The Role of Large-Scale Containment Model Tests in Nuclear Power Plant Safety and Risk Analyses

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• 1946: Atomic Energy Act est. Atomic Energy Commission
• 1950: WASH 3: Defined Exclusion Zone
  \[ R \text{ (miles)} = 0.1 [P \text{ (kWt)}]^{1/2} \]
  For a 3000 MWt plant, \( R = 17.3 \text{ m (27.8 km)} \)
• 1957: Shippingport Atomic Power Station, 20 miles from Pittsburgh
• ‘Defense in Depth’
  – Accident prevention
  – Redundancy of safety systems
  – Containment
  – Accident management
  – Remote siting/emergency planning (sheltering and evacuation)
• 1962: 10CFR100 (Maximum Credible, Design Basis Accident)
• 1972: WASH-1250 (Definition of severe accidents, PRA)
• 1975: WASH-1400 (Containment capacity)
• 1979: Three-Mile Island accident
• 1981: SNL Background Study on Containment Capacity
• 1990: NUREG-1150 (PRA for 5 representative plants)
The primary purposes of the containment system are:

- to contain any radioactive material that may be released from the primary system in case of an accident.
- to protect the nuclear system from weather and other external threats such as missiles produced by earthquakes, tornadoes, wind, and in some cases aircraft impact.
- to act as a supporting structure for operational equipment such as cranes.
• 1971: General Design Criteria, Appendix A of 10 CFR 50:

  Criterion 1, *Quality standards and records*, requires, in part, that:

  “Structures, systems, and components important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. Where generally recognized codes and standards are used, they shall be identified and evaluated to determine their applicability, adequacy, and sufficiency and shall be supplemented or modified as necessary to assure a quality product in keeping with the required safety function.”

  Criterion 16, *Containment Design* states:

  “Reactor containment and associated systems shall be provided to establish an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment and to assure that the containment design conditions important to safety are not exceeded for as long as postulated accident conditions require.”

  Criteria 50 through 57 give specific requirements for reactor containment
Containment Building

- One of the multiple barriers between the radioactive fission products and the public.
- Designed to withstand high pressures (45-60 psig) and temperatures (>300 F)
- Includes basic structure (steel, reinforced or prestressed concrete) and operational components (equipment hatch, personnel airlocks, piping and electrical penetrations)
Detailed assessment of the risks of severe accidents at five plants. 

- **CCFP**, conditional containment failure probability,

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

- **CFF**, containment failure frequency,

\[
CFF = \sum_{i=1}^{n} S_i \cdot C_i
\]

- **CDF** is the total core damage frequency,
- **\(S_i\)** is the frequency of accident sequence **\(i\)**,
- **\(C_i\)** is the conditional probability of containment failure given accident sequence **\(i\)**, fragility
- **\(n\)** is the total number of accident sequences.

- Containment capacity estimates based on expert elicitation
IPE Fragility Curves for Large, Dry PWRs
• Objective:
  – Evaluate methods used to predict the performance of light water reactor containment systems when subjected to loads beyond those specified in the design codes.
  – **NOT** to determine the pressure carrying capacity of actual containments by testing scale models.

• Two types of loadings are being considered:
  – Severe Accident Loadings (static pressurization and elevated temperature)
  – Earthquakes greater than the Safe Shutdown Earthquake (SSE) - analysis only

• An integrated program of testing models of containment structures and components (both scaled and full-size specimens) coupled with detailed pre- and posttest analyses
• Pneumatic pressure tests of large-scale models of representative containment structures and full scale tests of components (penetrations, etc.).

• Models of three types of containments used in current nuclear construction:
  – free-standing steel containments,
  – steel lined reinforced concrete containments and steel lined,
  – prestressed concrete containments.

• Guiding principles
  – models would incorporate representative features of the prototypes,
  – would not knowingly preclude a potential failure mode
  – and would not incorporate details which were unique to the model and not representative of the prototype.
• **Scope:**
  – Scale-model Containment Overpressurization Tests
    • Steel: four 1:32-scale, one 1:8-scale, one 1:10-scale
    • Reinforced Concrete: one 1:6-scale
    • Prestressed Concrete: one 1:4-scale
  – Penetration Tests (hatches, electrical & piping penetrations, seals & gaskets)
  – Degraded Containment Analyses
  – Seismic Analyses of scale model tests

• **Related Efforts:**
  – Impact Tests (aircraft, turbine missiles)
1:32-Scale SCV Models
1:32-Scale SCV Models
1:8-Scale SCV Model

- Designed and built to ASME Code
- Design Pressure, $P_d = 40$ psig (.27 MPa)
- 800 channels of Instrumentation
- Failed catastrophically at 195 psig (1.34 MPa), - 5Pd
- ‘Free-field’ strain was 2.5 to 3% at failure
1:8-Scale SCV Model - Pretest
1:8-Scale SCV Model

- Stiffener Detail
1:8-Scale SCV Model - Summary

MODEL W/ VIEW OF EH1/CRACKED STIFFENER AT 190 psig

<table>
<thead>
<tr>
<th>TEST</th>
<th>PREDICTION</th>
<th>POST-TEST ANALYSIS</th>
</tr>
</thead>
<tbody>
<tr>
<td>Capacity (psig)</td>
<td>195</td>
<td>210</td>
</tr>
<tr>
<td>Failure Mechanism</td>
<td>Rupture</td>
<td>Leakage</td>
</tr>
<tr>
<td>Location</td>
<td>Near EH1 or EH2 'O' Ring seals</td>
<td>Near EH1 or EH2</td>
</tr>
</tbody>
</table>

RESPONSE

AERIAL VIEW OF SITE AFTER RUPTURE

Ovalization Increased Rapidly After Cylinder Membrane Yielding Occurred
1:8-Scale SCV Global Response
Steel Containment Vessel Model

- Japanese Improved BWR Mark II supplied by NUPEC
- Scale: 1:10 on geometry; 1:4 on thickness
- Diameter: 2900 mm (9.5’); Overall Height: 5900 mm (19.5’); Internal Volume: 21 m³ (740 ft³)
- Weight: 13,000 kg (28,634 lb)
- Design Pressure $P_{da}=0.31$ MPa (45 psig)-actual
  $P_{ds}=0.78$ MPa (112.5 psig)-scaled
- Materials: SGV480 ($F_y=265$ MPa, 38 ksi) ~ SA-516 Grade 70; SPV490 ($F_y=490$ MPa, 71ksi) ~ SA-537 Class 2
- Contact Structure
  - Weight - 9 metric tons (20,000 lbs)
  - Material: SA-516-70 ($F_y=38$ ksi)
  - Nominal thickness = 38.1 mm (1.5 in.)
- Low Pressure Test: $1.50 P_{ds}=1.17$ MPa (169 psig)
- High Pressure Test Date: Dec. 9 - 13, 1996
- Instrumentation:
  - SCV External: 113 Strain Gages, 6 Displacement Transducers
  - SCV Internal: 151 Strain Gages, 57 Displacement Transducers
  - CS: 15 Strain Gages, 10 Gap LVDT's, 59 Contact Probes
- Failure Pressure~Mode:
  - $6 P_{ds}: 4.7$ MPa (676 psig)~tearing and leakage in HAZ of SPV 490 adjacent to E/H insert plate.
SCV Model Pre- and Posttest

1/10th Scale

Failure Pressure: 676 psig (6xDesign)

Tested: Dec. 9-13, 1996
SCV Middle Stiffener - Posttest
SCV Round Robin Analysis

- Agenzia Nazionale per la Protezione del Ambienti (ANPA) (Italy)
- Argonne National Laboratory
- Bhabha Atomic Research Centre (India)
- General Dynamics, Electric Boat Division
- Japan Atomic Energy Research Institute (JAERI)
- Staatliche Material Prüfungsanstalt (MPA), Universitat Stuttgart (Germany)
- Nuclear Power Engineering Corporation (NUPEC)
- Sandia National Laboratory
## Summary of SCV RR Pretest Results

<table>
<thead>
<tr>
<th>Organization</th>
<th>Failure Pressure</th>
<th>Failure Location</th>
<th>Failure Mode</th>
</tr>
</thead>
<tbody>
<tr>
<td>Test</td>
<td>(4.5?)-4.7 MPa</td>
<td>(“Rat-hole”) E/H Insert Plate</td>
<td>Material Failure</td>
</tr>
<tr>
<td>ANL</td>
<td>4.9 - 5.5 MPa</td>
<td>Knuckle</td>
<td>Material failure</td>
</tr>
<tr>
<td>ANPA</td>
<td>10.9 MPa</td>
<td>Drywell Head</td>
<td>Buckling</td>
</tr>
<tr>
<td>BARC</td>
<td>11.5 - 12.0 MPa</td>
<td>Drywell Head</td>
<td>Material failure or buckling</td>
</tr>
<tr>
<td>GD/EB</td>
<td>4.7 MPa</td>
<td>Thinned Liner @ Equipment Hatch</td>
<td>Material failure</td>
</tr>
<tr>
<td>JAERI</td>
<td>&gt;4 MPa</td>
<td>Drywell Head</td>
<td>Buckling</td>
</tr>
<tr>
<td>MPA</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>NUPEC</td>
<td>4 - 7.3 MPa</td>
<td>Thinned Liner @ E/H Knuckle</td>
<td>Material failure</td>
</tr>
<tr>
<td>SNL</td>
<td>4.5 MPa</td>
<td>Thinned Liner @ Equipment Hatch</td>
<td>Material failure</td>
</tr>
</tbody>
</table>
Generally the other participants in the Round Robin analyses predicted lower strains than the model experienced.

Note: The data from gage HCP-I-UCS-36 was converted from a radial displacement to a hoop strain by dividing the displacement by the radius at the gage elevation.
Generally the participants in the Round Robin analyses (8 participants) for the 1/10th scale SCV predicted lower strains than the model experienced.

Global behavior of complex structures such as the SCV is dominated by the response of the material at low strains (below 2%).

Residual strains and coupon testing techniques can influence the stress-strain relationships used in pretest analyses.

Local behavior predictions need to include the effects of material property changes due to welding in areas where high strains can occur, such as around the equipment hatch.
Three out of four of the 1:32-scale models failed catastrophically.

Failure was initiated at strain concentrations caused by penetrations or stiffeners.

The 1:8-scale model failed catastrophically at 195 psig (1.34 MPa), \(-5P_d\). Failure initiated at a eccentric junction of stiffeners surrounding the Equipment Hatch.

‘Free-field’ strain was 2.5 to 3% at failure

Pretest analyses provided good agreement with the observed global behavior.

However, posttest analyses were required to ‘predict’ the strain concentration at a stiffener that caused the failure.
1:6-Scale RCCV Model

- Designed and built to ASME Code
- Design Pressure, $P_d = 46$ psig (.32 MPa)
- 1200 channels of instrumentation
- Failure Pressure was 145 psig (1.00 MPa) - 3Pd
- ‘Free-field’ strain was 1.5 to 2.0% at failure
1:6-Scale RCCV Model

- Construction
TEST OF A REINFORCED CONCRETE CONTAINMENT MODEL

THE 1/6-SCALE REINFORCED CONCRETE MODEL

THE POSTTEST LINER TEAR AT THE JUNCTION OF THE LINER/INSERT PLATE

COMPARISON OF ANALYTICAL & EXPERIMENTAL RESULTS OF HOOP REBAR STRAINS AT CYLINDER MIDHEIGHT

POSTTEST ANALYSIS OF LINER AND INSERT PLATE AT 145.5 psig
1:6-Scale RCCV Model
1:6-Scale RCCV Model – Test Results

- Failure caused by excessive leakage through tears in the steel liner associated with studs and discontinuities.
- Failure Pressure was 145 psig (1.00 MPa) - 3Pd.
- ‘Free-field strain was 1.5 to 2.0% at failure
- As for steel models, pretest analyses provided good agreement with global test results, however, no one predicted the mechanism that caused the main liner tear.
- Posttest Analyses and additional 'Separate Effects' Tests were required to fully understand the primary liner tearing mechanism. At 145 psig (1.0 MPa), strain concentrations of 10-15 times the free field strain were calculated at the base of the studs adjacent to the insert plate.
- Test results are not necessarily representative of actual containments and each case should be examined independently.
1:6-Scale RCCV Model - Pretest Analyses

• ‘Round-Robin’ Pretest Analyses - Organizations from the U.S., United Kingdom, France, Italy, Germany and Japan.
• Predicted ‘Best Estimate’ capacities for the model varied from 130 to 190 psig (0.90 to 1.31 MPa).
• Range in failure predictions mainly due to differences in interpretation of failure rather than differences in the analysis results.
• Generally good agreement between predicted global strains and displacements and test results.
Containment Technology Test Facility
Prestressed Concrete Containment Vessel Model

- Model of OHI-3 in Japan, PWR, 2-buttress, supplied by NUPEC
- Scale: 1:4 overall (except free-field liner anchor spacing)
- Design Pressure, \( P_d \): 0.39MPa (56.9 psig)
- Materials:
  - Liner: SGV410, \( F_y = 225 \) MPa (33 ksi), \( F_t = 410 \) MPa (59 ksi)
  - Anchor: SS400, \( F_y = 235 \) MPa (34 ksi), \( F_t = 392 \) MPa (57 ksi)
  - Tendons: JIS G3536 (custom), \( P_t > 630 \) kN (142kips), \( P_y > 190 \) kN (128kips)
  - Rebar: JIS G3112: SD490, \( F_y = 490 \) MPa (71ksi); SD390, \( F_y = 390 \) MPa (56ksi); SD345, \( F_y = 345 \) MPa (50ksi)
  - Basemat: Main Bars-SD490, Shear Bars-SD390
  - Shell: Main Bars-SD390, Ties-SD345
  - Concrete: Basemat 29.42MPa (\( F_c' = 4.2 \) ksi); Wall 44.13MPa (\( F_c' = 6.4 \) ksi)
- Prestressing Levels: (before/after anchoring)
  - Meridional: 113.1/105.8 kips; Hoop: 101.9/78.7 kips
- ILRT: \( 0.9P_d = 0.36 \) MPa (51psig); SIT: \( 1.125P_d = 0.45 \) MPa (64psig)
- Limit State Test Date: September 26-29, 2000
  - First Leak detected at 2.5 \( P_d = 0.98 \) MPa (142 psig)
  - Terminated at 3.3 \( P_d = 1.29 \) MPa (187.9 psig)
- Structural Failure Mode Test: November 14, 2001
  - Catastrophic Rupture @ 3.6 \( P_d = 1.42 \) MPa (206.4 psig)
- Instrumentation: Total 1560 channels
  - Strain Gages: 559 Liner, 391 Rebar, 37/156 Tendons, 94 Concrete
  - Load Cells: 68
  - Displacements: 101
  - Acoustic: 54
  - Temperature & Pressure: 100
- Predicted Failure (based on Final pretest analysis):
  - \( \sim 3.25P_d \) (1.28 MPa [185 psi]) - liner tearing @ E/H
PCCV Pretest Round Robin Participants

- Argonne National Laboratory (ANL) (U.S.)
- Atomic Energy of Canada Limited (AECL) (Canada)
- Commissariat A L’Energie Atomique/Saclay/DRN (France)
- Electricite de France (EDF) (France)
- Institute of Nuclear Energy Research (INER) (Repub. of China)*
- Institut de Protection et de Sûreté Nucléaire (IPSN) (France)
- Japan Atomic Energy Research Institute (JAERI) (Japan)*
- Japan Atomic Power Company / PWR Utility Research Group (Japan)
- Korea Institute of Nuclear Safety (KINS) (Repub. of Korea)
- Korea Power Engineering Company (KOPEC)
- Nuclear Installations Inspectorate (U.K.)
- Nuclear Power Engineering Corporation (NUPEC) (Japan)
- Nuclear Safety Institute (IBRAE) (Russia)*
- PRINCIPIA-EQE SA (Spain)
- Research and Development Institute of Power Engineering (Russia)
- Sandia National Laboratories (SNL)/ANATECH (U.S.)
- University of Glasgow (U.K.)
## Summary of PCCV RR Pretest Results

<table>
<thead>
<tr>
<th>Pressure (MPa)</th>
<th>Failure Mode</th>
</tr>
</thead>
<tbody>
<tr>
<td>ANL 1.51-1.62</td>
<td>local liner tear/hoop tendon failure @ El. 6.4 m</td>
</tr>
<tr>
<td>AECL 0.94-1.24</td>
<td>complete cracking/axisymmetric yield</td>
</tr>
<tr>
<td>CEA 1.60-1.70</td>
<td>numerically unstable</td>
</tr>
<tr>
<td>EDF 1.95</td>
<td></td>
</tr>
<tr>
<td>INER 0.81</td>
<td></td>
</tr>
<tr>
<td>JAERI</td>
<td>buckling @ dome or local fracture by bending in cylinder</td>
</tr>
<tr>
<td>JAPC 1.45-1.55</td>
<td>hoop tendon/rebar/liner rupture @ El. 7 m</td>
</tr>
<tr>
<td>KINS 1.25-1.44</td>
<td>tendon rupture</td>
</tr>
<tr>
<td>KOPEC 1.30-1.51</td>
<td>tendon rupture (@3.55% strain)</td>
</tr>
<tr>
<td>HSE/NNC 1.08</td>
<td>liner tear w/ extensive concrete cracking @ buttress</td>
</tr>
<tr>
<td>NUPEC 1.49-1.57</td>
<td>tendon rupture</td>
</tr>
<tr>
<td>IBRAE 1.26</td>
<td>tendon rupture</td>
</tr>
<tr>
<td>Principia 1.30</td>
<td>tendon yielding</td>
</tr>
<tr>
<td>RINSC 1.50</td>
<td>hoop failure of vessel</td>
</tr>
<tr>
<td>ANATECH/SNL 1.25, 1.40</td>
<td>liner tearing (16%) @ E/H, tendon rupture</td>
</tr>
<tr>
<td>Test 0.98</td>
<td>1.5% mass/day leak through liner tear @ E/H</td>
</tr>
<tr>
<td>1.30</td>
<td>limit of pressurization capacity during LST</td>
</tr>
<tr>
<td>1.42</td>
<td>hoop tendon and rebar rupture during SFMT</td>
</tr>
</tbody>
</table>
Global Axi-Symmetric Model

Added tendon friction ties in “2000 pretest model”

Typical output locations

Detailed wall-base model

Basemat

“No-tension” springs
Vertical stress applied to top plane of $\sigma_z = f(p)$

Hoop prestressing elements

270°

El. 4.67 m

Bottom plane $\Delta Z = 0$

M/S

E/H Test pressure applied uniformly to inside surface

A/L

Hoop prestressing elements

El. 8.96 m

At top plane $\theta_0$ (rotation about tangential axis) = 0

90°
M/S Penetration Model

- Strain concentration at insert plate
PCCV Limit State Test (LST)
PCCV LST Leak Rates

PCCV LST - Calculated Leak Rate

Leak Rate (% mass/day)

Time (day/hour)

PCCV LST - Estimated Leak Rate

Leak Rate (% mass/day)

Time (day/hour)

PCCV LST - Terminal Leak Rate @ 180 psig

Leak Rate @ 170 psig

Leak Rate @ 180 psig

Time (day/hour)
PCCV LST Liner Tears
### PCCV LST Acoustic Response

#### Acoustic Signals @ E/H

**Limit State Test - Concrete Cracking/Crushing Events**

<table>
<thead>
<tr>
<th>Pressure (xPd, Pd = 0.39MPa = 57psig)</th>
<th>Number of Events (since previous pressure step)</th>
</tr>
</thead>
<tbody>
<tr>
<td>2.3Pd</td>
<td>34</td>
</tr>
<tr>
<td>2.4Pd</td>
<td>32</td>
</tr>
<tr>
<td>2.5Pd</td>
<td>25</td>
</tr>
</tbody>
</table>

**Acoustic Signals**

- **AS0**: Magnitude (Volt) vs Frequency (Hz)
- **AS1**: Magnitude (Volt) vs Frequency (Hz)
- **A52**: Magnitude (Volt) vs Frequency (Hz)
- **A53**: Magnitude (Volt) vs Frequency (Hz)
• Structural Failure Mechanism Test:
  – Justification: LST did not completely satisfy pre-test objective of providing data to validate response predictions ‘well into the in-elastic regime’. 
Round Robin Predictions

- AECL
- ANL
- CEA
- EDF
- Glasgow
- HSE
- IBRAE-2d
- IBRAE-3d
- INER
- JAERI
- JAPC
- KINS
- KOPEC
- NUPEC
- PRINCIPIA
- RINSC
- SNL/ANATECH
- DL-R-Z6-05
- DT-R-Z6-01

Displacement, mm vs. Pressure, MPa (divisions are multiples of $P_d$)
3D Shell Response @ 1.38 MPa (3.51Pd)

Tendon Rupture (Strain ~4%)
PCCV - Conclusions

- Round Robin included 15 participants with failure pressures ranging from 0.81 to 1.98 MPa – Test 1.42 MPa.
- Large amounts of data on elastic and in-elastic response of representative models of containment vessels were obtained for comparison with analyses.
- Significant plastic ‘free-field’ strains were developed before failure, with considerable margin between design and failure pressures.
  - ‘Free-field’ strains at failure were considerably less than material ultimate strains
  - In-situ material properties may vary significantly from sample or coupon tests
- Model capacities were limited by local strain concentrations.
- Existing non-linear analytical methods are generally adequate for predicting global response, however prediction of local failure modes is much more difficult.
- Combined severe accident temperature and pressure loading needs to be addressed (by analysis?)
- Posttest analyses have reproduced the local mechanisms that caused the failure.
- Structural failure modes, representative of actual containment vessels were demonstrated.
Containment Bellows Tests
- 1:10 scale geometry
- 1:8 scale concrete wall thickness
- 1:4 scale liner thickness & anchors
- Dome truncated & 420 metric tons lead weighs attached
- Input accelerations scaled
  - Magnitudes multiplied by 0.75
  - Frequency increased by 2.56 (i.e., time scaled compressed by factor of 2.56)
- 1:8 scale geometry
- 1:10 scale concrete wall thickness
- 1:4 scale liner thickness & anchors
- Dome truncated & lead weighs attached at top
- Input accelerations scaled similar to PCCV model scaling
Aircraft Impact Test
Turbine Missile Impact Tests

Turbine Missile Concrete Impact Test Series Test #3
Containment Vulnerability Studies
‘Water Slug’ Tests
OECD/NEA/CSNI ISP 48 on Containment Capacity

- Proposed to CSNI by NRC in 2002
- Objective:
  - Extend the understanding of capacities of actual containment structures based on results of the recent PCCV test and other previous research. The PCCV results showed a leakage failure that began at about 2.5 times the design pressure. The subsequent structural failure mode test (SFMT) showed a global failure due to exceeding hoop tendon capacity at about 3.6 times design pressure. Two questions about actual structures are obvious:
    - Would the onset of leakage be later and much closer to the burst pressure?
    - How would including the effect(s) of accident temperatures change the outcome?
ISP 48 on Containment Capacity

• Phase 3: Combined Mechanical + Thermal Loading
  – Case 1 (Steady State)
    • Monotonically increasing static pressure and temperature (saturated steam)
    • Each participant performs heat transfer calculations or reads gradients provided by SNL.
  – Case 2 (Modified Station Blackout Scenario)
    • NRC/SNL/DEA proposal plus hydrogen detonation defined by IRSN
    • SNL will perform heat transfer calculation using full-scale axisymmetric model w/ 12 nodes through the thickness.
    • Apply resulting gradients to 1:4-scale model
Case 1: Pressure-Temperature Relationship

- Saturated Steam
Case 1: Pressure-Temperature Time Histories

- Saturated Steam
  - Pseudo-time history based on SFMT pressurization rate (5 psi/min)
Case 2: Pressure-Temperature Time Histories

- Large, Dry PWR SBO, no containment leakage

![Graph showing pressure and temperature time histories](image-url)
Fragility Analysis of Degraded Containment

- Use Latin Hypercube Sampling and nonlinear finite element analysis to generate curves.
- Fragility curves provide interface between structural analysis and risk analysis.
Fragility Analysis of Degraded Containment

- Use Latin Hypercube Sampling and nonlinear finite element analysis to generate fragility curves.
- Fragility curves provide interface between structural analysis and risk analysis (PRA) – Determine change in risk due to degradation during a severe accident.
- Currently exploring degraded containment effects in MELCOR and MACCS analyses – need Leak Rate or Area vs Pressure.
Containment Performance Model

- Integration of Containment Integrity Research results into Risk-Informed Regulatory Framework.
- Support regulatory action for existing fleet of NPP’s and next generation
  - Maintenance/Inspection/License Extension
- Provide a framework to tie together containment design requirements, capacity tests and analysis.
  - Containment performance typically defined in terms of leak rate
  - Containment response/capacity defined in terms of pressure
- Describe containment performance in a format useful for probabilistic risk assessments.
- Demonstrate effects of degradation on performance.
Strain-Based Failures for RC Containment

![Graph showing strain-based failures for RC containment. The graph plots failure function against pressure (MPa gauge). Different failure modes are color-coded and include Leak Steam Penetration, Leak Wall-Basemat Junction, Leak Hatch, Leak Springline, Leak Base 50% Corrosion, Leak Base 65% Corrosion, Leak Midheight 50% Corrosion, Leak Midheight 65% Corrosion, Rupture No Corrosion, Rupture Base 50% Corrosion, Rupture Base 65% Corrosion, Rupture Midheight 50% Corrosion, Rupture Midheight 65% Corrosion, and Catastrophic Rupture. The design pressure is indicated on the graph.]
Containment Performance Model

What we ‘know’ from Design:

![Graph showing Leak Rate (%mass/day) vs Pressure, with key points and labels: ILRT @ 0.9 Pd, 0.9 La, SIT @ 1.125 Pd, Structural Capacity Limit, Compliance-based Design (e.g. ASME B&PV Code).]
Containment Performance Model

What we ‘know’ from Analysis:

- Liner Tearing
- Concrete Cracking
- Structural Capacity Limit

Leak Rate (% mass/day) vs Pressure

- 0 P_d
- 1 P_d
- 2 P_d
- 3 P_d
Containment Performance Model

What we ‘know’ from Testing:

Leak Rate (% mass/day) vs. Pressure

- First ‘detectable’ leak below 1.0% at a pressure of 3 P_d.

Pressure scales:
- 0 P_d
- 1 P_d
- 2 P_d
- 3 P_d

Leak Rate scales:
- 0.1
- 1.0
- 10.0
- 100.0
- 1000.0

Note: The graph illustrates the relationship between leak rate and pressure, showing the first detectable leak at a specific pressure level.
Containment Performance Model

Effect of other components (tests):

Leak Rate (%mass/day)

Personnel Airlock Seal

Composite Performance Model

Equipment Hatch Seal

Bellows

Containment Shell

Pressure

0 1 P_d 2 P_d
Containment Performance Model

Effects of Temperature, Degradation:

Develop for Use in Accident Analyses (among other uses)
Containment Performance Model

- Are current analytical methods/results and test results adequate to develop a ‘continuous’ containment performance model?
- How can we illustrate the demand (e.g. ‘pressurization rate’) for comparison with the performance model and can we determine an equilibrium condition?
- What research/analyses/experiments are required to fill the gaps in our knowledge?
  - Can we relate strains or displacements to leak rates?
Future Containment Research Issues

• Integration of Containment Integrity Research results into Risk-Informed Regulatory Framework.
  – Containment Performance Model

• Support regulatory action for existing fleet of NPP’s and next generation (NP2010, NGNP, GENIV)
  – Maintenance/Inspection/License Extension
  – Performance (vs. Compliance)-based codes
  – Evolving demand on ‘containment’ function
    • Confinement vs. Containment
    • Long-term thermal loading
    • External threats