- Effectiveness of safety equipment
Key Parameters for Evaluating DCH Potential

• Sequence
  – Melt composition (amount of unoxidized metals)
  – Melt mass
  – Vessel pressure and failure area
  – Water availability (via containment sprays, etc.)

• Containment design
  – Subcompartment configuration
  – Cavity flow paths
  – Water availability (flooded height)
Recent results suggest very low potential for DCH in large dry or subatmospheric containments.

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

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

- Large scale, prototypic tests:
  - 100 to 2000 kg (220 to 4400 lbs) prototypic corium
  - 30 cm x 30 cm to 120 cm x 120 cm (1 ft x 1 ft to 4 ft x 4 ft) concrete basemat area
  - UO$_2$, ZrO$_2$, and Zr corium materials heated up to 2350 K (3770 °F)
  - Electrodes to simulate decay heat
  - Water added after corium melts

- Observed:
  - High initial heat transfer from corium
  - Significantly lower heat removal after crust forms on upper surface
  - Voiding in corium region beneath crust
  - Pool swelling followed by eruptions enhances heat removal.

- CEA-sponsored VULCANO underway (with ~30-50 kg prototypic materials)
Several Factors Influence MCCI

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

Limestone concrete ablates more rapidly and generates more gases.

Phenomena

Typical chemical composition (wt.%)

<table>
<thead>
<tr>
<th>Oxide</th>
<th>Basaltic Concrete</th>
<th>Limestone Concrete</th>
<th>Limestone/Common Sand Concrete</th>
</tr>
</thead>
<tbody>
<tr>
<td>SiO₂</td>
<td>54.73</td>
<td>3.60</td>
<td>35.70</td>
</tr>
<tr>
<td>CaO</td>
<td>8.80</td>
<td>45.40</td>
<td>31.20</td>
</tr>
<tr>
<td>Al₂O₃</td>
<td>8.30</td>
<td>1.60</td>
<td>3.60</td>
</tr>
<tr>
<td>MgO</td>
<td>6.20</td>
<td>5.67</td>
<td>0.48</td>
</tr>
<tr>
<td>Fe₂O₃</td>
<td>6.25</td>
<td>1.20</td>
<td>1.44</td>
</tr>
<tr>
<td>K₂O</td>
<td>5.38</td>
<td>0.68</td>
<td>1.22</td>
</tr>
<tr>
<td>TiO₂</td>
<td>1.05</td>
<td>0.12</td>
<td>0.18</td>
</tr>
<tr>
<td>Na₂O</td>
<td>1.80</td>
<td>0.08</td>
<td>0.82</td>
</tr>
<tr>
<td>MnO</td>
<td>-</td>
<td>0.01</td>
<td>0.03</td>
</tr>
<tr>
<td>Cr₂O₃</td>
<td>-</td>
<td>0.004</td>
<td>0.014</td>
</tr>
<tr>
<td>H₂O</td>
<td>5.00</td>
<td>4.10</td>
<td>4.80</td>
</tr>
<tr>
<td>CO₂</td>
<td>1.50</td>
<td>35.70</td>
<td>22.00</td>
</tr>
</tbody>
</table>
Presence of Water Does Not Guarantee Coolability

Water can cool released gases and retain some released fission products
EPR Relies on Large Spreading Area to Guarantee Coolability

Reactor Cavity

Spreading Area
Incorporating MCCI Benefits in Event Mitigation Strategy – EPR Example

- Refractory layer ensures melt discharge from cavity only occurs at the gate

- Admixture of concrete constituents during MCCI conditions melt to facilitate spreading

- Heavy and light oxides fully miscible – oxide layer eventually rises above metallic layer

- Metals react with $H_2O$ and $CO_2$ with $H_2$ and $CO$ as products
Phenomena

Hydrogen Combustion Issues

\[2 \text{H}_2 + \text{O}_2 \rightarrow 2 \text{H}_2\text{O} + 57.8 \text{ kcal /mole H}_2 \text{ consumed} \]
(229 Btu /mole H\textsubscript{2} consumed)

• Under what conditions will hydrogen combustion occur?
• Are pressure loads associated with hydrogen combustion sufficient to threaten containment integrity?
Hydrogen ignition significantly increased TMI-2 containment pressure

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

- Deflagration waves
  - requires low energy ignition source
  - requires $[\text{H}_2] > 4 \text{ vol}\%$ and $[\text{H}_2\text{O}] < 60 \text{ vol}\%$.
  - travel subsonically ($< 35 \text{ m/s or } < 120 \text{ ft/s}$)
  - heat unburned gases to temperatures high enough for chemical reactions to occur
  - produce quasi-static containment loads

- Detonation waves
  - requires high energy ignition source
  - requires $[\text{H}_2] > 18 \text{ vol}\%$
  - travel supersonically (at least 2200 m/s or 7200 ft/s)
  - heat unburned gases by compression
  - produce dynamic or impulsive containment loads in addition to static loads (can generate missiles and challenge containment steel shell).
Shapiro and Moffette Diagram Depicts Hydrogen: Air: Steam Flammability Limits

Limits vary with:

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

Mixture non-flammable if:

- $[\text{H}_2] < 4 \text{ vol\%}$,
- $[\text{O}_2] < 5 \text{ vol\%}$, or
- $[\text{H}_2\text{O}] > 60 \text{ vol\%}$
RUT Experimental Data Provides Insights about Hydrogen Ignition

- Series of tests with dynamic hydrogen injection and spark ignition
  - Up to 480 m³ (17,000 ft³)
  - 0.6-1.0 kg/s (1.3 – 2.0 lb/s) and 0.1-0.2 kg/s (0.2-0.4 lb/s) H₂ 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 H₂ concentration

- Results used to optimize number and location of igniters and develop H₂ combustion criteria
  - $\sigma$ criterion to estimate risk of flame acceleration
  - $7\lambda$ criterion to assess non-occurrence of DDT
Localized Effects May Be Important

- Higher concentrations of hydrogen
  - near release points
  - under ceilings or dome due to density stratification,
  - near steam removal locations, such as ice condensers, suppression pools, and fan coolers
  - within smaller volume compartments
- Equipment susceptible to high pressure or temperature
- Ignition sources
  - structures / regions at higher temperature
  - electrical equipment sparks
10CFR50.44 Hydrogen Control Requirements Instituted after TMI-2

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

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

• Design dependent
  – Pedestal door, drywell floor, sump, and downcomer entrance size and geometry

• Sequence dependent
  – Melt mass released
  – Melt composition
  – Melt superheat
  – RCS pressure
  – Presence of water
Phenomena

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

- Mark I Liner failure studies grouped cases by key parameters affecting liner failure
  - Pressure
  - Drywell Flooding
  - Vessel Failure Mode
- Studies recommended several actions to
  - Improve success for vessel depressurization
    - Revised procedures
  - Improve success for drywell flooding
    - Availability of alternate water sources to drywell spray header
    - Revised criteria for initiation of containment sprays
    - Improved diesel pump and spray nozzle designs
Case Study: DCH in Westinghouse Plants with Large Dry Containments or Subatmospheric Containments
DCH Resolution Methodology

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

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

Case Study and Problem

Failure probability

Pressure (MPa)

Pressure (psig)

Robinson (Large Dry)
Zion (Large Dry)
ANO-2 (Large Dry)
D.C. Cook (Ice Condenser)
Surry (Subatmospheric)
Case Study and Problem

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

- No intersections of load distributions with fragility distributions for most plants (CCFP ~ 0).
- Finite, but negligible, intersection predicted for H.B. Robinson (broad containment fragility distribution and dome transport characteristics).
Problem: How would DCH analysis change if a Mark I containment were considered?
Problem: How would EPR containment integrity evaluations differ?

- Double-walled containment with ventilation and filtering system
- Containment heat-removal system
- Spreading Area
- Protection of the Basemat
- 4-train redundancy of main safeguard systems
- In Containment Refueling Water Storage Tank (IRWST)
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?
References

General


In-Vessel and Ex-Vessel Steam Explosions


Direct Containment Heating (DCH)

References

DCH (continued)


References

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.


• 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.


References

Mark I Liner Failure


Hydrogen Combustion

Accident Progression Analysis (P-300)

6. CET Development

April 2016
Session Objectives

• To Understand the basic steps and information needs in the CET development process
Level of Detail in CET Varies

- CET models can be very simple or very complex
  - WASH-1400, many IPE’s only consist of 6 to 12 top events in event tree
  - NUREG-1150 APET’s comprised 75 to 125 top events
    - Not displayed graphically
CET Details Determined by Purpose of Level-2 Analysis

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

- Either explicitly or implicitly CET needs to:
  - Delineate boundary conditions (i.e., details of level-1 CD sequence, containment isolation, etc.)
  - Update/establish status of containment systems (e.g., Recovery of AC power)
  - Model progression of accident with respect to actual core damage and RPV/RCS failure
  - Model resulting loads on containment structure
  - Assign probability of release/source-term to each accident sequence
Level-2 Analysis Typically Represented as an Event Tree

- Event trees appropriate modeling choice for chronological progression of a sequence of event

- Ideally, Level-2 analysis would be incorporated into expanded level-1 models (i.e., single integrated ET)
  - Direct linking would better accommodate dependencies and obviate much manual manipulation of intermediate results

- Single integrated model, often not practical
  - Level-2 analyst usually different from Level-1 analyst
  - Modeling and bookkeeping requirements very extensive
  - Large, integrated models more difficult to quantify and review
Potential CET Top Event Sources

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

• Bypass Events
  – Vessel failure not required for release
    • Event V or Interfacing System LOCA (ISLOCA)
    • SGTRs
      – Largely determined by level-1 CD sequence information

• Early Failures
  – Early - usually in relation to the timing of vessel failure (i.e., before, during or shortly after vessel failure)
  – Typically within a few hours of the start of core damage

• Late Failures
  – Several hours after vessel failure
Containment Failure

• If containment is not bypassed, need to assess the likelihood and mode of containment failure

• Containment failure mechanisms are scenario dependent
  – Mode of RPV failure has major impact on magnitude of containment challenges
    • e.g., Does RPV fail while RCS is at high pressure or low pressure?
Analyze Containment Loads

- Many challenges need to be considered
  - Internal pressure rises (usually considered “static”)
  - High temperatures
  - Thermo-mechanical erosion of concrete structures (molten core concrete interaction)
  - Localized dynamic loads (e.g. shock waves and internally generated missiles)

- Analyses often distinguish between catastrophic failures and leaks

- Location of failure is also important
  - E.g., wetwell versus drywell
** Loads Can be Characterized at Different Levels of Detail **

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

- Both approaches have been used
Estimate Challenges to Containment Integrity (for example)

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

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

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

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

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

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

<table>
<thead>
<tr>
<th></th>
<th>Description</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>Time of containment failure</td>
</tr>
<tr>
<td>2</td>
<td>Period in which sprays operate</td>
</tr>
<tr>
<td>3</td>
<td>Occurrence of CCI</td>
</tr>
<tr>
<td>4</td>
<td>RCS press before VB</td>
</tr>
<tr>
<td>5</td>
<td>Mode of VB</td>
</tr>
<tr>
<td>6</td>
<td>SGTR</td>
</tr>
<tr>
<td>7</td>
<td>Amount of core available for CCI</td>
</tr>
<tr>
<td>8</td>
<td>Fraction of Zr oxidized in vessel</td>
</tr>
<tr>
<td>9</td>
<td>Fraction of core in HPME</td>
</tr>
<tr>
<td>10</td>
<td>Size or type of containment failure</td>
</tr>
<tr>
<td>11</td>
<td># of large holes in RCS after VB</td>
</tr>
<tr>
<td>12</td>
<td>Early ice condenser function</td>
</tr>
<tr>
<td>13</td>
<td>Late ice condenser function</td>
</tr>
<tr>
<td>14</td>
<td>Status of air return fans</td>
</tr>
</tbody>
</table>
**Example: SEQSOR Characteristic 1 - Containment Failure Time**

<p>| | | |</p>
<table>
<thead>
<tr>
<th></th>
<th></th>
<th></th>
</tr>
</thead>
<tbody>
<tr>
<td>A</td>
<td>V-Dry</td>
<td>Event V, releases not scrubbed by fire suppression sprays</td>
</tr>
<tr>
<td>B</td>
<td>V-Wet</td>
<td>Event V, releases scrubbed by fire suppression sprays</td>
</tr>
<tr>
<td>C</td>
<td>CF-E</td>
<td>Containment failure during core degradation</td>
</tr>
<tr>
<td>D</td>
<td>CF-VB</td>
<td>Containment failure at vessel breach</td>
</tr>
<tr>
<td>E</td>
<td>CF-L</td>
<td>Late containment failure (during initial CCI, nominally a few hours after VB)</td>
</tr>
<tr>
<td>F</td>
<td>CF-VL</td>
<td>Very late containment failure (from 12 to 24 hours after VB)</td>
</tr>
<tr>
<td>G</td>
<td>NoCF</td>
<td>No containment failure</td>
</tr>
</tbody>
</table>
Parametric Source Term Code

- XSOR codes written specifically for NUREG-1150 plants
- Parametric Source Term (PST) code developed in 1996 under Accident Sequence Precursor (ASP) program
  - PST developed to provide source terms for all U.S. PWRs
  - Estimates source terms for 9 release classes comprising approximately 60 isotopes
### PST Input Uses 10-Character Vector

<table>
<thead>
<tr>
<th>No.</th>
<th>Description</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>Containment Failure Mode</td>
</tr>
<tr>
<td>2</td>
<td>Status of Containment Heat Removal Systems</td>
</tr>
<tr>
<td>3</td>
<td>Occurrence of Core Concrete Interactions</td>
</tr>
<tr>
<td>4</td>
<td>RCS Pressure at Vessel Breach</td>
</tr>
<tr>
<td>5</td>
<td>Mode of Vessel Breach</td>
</tr>
<tr>
<td>6</td>
<td>Occurrence of SGTR</td>
</tr>
<tr>
<td>7</td>
<td>Presence of Water in Reactor Cavity</td>
</tr>
<tr>
<td>8</td>
<td>Amount of Oxidation in Vessel</td>
</tr>
<tr>
<td>9</td>
<td>Containment Failure Size</td>
</tr>
<tr>
<td>10</td>
<td>Core Damage Time</td>
</tr>
</tbody>
</table>
**Example: PST Characteristic 1 - Containment Failure Mode**

<table>
<thead>
<tr>
<th>ID</th>
<th>Definition</th>
</tr>
</thead>
<tbody>
<tr>
<td>A</td>
<td>Containment bypass</td>
</tr>
<tr>
<td>B</td>
<td>Containment not isolated</td>
</tr>
<tr>
<td>C</td>
<td>Early containment failure <em>(near time of vessel breach)</em></td>
</tr>
<tr>
<td>D</td>
<td>Late containment failure</td>
</tr>
<tr>
<td>E</td>
<td>No containment failure</td>
</tr>
</tbody>
</table>
Most Level-2 Analyses Involve a Mix of Supporting Information

• Plant-specific code calculation
  – MAAP, MELCOR, SCDAP/RELAP5

• Analyses from other prior PRAs or severe accident studies
  – NUREG-1150, IPEs

• Engineering analyses of specific issues
  – Threat from hydrogen combustion

• Experimental data
  – Debris coolability
<table>
<thead>
<tr>
<th>Accident Progression</th>
<th>Phenomena</th>
<th>LP Dry</th>
<th>Ice Cond</th>
<th>Mark-I</th>
<th>Mark-II</th>
<th>Mark-III</th>
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<td>Bypass</td>
<td>ISLOCA</td>
<td>Yes</td>
<td>Yes</td>
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<td>SGTR</td>
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<td>Yes</td>
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<td>Induced SGTR</td>
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<td>Induced Isol Cond tube failure</td>
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<td>No</td>
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<td>BWR/2&amp;3</td>
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<tr>
<td>CF before VB</td>
<td>Isolation Failure (includes pre-existing leak)</td>
<td>Yes</td>
<td>Yes</td>
<td>Yes</td>
<td>Yes</td>
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<td>Venting</td>
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<td>No</td>
<td>No</td>
<td>Yes</td>
<td>Yes</td>
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<tr>
<td>Over Pressure</td>
<td>Steam</td>
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<td>Yes</td>
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<td></td>
<td>H2 combustion</td>
<td>Yes</td>
<td>Yes (SBO)</td>
<td>inerted</td>
<td>inerted</td>
<td>Yes (SBO)</td>
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<tr>
<td>CF at VB</td>
<td>LP-RCS IVSE (FCI)</td>
<td>Yes</td>
<td>Yes</td>
<td>Yes</td>
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<td></td>
<td>EVSE (FCI)</td>
<td>Yes</td>
<td>Yes</td>
<td>Yes</td>
<td>Yes</td>
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<td></td>
<td>H2 combustion</td>
<td>Yes</td>
<td>Yes</td>
<td>inerted</td>
<td>inerted</td>
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<td>Liner (Shell) Melt-Thru</td>
<td>No</td>
<td>No</td>
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<td>HP-RCS IVSE (FCI)</td>
<td>Yes</td>
<td>Yes</td>
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<td>HPME (RPV blowdown) DCH</td>
<td>Yes</td>
<td>Yes</td>
<td>Yes</td>
<td>Yes</td>
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<td>Steam</td>
<td>Yes</td>
<td>Yes</td>
<td>Yes</td>
<td>Yes</td>
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<td>H2 combustion</td>
<td>Yes</td>
<td>Yes (SBO)</td>
<td>inerted</td>
<td>inerted</td>
<td>Yes (SBO)</td>
</tr>
<tr>
<td></td>
<td>Direct Impingement</td>
<td>Yes</td>
<td>Yes</td>
<td>No</td>
<td>Yes</td>
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<tr>
<td>CF after VB</td>
<td>Venting</td>
<td>No</td>
<td>No</td>
<td>Yes</td>
<td>Yes</td>
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</tr>
<tr>
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<td>Over Pressure (CCI)</td>
<td>Steam</td>
<td>Yes</td>
<td>Yes</td>
<td>Yes</td>
<td>Yes</td>
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<tr>
<td></td>
<td>Non-Cond.</td>
<td>Yes</td>
<td>Yes</td>
<td>Yes</td>
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<td></td>
<td>H2 combustion</td>
<td>Yes</td>
<td>Yes</td>
<td>Yes</td>
<td>Yes</td>
<td>Yes</td>
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<tr>
<td></td>
<td>Basemat melt-thru</td>
<td>Yes</td>
<td>Yes</td>
<td>Yes</td>
<td>Yes</td>
<td>Yes</td>
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<tr>
<td>Scenario</td>
<td>Description</td>
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<td>-------------------------------</td>
<td>-----------------------------------------------------------------------------</td>
<td></td>
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<tr>
<td>Dry Cavity</td>
<td>Some steam produced, but core concrete interaction (CCI) can produce H2 and non-condensible gas</td>
<td></td>
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<tr>
<td>Wet Cavity</td>
<td>coolable geometry: Large amount of steam but no CCI</td>
<td></td>
<td></td>
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<tr>
<td></td>
<td>non-coolable: Steam plus H2 and non-cond. gas (from CCI)</td>
<td></td>
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<tr>
<td>Ice Condenser and Mark III</td>
<td>H2 combustion: possible only if igniters have failed (i.e., SBO)</td>
<td></td>
<td></td>
<td></td>
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</tr>
<tr>
<td>Direct Impingement</td>
<td>Depends on geometry of reactor cavity</td>
<td></td>
<td></td>
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<tr>
<td></td>
<td>[i.e., does a direct path (instrument tunnel) exist for molten core to contact containment wall?]</td>
<td></td>
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<tr>
<td>Over Pressure</td>
<td>Steam - requires failure of containment heat removal (CHR)</td>
<td></td>
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<tr>
<td>IVSE</td>
<td>In-Vessel Steam Explosion (also see alpha-mode, below)</td>
<td></td>
<td></td>
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<td></td>
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<tr>
<td>EVSE</td>
<td>Ex-Vessel Steam Explosion</td>
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<td></td>
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</tr>
<tr>
<td>FCI</td>
<td>Fuel-Coolant Interaction: Such interactions can lead to steam explosions (encompasses both IVSE and EVSE)</td>
<td></td>
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<tr>
<td>alpha-mode</td>
<td>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</td>
<td></td>
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<tr>
<td>BWR/2&amp;e3</td>
<td>Only BWR /2 and early /3 designs include isolation condensers</td>
<td></td>
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</table>
SPAR Level-2 Models (SAPHIRE ver. 7)

April 2016
Work Began July 2008

- July 2008: Work began to develop integrated Level-1/Level-2 SPAR model using SAPHIRE ver. 7
- SOW specified three models
  - Surry
  - Peach Bottom
  - Sequoyah
Level-2 Modeling Relies on Series of Event Trees Linked Together

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

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

<table>
<thead>
<tr>
<th>Core Damage Sequence</th>
<th>Level 1 Core Damage Frequency</th>
<th>Containment Spray System</th>
<th>Containment Spray Recirc (no heat removal)</th>
<th>Cont. Spray Heat Removal (CSS &amp; RHRS)</th>
<th>Low Press Injection Late</th>
<th>Low Press Recirculation Late</th>
<th>Containment Isolation</th>
<th>#</th>
<th>End-States</th>
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<tbody>
<tr>
<td>L1-CDS</td>
<td>L1-CDF</td>
<td>CSS</td>
<td>CSR</td>
<td>CSHR</td>
<td>LPI</td>
<td>LPR</td>
<td>CISO</td>
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</tbody>
</table>

CST-ET - Sequoyah Containment Systems Transfer Event Tree

2009/01/22
CST-ET End States Transfer to PDS-ET

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

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

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

#### Table: Core Damage Sequence and Containment Status

<table>
<thead>
<tr>
<th>Core Damage Sequence</th>
<th>Containment Isolation Status</th>
<th>Containment Bypass</th>
<th>Type of Accident</th>
<th>Station Blackout</th>
<th>AC Power Recovery</th>
<th>Consequential LOCA</th>
<th>Secondary Heat Removal</th>
<th>RCS Pressure at CD</th>
<th>CS Injection of Radioactive Material</th>
<th>Containment Heat Removal via CS</th>
<th>In-Vessel Inject before RPXysis</th>
<th>End-States</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>2</td>
<td>3</td>
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<td>11</td>
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<tr>
<td>CDS</td>
<td>CONSOLIDATE</td>
<td>CONEBYPASS</td>
<td>ACCIDENT</td>
<td>SBO</td>
<td>POWRESS</td>
<td>CONSLOCA</td>
<td>SECHETREM</td>
<td>RCSPRESS</td>
<td>CONTSPRAY</td>
<td>CONTSPRAY</td>
<td>IMESSIONIUY</td>
<td></td>
</tr>
</tbody>
</table>

#### Diagram:

The diagram illustrates the progression of events leading to core damage sequences, containment isolation status, and consequent actions. It covers various scenarios including Core LOCA, Double and Large LOCA, and In-Vessel Inject before RPXysis, leading to end-states.

---

**PDS-ET - Sequoyah Plant Damage State Event Tree**

2008/12/08

April 2016

Accident Progression Analysis (P-300)
PDS-ET Logic Rules Example

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

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

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

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

| Default to No-Bypass |
| else |
| /CONBYPASS = DE-N-NOBYPASS; |
| | complimented, so set to zero |
CONBYPASS[1] = skip(CONBYPASS);
CONBYPASS[2] = skip(CONBYPASS);
CONBYPASS[3] = skip(CONBYPASS);
endif
PDS Serves as an Intermediate Calculation Point

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

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

| 2 |
| CONBYPASS - Containment Bypass |
| Branch[0] = No Bypass (N) |
| Branch[1] = ISLOCA (I) |
| Branch[2] = Large Early Release SGTR (L) |
| Branch[3] = SGTR but not a Large Early Release (S) |

Define Partition Macros (PM) for top event parameters

PM-NO-BYPASS = SYSTEM(/DE-N-NOBYPASS);
PM-ISLOCA = SYSTEM(DE-ISLOCA);
PM-LER-SGTR = SYSTEM(DE-LER-SGTR);
PM-SGTR = SYSTEM(DE-NLR-SGTR);

If PM-NO-BYPASS then
  GlobalPartition = "PDS-?N";
Elsif PM-ISLOCA then
  GlobalPartition = "PDS-?I";
Elsif PM-LER-SGTR then
  GlobalPartition = "PDS-?L";
Elsif PM-SGTR then
  GlobalPartition = "PDS-?S";
Else
  GlobalPartition = "PDS-?Z";
endif
# PDS Results for Sequoyah

<table>
<thead>
<tr>
<th>PDS #</th>
<th>PDS Vector</th>
<th>Frequency</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td></td>
<td><strong>SBO</strong></td>
</tr>
<tr>
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<td><strong>PDS</strong></td>
</tr>
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<td>PDS-INTBENNHNNN</td>
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<td>PDS-INTBELNMNNN</td>
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<td>SBO-24</td>
<td>PDS-INTBNLNMNNN</td>
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April 2016 Accident Progression Analysis (P-300)
Each PDS Vector Becomes an IE

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

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

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<tr>
<td>RCSFAIL</td>
<td>Mode of Induced RCS failure</td>
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<tr>
<td>SGTRPATH</td>
<td>Path of release from SGTR</td>
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<tr>
<td>INVCOOL</td>
<td>Status of core debris cooling in-vessel</td>
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<tr>
<td>CF-EARLY</td>
<td>Mode of Early Cont. Failure</td>
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<tr>
<td>RS-EARLY</td>
<td>Early status of Recirc. Spray</td>
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<tr>
<td>EXVCOOL</td>
<td>Status of core debris cooling ex-vessel</td>
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<tr>
<td>CONHETRE</td>
<td>Status of Cont. Heat Removal</td>
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<tr>
<td>CF-LATE</td>
<td>Mode of Late Cont. Failure</td>
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CET Logic Based on PDS Vector

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

if "PDS-F**" then
  /CONBYPAS = SKIP(CONBYPAS);
  CONBYPAS[1] = SKIP(CONBYPAS);
  CONBYPAS[2] = SKIP(CONBYPAS);
  CONBYPAS[3] = SKIP(CONBYPAS);
  CONBYPAS[4] = SYS-TRUE;
elseif "PDS-?I**" then
  /CONBYPAS = SKIP(CONBYPAS);
  CONBYPAS[1] = SKIP(CONBYPAS);
  CONBYPAS[2] = SYS-TRUE;
  CONBYPAS[3] = SKIP(CONBYPAS);
  CONBYPAS[4] = SKIP(CONBYPAS);
elseif "PDS-?L**" then
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  CONBYPAS[2] = SYS-TRUE;
  CONBYPAS[3] = SKIP(CONBYPAS);
  CONBYPAS[4] = SKIP(CONBYPAS);
elseif "PDS-?S**" then
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  CONBYPAS[1] = SKIP(CONBYPAS);
  CONBYPAS[2] = SKIP(CONBYPAS);
  CONBYPAS[3] = SYS-TRUE;
  CONBYPAS[4] = SKIP(CONBYPAS);
else
  /CONBYPAS = SYS-FALSE;
  CONBYPAS[1] = SKIP(CONBYPAS);
  CONBYPAS[2] = SKIP(CONBYPAS);
  CONBYPAS[3] = SKIP(CONBYPAS);
  CONBYPAS[4] = SKIP(CONBYPAS);
endif
CET End States Transfer to STC-ET

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

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<td>Medium Early</td>
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7. Severe Accident Simulation Codes

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

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

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<th>Developer/Sponsor</th>
<th>Design Philosophy</th>
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<td>ISL/NRC/United States INL/DOE</td>
<td>Detailed mechanistic models</td>
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<td>SCDAP/RELAP5-3D®</td>
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<td>FAI / EPRI/ United States</td>
<td>Simplified, parametric models</td>
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*Presentation focuses on US severe accident analysis codes.*
Approximate Accident Phenomena Covered by U.S. Severe Accident Computer Codes

Integrated Codes
- MAAP
- MELCOR

Detailed Mechanistic Codes
- SCDAP/RELAP5-3D
- VICTORIA
- CONTAIN

Accident Progression Phenomena
- Thermal hydraulics
- Core melting
- Release from fuel
- Transport in RCS
- RCS failure
- Concrete interactions
- Release from debris
- Transport in containment
- Containment loads
- Containment performance
- Off-site consequences
SCDAP/RELAP5-3D© Embodies Understanding of Severe Accident Processes

Model Development and Assessment Based on Data from:
- LOFT
- PBF
- CORA/Quench
- RASPLAV

Experiments and Analyses

SCDAP/RELAP5-3D©

Applications

Severe Accident Resolution
(DCH, SGTR)

Severe Accident Mitigation Strategies
(Depressurization, Water Addition)

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

DOE Research Reactors (ATR, etc.)

LWR

EPR

AP600

Existing LWRs

Indian Point

Surry

TMI-2

Non-LWR

GENIV Reactors
(NGNP, etc.)

CANDU Reactors

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

- Thermal hydraulic behavior
- Fission product release, hydrogen production, heat generation, and geometry
- Interphase/field mass transfer
- Coolant temperatures, flows, and composition; convective and radiative heat transfer
- Coolant temperatures, flows, and compositions
- Heat generation
- Surface geometry and temperature
- Fuel rod, control rod, debris, vessel, and structure behavior
- Radionuclide deposition and decay (with VICTORIA Interface)
PVM linkage provides options not available with other analysis tools
CONTAIN provides mechanistic containment analyses tool.

**AEROSOLS**
- Particle size distribution
- Material composition
- Deposition

**FISSION PRODUCTS**
- Radioisotope inventory
- Decay and heating
- Release and acceptance

**THERMAL HYDRAULICS**
- Gas and liquid flow
- Heat transfer
- Thermodynamics
- Engineered safety features
- Debris fields

- Heat to gas, walls, pool
- Intercell transport
- Distribution of fission products
- Transport of gas or fission products
- Evaporable coolant inventory
- Deposition/ agglomeration rates
MELCOR Code Physics Description

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

**Generic Models** (no “built-in” nodalization)

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

4/19/2016 Accident Progression Analysis (P-300)
MELCOR User Interface
MELCOR Models Fission Facilities

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

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

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

- Aerosols and fission products are transported both in the vapor and liquid phases.
MELCOR Models Fission Facilities (cont.)

A core model for fuel/cladding response

A cavity model for debris concrete reactions (dry well below RPV)
MELCOR Models Fission Releases and Transport

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

- Aerosol phenomenological sources from other packages, such as release from fuel rods or during core-concrete interactions, and/or arbitrary user-specified sources;
- Condensation and evaporation of water and fission products to and from aerosol particles;
- Particle agglomeration (or coagulation), whereby two particles collide and form one larger particle;
- Particle deposition onto surfaces or settling through flow paths into lower control volumes;
- Advection of aerosols between control volumes by bulk fluid flows
- Removal of aerosol particles by Engineered Safety Features (ESFs), such as filter trapping, pool scrubbing, and spray washout
MELCOR Models Fission Releases and Transport (cont.)

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

<table>
<thead>
<tr>
<th>Reaction</th>
<th>Notes</th>
</tr>
</thead>
<tbody>
<tr>
<td>$\text{CsI(g)} \rightarrow \text{CsI(ad)}$</td>
<td>rate constant for adsorption is supplied through input</td>
</tr>
<tr>
<td>$\text{CsI(ad)} \rightarrow \text{CsOH(ad)} + \text{H}_2\text{O(s)}$</td>
<td>instantaneous and complete transfer between classes when water is present. Note that the water mass is not included in the model; water mass is not explicitly conserved.</td>
</tr>
<tr>
<td>$\text{CsOH(g)} \rightarrow \text{CsOH(ad)}$</td>
<td>rate constant for adsorption supplied or condensation limited</td>
</tr>
<tr>
<td>$\text{CsOH(ad)} \rightarrow \text{CsOH(g)}$</td>
<td>reaction with zero rate constant below $T_1$</td>
</tr>
<tr>
<td>$\text{H}_2\text{O(s)} \leftrightarrow \text{H}_2\text{O(g)}$</td>
<td>positive value or instantaneous above $T_1$, controlled by condensation/evaporation</td>
</tr>
</tbody>
</table>
A MELCOR Model of a BWR that includes Reactor Building, Plus All Emergency Cooling Systems was used for analyzing Fukushima Unit 1

- Reactor Service Floor (Steel Construction)
- Concrete Reactor Building (secondary Containment)
- Reactor Core
- Reactor Pressure Vessel
- Containment (Dry well)
- Containment (Wet Well) / Condensation Chamber
- Fresh Steam line
- Main Feedwater
- Spent Fuel Pool
MELCOR Core Zones Modeled

Axial Zone

Top Guide & Upper Tie Plate

Height of Fuel Assemblies

Heated Fuel Length

Lower Tie Plate & Nose Pieces

Lower Core Plate & "Elephant's Foot"
Time = 45 hrs
Fukushima Unit 1 MELCOR Calculated/Defined
RPV Water Injection Rate-SAND2012-6173
Fukushima Unit 1 Fuel Temperatures – SAND2012-6173

Temperature step-change to 0 K occurs when fuel begins to relocate.
Fukushima Unit 1 Hydrogen Generation from Cladding Stainless Steel and B4C – SAND2012-6173
Fukushima Unit 1 Lower Head Fuel Temperatures – SAND2012-6173

Graph showing temperature over time with a dashed line indicating the steel melting temperature.
Fukushima Unit 1 Accumulation of Fuel in Lower Plenum 139,000 kg on concrete – SAND2012-6173
Core Mass Draining from Vessel

- UO₂: 68,700 kg
- Zr: 25,500 kg
- ZrO₂: 16,500 kg
- SS: 28,700 kg
- SS-Ox: 260 kg
- Total: 139,600 kg

Graph showing the mass drainage of different materials over time (in hours). The materials are plotted against time (hr) on the x-axis and mass (kg) on the y-axis. The materials include UO₂, Zr, ZrO₂, stainless steel, and s.s. oxide.
Fukushima Unit 1 Core Condition –SAND2012-6173

- Debris accumulating on lower head, before lower head failure
- After lower head failure

- TAF
- BAF
- Lower core plate
- Lower plenum

4/19/2016  Accident Progression Analysis (P-300)
Fukushima Unit 1 MCCI Interaction–SAND2012-6173

4/19/2016 Accident Progression Analysis (P-300)
Fukushima Unit 1 MCCI and other hydrogen production – SAND2012-6173
Fukushima Unit 1 CsI Most Retained in Suppression pool with at 1-2% Release – SAND2012-6173
Fukushima Unit 2 Significant Fuel Melt – SAND2012-6173
Fukushima Unit 2 Significant Fuel Damage and Fission Product Release – SAND2012-6173
Fukushima Unit 3 hydrogen production little damage to fuel – SAND2012-6173
Fukushima Unit 3 Fuel Retained in the RPV retained damage 58% of Noble Gases Released –SAND2012-6173
MELCOR Role Evolving

- Original role for PRAs required simpler, fast-running code
  - Uncertainties assessed through sensitivity studies
  - Substantial user flexibility allowed for parametric studies

- Recent role uses more detailed models
  - NRC consolidating to one code
  - Assessments against more detailed codes used to determine required model complexity
  - More mechanistic models implemented as necessary

- Recent role using more flexible modeling geometry
  - More generic modeling without “built-in” nodalization
  - Control volume approach used to define plant system

- Application NOT limited to LWR reactor accident analysis
MELCOR Modeling Improvements Assessed with Mechanistic Codes

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

- Severe accident response and source term
- Containment response to design basis accident
MAAP Designed for Full-Plant Calculations

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

- Natural Circulation
- Heat Transfer to Upper Plenum
- Upper Head Injection
- Condensation
- Letdown Flow
- ESF Injection
- Core Heatup & Melt Progression
- Natural Circulation
- Zr/H₂O Reaction
- Fission Products Decay
- Loss through Break in Unbroken Loop
- Loss through Break in Broken Loop
- Heat Transfer from Core
- Heat Transfer from Debris in Lower Plenum
- Fluid Loss through RPV Penetration
- Heat Transfer to Surge Line
- PZR Sprays
- PZR Heaters
- Release through PORV
- Primary System Fluid Volume Change
- Release through High Pressure Vent

4/19/2016 Accident Progression Analysis (P-300)
MAAP Modeled Phenomena
Representative MAAP PWR Analysis Considers Gas Nodes, Heat Structures, and Water Nodes

- Cold Leg Steam Generator Shell
- Hot Leg Steam Generator Shell
- Cold Leg Tubes
- Hot Leg Tubes
- Pressurizer
- Reactor Dome
- Upper Plenum
- Core Node
- Downcomer
- Intermediate Leg

3 ‘Unbroken’ Loops
1 ‘Broken’ Loop (Nodalization Same as Unbroken Loop)
MAAP4 Melt Progression Phenomena

<table>
<thead>
<tr>
<th>Temperature (°F)</th>
<th>Temperature (K)</th>
<th>Phenomenon</th>
</tr>
</thead>
<tbody>
<tr>
<td>5144</td>
<td>3113</td>
<td>Melting of UO₂</td>
</tr>
<tr>
<td>4780</td>
<td>2911</td>
<td>Melting of ZrO₂</td>
</tr>
<tr>
<td>4670</td>
<td>2850</td>
<td>Melting of U-Zr-O Ceramic</td>
</tr>
<tr>
<td>4400</td>
<td>2700</td>
<td>Melting of B₄C</td>
</tr>
<tr>
<td>4346</td>
<td>2670</td>
<td>Formation of α - Zr(O)/UO₂ &amp; U/UO₂ Monotectics</td>
</tr>
<tr>
<td>3581</td>
<td>2245</td>
<td>Melting of α - Zr(O)</td>
</tr>
<tr>
<td>3446</td>
<td>2170</td>
<td>Formation of α - Zr(O)/UO₂ Eutectics</td>
</tr>
<tr>
<td>3365</td>
<td>2125</td>
<td>Melting of Zircaloy-4</td>
</tr>
<tr>
<td>2600</td>
<td>1700</td>
<td>Melting of Stainless Steel (SS)</td>
</tr>
<tr>
<td>2510</td>
<td>1650</td>
<td>Melting of Inconel</td>
</tr>
<tr>
<td>2240</td>
<td>1500</td>
<td>Inconel/Zircaloy Liquefaction; B₄C-SS Interaction</td>
</tr>
<tr>
<td>1736</td>
<td>1220</td>
<td>Formation of Fe/Zr &amp; Ni/Zr Eutectics</td>
</tr>
<tr>
<td>1520</td>
<td>1100</td>
<td>Melting of Ag-In-Cd</td>
</tr>
<tr>
<td>1430</td>
<td>1050</td>
<td></td>
</tr>
</tbody>
</table>
MAAP Considers Unique BWR RCS Phenomena
MAAP Modeled BWR Containment Phenomena
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:

<table>
<thead>
<tr>
<th>Phenomenon</th>
<th>SCDAP/RELAP5-3D</th>
<th>MAAP</th>
<th>MELCOR</th>
</tr>
</thead>
<tbody>
<tr>
<td>RCS Depressurization</td>
<td>Ransom/Trapp critical flow model (results consistent with ROSA/AP600 data)</td>
<td>Single phase critical flow model (unexplained mass retained in RCS)</td>
<td>Two-phase critical flow model (with user supplied discharge coefficients)</td>
</tr>
<tr>
<td>Model</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Fuel melting</td>
<td>At 2870 K / 4710 °F due to eutectic formation</td>
<td>At 3100 K / 5120 °F (UO$_2$ melting temperature)</td>
<td>At user-specified temperature.</td>
</tr>
<tr>
<td>Hydrogen generation</td>
<td>Throughout core degradation</td>
<td>Until first relocation</td>
<td>Until cladding failure temperature.</td>
</tr>
<tr>
<td>Relocation to vessel</td>
<td>If crust cannot support molten material</td>
<td>When melting temperature is predicted</td>
<td>When fuel melting occurs, material relocates to core plate and is retained until core plate reaches user-specified temperature.</td>
</tr>
<tr>
<td>Debris-to-vessel heat</td>
<td>No enhanced debris cooling (model developed and data now available)</td>
<td>Enhanced cooling from water in user-specified gaps with user-specified heat transfer</td>
<td>No enhanced debris cooling (model developed, and data now available)</td>
</tr>
<tr>
<td>transfer</td>
<td></td>
<td></td>
<td></td>
</tr>
</tbody>
</table>
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.

\[ Temperature \ (°F) \]

\[ Temperature \ (K) \]

\[ Time \ (s) \]

Lower head debris bed begins to form
In-core pool begins superheating
First in-core pool formation

MELCOR
MAAP
SCDAP
MAAP and MELCOR predict much lower total hydrogen generation.
Case Study 2

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

MELCOR and MAAP predict lower debris heat load on vessel wall

*Vessel Temperature (K)*

- 4000
- 3000
- 2000
- 1000
- 0

*Temperature (°F)*

- 4000
- 3000
- 2000
- 1000
- 0

Time (s)

MELCOR and MAAP predict lower debris heat load on vessel wall
Summary and Discussion

• Selection of mature US severe accident analysis codes available.
  – Codes differ in modeling approaches
  – Codes have undergone fairly extensive code-to-data comparisons.
  – Insights from code calculations have played a key role in resolving accident management issues

• Analysis reviews must consider impact of code modeling assumptions and approaches
Regulatory Considerations (SECY-93-087)

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

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

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

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

Uncertainty Convolution: Deterministic vs. Probabilistic

Deterministic Treatment

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

p = 95%

Probabilistic Treatment

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

p = 100%

p = 95%

4/19/2016

Accident Progression Analysis (P-300)
Severe Accident Phenomena (EPR)

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

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

4/19/2016  Accident Progression Analysis (P-300)
CGCS Analysis: Tolerance Limit of H2 Concentration

- Licensing limit is 10%
Study Questions

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

Code References


References

Code References (continued)


• “MELCOR Project Status NRC Severe Accident Code Consolidation,” presented at USNRC CSARP-2001 Meeting, May 7-9, 2001, Bethesda, Maryland.


• 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, 2002, Arlington, VA.

References (continued)

Code References (continued)


Tier 2 NRC Recommendations

• Spent fuel pool makeup capability *(Recommendation 7.2, 7.3, 7.4, and 7.5)*
• Emergency preparedness regulatory actions *(Recommendation 9.3)*
• Other External Hazards Reevaluation (tornados, hurricanes, drought, etc.) *(additional Issue)*
Tier 3 NRC Recommendations

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

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

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

- Identify and understand factors affecting radionuclide release and transport during a severe accident.
- Identify and describe differences between various methods and approaches used to estimate severe accident releases.
Inventory Characterized in Terms of Decay Rates

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

- 1 Becquerel (Bq) = 1 dps, or
- 1 Ci = $3.7 \times 10^{10}$ Bq
Categories of Fission Product Inventory

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

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

(England & Rider, 1994)
(Rudling and Peterson -2012)
Most Volatile Radionuclides Reside in Reactor Core

<table>
<thead>
<tr>
<th>Location</th>
<th>Inventory, Ci</th>
<th></th>
<th></th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Noble Gases</td>
<td>Iodine (I)</td>
<td></td>
</tr>
<tr>
<td></td>
<td>(Xe, Kr)</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Core</td>
<td>4.0E+8</td>
<td>7.5E+8</td>
<td></td>
</tr>
<tr>
<td></td>
<td>(1.48E+19 Bq)</td>
<td>(2.775E+19 Bq)</td>
<td></td>
</tr>
<tr>
<td>Gap between UO₂ fuel and Zr cladding</td>
<td>3.0E+7</td>
<td>1.4E+7</td>
<td></td>
</tr>
<tr>
<td></td>
<td>(1.11E+18 Bq)</td>
<td>(6.29E+17 Bq)</td>
<td></td>
</tr>
<tr>
<td>Spent fuel storage pool</td>
<td>1.0E+6</td>
<td>5.0E+5</td>
<td></td>
</tr>
<tr>
<td></td>
<td>(3.7E+16 Bq)</td>
<td>(5.18E+15 Bq)</td>
<td></td>
</tr>
<tr>
<td>Primary coolant³</td>
<td>1.0E+4</td>
<td>6.0E+2</td>
<td></td>
</tr>
<tr>
<td></td>
<td>(3.7E+14 Bq)</td>
<td>(2.22E+13 Bq)</td>
<td></td>
</tr>
</tbody>
</table>

³Nominal value, varies depending on fuel leakage.
### Average Annual Plant Release Considerably Lower than Accident Releases

<table>
<thead>
<tr>
<th></th>
<th>Noble Gases, Ci</th>
<th>Iodine, Ci</th>
</tr>
</thead>
<tbody>
<tr>
<td>Average annual reactor release (1975-1979)</td>
<td>1.00 (3.7E+10 Bq)</td>
<td>0.13 (4.81E+9 Bq)</td>
</tr>
<tr>
<td>TMI-2 accident (March 1979)</td>
<td>2.50E+6 (9.25E+16 Bq)</td>
<td>15 (5.55E+11 Bq)</td>
</tr>
<tr>
<td>Chernobyl accident (April 1986)</td>
<td>1.90E+8 (7.03E+18 Bq)</td>
<td>4.5E+7 (1.665E+18 Bq)</td>
</tr>
</tbody>
</table>
Radionuclide Inventory Time-Dependent

\[ \frac{dA_i(t)}{dt} = -\Lambda_i(t)A_i(t) + Q_n(t) \]

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

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

1 Group definitions vary in different approaches.
2 For representative large (3300 MWt) LWR 30 minutes after shutdown.
*In highly oxidizing environment, Ru is volatile
Radiological Impact of Isotopes Differ-
Overall Exposure of 600 Rem or 6 Sv
Considered Potentially Fatal

Characterization

Early Bone Marrow Dose
24 hour exposure

Early Lung Dose

Total Latent
Cancer deaths

Assumes unit release of each element.
### Representative Isotope Used to Characterize Group Decay

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

- Fuel Release
  - Oxidation
  - Temperature-induced
  - Revolatilization and revaporization

- RCS
  - Containment bypass leakage
  - Decay
  - Deposition

- Containment
  - Leakage
  - Deposition
  - Decay

- Environment
  - Leakage
  - MCCI release
  - Revolatilization and revaporization

Phenomena
AP1000 Radionuclide Containment

AP1000 Passive Containment Cooling System

<table>
<thead>
<tr>
<th>S/G's</th>
<th>Rx Clnt</th>
<th>Containment</th>
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</thead>
<tbody>
<tr>
<td>I-131 Eq 6.24E+03 Ci/gm</td>
<td>Kr-87 Eq 2.51E+03 Ci/gm</td>
<td>Cladding Failure 0.99%</td>
</tr>
<tr>
<td>I-131 Eq 3.08E+00 Ci/gm</td>
<td>Kr-87 Eq 5.08E+00 Ci/gm</td>
<td></td>
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</table>

<table>
<thead>
<tr>
<th>Iodine</th>
<th>Noble Gas</th>
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</thead>
<tbody>
<tr>
<td>1.13E-05 Ci/s</td>
<td>6.177E-06 Ci/s</td>
</tr>
<tr>
<td>RM3</td>
<td>3.3E+00 mCi/h</td>
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</table>

<table>
<thead>
<tr>
<th>Iodine</th>
<th>Noble Gas</th>
</tr>
</thead>
<tbody>
<tr>
<td>1.00E+00 Ci/s</td>
<td>0.00E+00 Ci/s</td>
</tr>
<tr>
<td>RM1</td>
<td>1.10E+00 mCi/h</td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>Pressure</th>
<th>Temperature</th>
</tr>
</thead>
<tbody>
<tr>
<td>14.7 bar</td>
<td>220°C</td>
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</table>

<table>
<thead>
<tr>
<th>Iodine</th>
<th>Noble Gas</th>
</tr>
</thead>
<tbody>
<tr>
<td>1.81E-08 Ci/s</td>
<td>0.00E+00 Ci/s</td>
</tr>
<tr>
<td>RM1</td>
<td>1.10E+00 mCi/h</td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>Iodine</th>
<th>Noble Gas</th>
</tr>
</thead>
<tbody>
<tr>
<td>0.00E+00 Ci/s</td>
<td>0.00E+00 Ci/s</td>
</tr>
<tr>
<td>RM1</td>
<td>1.10E+00 mCi/h</td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>Iodine</th>
<th>Noble Gas</th>
</tr>
</thead>
<tbody>
<tr>
<td>1.40E-03 Ci/s</td>
<td>0.00E+00 Ci/s</td>
</tr>
<tr>
<td>RM1</td>
<td>1.10E+00 mCi/h</td>
</tr>
</tbody>
</table>

4/19/2016 Accident Progression Analysis (P-300)
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
## Phenomena

### Plant Features Significantly Reduce Release

<table>
<thead>
<tr>
<th>Design Feature</th>
<th>Decontamination Factor$^1$</th>
</tr>
</thead>
<tbody>
<tr>
<td>Containment Sprays</td>
<td>100 to 1000</td>
</tr>
<tr>
<td>Ice Condensers</td>
<td>1 to 20 with ice present</td>
</tr>
<tr>
<td>Suppression pools</td>
<td>1 to 4000</td>
</tr>
<tr>
<td>Overlying water layers</td>
<td>1 to 4000</td>
</tr>
</tbody>
</table>

$^1$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 I₂ formation and vaporization (if heated) at later time periods.
Several Methods Available for Estimating Severe Accident Release

- Detailed methods
  - MELCOR
  - SCDAP/RELAP5/VICTORIA/CONTAIN
  - MAAP
- Less-detailed methods
  - TID
  - XSOR
  - Parametric Source Term (PST)
  - Alternate Approach (Revised Source Term, RST, or Alternate Source Term from NUREG-1465)
Source Terms Initially Based on TID-14844

• Based on a postulated core melt accident and 1962 understanding of fission product behavior.
• As codified in Reg. Guides 1.3 and 1.4, assumed source term consists of an instantaneous release of:
  – 100% of core inventory of noble gases
  – 50% of core inventory of iodine
    • half assumed to subsequently deposit on containment surfaces
    • 91% elemental, 5 % particulate, and 4% organic
• Assumed source term affected the site selection process and the design of engineered safety features, such as containment isolation valves, containment sprays, and filtration systems.
NUREG-1150 Release and Transport Estimated with XSOR Codes

• Developed for five NUREG-1150 plants

• Doesn't consider knowledge gained from severe accident research since 1990.

• XSOR method decomposed source term into release fractions for various time periods and release barriers and quantified release fractions using expert opinion
  – Approach is time-consuming.
  – Approach isn’t reproducible.
NUREG-1465 Proposes More Realistic Source Term

- Developed more realistic source term for regulating future LWRs and for evaluating proposed changes to existing plants
  - Considers chemical and physical form
  - Provides safety and cost benefits
- Releases based on severe accident research and range of PWR and BWR STCP, MAAP, and MELCOR calculations
  - Comparisons with MELCOR comparisons suggest considerable margin between RST and best-estimate MELCOR predictions.
- Proposes time-dependent releases grouped into five phases:
  - DBA source term considers coolant, gap, and early-in-vessel releases
  - Severe accident source term considers coolant, gap, early in-vessel, ex-vessel, and late ex-vessel releases
- Implementation requires revised Part 20 dose methodology (TEDE criterion) and evaluate dose for accident’s “worst two hour interval.”
- Codified in Regulatory Guide 1.183
NUREG-1465 provides Time-dependent Releases

<table>
<thead>
<tr>
<th></th>
<th>Gap and Coolant</th>
<th>Early In-vessel</th>
<th>Ex-Vessel</th>
<th>Late Ex-vessel</th>
</tr>
</thead>
<tbody>
<tr>
<td>Duration, hours</td>
<td>0.5</td>
<td>1.3</td>
<td>2.0</td>
<td>10.0</td>
</tr>
<tr>
<td>Noble gases</td>
<td>0.05</td>
<td>0.95</td>
<td>0</td>
<td>0</td>
</tr>
<tr>
<td>Halogens(^1)</td>
<td>0.05</td>
<td>0.35</td>
<td>0.25</td>
<td>0.01</td>
</tr>
<tr>
<td>Alkali metals</td>
<td>0.05</td>
<td>0.25</td>
<td>0.35</td>
<td>0.01</td>
</tr>
<tr>
<td>Tellurium group</td>
<td>0</td>
<td>0.05</td>
<td>0.25</td>
<td>0.005</td>
</tr>
<tr>
<td>Barium, strontium</td>
<td>0</td>
<td>0.02</td>
<td>0.1</td>
<td>0</td>
</tr>
<tr>
<td>Noble Metals</td>
<td>0</td>
<td>0.0025</td>
<td>0.0025</td>
<td>0</td>
</tr>
<tr>
<td>Lanthanides</td>
<td>0</td>
<td>0.0002</td>
<td>0.005</td>
<td>0</td>
</tr>
<tr>
<td>Cerium group</td>
<td>0</td>
<td>0.0005</td>
<td>0.005</td>
<td>0</td>
</tr>
</tbody>
</table>

\(^1\)If coolant pH greater than or equal to 7, then 95% particulate, ~5% elemental and ~0.15% organic.
Pilot plant applications demonstrate that RST reduces regulatory requirements and enhances safety

• Time-dependent source term allows:
  – delayed automatic isolation function for containment isolation valves
  – increased allowable containment and/or penetration leakage rates

• Realistic iodine chemical species allows:
  – relaxation of charcoal filtration system requirements
  – relaxation of control room habitability requirements
  – requirements for post-accident pH control of iodine particulates dissolved in water (to prevent elemental iodine formation).
SOARCA

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

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

References for Additional Reading


References for Additional Reading (continued)


Accident Progression Analysis (P-300)

9. PRA Integration and Quantification
Session Objectives

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

• Integration of Level-1 and Level-2
• Uncertainty
• Level-2 Results
Level-1/2 PRA Integration

IEs
RxTrip
LOCA
LOSP
SGTR
etc.

Level-1 Event Tree

Bridge Event Tree (containment systems)

Level-2 Containment Event Tree

Containment failure modes and source terms (to Level-3 analysis)
Level-1/Level-2 Analysis Approach

• Assignment of core damage (CD) sequences into appropriate plant damage state (PDS) bins

• Assessment of challenges associated with each PDS bin (typically using computer codes)

• Characterization of the containment’s capacity to withstand the identified challenges (i.e., fragility)

• Combining the uncertainties associated with the previous two analyses to estimate probability of containment failure (for a given PDS)

• Combining the uncertainties associated with CD frequency with those associated with conditional containment failure probabilities to estimate containment failure frequency
Level-1 CD Sequences Mapped Into PDSs

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

• Sometimes called “binning” of CD sequences
• Bridge Tree typically straightforward extension/expansion of Level-1 event trees
  – Extends consideration beyond core damage
  – Determines status of containment systems
• Every core damage sequence propagated through bridge tree
<table>
<thead>
<tr>
<th>Core Damage</th>
<th>Emergency Coolant Injection</th>
<th>Emergency Coolant Recirculation</th>
<th>Containment Spray Injection</th>
<th>Containment Spray Recirculation</th>
<th>Cooling to RCP Seals</th>
<th>Auxiliary Feedwater to SG</th>
<th>#</th>
<th>PDS</th>
</tr>
</thead>
<tbody>
<tr>
<td>CD</td>
<td>ECI</td>
<td>ECR</td>
<td>CSI</td>
<td>CSR</td>
<td>RCP-SL</td>
<td>AFW</td>
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</tr>
</tbody>
</table>
Each CD Cut Set Unique

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

• To accommodate different impacts on Level-2 analysis, each CD cut set explicitly mapped into a PDS (sometimes referred to as binning)

• Two approaches to binning Level-1 cut sets into PDSs
  – Two step process (often performed using “If-Then” rules)
    1 - assign PDS vector identifier to each CS
    2 - map CS into PDS based on best match of vector
  – One step process (often manually performed)
    • Directly bin each CS into a PDS (this process does not necessarily need the vector framework)
Simple Binning Example

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

\[ S_2D = IE-S_2 \times ECCS-Pump-F + IE-S_2 \times LOSP \times EAC-F. \]
<table>
<thead>
<tr>
<th></th>
<th>Simple PDS Scheme for PWR (Status of ...)</th>
</tr>
</thead>
</table>
|   | **RCS integrity at start of CD** | **I** – Intact  
|   |   | **S** – Small hole  
|   | **ECCS** | **A** – Available  
|   |   | **U** – Unavailable  
|   | **CHR** | **A** – Available  
|   |   | **U** – Unavailable  
|   | **AC Power** | **A** – Available  
|   |   | **U** – Unavailable  
|   | **RWST** | **A** – Available for injection  
|   |   | **I** – Injected into containment  
|   |   | **U** – Unavailable for injection  
|   | **Heat Removal from S/G** | **A** – Available  
|   |   | **U** – Unavailable  
|   | **RCP seal cooling** | **A** – Available  
|   |   | **U** – Unavailable  
|   | **Containment Fan Coolers** | **A** – Available  
|   |   | **U** – Unavailable  

April 2016 Accident Progression Analysis (P-300) 09 - 12
Different PDS Vectors for CS#1 and CS#2

<table>
<thead>
<tr>
<th></th>
<th>1 RCS</th>
<th>2 ECCS</th>
<th>3 CHR</th>
<th>4 AC</th>
<th>5 RWST</th>
<th>6 S/G</th>
<th>7 RCP seals</th>
<th>8 Fans</th>
</tr>
</thead>
<tbody>
<tr>
<td>CS#1</td>
<td>S</td>
<td>U</td>
<td>A</td>
<td>A</td>
<td>A</td>
<td>A</td>
<td>A</td>
<td>U</td>
</tr>
<tr>
<td>CS#2</td>
<td>S</td>
<td>U</td>
<td>U</td>
<td>U</td>
<td>A</td>
<td>A*</td>
<td>U</td>
<td>U</td>
</tr>
</tbody>
</table>

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

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

- Uncertainty analysis (i.e., Monte Carlo or Latin Hypercube) generates probability histogram for each PDS
- Each PDS then used as input to (i.e., serves as the initiating event) the CET
  - CET can be manually tailored for each PDS
    - Each PDS associated with a unique CET
      - Note that vector framework NOT necessary
    - Single “general-purpose” CET can be modified during processing
      - Incorporates various “If-Then” logic rules
        - Vector framework not absolutely necessary but very useful
**PDSs Are Level-2 “Initiating Events”**

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

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

<table>
<thead>
<tr>
<th>Plant Damage State</th>
<th>Vessel Breach</th>
<th>Containment Cooling</th>
<th>Early Containment Failure</th>
<th>Late Containment Failure</th>
<th>#</th>
<th>CET-ES</th>
</tr>
</thead>
<tbody>
<tr>
<td>PDS</td>
<td>VB</td>
<td>CC</td>
<td>ECF</td>
<td>LCF</td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

**Event Tree Diagram**

```
1  2  3  4  5  6  7
LERF
LERF
```
Split Fractions for LERF-ET

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

Quantify LERF
What is conditional probability of LER given S2D1?
CET Output Organized

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

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

- Classical
  - Requires a statistical basis
  - Generates confidence intervals only (not probability distributions)

- Bayesian
  - Implies a degree of belief
    - able to accommodate sparse data and engineering judgement
  - Needed to produce and propagate probability distributions in a PRA (i.e., all PRAs employ Bayesian techniques and interpretations)
Uncertainty Often Classified by Type

- **Aleatory** - Stochastic, random or tolerance uncertainty
  - A product of the assumed model
    - i.e., a binomial or Poisson process
  - Can also include variability in boundary conditions

- **Epistemic** - State of knowledge, subjective or confidence uncertainty
  - A produced by a lack of data
    - Similar to a classical statistical confidence
    - Bayesian interpretation is the degree of belief
Aleatory Uncertainties

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

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

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

• Analytical methods simply not feasible
• Monte Carlo or Latin Hypercube are currently the only practical approaches to propagating uncertainty
  – Select random values from input parameter distributions, quantify model, repeat many times
    • repeating mathematical “experiment” over and over produces a frequency histogram on the output
• Quantification done step-wise
  – Distributions on intermediate results (e.g., CDF or PDS) are then inputs to subsequent steps
Example Monte Carlo Sampling (5 Samples) on input parameter $\lambda$

![Diagram showing Monte Carlo sampling of $\lambda$ with cumulative probability distribution.](image-url)
Latin Hypercube Sampling (one $\lambda$ selected from each equal-probability area)

$\text{CumPr}(\lambda < \lambda')$

Equal probability regions

$\lambda$
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

\[
\begin{align*}
\text{Probability of Early Containment Failure Given Total Core Damage} & = \frac{(0 \times 1E-5 + .5 \times 1E-4)}{1.1E-4} = .4 \\
\text{Probability of Containment Bypass Given Total Core Damage} & = \frac{(1 \times 1E-5 + 1 \times 1E-4)}{1.1E-4} = .2 \\
\text{Probability of Late Containment Failure Given Total Core Damage} & = \frac{(0 \times 1E-5 + .2 \times 1E-4)}{1.1E-4} = .2 \\
\text{Probability of No Containment Failure Given Total Core Damage} & = \frac{(0 \times 1E-5 + .2 \times 1E-4)}{1.1E-4} = .2 \\
\end{align*}
\]
Two Measures Typically Cited for Assessing Containment Performance

Conditional Containment Failure Probability

\[ \text{CCFP} = \sum_{i=1}^{n} \frac{S_i}{C_i} \]

\[ S_i = \text{frequency for accident sequence, } i \]

\[ C_i = \text{containment conditional failure probability given accident sequence, } i \]

\[ n = \text{total number of accident sequences} \]

Containment Failure Frequency

\[ \text{CFF} = \sum_{i=1}^{n} S_i C_i \]
**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
### ACCIDENT PROGRESSION BIN

<table>
<thead>
<tr>
<th>Event Description</th>
<th>LOSP Frequency</th>
<th>ATWS Frequency</th>
<th>Transients Frequency</th>
<th>LOCAs Frequency</th>
<th>Bypass Frequency</th>
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<tbody>
<tr>
<td>VB, early CF (during CD)</td>
<td>0.014</td>
<td>0.003</td>
<td>0.002</td>
<td></td>
<td>0.005</td>
</tr>
<tr>
<td>VB, alpha, early CF (at VB)</td>
<td>0.002</td>
<td>0.003</td>
<td>0.002</td>
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<td>0.002</td>
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<tr>
<td>VB &gt; 200 psi, early CF (at VB)</td>
<td>0.064</td>
<td>0.023</td>
<td>0.014</td>
<td>0.031</td>
<td>0.035</td>
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<tr>
<td>VB &lt; 200 psi, early CF (at VB)</td>
<td>0.054</td>
<td>0.020</td>
<td>0.004</td>
<td>0.014</td>
<td>0.023</td>
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<td>VB, late CF</td>
<td>0.153</td>
<td>0.001</td>
<td>0.001</td>
<td></td>
<td>0.038</td>
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<tr>
<td>VB, BMT, very late CF</td>
<td>0.065</td>
<td>0.151</td>
<td>0.039</td>
<td>0.260</td>
<td>0.171</td>
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<tr>
<td>Bypass</td>
<td>0.001</td>
<td>0.134</td>
<td>0.006</td>
<td></td>
<td>0.996</td>
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<tr>
<td>VB, No CF</td>
<td>0.200</td>
<td>0.471</td>
<td>0.137</td>
<td>0.301</td>
<td>0.269</td>
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<tr>
<td>No VB, early CF (during CD)</td>
<td>0.038</td>
<td>0.001</td>
<td>0.005</td>
<td>0.002</td>
<td>0.011</td>
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<tr>
<td>No VB</td>
<td>0.384</td>
<td>0.171</td>
<td>0.785</td>
<td>0.367</td>
<td>0.371</td>
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</tbody>
</table>

**PLANT DAMAGE STATE**

(Mean Core Damage Frequency)

- **LOSP**: (1.36E-05)
- **ATWS**: (2.07E-06)
- **Transients**: (2.32E-06)
- **LOCAs**: (3.52E-05)
- **Bypass**: (2.39E-06)
- **Frequency Weighted Average**: (5.58E-05)

**BMT = Basemat Meltdown**

**CF = Containment Failure**

**VB = Vessel Breach**

**CD = Core Degradation**
NUREG-1150 Results Indicate BWR Early Containment Failures More Likely

NUREG-1150 relative probability of containment failure modes from internal events
IPEs Suggest that Late Failures Dominate in BWRs and PWRs
General Insights From Containment Response Analyses

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

• How are the Level-1 and Level-2 portions of a PRA linked?
• What are the two types of uncertainty?
• How is uncertainty propagated through the analysis?
Accident Progression Analysis (P-300)

Example: Palisades IPE (Jan. 1993)
Example: Palisades IPE

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

\[ \sum_{m=1}^{i} Pr(CDES_m) = CDF \]
\[ \sum_{n=1}^{j} Pr(PDS_n) = CDF \]
\[ \sum_{p=1}^{k} Pr(CET_p) = CDF \]
\[ \sum_{q=1}^{r} Pr(RC_q) = CDF \]
Palisades IPE Used PDS Bridge Tree to Map CD Sequences Into PDS

• CET developed first, PDS-BT then developed to satisfy information needs of CET

• Total CDF conserved in binning to PDS’s
  – i.e., Total CDF = Σ_{m=1,i} PDS_m

• PDS-BT incorporated as an extension of the Level-1 core damage event tree

• PDS-BT primarily used as a sorting mechanism
  – Most branch choices dictated by previous events
  – Presence of water in containment was exception (PDS-BT top event SII)
    • In some CD sequences, operation of ECCS does not guarantee water in containment (i.e., ISLOCA, SGTR)
# Palisades IPE PDS Characteristics

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

A1 - Large LOCA (d > 18 in.)
A2 - Medium LOCA (2 in. < d < 18 in.)
B - Small LOCA (1/2 in. < d < 2 in.)
C - Interfacing System LOCA
D - SGTR
T - Transient
Z - ATWS
Palisades IPE PDS Char. #’s 2, 3 & 4

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

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

4 Pressurizer PORV
   M - PORV Available
   N - PORV Unavailable
Palisades IPE PDS Char. #5 (Cont. Sys.)

P - Containment sprays and air coolers available
Q - Cont. sprays avail. and cont. air coolers NOT avail.
R - Only cont. air coolers avail., RWST contents in cont.
S - Only cont. air coolers avail., RWST contents NOT in cont.
V - No cont. systems avail., RWST contents in cont.
W - No cont. systems avail., RWST contents NOT in cont.
X - Late (post VB) operation of only HPSI/LPSI
Palisades PDS’s Grouped to Reduce Number of CET Analyses

- Initial development resulted in 392 possible PDS’s
- IPE judged preemptive protective actions were unlikely
  - All core damage timing assumed to be early
  - Reduced number of possible PDS’s to 196
- Illogical PDS’s were also removed from the list (reduced number to 168)

- Truncation (at 1E-9) during the CD/PDS quantification further reduced the list to 70 PDS’s
  - Still too many PDS’s

- PDS’s collapsed on PORV availability
  - For each remaining PDS PORV availability calculated by taking a weighted average (53 PDS’s left)
<table>
<thead>
<tr>
<th>Heading</th>
<th>Description</th>
</tr>
</thead>
<tbody>
<tr>
<td>2ND</td>
<td>AFW available to both steam generators</td>
</tr>
<tr>
<td>CSI</td>
<td>Containment spray system available in injection mode</td>
</tr>
<tr>
<td>CSR</td>
<td>Containment spray system available in recirculation mode</td>
</tr>
<tr>
<td>PRV</td>
<td>One pressurizer PORV available to depressurize RCS</td>
</tr>
<tr>
<td>SII</td>
<td>RWST water is in containment</td>
</tr>
<tr>
<td>FC</td>
<td>Containment air coolers available</td>
</tr>
<tr>
<td>SIL</td>
<td>Safety injection available after vessel failure</td>
</tr>
<tr>
<td>Core Damage Sequence</td>
<td>APW available to both S/Gs</td>
</tr>
<tr>
<td>----------------------</td>
<td>-----------------------------</td>
</tr>
<tr>
<td>CD</td>
<td>2ND</td>
</tr>
</tbody>
</table>

Yes

No

_PALISADES-IPE-BT - Palisades IPE (Jan. 1993) Plant Damage State Bridge Tree_
### Top 18 PDSs from Palisades IPE

<table>
<thead>
<tr>
<th>PDS</th>
<th>Freq</th>
<th>PDS</th>
<th>Freq</th>
</tr>
</thead>
<tbody>
<tr>
<td>BEGP</td>
<td>1.11E-5</td>
<td>BEGS</td>
<td>7.22E-7</td>
</tr>
<tr>
<td>TEJP</td>
<td>9.40E-6</td>
<td>TEJQ</td>
<td>3.70E-7</td>
</tr>
<tr>
<td>TEJW</td>
<td>9.02E-6</td>
<td>CEJW</td>
<td>3.70E-7</td>
</tr>
<tr>
<td>TEJV</td>
<td>6.89E-6</td>
<td>A2EGR</td>
<td>2.42E-7</td>
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<tr>
<td>ZEGP</td>
<td>4.20E-6</td>
<td>BEGV</td>
<td>2.33E-7</td>
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<td>BEGR</td>
<td>2.97E-6</td>
<td>TEJS</td>
<td>3.32E-7</td>
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<td>TEJR</td>
<td>2.42E-6</td>
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<td>1.10E-7</td>
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<td>DEJP</td>
<td>1.33E-6</td>
<td>A2EGP</td>
<td>1.00E-7</td>
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<tr>
<td>DEJS</td>
<td>1.04E-6</td>
<td>A1EGR</td>
<td>9.72E-8</td>
</tr>
</tbody>
</table>
CET Top Event Quantification Focus on Probability of Containment Failure

• Need to know how strong is the containment structure
• Need to identify the likely failure location
• Need to identify the size of any potential containment failure
Palisades IPE Containment Structural Response and Failure Characterization

• Purpose
  – To establish best estimate probabilistic measure of containment fragility
  – Identify failure mode (i.e., leak or rupture) given a predicted failure due to quasi-static overpressure event

• Approach
  – Two dimensional axi-symmetric finite element analysis of the total containment structure
    • Provided detailed information on potential weak links (discontinuities)
  – Detailed analyses of the weak links
Containment Structural Evaluation Comprised Two Parts

• Palisades Finite Element Model (PFEM) mesh consisted of five major sections
  – dome, ring girder, cylinder wall, basemat and soil
  – Analysis performed by plant Engineer/Constructor (Bechtel)

• Leakage at major penetrations was evaluated using EPRI developed method (EPRI NP-6260-M)
  – Penetrations less than 24-inches diameter were judged not to constitute a weak link in a concrete containment
  – Electrical penetrations also judged to not be a concern (based on NUREG-1037 analysis)
Structural Evaluations Identified Potential Weak Links

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

• Fragility curve provides cumulative probability of containment failure as a function of internal pressure
  – Seven weak link fragility curves combined into composite (total containment) fragility curve
  – PrF(p) = 1- $\prod_{i=1,n}[1-PrF_i(p)]$
  – 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
Palisades IPE CET Features

- CET and PDS’s developed together such that PDS’s contain ONLY plant system information, and CET addresses ONLY effect of severe accident physical processes
  - Plant system dependencies accounted for
  - CET focused on containment performance and fission product release

- Single, general-form CET
  - Consistent treatment of PDS’s
  - Consistent binning of CET endstates into source terms
Only Dominant PDS’s Used in CET Analysis

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

PDS - Plant Damage State
BYE - Early Cont. Bypass
CIS - Cont. Isolation
BYL - Late Cont. Bypass
RIV - Recovery after CD but before VB
UDD - Upward debris dispersal at VB
CAE - Early relocation of core debris to aux. bldg.
CIE - Cont. intact early
LVE - Large volatile fission product release early
CAL - Late relocation of core debris to aux. bldg.
CIL - Cont. intact late
CCI - Core concrete interaction resulting in large fission product release
LVL - Large volatile fission product release late
CET Top Events Modeled Using Fault Trees

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

- Core debris enters Auxiliary building Early
  - Early: soon after vessel failure

- Core debris enters the auxiliary building via ESF (sump) recirculation line
  - Core debris falls to cavity floor, is not quenched and flows into sump via drain lines.
  - Core debris falls to cavity floor, is quenched, but not in a coolable geometry
    - Core debris reheats and flow into sump
  - RV fails at high pressure causing catastrophic failure of cavity floor.
RV lower head fails, core debris enters ESF sump either via the two floor drains, or through catastrophic failure of cavity floor.
CAE Top Event Fault Tree

- Core Debris Transport to Aux Bldg - Early
  - Flow Path to Sump for Debris
    - C276
  - Debris in Cavity is Molten
    - C16
      - Core Debris in Cavity Quenched but not Cooled Early
        - C82
      - Core Debris in Cavity Not Promptly Quenched
        - NPRMTQUENC
          - Core Debris in Cavity Promptly Quenched
            - PRMTQUENCH
Core Debris in Cavity Quenched but Not Cooled Early

Core Debris in Cavity Not Cooled Early

Core Debris in Cavity Promptly Quenched

Cavity Debris not Cooled Early

Debris Config in Cavity Non-Coolable

Debris in Cavity Not Coolable

No Water on Debris in Cavity Early

Debris Config in Cavity Coolable

C82

PRMTQUENCH

C83

NCCVDBCNFG

14

C84A

NCCVDBCNFG

NCCVDBCNFG
Flow Path to Sump for Debris

C276

Cavity Sump Drains Open (no mod)  CAVSMPDRNS

High Pressure Injection Fails Cavity Floor  C276A

Catastrophic Cavity Floor Failure at Vessel Failure  CAVFLRFAIL

Reactor Vessel Failure at High Pressure  RVFAILHP  57
CAVFLRFAIL – Cavity Floor Fails

• Structural analysis estimates a failure pressure of 370 psid (2.55 MPa)
  – Assumed to be a mean value
  – Standard deviation of 10% assumed

• Analyses of Palisades severe accidents produced peak cavity pressure estimates for three class of PDS
  – High, medium, and low RCS pressures
**CAVFLRFAIL – Cavity Floor Failure Probability Estimate**

Probability of Cavity Floor Failure depends upon RCS Pressure (at time of RPV failure) – Estimated by convolution of peak cavity pressure distribution and floor failure pressure distribution.

<table>
<thead>
<tr>
<th>RCS Pressure (MPa)</th>
<th>RCS Pressure Class (at RPV failure)</th>
<th>Prob of Cavity Floor Failure</th>
<th>Applicable PDS</th>
</tr>
</thead>
<tbody>
<tr>
<td>17.0</td>
<td>High</td>
<td>0.53</td>
<td>T w/o creep rupture</td>
</tr>
<tr>
<td>7.0</td>
<td>Medium</td>
<td>0.196</td>
<td>B and D</td>
</tr>
<tr>
<td>3.0</td>
<td>Low</td>
<td>2.71E-3</td>
<td>A1, A2, C, and T w/ creep rupture</td>
</tr>
</tbody>
</table>
## Other BE Quantified in a Variety of Ways

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

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

<table>
<thead>
<tr>
<th>PDS Initiator</th>
<th>Containment System Status</th>
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<tr>
<td></td>
<td>P or Q (Cavity Flooded)</td>
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<tr>
<td>A1</td>
<td>0.99</td>
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<tr>
<td>A2</td>
<td>0.99</td>
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<tr>
<td>B</td>
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<tr>
<td>C</td>
<td>0.99</td>
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<tr>
<td>D</td>
<td>0.50</td>
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<tr>
<td>T (w/ induced failure)</td>
<td>0.75</td>
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<tr>
<td>T (w/o induced failure)</td>
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## Accident Progression Analysis (P-300)

### Basic Events

<table>
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<th>Value</th>
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<tbody>
<tr>
<td>LLOCA</td>
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<tr>
<td>SLOCA</td>
<td>0.0</td>
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<tr>
<td>CSP</td>
<td>0.0</td>
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<tr>
<td>SIRWT</td>
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<tr>
<td>LBOCA</td>
<td>0.0</td>
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<tr>
<td>SGTR</td>
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<tr>
<td>CAC</td>
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<td>SILATE</td>
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<tr>
<td>MILOCA</td>
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<tr>
<td>TRANSIENT</td>
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<tr>
<td>SECONDCOOL</td>
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### Recovery Events

<table>
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<tr>
<td>CACRECDOV</td>
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<tr>
<td>CSPRECOV</td>
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<tr>
<td>CSPRECIV</td>
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<tr>
<td>SECCLECOV</td>
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### Operator Action Events

<table>
<thead>
<tr>
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<tbody>
<tr>
<td>OPDEPRESS</td>
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### Phenomenological Events

<table>
<thead>
<tr>
<th>Event</th>
<th>Value</th>
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<tbody>
<tr>
<td>CAFAIL</td>
<td>0.100</td>
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<tr>
<td>CAVFLRFAIL</td>
<td>0.530</td>
</tr>
<tr>
<td>CAVITYFAIL</td>
<td>0.0215</td>
</tr>
<tr>
<td>CAUSMHDRNS</td>
<td>1.000</td>
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<tr>
<td>CAVTYRETN</td>
<td>0.001</td>
</tr>
<tr>
<td>CIS</td>
<td>0.925</td>
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<tr>
<td>CNACA1TMG1</td>
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</tr>
<tr>
<td>CNACA1TMG2</td>
<td>0.010</td>
</tr>
<tr>
<td>CRDRAFTBD</td>
<td>0.080</td>
</tr>
<tr>
<td>CRUIMMPNG</td>
<td>0.010</td>
</tr>
<tr>
<td>CSFAIL</td>
<td>0.100</td>
</tr>
<tr>
<td>CSPFAILRIV</td>
<td>0.050</td>
</tr>
<tr>
<td>CVFLDOSYS</td>
<td>0.0165</td>
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<td>DCH</td>
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<tr>
<td>DRYOUTMPNG</td>
<td>1.000</td>
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<tr>
<td>EXVSLSTEKP</td>
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</tr>
<tr>
<td>FLAMMGA50</td>
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<td>FLAMMGA52</td>
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<tr>
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<tr>
<td>FRACMELBOC</td>
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</tr>
<tr>
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<td>0.492</td>
</tr>
<tr>
<td>INVSLSTEXP</td>
<td>0.0001</td>
</tr>
<tr>
<td>INVSLSH2</td>
<td>0.00792</td>
</tr>
<tr>
<td>MSLFRPSS</td>
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<tr>
<td>MSLBFALL</td>
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<tr>
<td>MSDFGSM</td>
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<tr>
<td>MVS04SPF</td>
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</tr>
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<td>NATCCNV</td>
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<tr>
<td>NCCVDBCNFG</td>
<td>0.250</td>
</tr>
<tr>
<td>NCUCDBCNFG</td>
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</tr>
<tr>
<td>PCSDEPRESS</td>
<td>0.710</td>
</tr>
<tr>
<td>PCSRETEN</td>
<td>0.550</td>
</tr>
<tr>
<td>PORUS</td>
<td>0.4149</td>
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<tr>
<td>PRMTQUENCH</td>
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</tr>
<tr>
<td>PLSRVFTC</td>
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</tr>
<tr>
<td>REVAPTNG</td>
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<td>RLFVFLOPEN</td>
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<tr>
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</tr>
<tr>
<td>SECVFLOPEN</td>
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</tr>
<tr>
<td>SGTFUBEFAIL</td>
<td>0.0034</td>
</tr>
<tr>
<td>SIFLOWFAIL</td>
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</tr>
<tr>
<td>STMSPIKE</td>
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<tr>
<td>SURGEFAIL</td>
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<td>VFTIMELO</td>
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<td>VSLSHTXFR</td>
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<tr>
<td>VSSLIMMPNG</td>
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<tr>
<td>VSSLTHRUST</td>
<td>0.00085</td>
</tr>
</tbody>
</table>

Values for each basic event documented for every PDS.
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)
<table>
<thead>
<tr>
<th>CET</th>
<th>Aggregated Freq (/yr)</th>
<th>Important PDS Contributors</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>1.20E-8</td>
<td>DEJP(100%\textsuperscript{a})</td>
</tr>
<tr>
<td>2</td>
<td>8.47E-7</td>
<td>DEJS(99%)</td>
</tr>
<tr>
<td>3</td>
<td>8.53E-7</td>
<td>DEJS(99%)</td>
</tr>
<tr>
<td>4</td>
<td>4.18E-7</td>
<td>CEJW(55%) DEJS(44%)</td>
</tr>
<tr>
<td>10\textsuperscript{b}</td>
<td>4.44E-8</td>
<td>TEJW(39%) TEJV(29%) TEJP(15%) TEJR(9%)</td>
</tr>
<tr>
<td>18</td>
<td>1.54E-7</td>
<td>BEGP(40%) TEJP(36%) ZEGP(22%)</td>
</tr>
<tr>
<td>22</td>
<td>7.31E-6</td>
<td>TEJW(53%) TEJV(40%)</td>
</tr>
<tr>
<td>23</td>
<td>1.43E-7</td>
<td>BEGR(46%) BEGV(36%) BEGS(11%) ZEGP(5%)</td>
</tr>
<tr>
<td>26</td>
<td>2.29E-7</td>
<td>TEJW(36%) TEJV(30%) TEJR(13%) BEGR(7%)</td>
</tr>
<tr>
<td>28</td>
<td>2.15E-7</td>
<td>TEJW(39%) TEJV(33%) TEJR(14%) BEGR(4%)</td>
</tr>
<tr>
<td>29</td>
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</tr>
<tr>
<td>30</td>
<td>5.43E-6</td>
<td>TEJW(36%) TEJV(27%) BEGR(14%) TEJR(8%) A2EGR(4%) BEGS(4%)</td>
</tr>
<tr>
<td>31</td>
<td>8.25E-6</td>
<td>TEJW(30%) TEJV(23%) TEJP(22%) BEGP(11%) TEJR(7%) ZEGP(3%)</td>
</tr>
<tr>
<td>32</td>
<td>2.01E-6</td>
<td>BEGP(40%) BEGR(34%) ZEGP(13%) BEGS(10%)</td>
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<td>33</td>
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</tr>
<tr>
<td>57\textsuperscript{c}</td>
<td>2.73E-8</td>
<td>TEJW(36%) TEJV(27%) BEGR(14%) TEJR(8%)</td>
</tr>
<tr>
<td>total</td>
<td>2.62E-5</td>
<td>Sum of dominant CET ES</td>
</tr>
<tr>
<td></td>
<td>2.68E-5</td>
<td>Total containment failure frequency</td>
</tr>
</tbody>
</table>

\textsuperscript{a} Contribution to ES frequency
\textsuperscript{b} Containment bypass
\textsuperscript{c} Containment isolation failure
Source Terms Calculated Using CPMAAP

- 41 cases selected for CPMAAP analysis
  - various combinations of PDS and CET-ES from list of dominant contributors to containment failure
    - For example:
      - DEJP-01 – SGTR with recovery in-vessel
      - DEJS-02 – SGTR with stuck open secondary SRV, upward debris dispersal and CCI in upper containment
      - CEJW-04 – ISLOCA outside containment
      - TEJW-10 – Blackout with creep induced SGTR
      - A1EGR-30 – LBLOCA with core to aux early and a large volatile release early
      - BEGP-31 – SBLOCA with core to aux early and late revaporization form aux building and CCI in aux bldg
<table>
<thead>
<tr>
<th>CET ES</th>
<th>Aggregated Freq (/yr)</th>
<th>Important PDS Contributors</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
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<tr>
<td>10&lt;sup&gt;b&lt;/sup&gt;</td>
<td>4.44E-8</td>
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<td>TEJW(36%) TEJV(27%) BEGR(14%) TEJR(8%) A2EGR(4%) BEGS(4%)</td>
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<tr>
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<tr>
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<tr>
<td>total</td>
<td>2.62E-5</td>
<td>Sum of dominant CET ES</td>
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<td>Total containment failure frequency</td>
</tr>
</tbody>
</table>

<sup>a</sup> Contribution to ES frequency
<sup>b</sup> Containment bypass
<sup>c</sup> Containment isolation failure
## Calculated Source Terms from CPMAAP (examples)

<table>
<thead>
<tr>
<th>PDS-ES</th>
<th>Nobel Gas</th>
<th>I</th>
<th>Cs</th>
<th>Te</th>
<th>Sr</th>
<th>Mo</th>
<th>La</th>
<th>Ce</th>
<th>Ba</th>
<th>Time of release (hr)</th>
<th>Warning Time (hr)</th>
<th>Release Duration (hr)</th>
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<td>2E-3</td>
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</tbody>
</table>

Typically, multiple PDSs selected for each ES/CPMAAP calculation with “worst-case” eventually selected to represent particular CET-ES.
Accident Progression Analysis (P-300)

11. NUREG/CR-6595, Rev.1
Session Objectives

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

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

- Early structural failure
- Containment bypass
- Containment isolation failure
- Late structural failure
- Containment venting
LERF Models developed for PWRs and BWRs

- PWRs with a large volume containment
- PWRs with an ice condenser containment
- BWRs with a Mark I containment
- BWRs with a Mark II containment
- BWRs with a Mark III containment
**CET from NUREG/CR-6595 (LERF)**

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

• Also encompasses subatmospheric containments
  – Both rely on large volumes and relatively high design pressures to mitigate consequences

• Initiating Event is Core Damage (CD) - Frequency and characteristics of CD sequences from Level-1 analysis

• Most split fractions determined from Level-1 PRA supplemented by additional analysis and information
  – Generic estimates provided only for probability of early containment failure
<table>
<thead>
<tr>
<th>Core Damage</th>
<th>Containment Isolated and Not Bypassed</th>
<th>RCS Depress.</th>
<th>Core Damage Arrested without VB</th>
<th>No Induced Steam Generator Tube Rupture</th>
<th>No Containment Failure at VB</th>
<th>No Potential for Early Fatalities</th>
<th>#</th>
<th>ES</th>
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<td>CD</td>
<td>NCI</td>
<td>HIPR</td>
<td>VB</td>
<td>I-SGTR</td>
<td>ECF</td>
<td>EF</td>
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<td></td>
</tr>
</tbody>
</table>

![Diagram showing the progression analysis with nodes and branches representing different scenarios and outcomes.](attachment:image.png)

- **CD**: Core Damage
- **NCI**: Nef Poso Cintaine
- **HIPR**: Hiperpressa
- **VB**: Vessel Break
- **I-SGTR**: Induced Steam Generator Tube Rupture
- **ECF**: Early Containment Failure
- **EF**: Early Fatalities

**LERF-LGDRY - PWR Large Dry LERF CET (NUREG/CR-6595)**
**NCI - No Containment Isolation (Nor Containment Bypass)**

- Includes:
  - Failure of containment to isolate
  - Interfacing system LOCA
  - SGTR Initiating Event
  - ATWS (pressure-)induced SGTR or RCS pipe failure
  - Loss of containment heat removal
    - i.e., containment failure before core damage
- Quantified using Level-1 information
**HIPR - RCS Not Depressurized**

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

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

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

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

- In simplified treatment, only RCS pressure explicitly considered
ECF - Low Pressure RCS

- ECF given Low Pressure Vessel Failure Includes:
  - In-vessel steam explosion
  - Rapid steam generation from core debris contacting water in the cavity
  - Hydrogen combustion

- Conditional probability of ECF estimated at 0.01
  - based on previous PRAs
ECF - High Pressure RCS

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

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

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

Failure ID – Success event description
NCI – Containment isolated and not bypassed
HIPR – RCS depressurized
VB – Core damage arrested without vessel breach
I-SGTR – No induced steam generator tube rupture
ECF – No containment failure at vessel breach
EF – No potential for early fatalities
NCI – Containment Not-Isolated or Bypassed

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

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

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

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

- No secondary side cooling (secondary side depressurized) makes an I-SGTR a concern
  - SGTR can be either temperature-induced (creep) or pressure-induced (typically an ATWS issue)
  - Dependent on state of SG tubes (flaws)
    - Crack depth (% through wall)
    - Flaw size distributions vary greatly among plants and crack size difficult to determine

- Conditional probability of I-SGTR given HIPR (i.e., no other induced RCS failure or depressurization)
  - I-SGTR is the worst case induced RCS failure
    - Containment bypass
    - Assume \( P(\text{I-SGTR}|\text{HIPR}) = 10\% \)
**ECF – Containment Failure at VB**

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

- Depends on magnitude and timing of release
- Typically comprises all early failures and bypass
- However, can discriminate on release size
  - Threshold of greater than 2.5% to 10% iodine
  - Requires estimation of source term
- Assume all early failures and bypass
  - Split fraction = 1.0
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<td>No Induced Steam Generator Tube Rupture</td>
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**Diagram:**

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    │         └── 0.9
    │             └── 0.01
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LERF CET for PWR Ice Condenser Containment

• Similar to that for large dry
  – Additional top event for H2 igniters

• Initiating Event is Core Damage (CD) - Frequency and characteristics of CD sequences from Level-1 analysis

• Most split fractions determined from Level-1 PRA supplemented by additional analysis and information
  – Generic estimates provided only for probability of early containment failure
Most Top Events Same as Lg Dry

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

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

• RCS pressure
• Amount and temp of core debris exiting vessel
• Size of vessel failure
• Operability of containment sprays
• Operation of igniters
• Amount of ice at time of VB
• Amount of water in vessel cavity
• Configuration of cavity
• Structural strength of containment building
No Cont. Failure at or before VB

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

- If igniters are operating, CF still possible from other causes
  - VB at low pressure, prob. of CF = 0.01
    - In-vessel steam explosion
    - Ex-vessel steam generation
  - VB at high pressure, prob. of CF = 0.05
    - HPME can result in direct impingement of corium on containment wall
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<tr>
<th>Core Damage</th>
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<th>Ignitors Operating Before TAF</th>
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Yes

No

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LERF
PWR Late Containment Failure

- To address accidents where evacuation not effective
  - E.g., Seismic and high wind
- Considers only
  - Core concrete interaction
  - H2 combustion (Ice Condenser only)
  - Basemat penetration not included
    - Assumed to result in “small” release
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<tr>
<th>Entry from PWR LERF CETs</th>
<th>Is Cavity Flooded</th>
<th>Is Core Debris Coolable</th>
<th>Is CHR Operating and Effective</th>
<th>No Late H2 Combustion</th>
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</tr>
<tr>
<td>9</td>
<td>LG-REL</td>
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</tbody>
</table>
```
**Flooded Cavity**

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

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

- Long-term operation of containment heat removal
  - Containment sprays
  - Two questions: Operating? Effective?

- If cavity is dry
  - Then core concrete interaction produces noncondensable and combustible gases
    - If CHR was operating, continued operation is questionable

- If cavity is flooded
  - Is core debris coolable?
    - Yes and CHR operating - then CCI does not occur and late CF prevented
    - Yes but CHR not operating – then eventual CF probable
    - No - then CCI occurs and CF probable
Late H2 Combustion

• Applies only to Ice Condenser
• Are igniters available?
• Did H2 combustion occur early?
• If No and No
  – Then CF (late H2 combustion) = 1.0
• If igniters are available
  – Then Late H2 Combustion = 0.0
Accident Progression Analysis (P-300)

Review

April 2015
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?