



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

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History

- 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)



Containment

- **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.



Containment Design

- 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,**

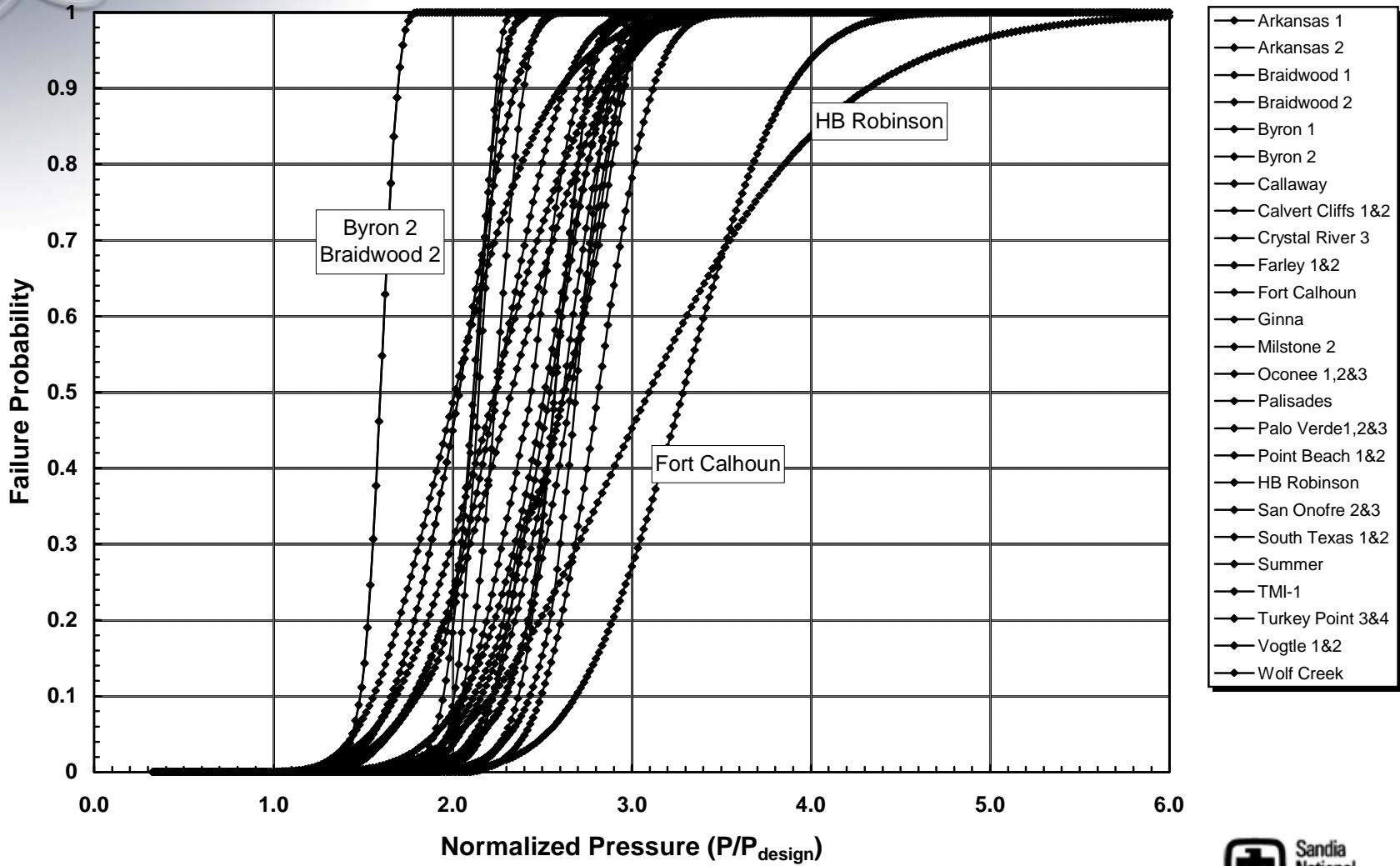
$$CCFP = \sum_{i=1}^n \frac{S_i}{CDF} \cdot C_i$$

- **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





Containment Integrity Research @ Sandia

- **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**



Containment Integrity Research @ Sandia

- 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.

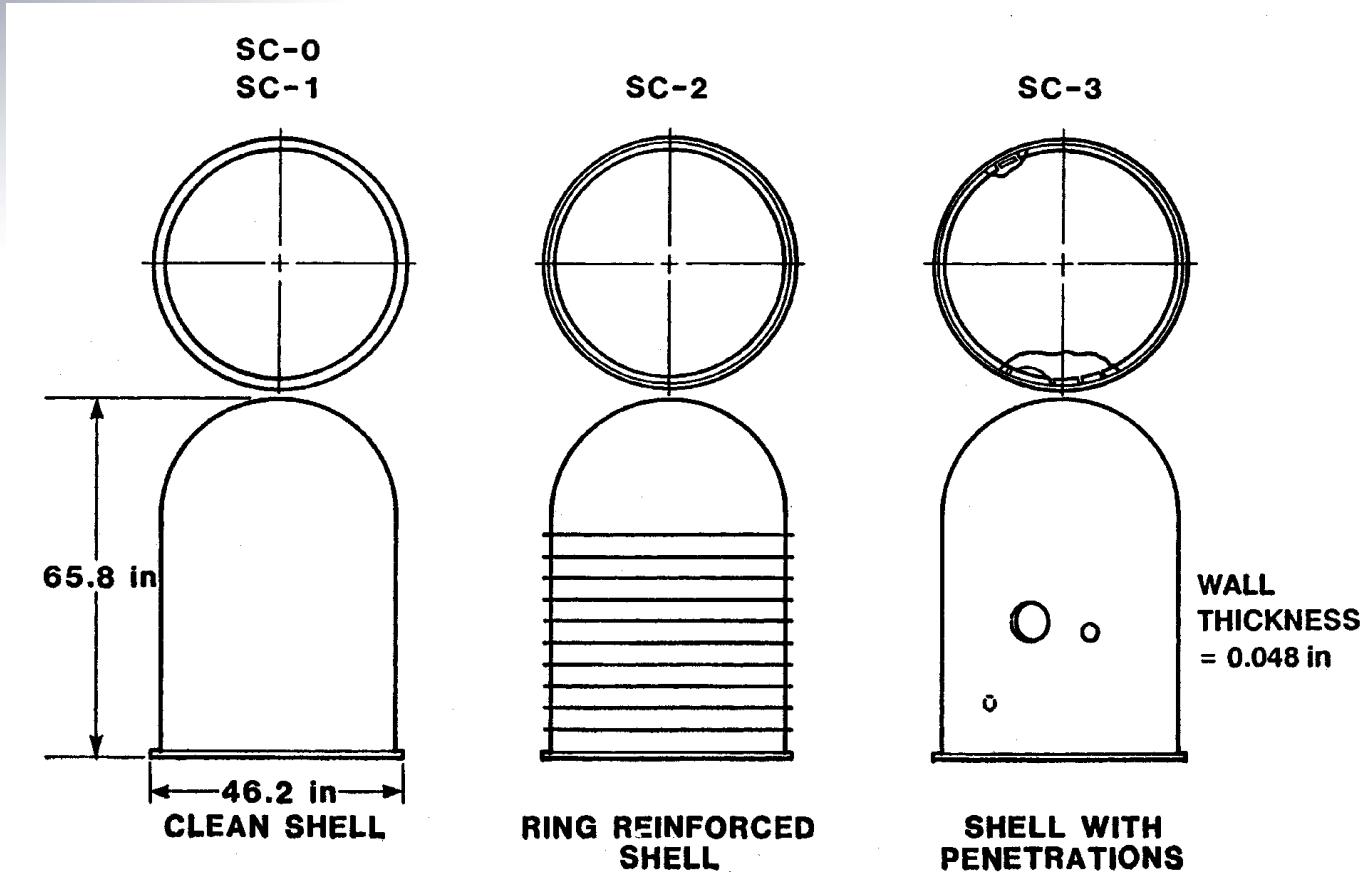


Containment Integrity Research @ Sandia

- **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)

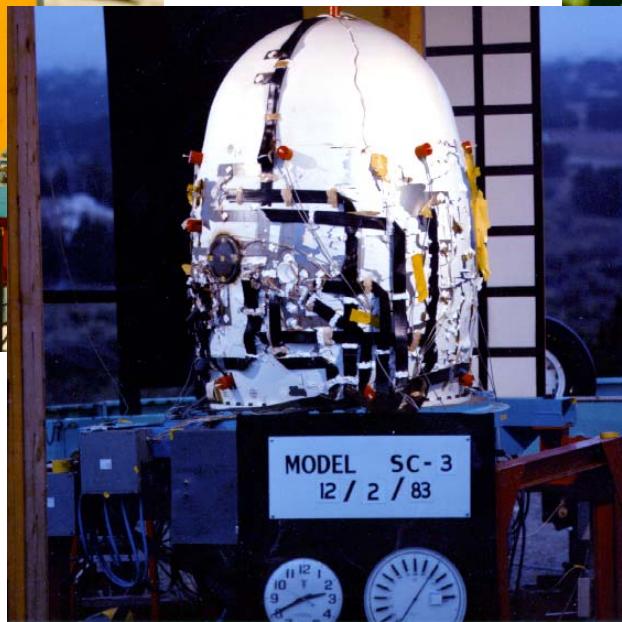
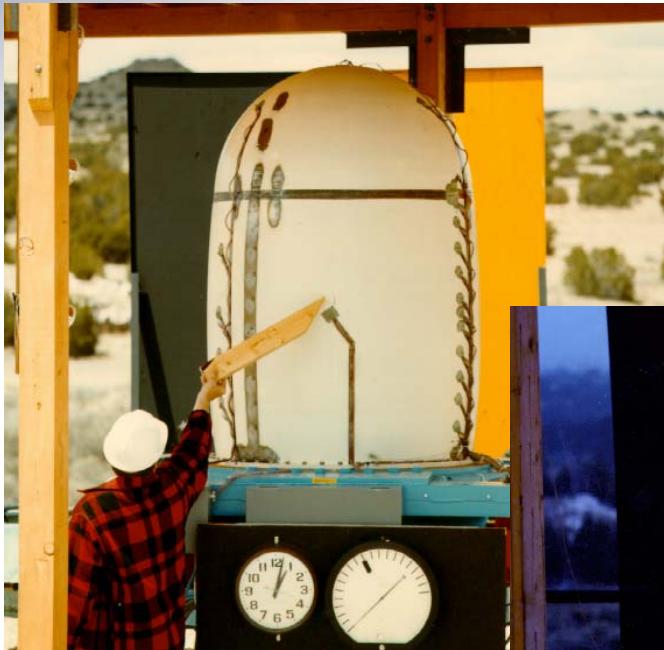


1:32-Scale SCV Models



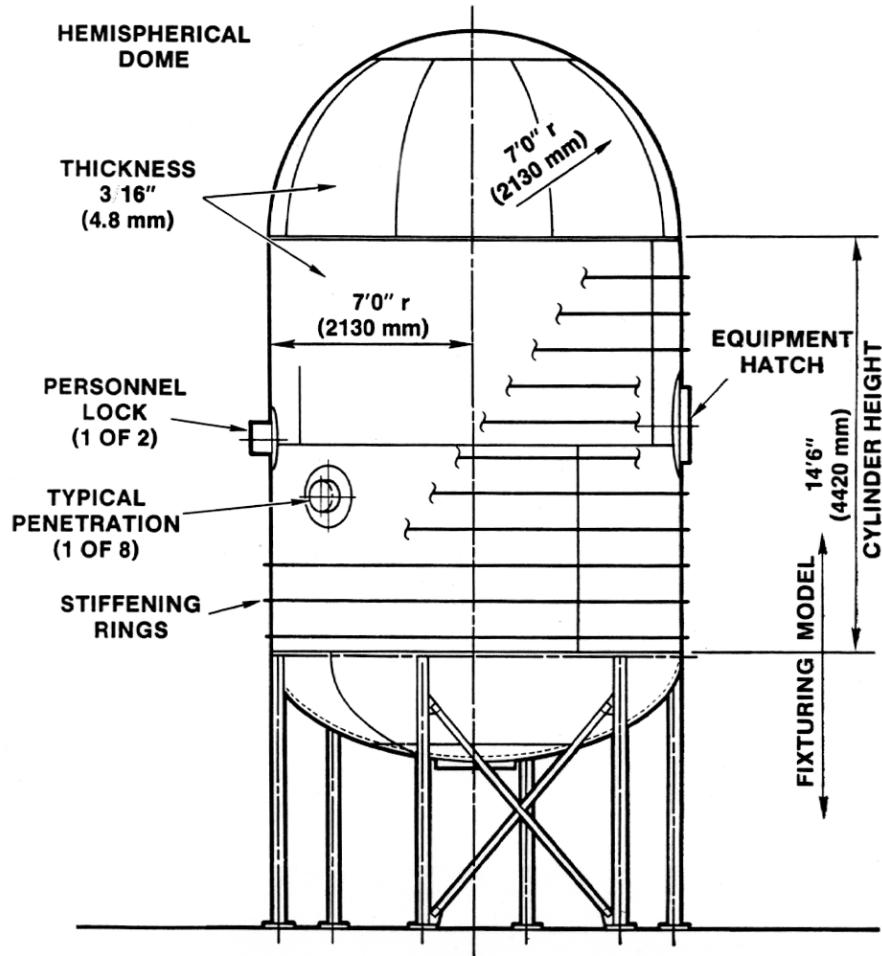


1:32-Scale SCV Models



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



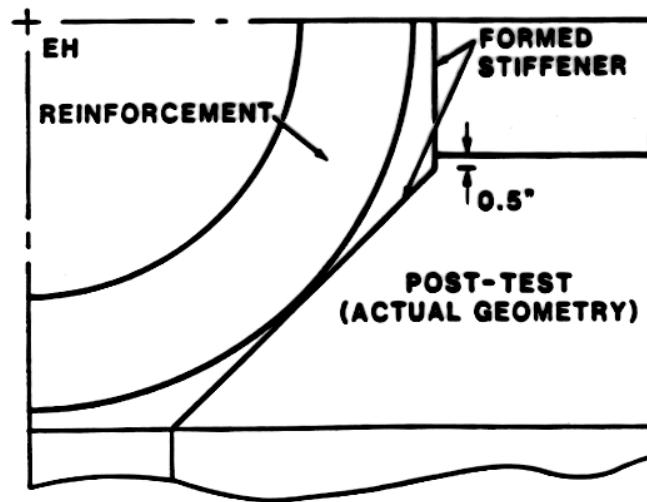
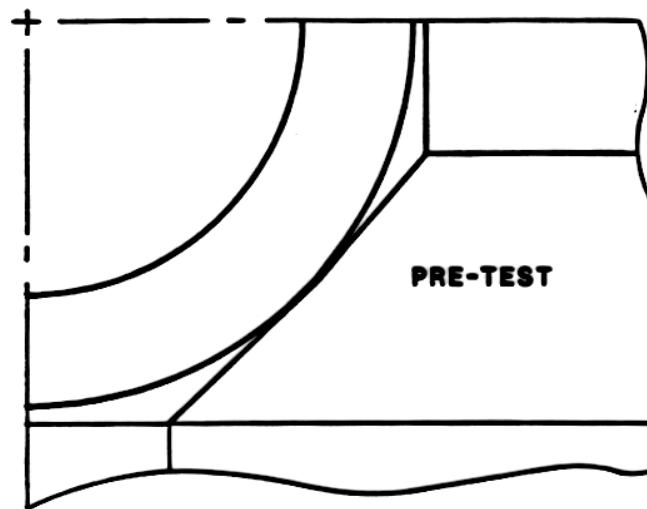


1:8-Scale SCV Model - Pretest

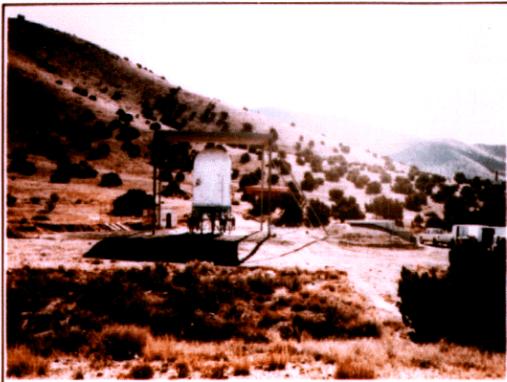


1:8-Scale SCV Model

- **Stiffener Detail**



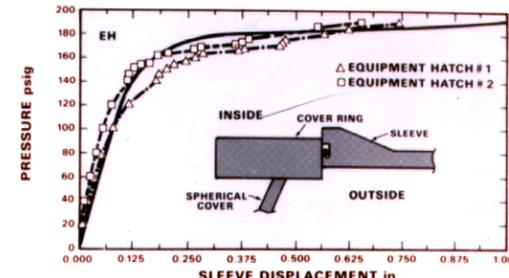
1:8-Scale SCV Model - Summary



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

	TEST	PREDICTION	POST-TEST ANALYSIS
Capacity [psig]	195	210	200
Failure Mechanism	Rupture	Leakage	Rupture
Location	Near EH1	Past EH1 or EH2 'O' Ring seals	Near EH1 or EH2

PERFORMANCE



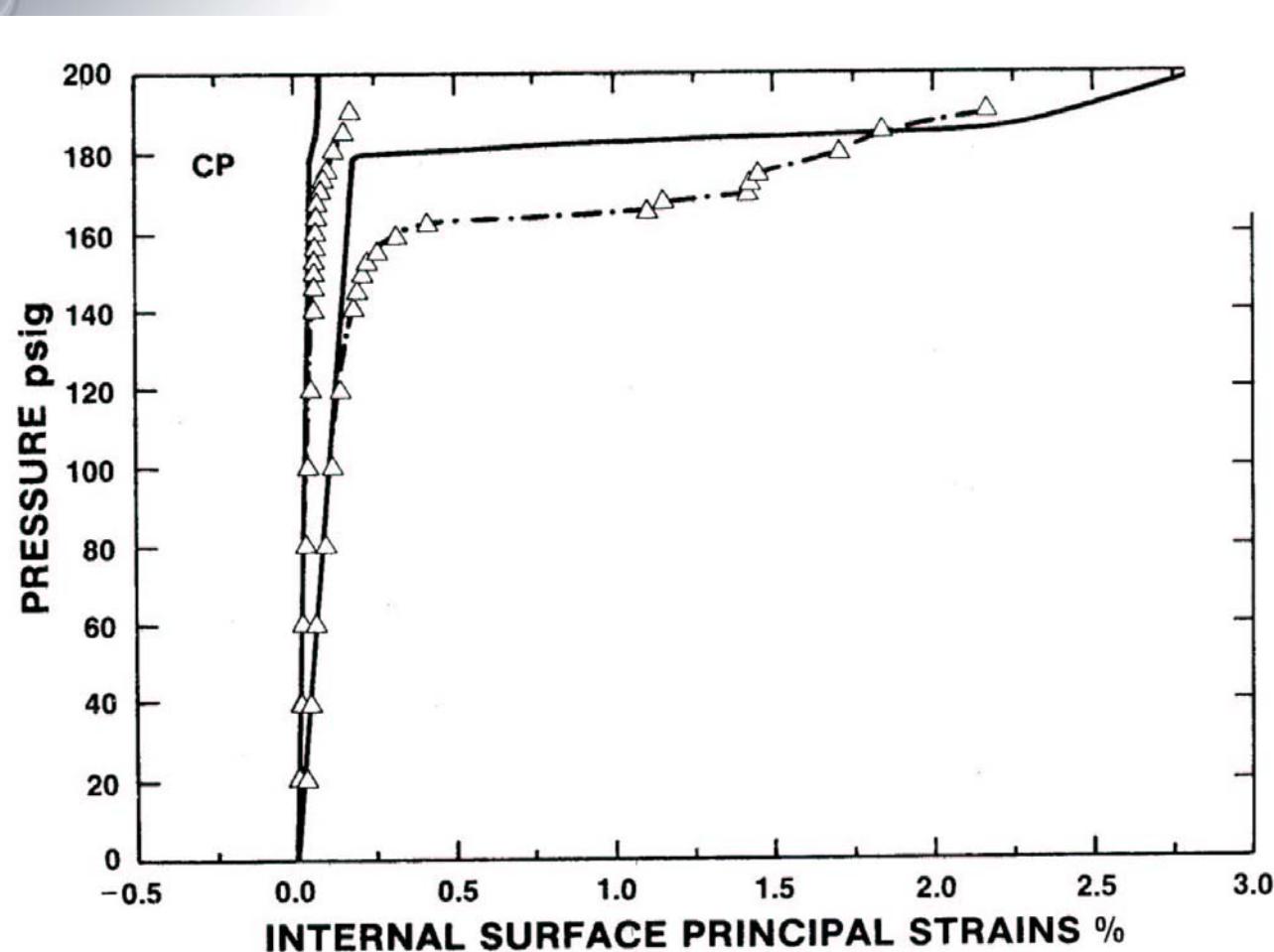
Ovalization Increased Rapidly After Cylinder Membrane Yielding Occurred

RESPONSE



AERIAL VIEW OF SITE AFTER RUPTURE

1:8-Scale SCV Global Response



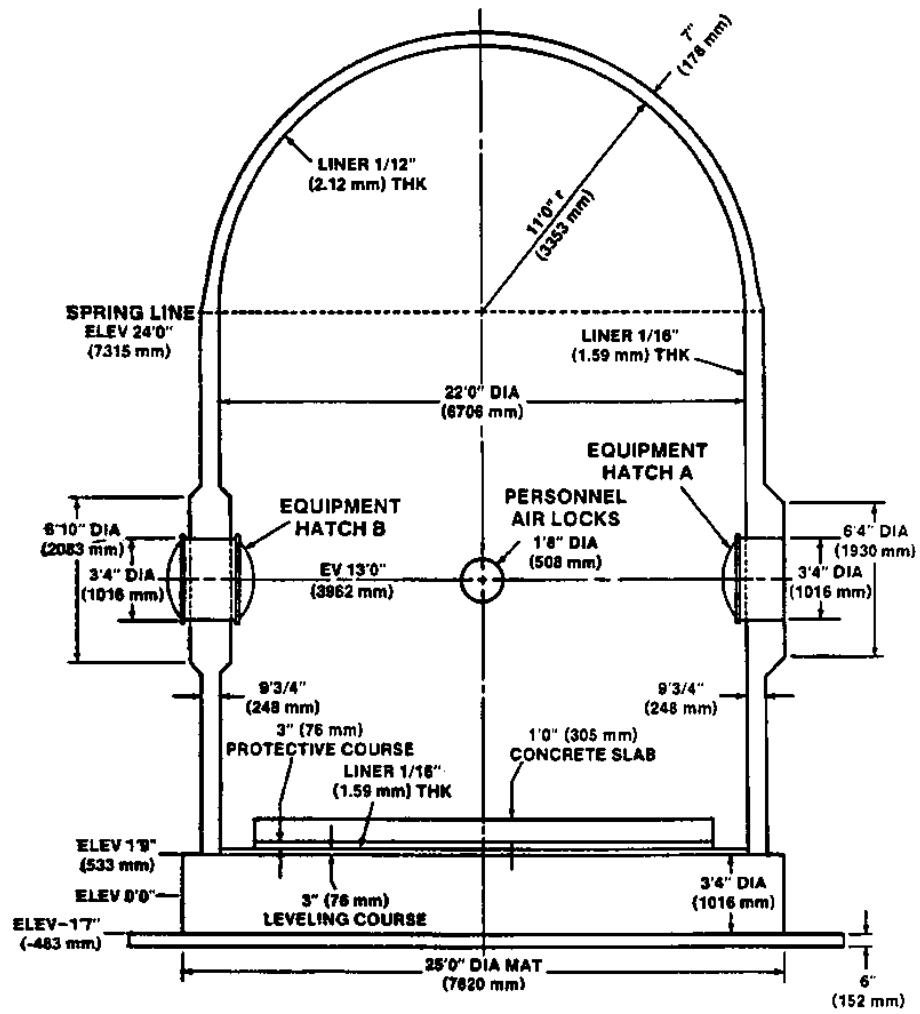


Steel Model Experiments - Summary

- Four out of five 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.

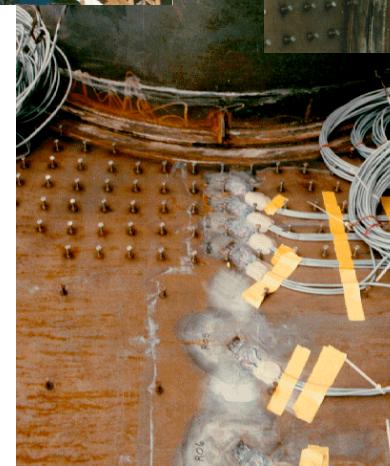
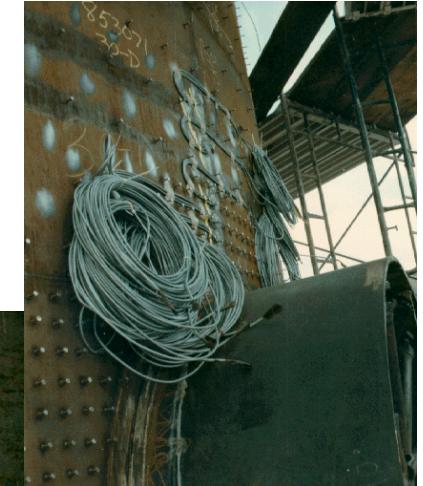
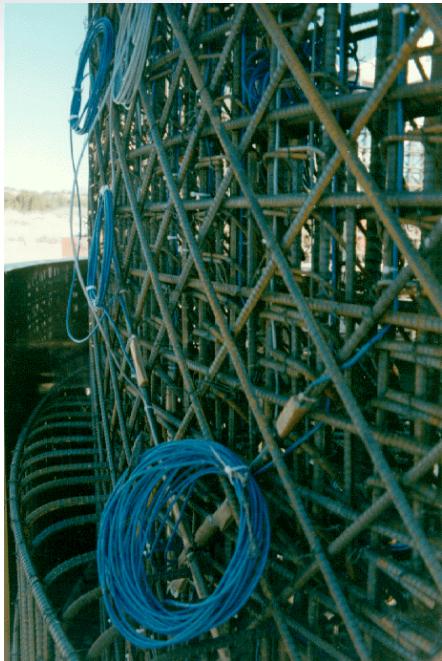
1:6-Scale RCCV Model

- 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

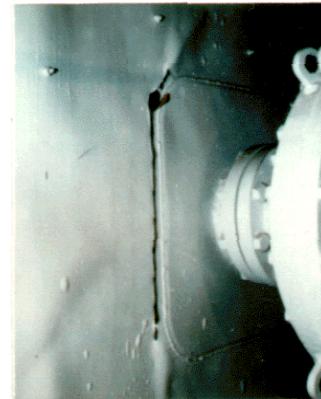


1:6-Scale RCCV Model - Summary

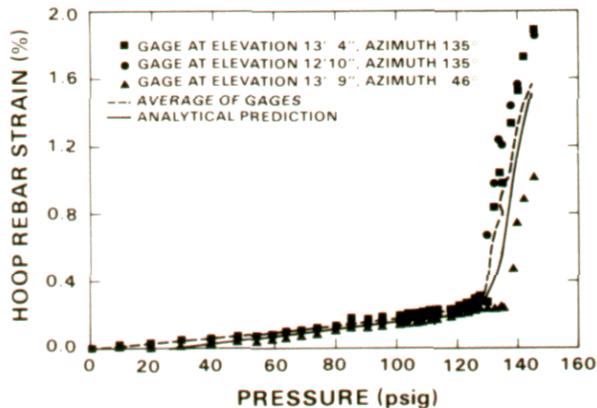
TEST OF A REINFORCED CONCRETE CONTAINMENT MODEL



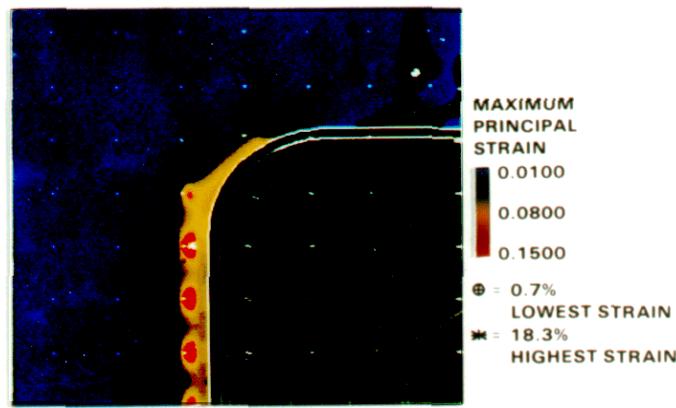
THE 1/6-SCALE
REINFORCED CONCRETE MODEL



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

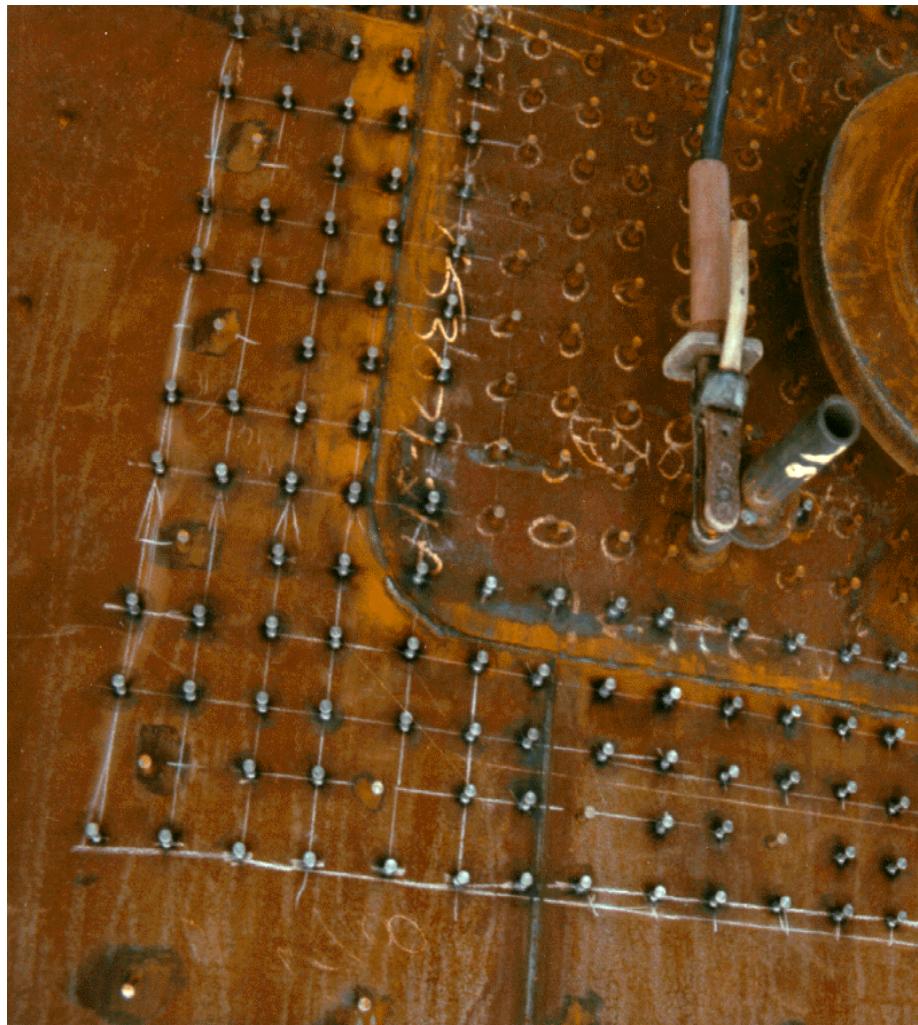
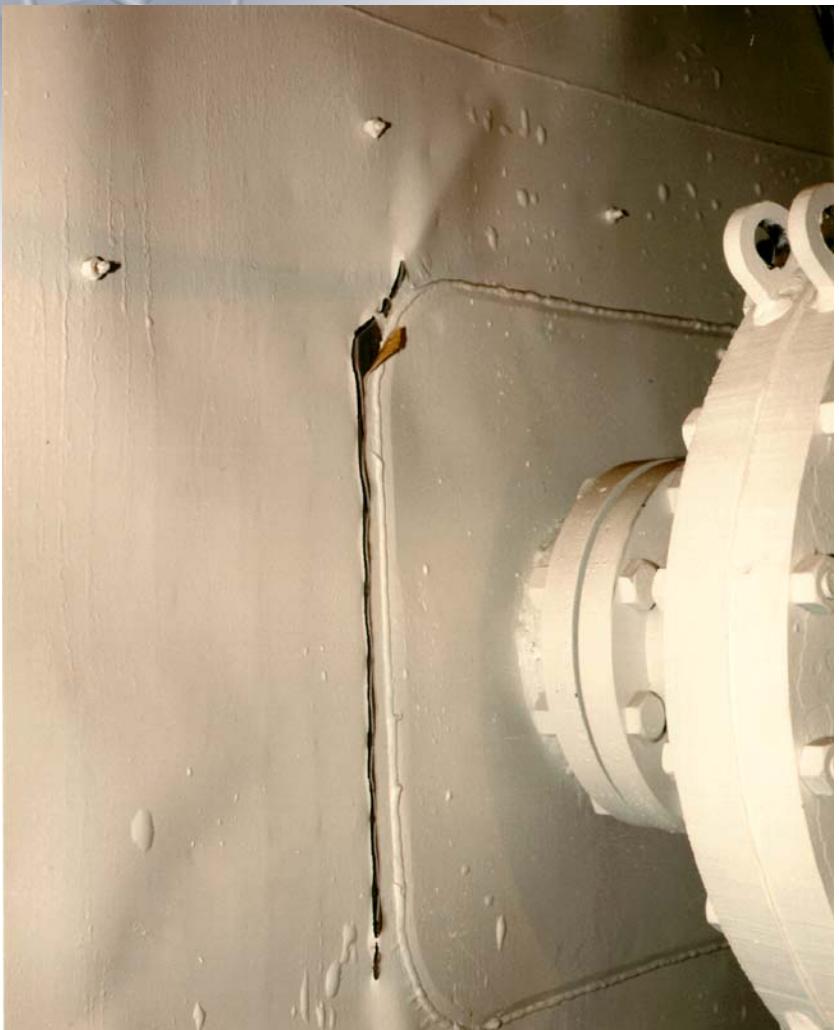


COMPARISON OF ANALYTICAL &
EXPERIMENTAL RESULTS OF HOOP REBAR
STRAINS AT CYLINDER MIDHEIGHT



POSTTEST ANALYSIS OF LINER
AND INSERT PLATE AT 145.5 psig

1:6-Scale RCCV Model





1:6-Scale RCCV Model - 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) - $3P_d$.
- '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.

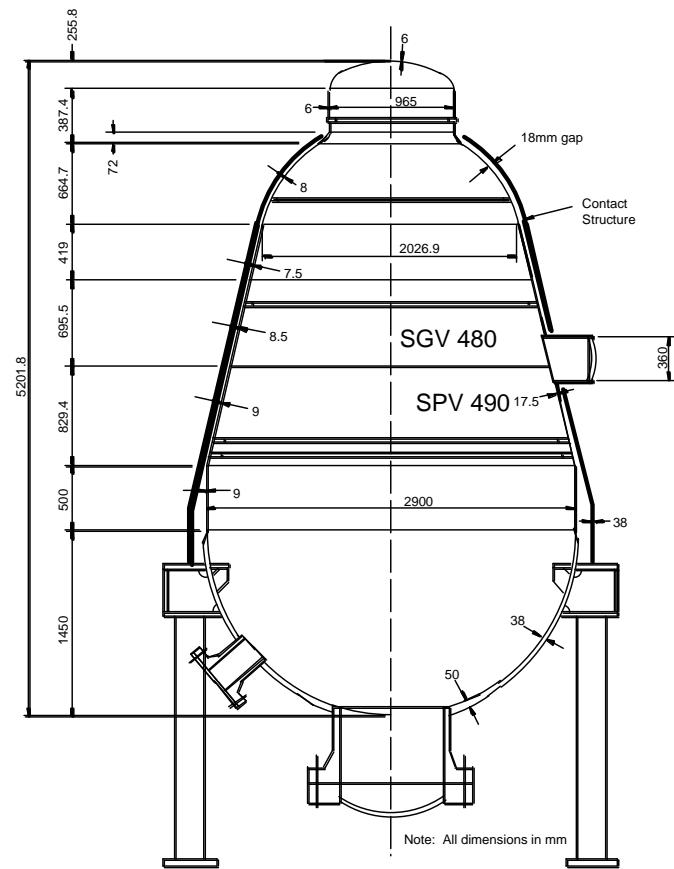


Containment Technology Test Facility

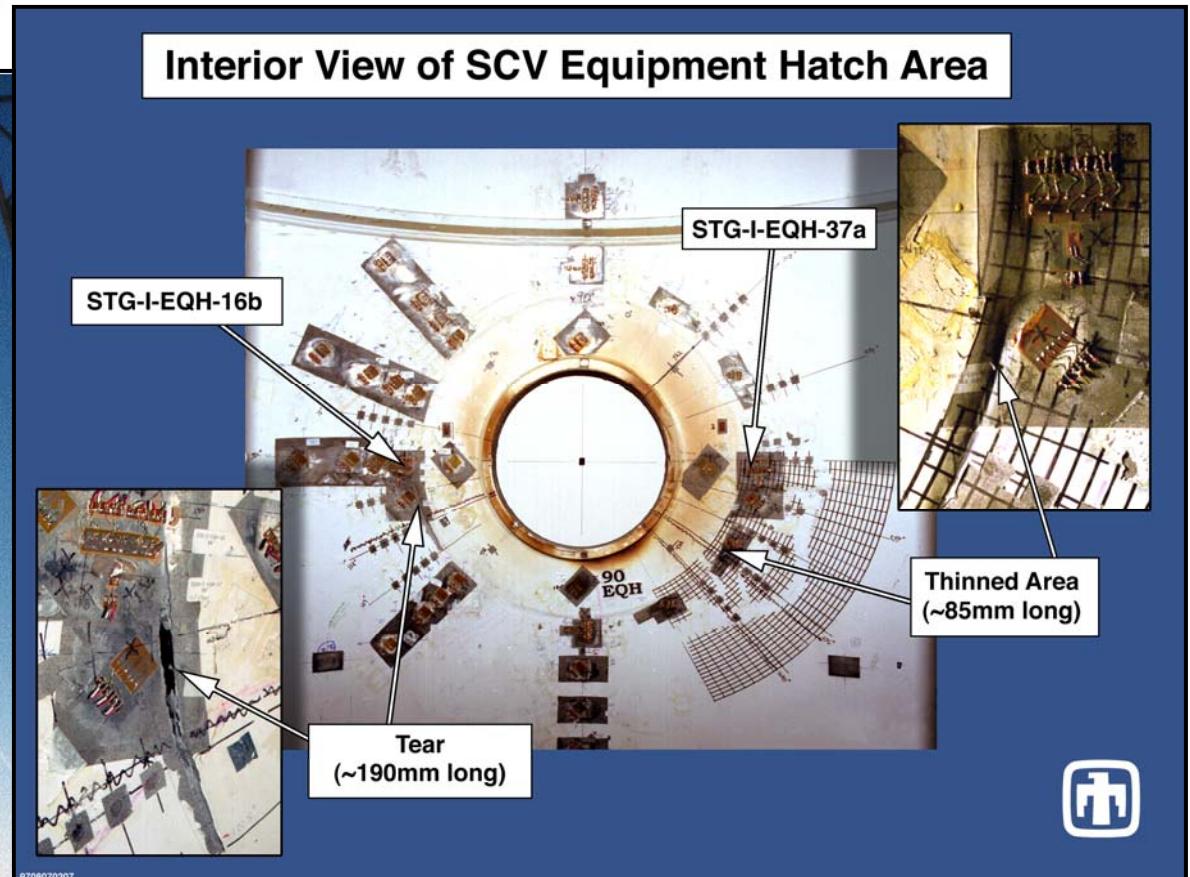
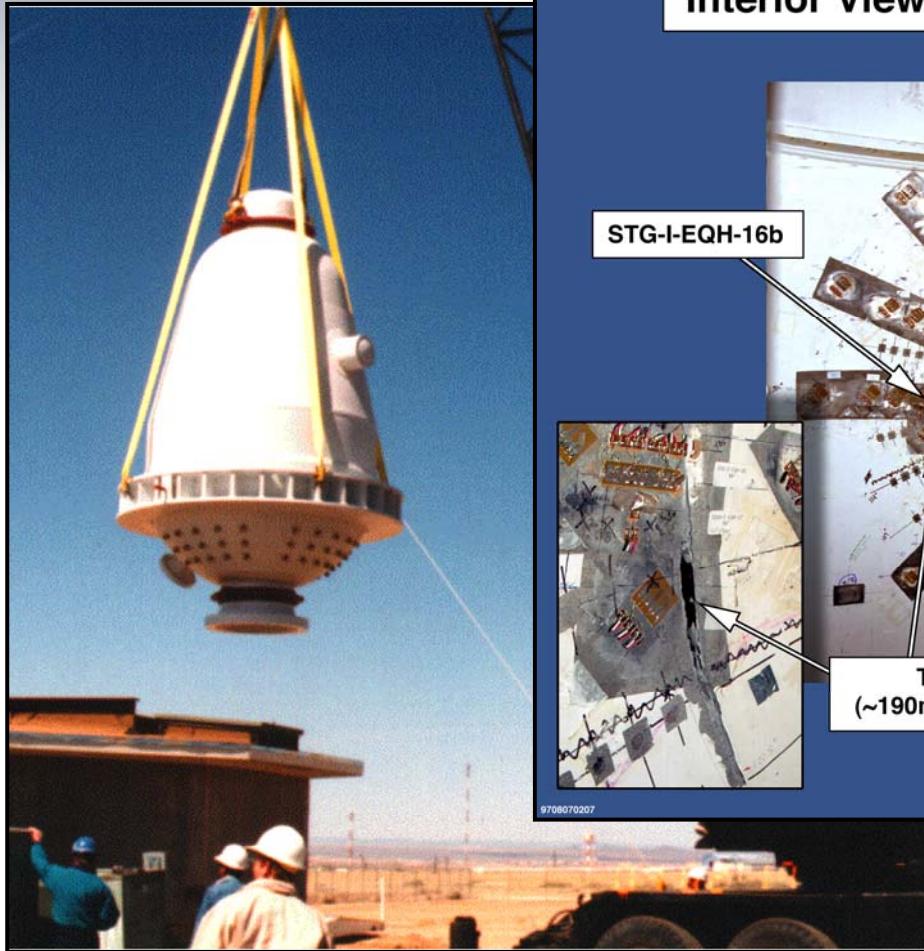


Steel Containment Vessel Model

- 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 $P_{da}=0.31 \text{ MPa (45 psig)-actual}$
 $P_{ds}=0.78 \text{ MPa (112.5 psig)-scaled}$
 - Materials: SGV480 ($F_y= 265 \text{ MPa, 38 ksi}$) ~ SA-516 Grade 70;
SPV490 ($F_y= 490 \text{ MPa, 71 ksi}$) ~ SA-537 Class 2
 - Contact Structure
 - Weight - 9 metric tons (20,000 lbs)
 - Material: SA-516-70 ($F_y = 38 \text{ ksi}$)
 - Nominal thickness = 38.1 mm (1.5 in.)
 - Low Pressure Test: $1.50 P_{ds} = 1.17 \text{ MPa (169 psig)}$
 - 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.



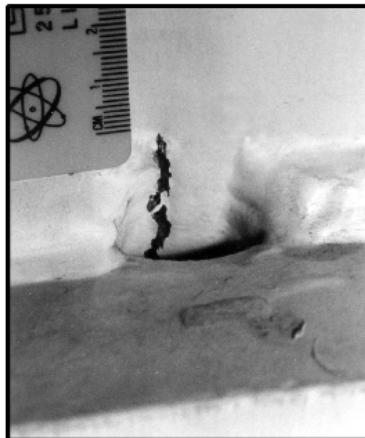
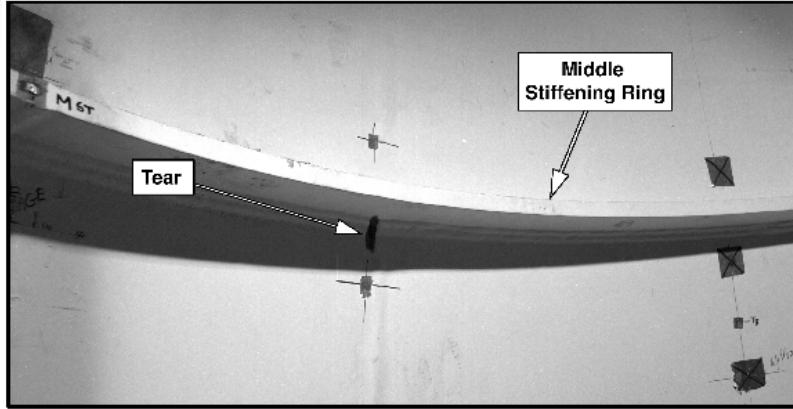
SCV Model Pre- and Posttest



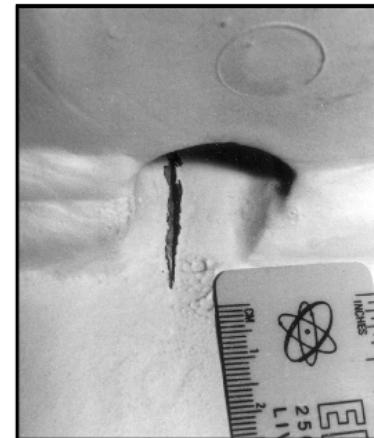
- 1/10th Scale
- Failure Pressure: 676 psig (6xDesign)
- Tested: Dec. 9-13, 1996



SCV Middle Stiffener - Posttest



Above



Below



SCV Round Robin Analysis

- **Agenzia Nazionale per la Protezione del Ambiente (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**

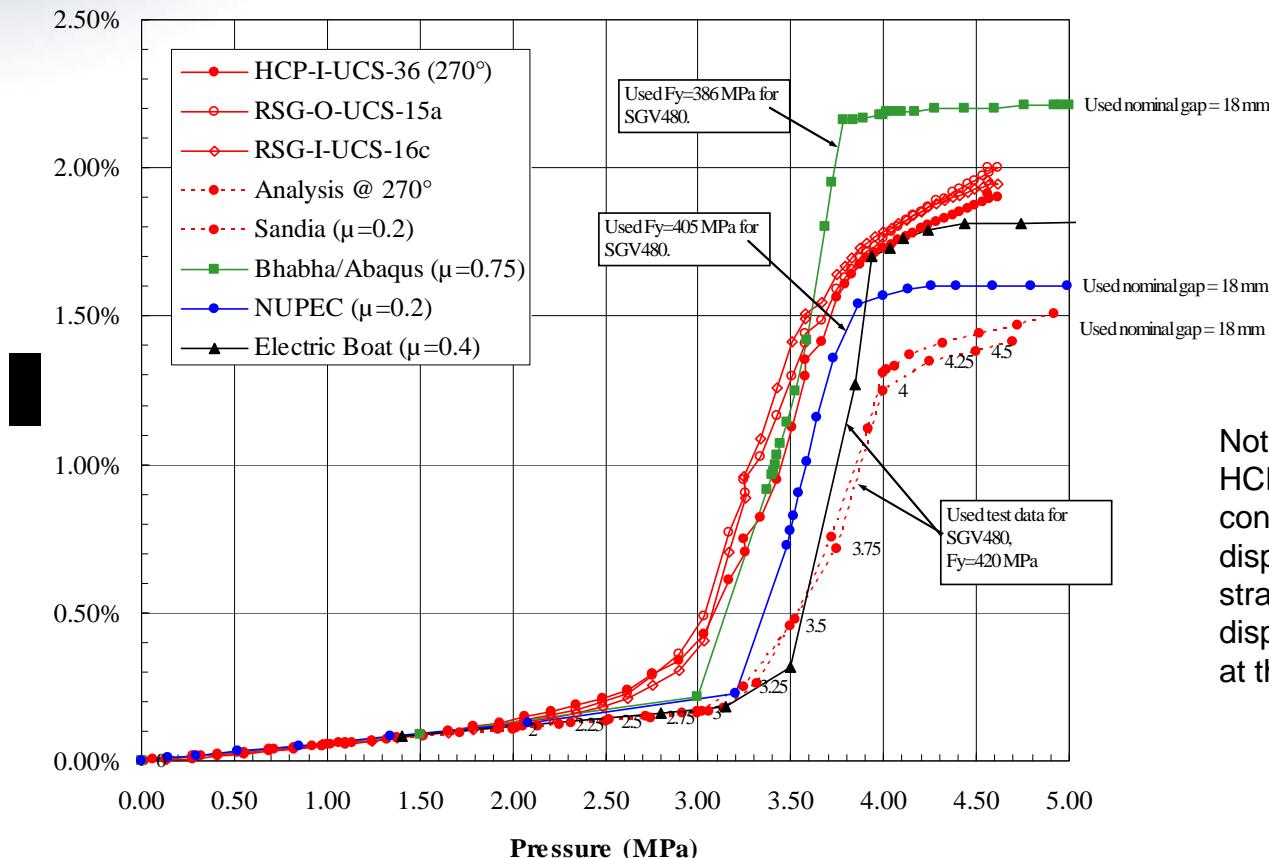


Summary of SCV RR Pretest Results

Organization	Failure Pressure	Failure Location	Failure Mode
Test	(4.5?)-4.7 MPa	(‘Rat-hole’) E/H Insert Plate	Material Failure
ANL	4.9 - 5.5 MPa	Knuckle	Material failure
ANPA	10.9 MPa	Drywell Head	Buckling
BARC	11.5 - 12.0 MPa	Drywell Head	Material failure or buckling
GD/EB	4.7 MPa	Thinned Liner @ Equipment Hatch	Material failure
JAERI	>4 MPa	Drywell Head	Buckling
MPA			
NUPEC	4 - 7.3 MPa	Thinned Liner @ E/H	Material failure
	7.3 - 11.8 MPa	Knuckle	
SNL	4.5 MPa	Thinned Liner @ Equipment Hatch	Material failure

SCV Posttest Analysis

- 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.



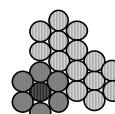
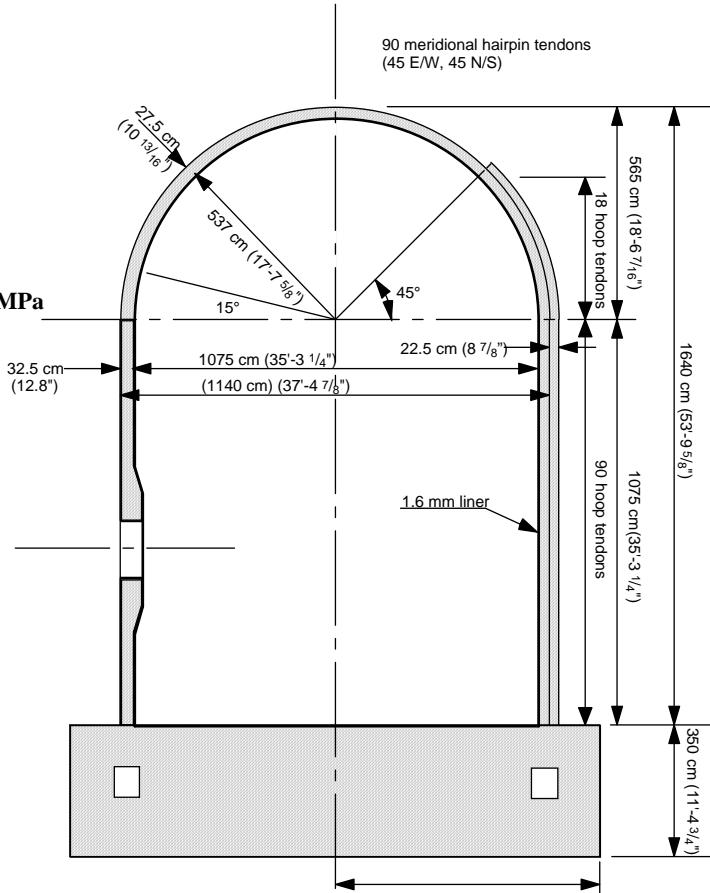
SCV Posttest Analysis

- **Conclusions:**

- 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.

Prestressed Concrete Containment Vessel Model

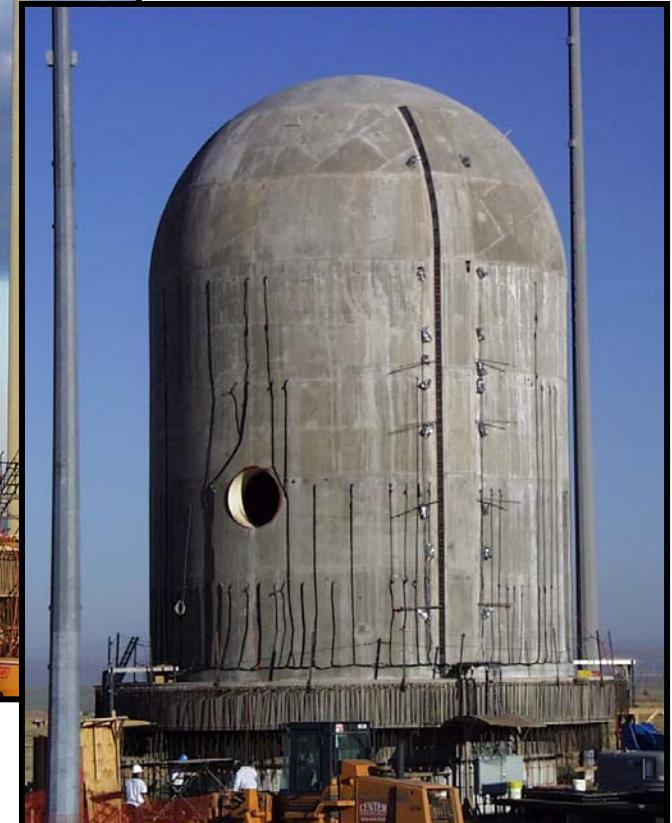
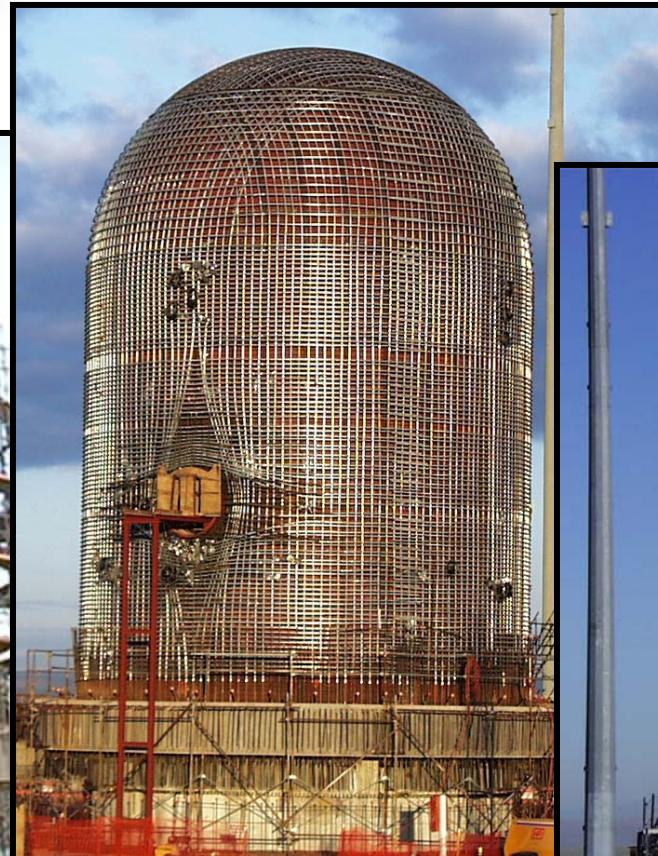
- Model of OHI-3 in Japan, PWR, 2-buttress, supplied by NUPEC
- Scale: 1:4 overall (except free-field liner anchor spacing)
- Design Pressure, P_d : 0.39MPa (56.9 psig)
- Materials:
 - Liner: SGV410, $F_y = 225$ MPa (33 ksi), $F_t = 410$ MPa (59 ksi)
 - Anchor: SS400, $F_y = 235$ MPa (34 ksi), $F_t = 392$ 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):
 - $\sim 3.25 P_d$ (1.28 MPa [185 psi]) - liner tearing @ E/H



PCCV Model Tendon (Actual Size)
3 - 13.7mm, twisted 7 wire strand, $A=3.393$ cm²



PCCV Model Construction



PCCV Pretest Round Robin Participants



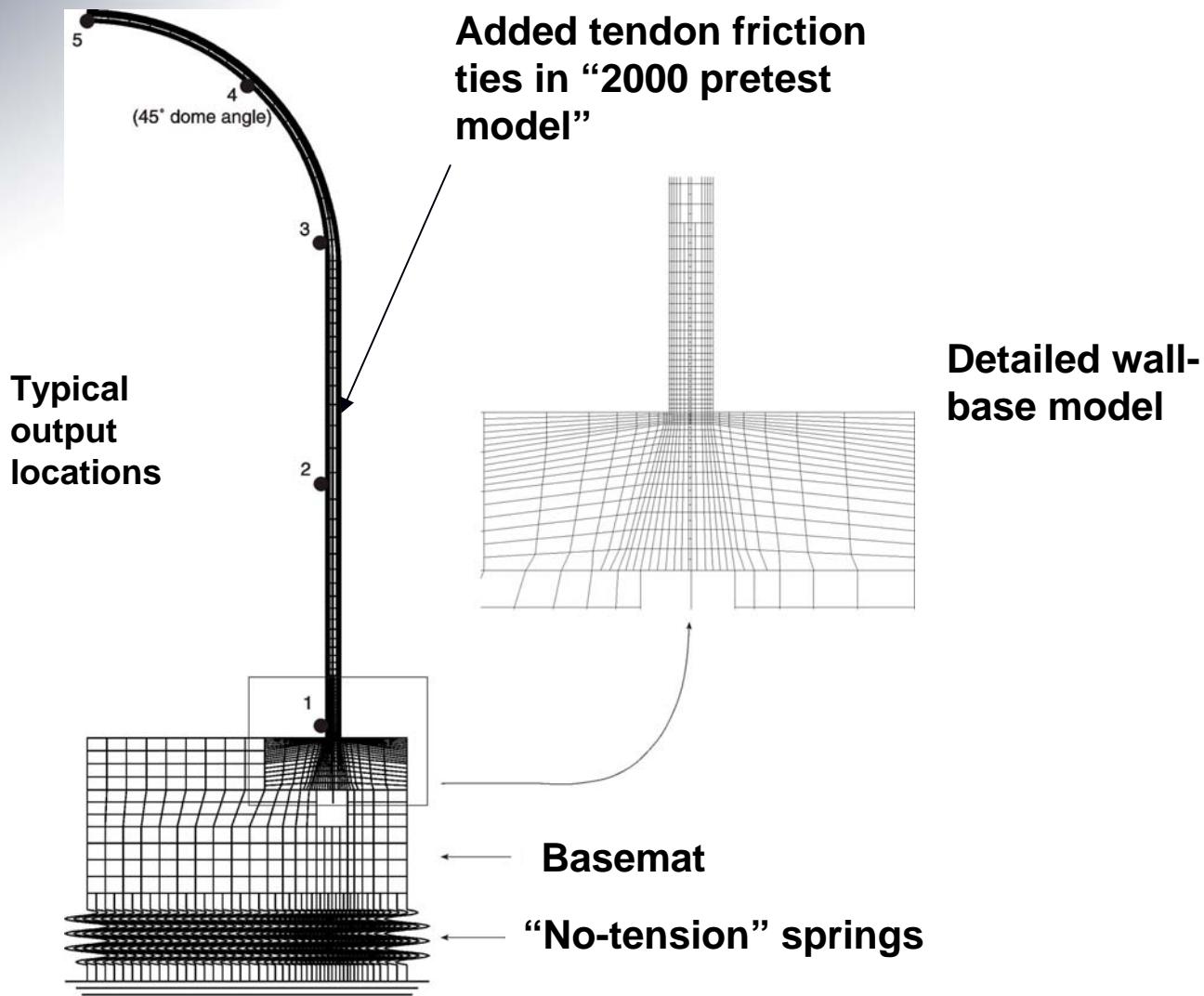
- Argonne National Laboratory (ANL) (U.S.)
- Atomic Energy of Canada Limited (AECL) (Canada)
- Commissariat A L'Energie Atomique/Saclay/DRN (France)
- Electricite de France (EDF) (France)
- Institute of Nuclear Energy Research (INER) (Repub. of China)*
- Institut de Protection et de Sûreté Nucléaire (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)/ANATEL
- University of Glasgow (U.K.)



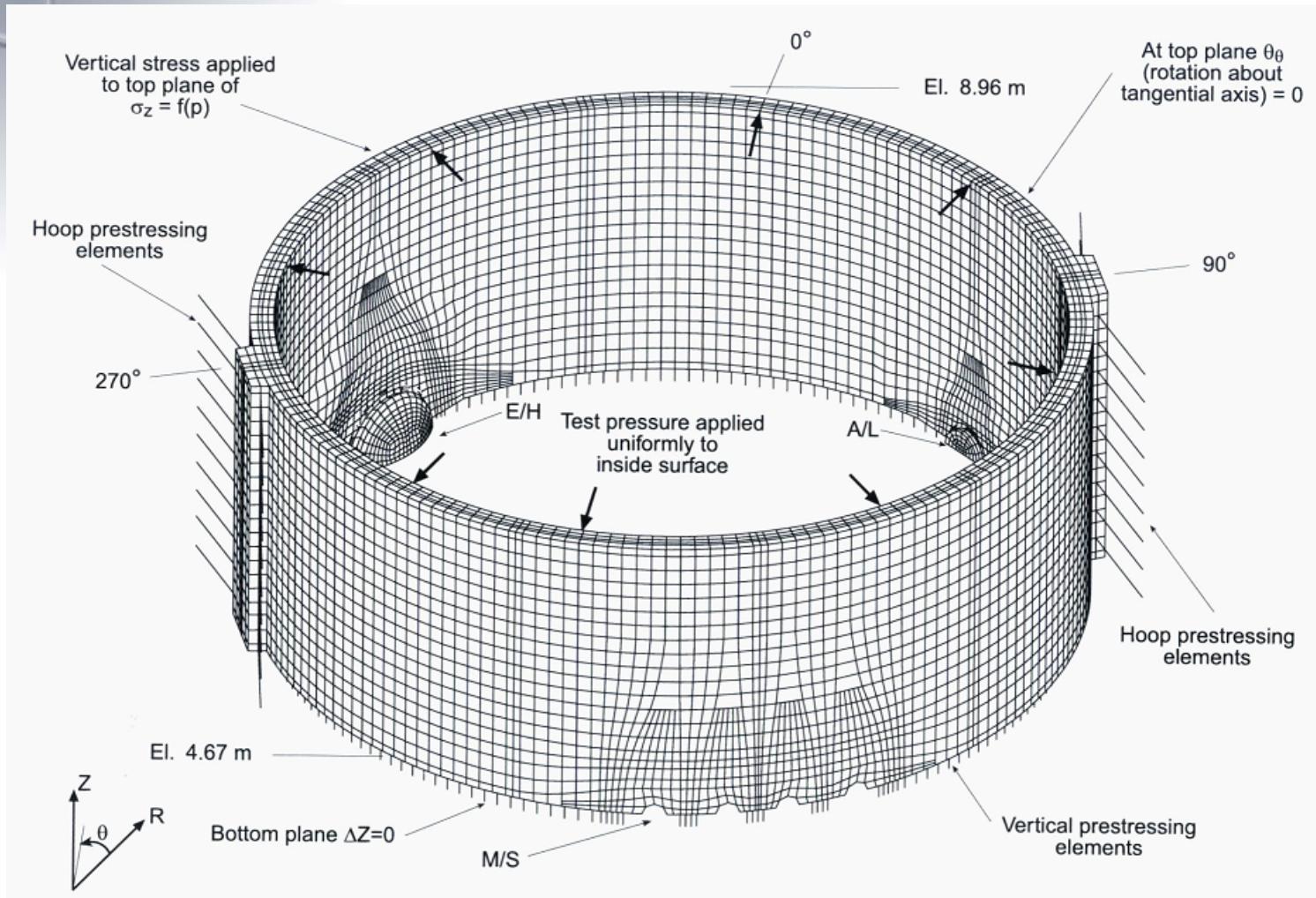
Summary of PCCV RR Pretest Results

	<u>Pressure (MPa)</u>	<u>Failure Mode</u>
• ANL	1.51-1.62	local liner tear/hoop tendon failure @ El. 6.4 m
• AECL	0.94-1.24	complete cracking/axisymmetric yield
• CEA	1.60-1.70	numerically unstable
• EDF	1.95	
• INER	0.81	
• JAERI		buckling @ dome or local fracture by bending in cylinder
• JAPC	1.45-1.55	hoop tendon/rebar/liner rupture @ El. 7 m
• KINS	1.25-1.44	tendon rupture
• KOPEC	1.30-1.51	tendon rupture (@3.55% strain)
• HSE/NNC	1.00	liner tear w/ extensive concrete cracking @ buttress
• NUPEC	1.49-1.57	tendon rupture
• IBRAE	1.26	tendon rupture
• Principia	1.30	tendon yielding
• RINSC	1.50	hoop failure of vessel
• ANATECH/SNL	1.25	liner tearing (16%) @ E/H
	1.40	tendon rupture
● Test	0.98	1.5% mass/day leak through liner tear @ E/H
	1.30	limit of pressurization capacity during LST
	1.42	hoop tendon and rebar rupture during SFMT

Global Axi-Symmetric Model

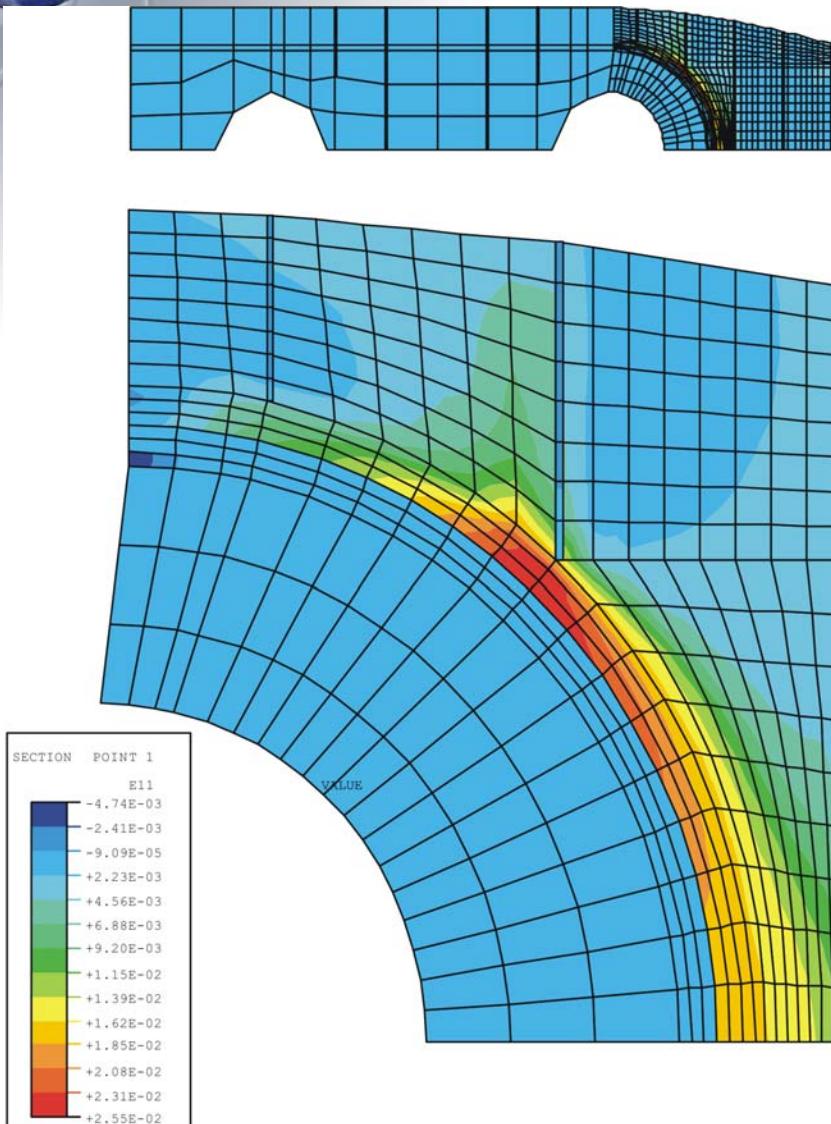


3DCM Model



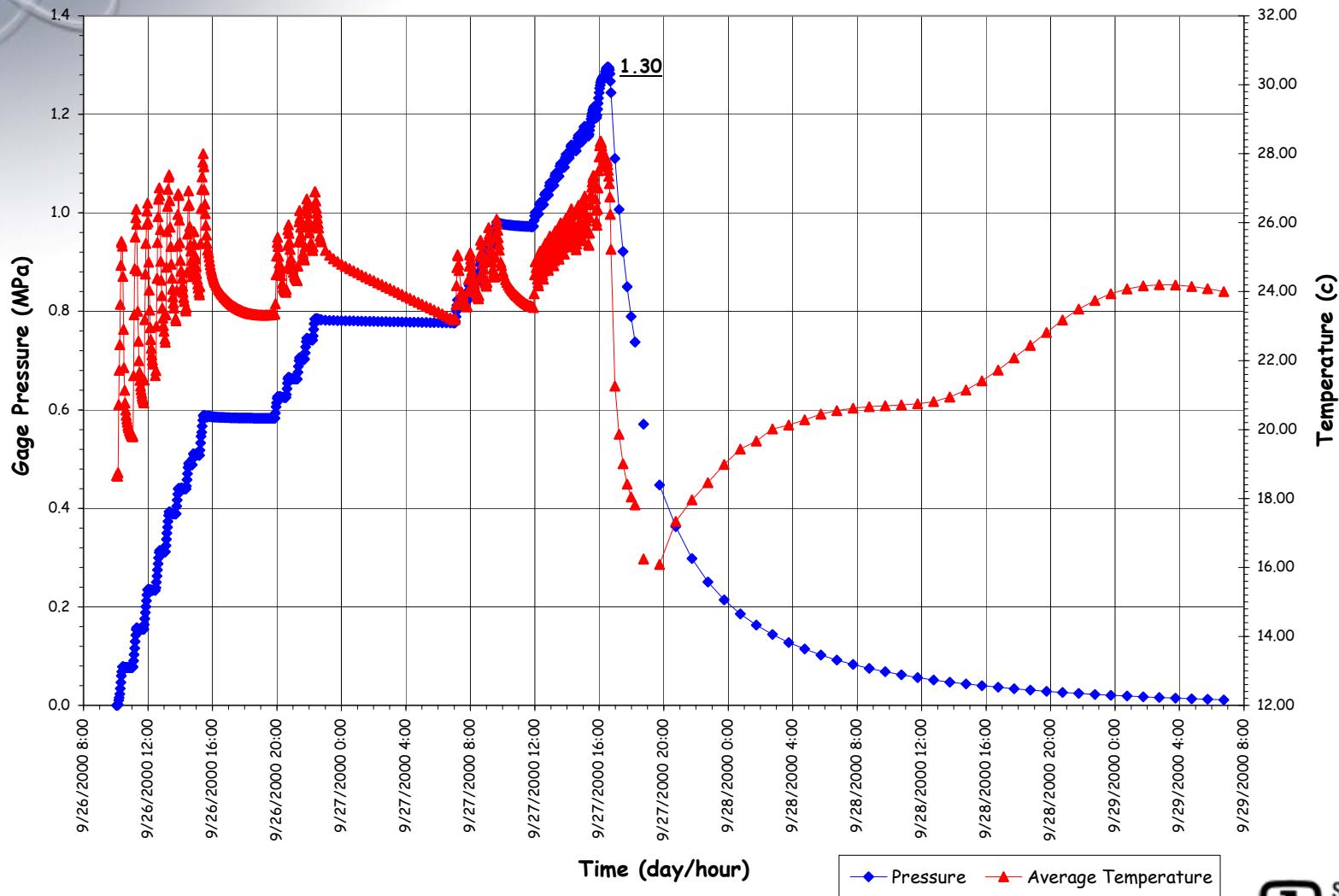


M/S Penetration Model

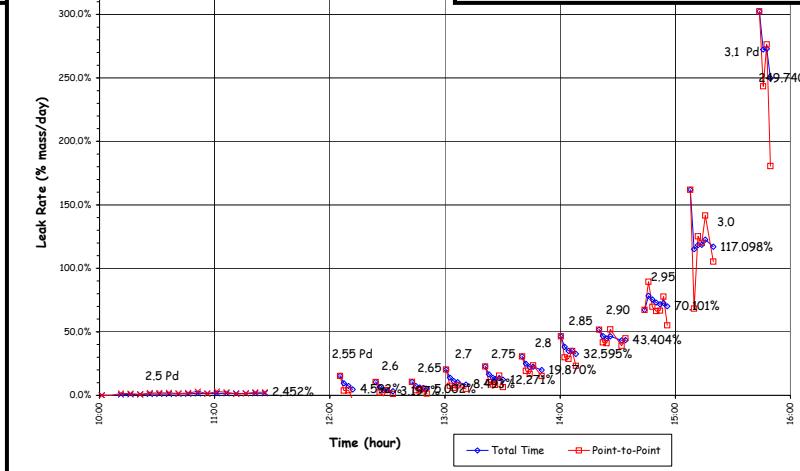
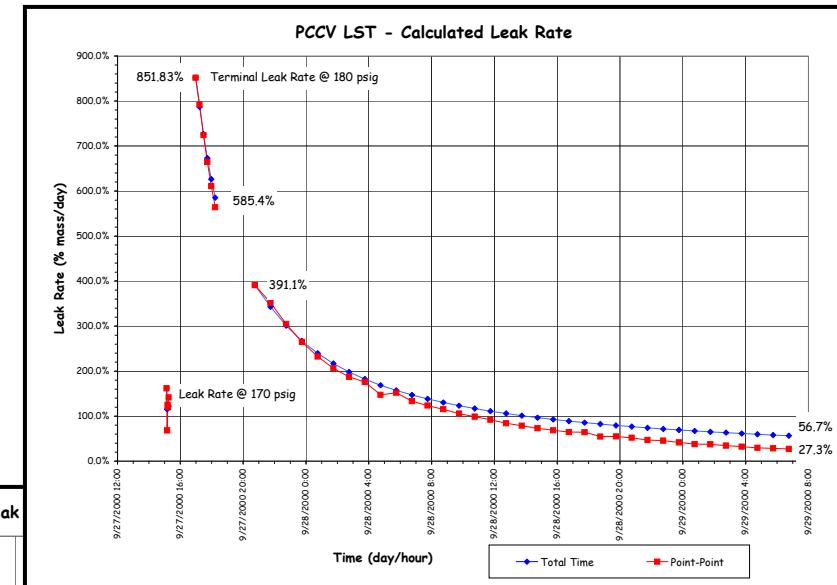
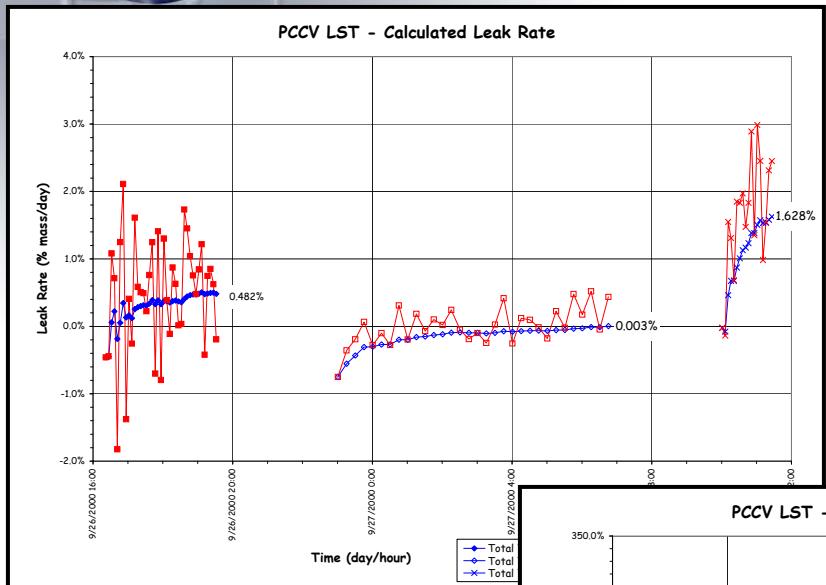


- Strain concentration at insert plate

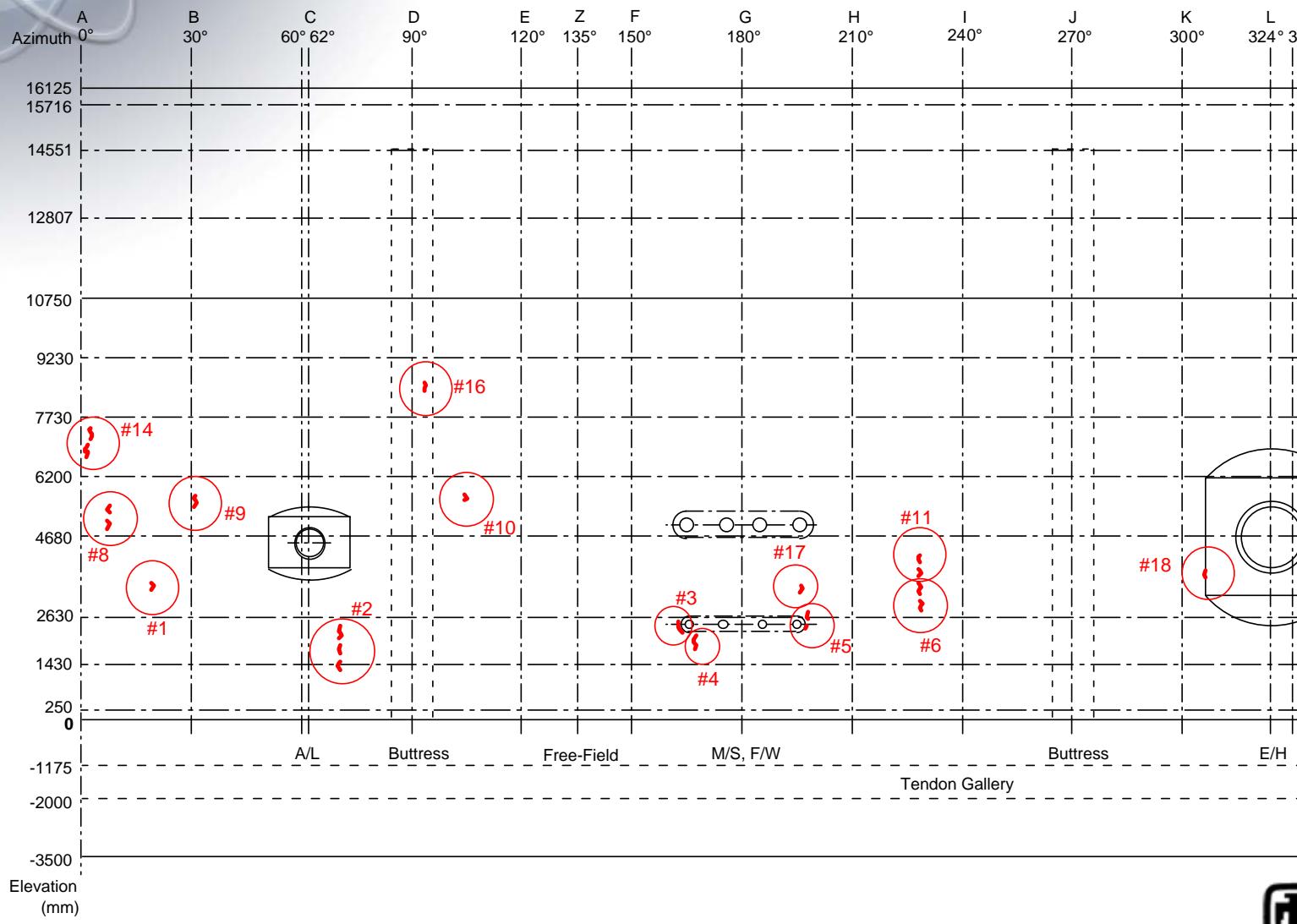
PCCV Limit State Test (LST)



PCCV LST Leak Rates



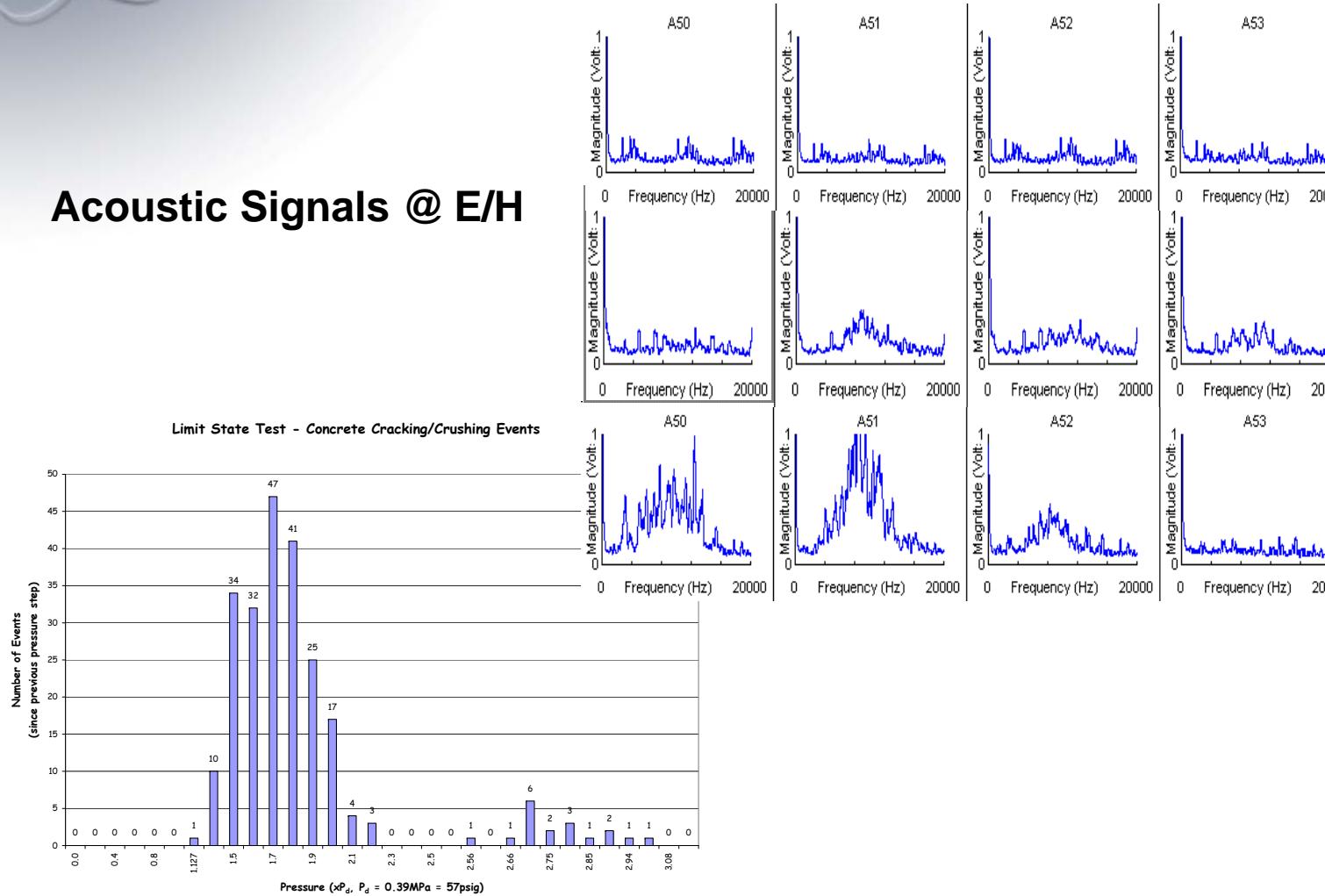
PCCV LST Liner Tears





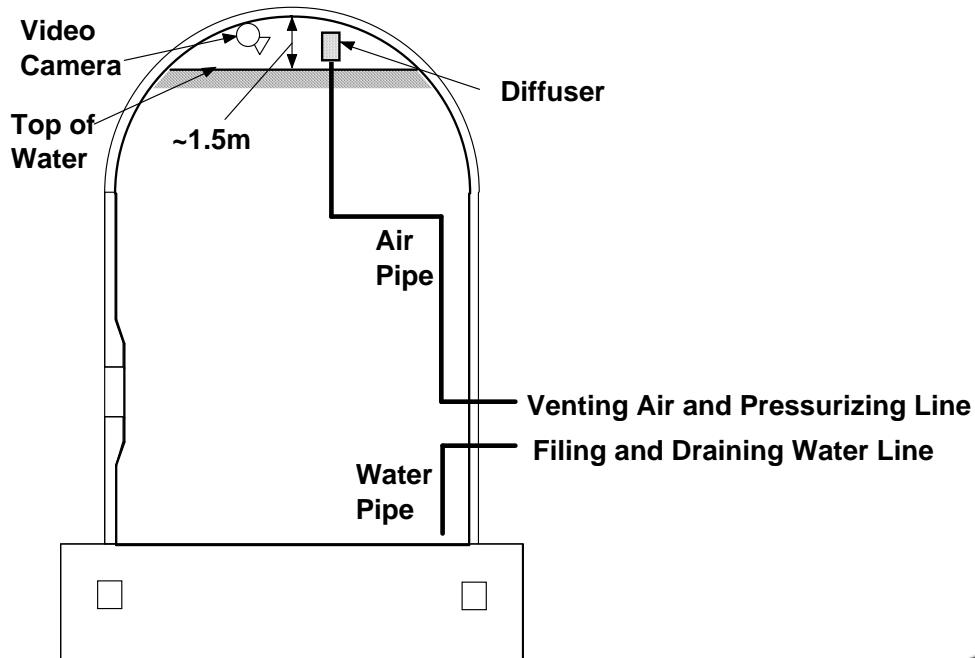
PCCV LST Acoustic Response

Acoustic Signals @ E/H



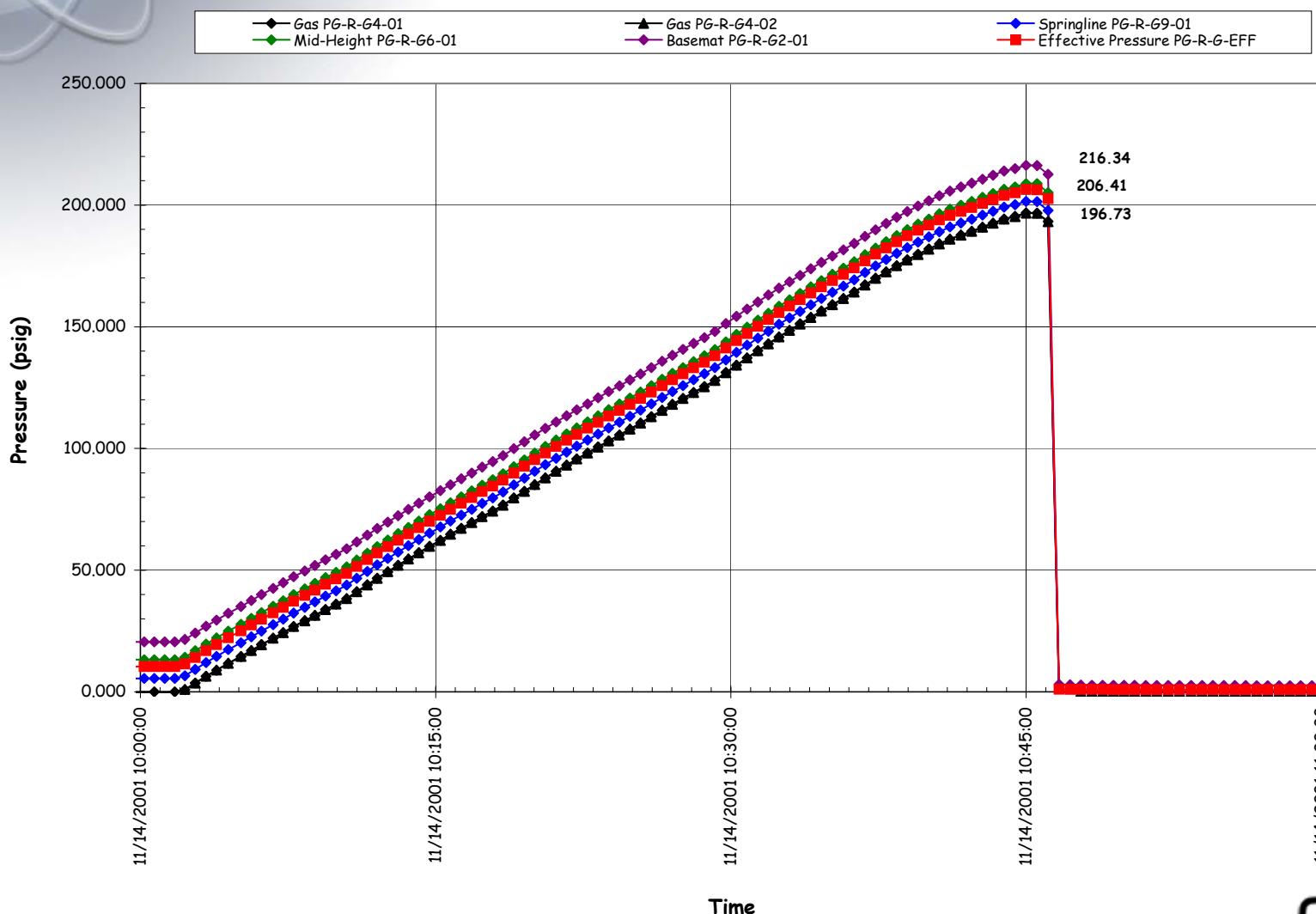
- **Structural Failure Mechanism Test:**

- Justification: LST did not completely satisfy pre-test objective of providing data to validate response predictions ‘well into the in-elastic regime’.





SFMT Pressure Time Histories





PCCV Model Structural Failure Mode Test

November 14, 2002, 10:46:12 AM



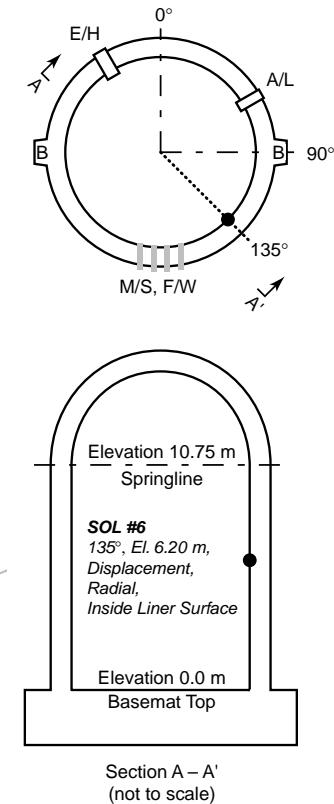
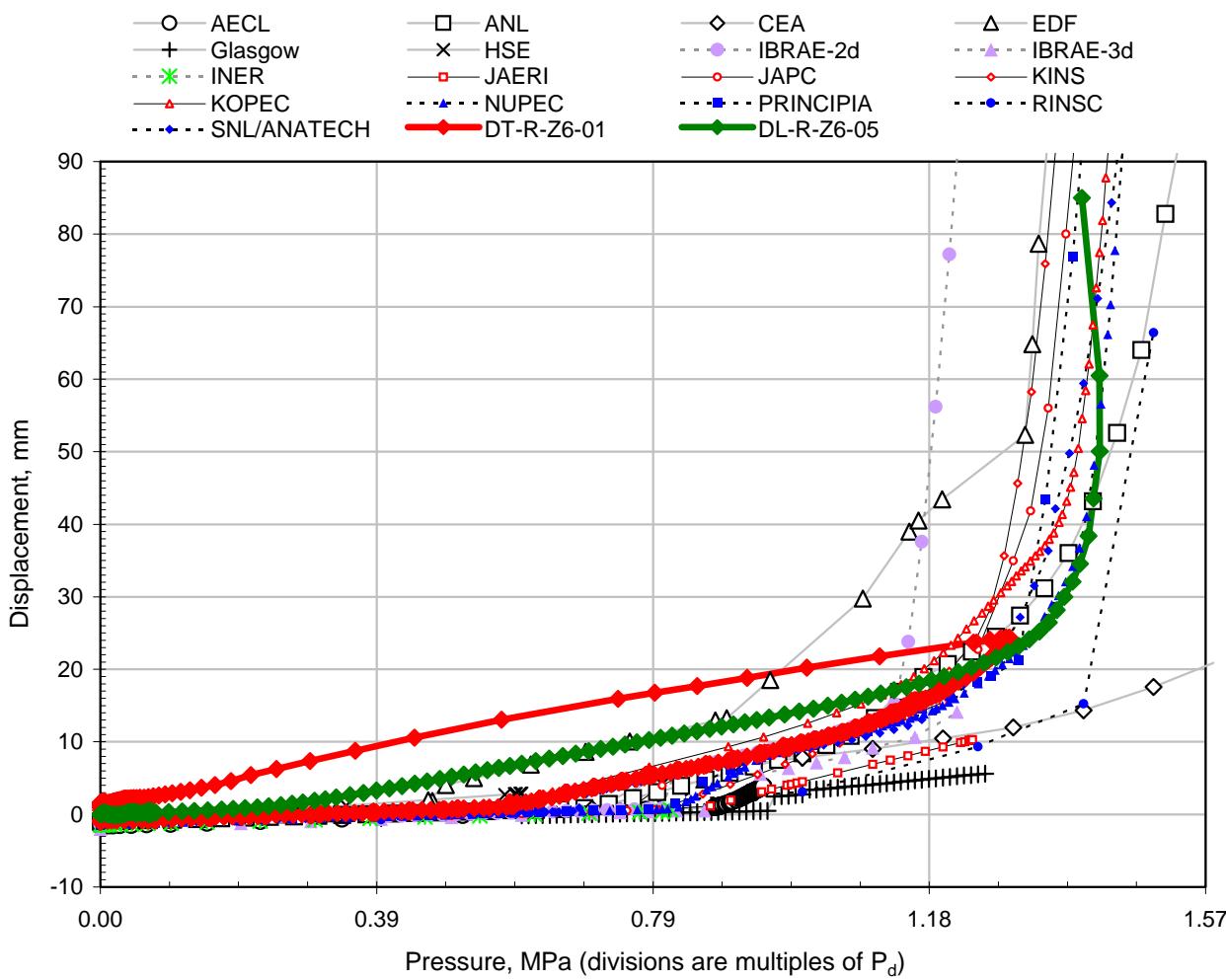
PCCV Model after SFMT

November 14, 2002





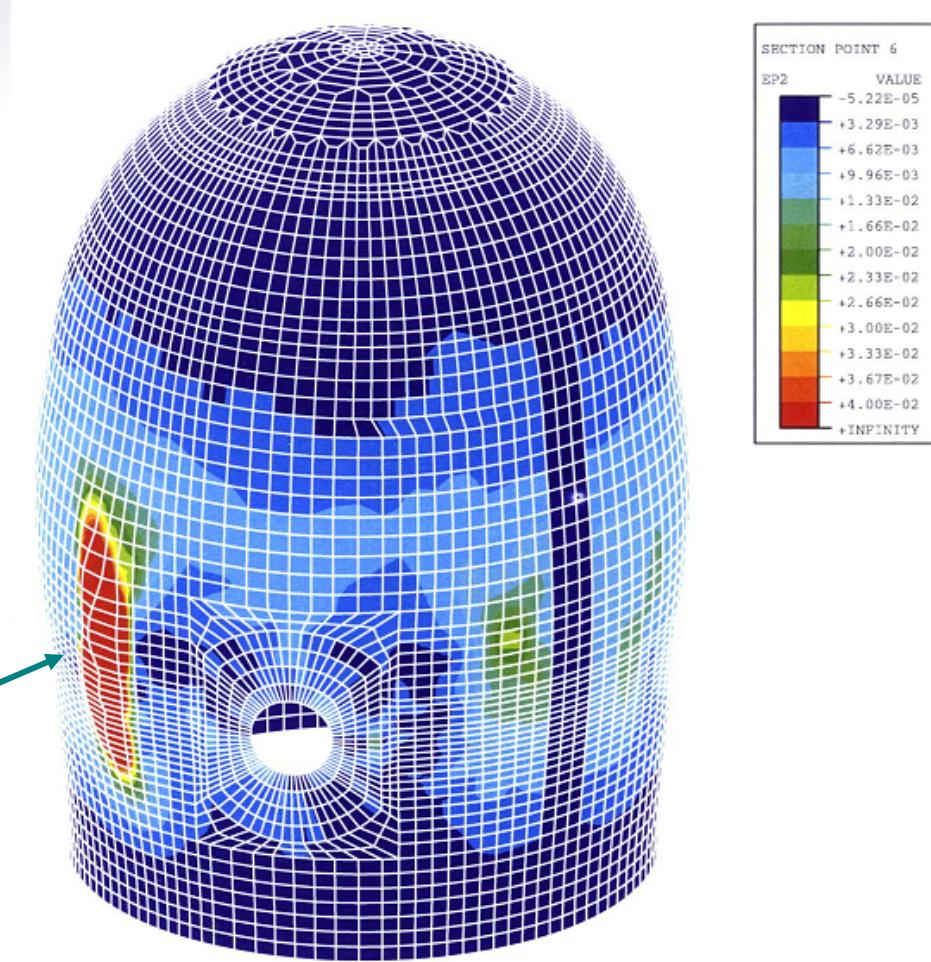
Round Robin Predictions





3D Shell Response @ 1.38 MPa (3.51Pd)

Tendon Rupture
(Strain ~4%)

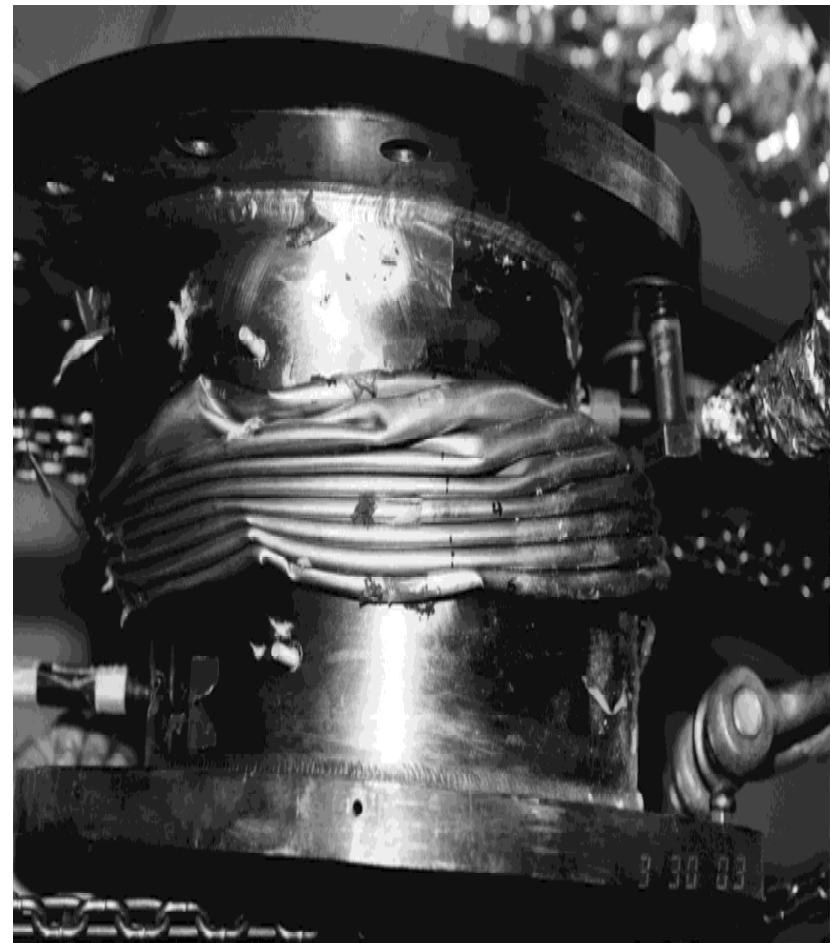
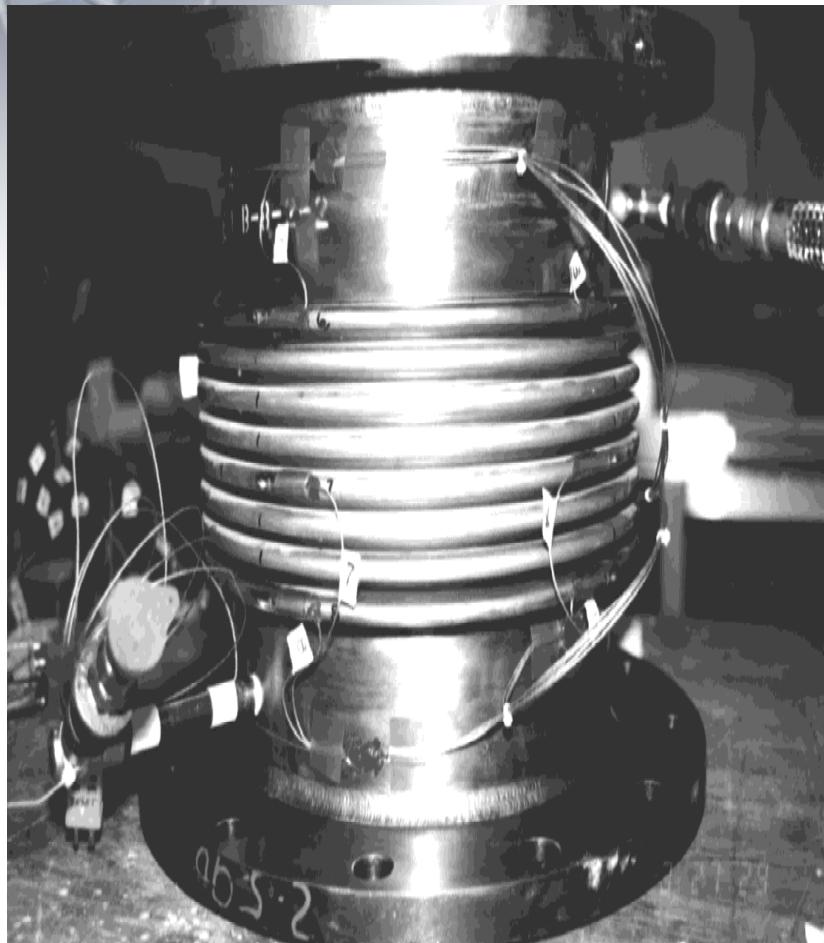


1.381 MPa

Overpressurization Tests - Conclusions

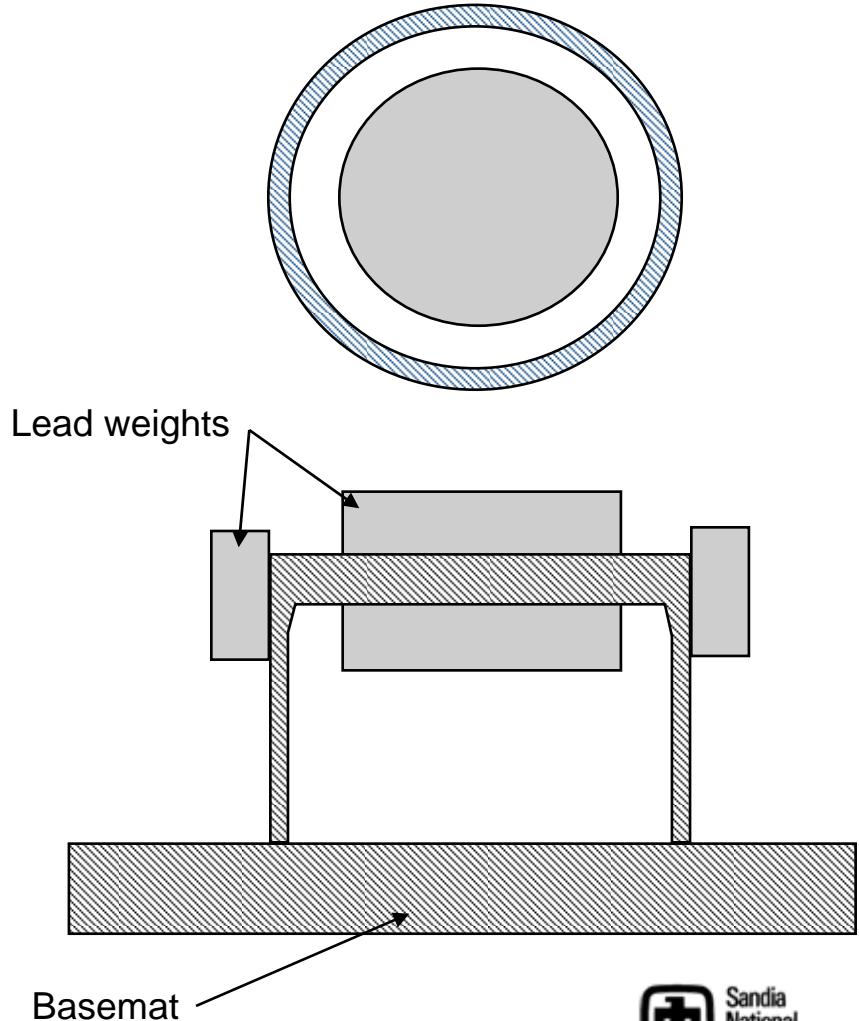
- Large amounts of data on elastic and in-elastic response of representative models of containment vessels were obtained for comparison with analyses.
- Significant plastic 'free-field' strains were developed before failure, with considerable margin between design and failure pressures.
 - 'Free-field' 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



PCCV Scaled Model

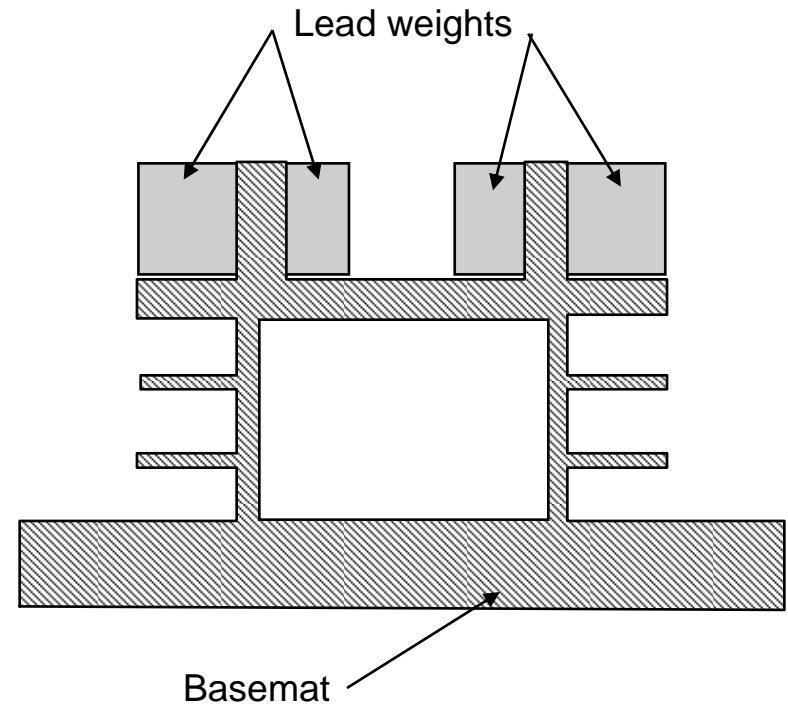
- **1:10 scale geometry**
- **1:8 scale concrete wall thickness**
- **1:4 scale liner thickness & anchors**
- **Dome truncated & 420 metric tons lead weighs attached**
- **Input accelerations scaled**
 - Magnitudes multiplied by 0.75
 - Frequency increased by 2.56 (i.e., time scaled compressed by factor of 2.56)





RCCV Scaled Model

- 1:8 scale geometry
- 1:10 scale concrete wall thickness
- 1:4 scale liner thickness & anchors
- Dome truncated & lead weights attached at top
- Input accelerations scaled similar to PCCV model scaling



Aircraft Impact Test



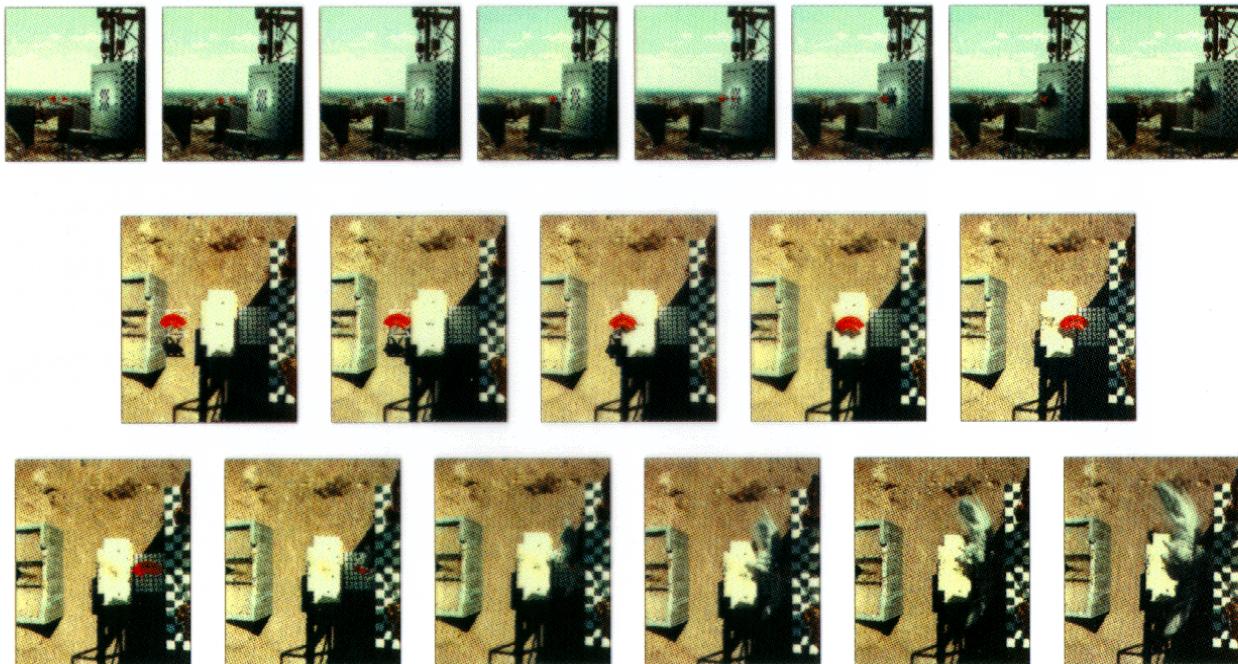
Sandia National Laboratories
Photometrics Division



Sandia
National
Laboratories

Turbine Missile Impact Tests

Turbine Missile Concrete Impact Test Series Test #3



Missile mass 147 kg, velocity 13.1 m/sec; Panel mass 153 tonnes, thickness 1.37m

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?



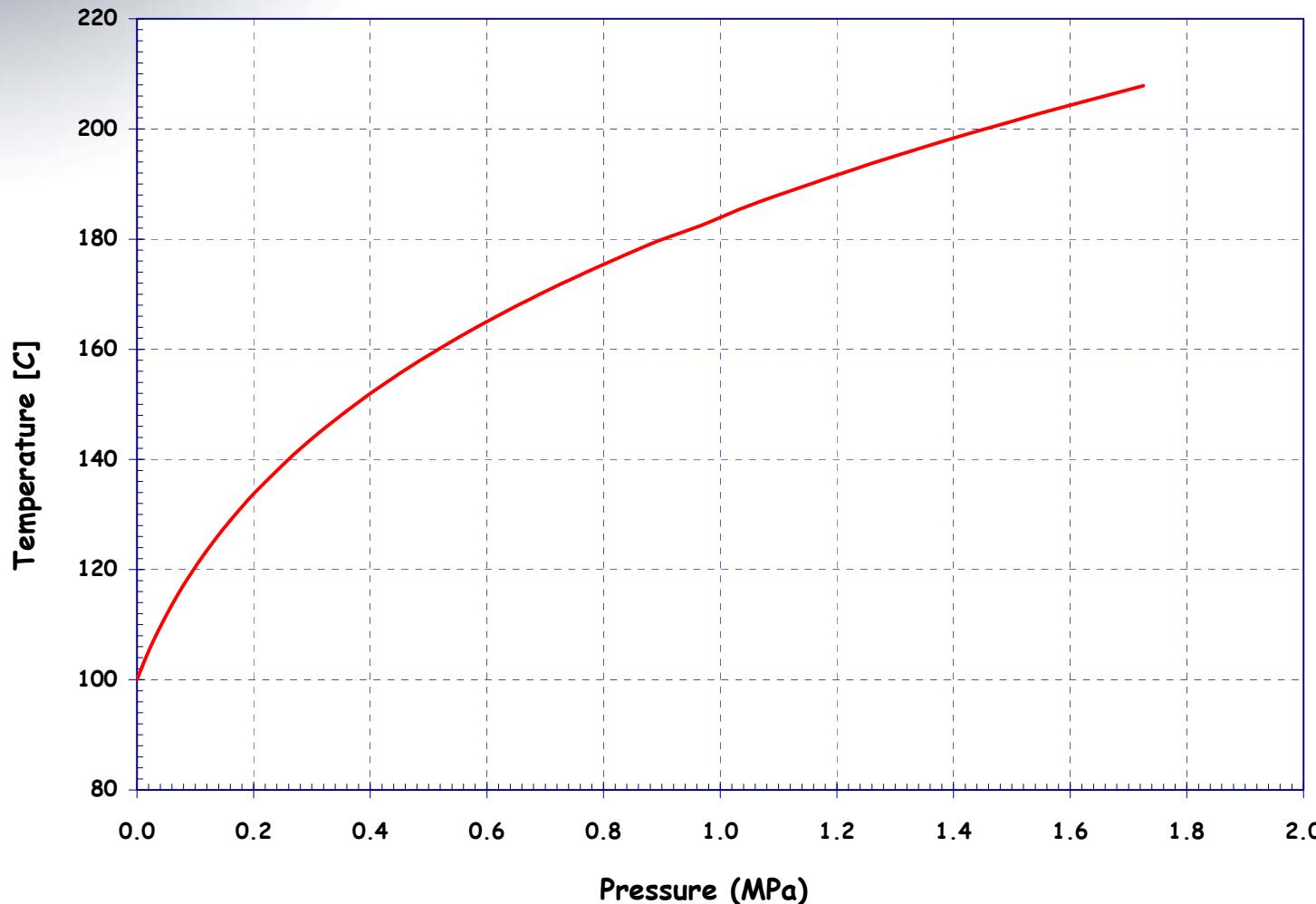
ISP 48 on Containment Capacity

- **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

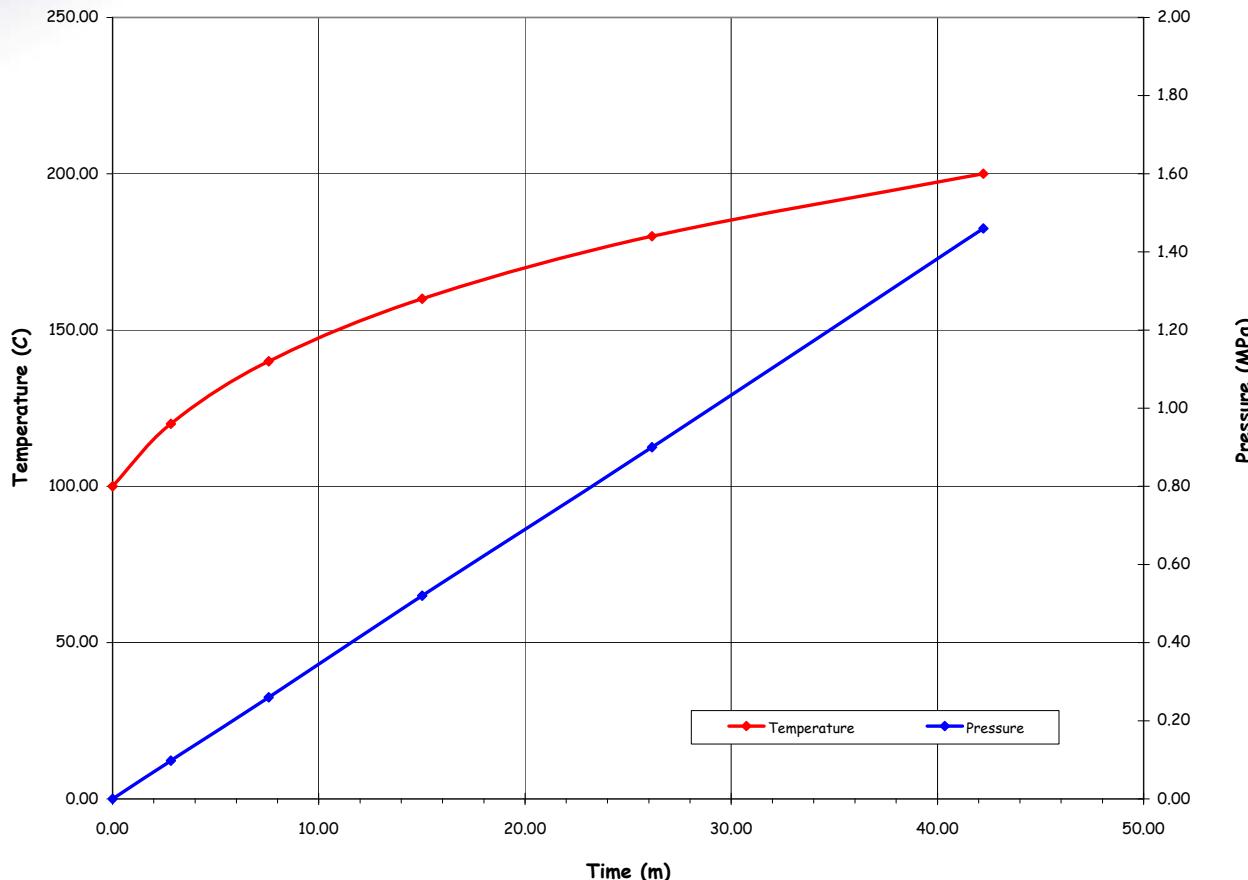




Case 1: Pressure-Temperature Time Histories

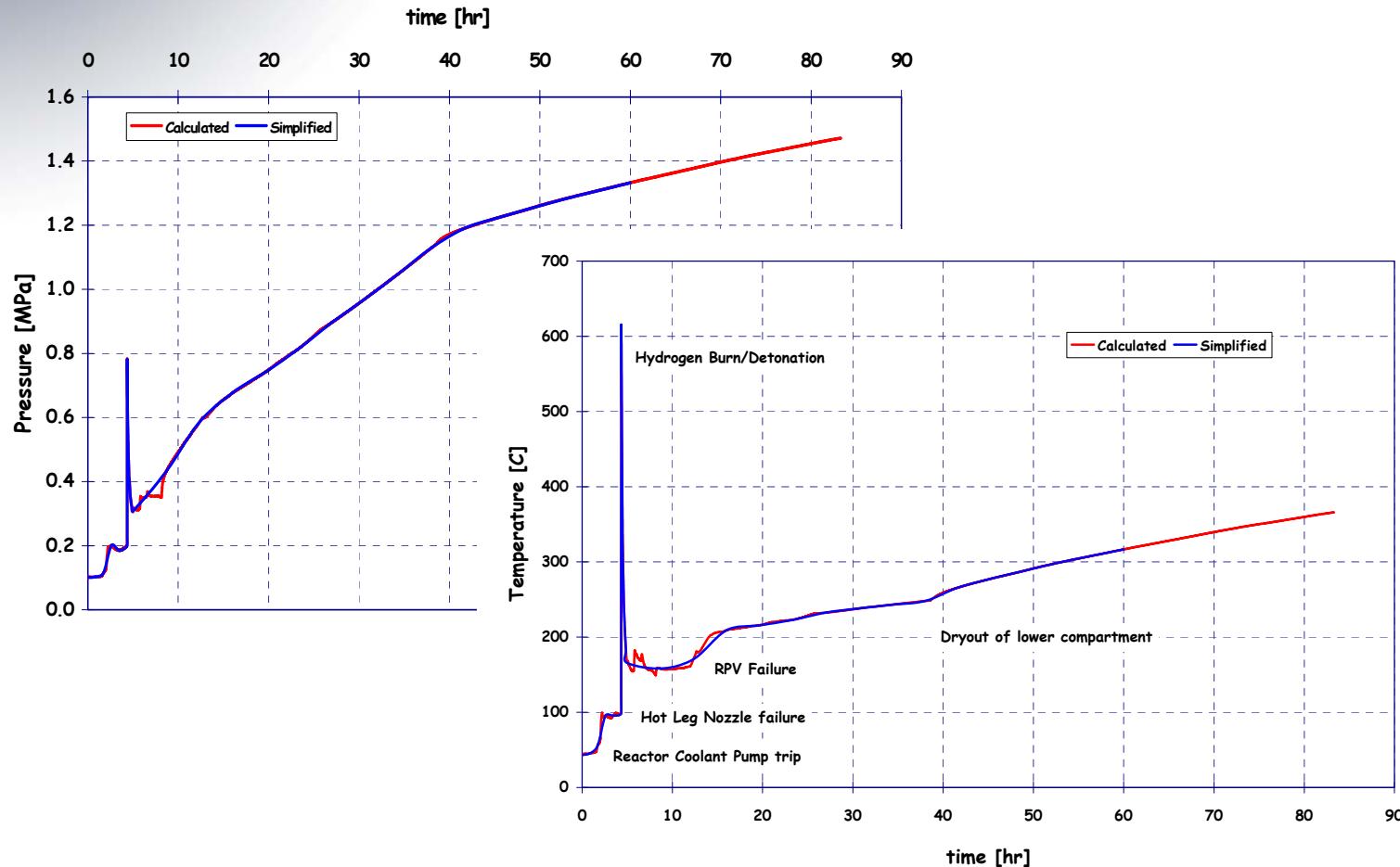
- Saturated Steam

- Pseudo-time history based on SFMT pressurization rate (5 psi/min)



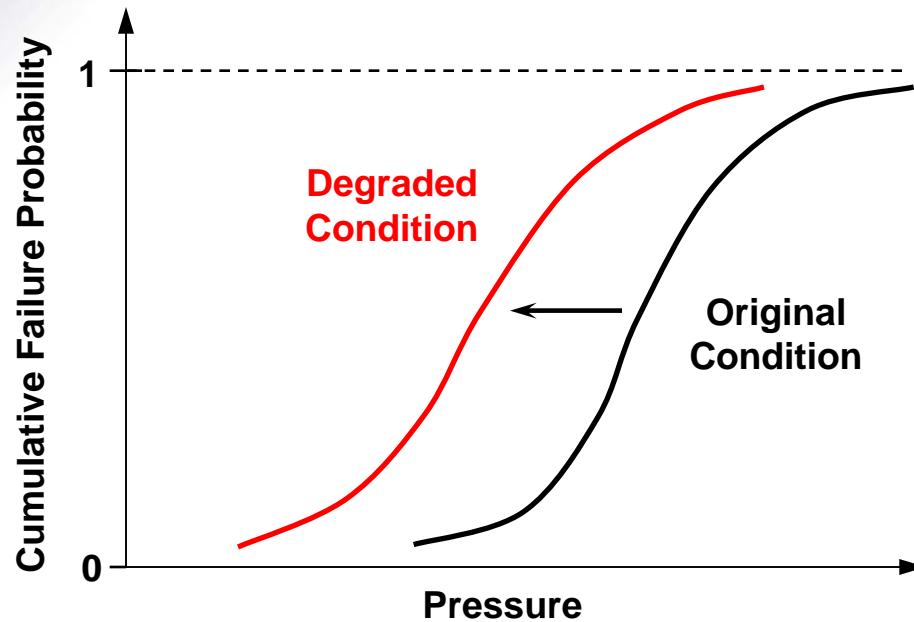
Case 2: Pressure-Temperature Time Histories

- Large, Dry PWR SBO, no containment leakage





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.



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



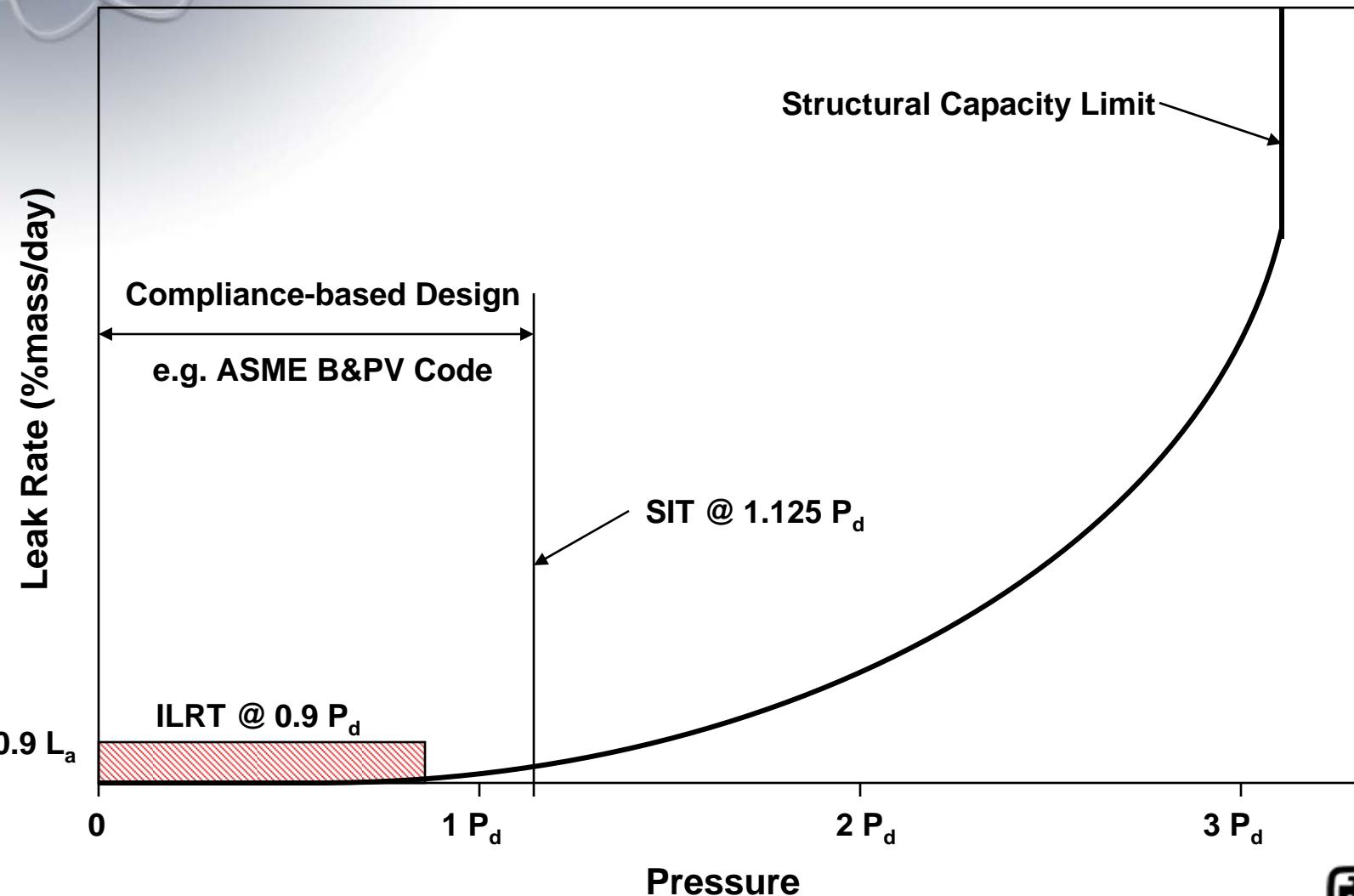
Containment Performance Model

- 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.



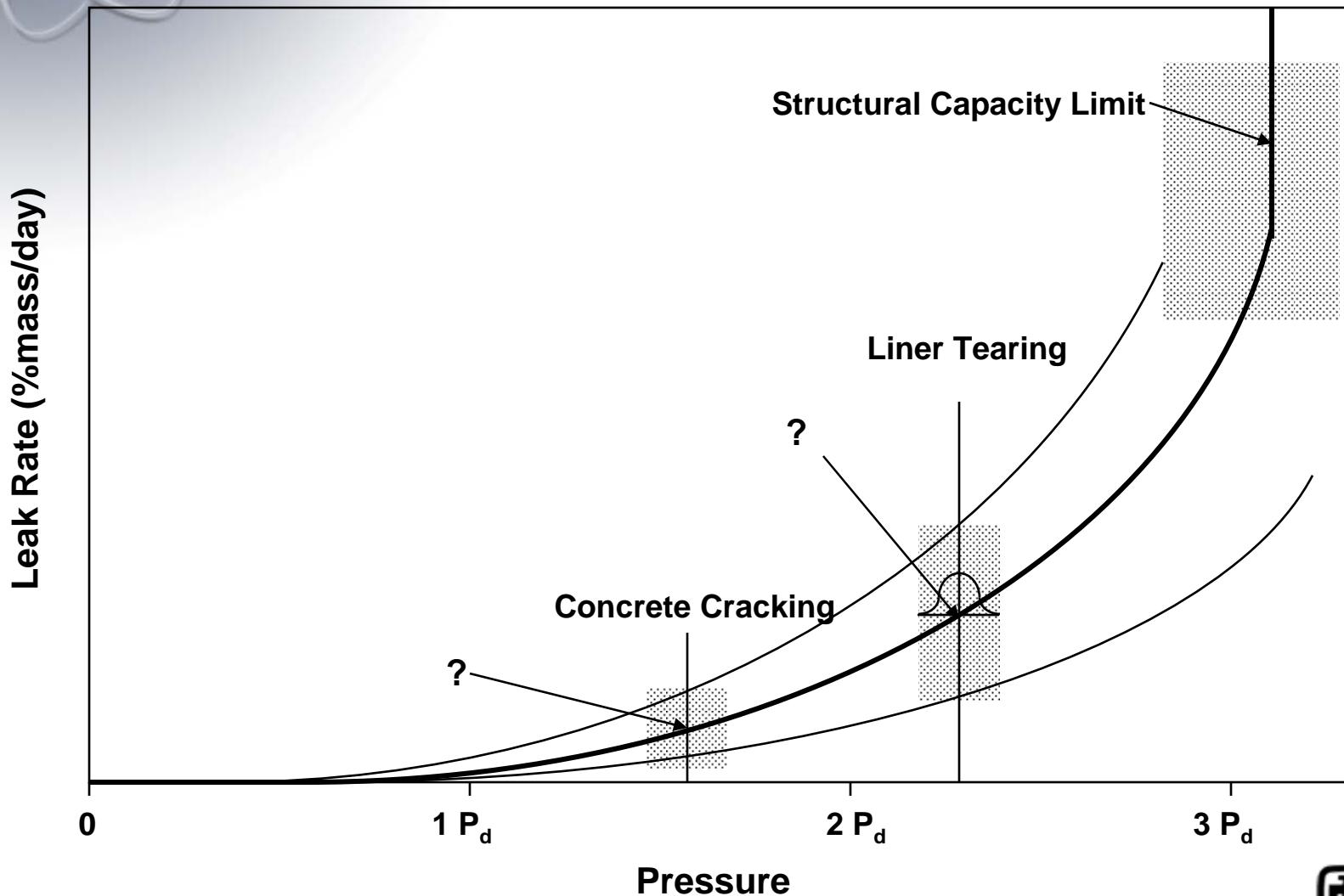
Containment Performance Model

What we 'know' from Design:



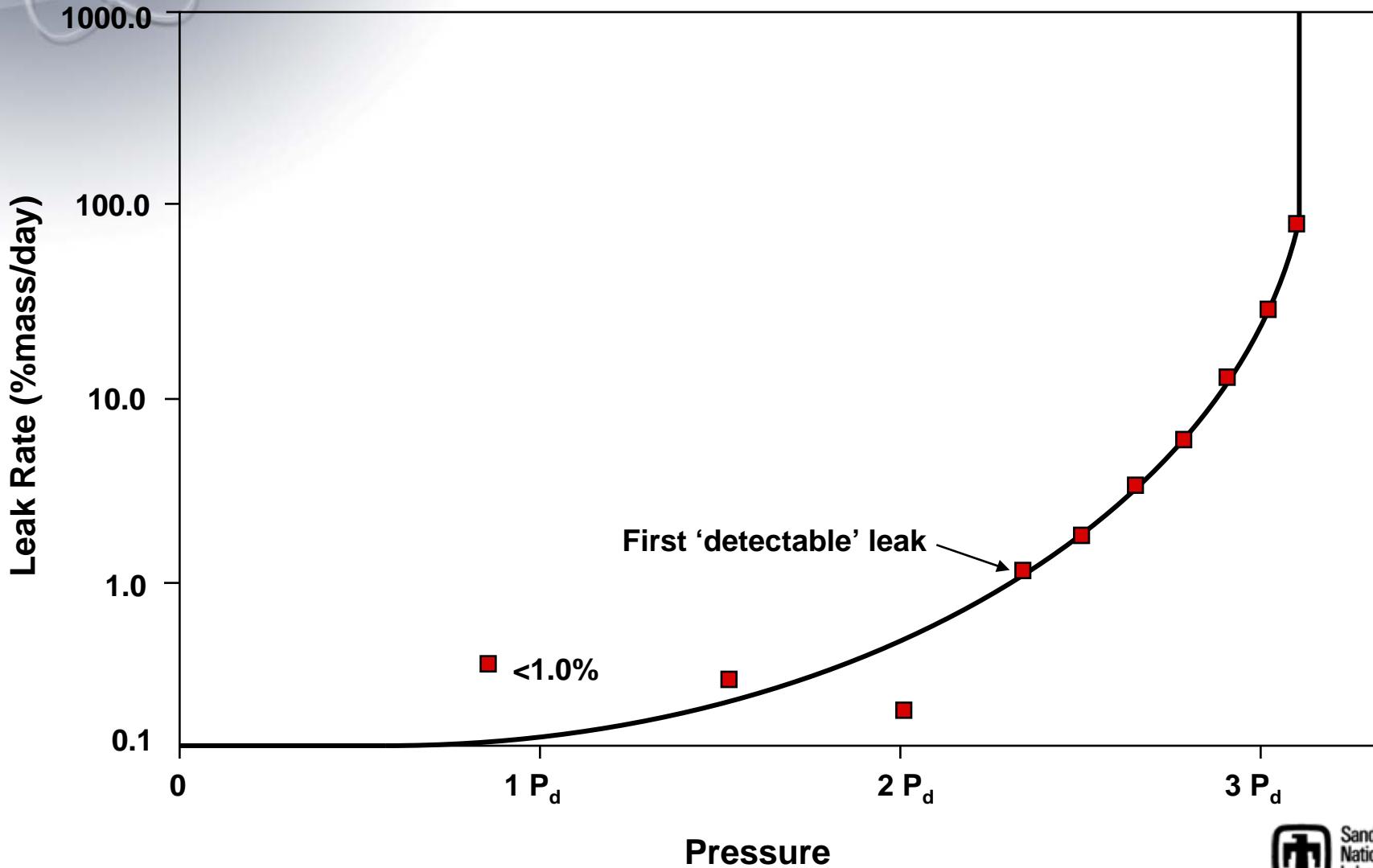
Containment Performance Model

What we 'know' from Analysis:



Containment Performance Model

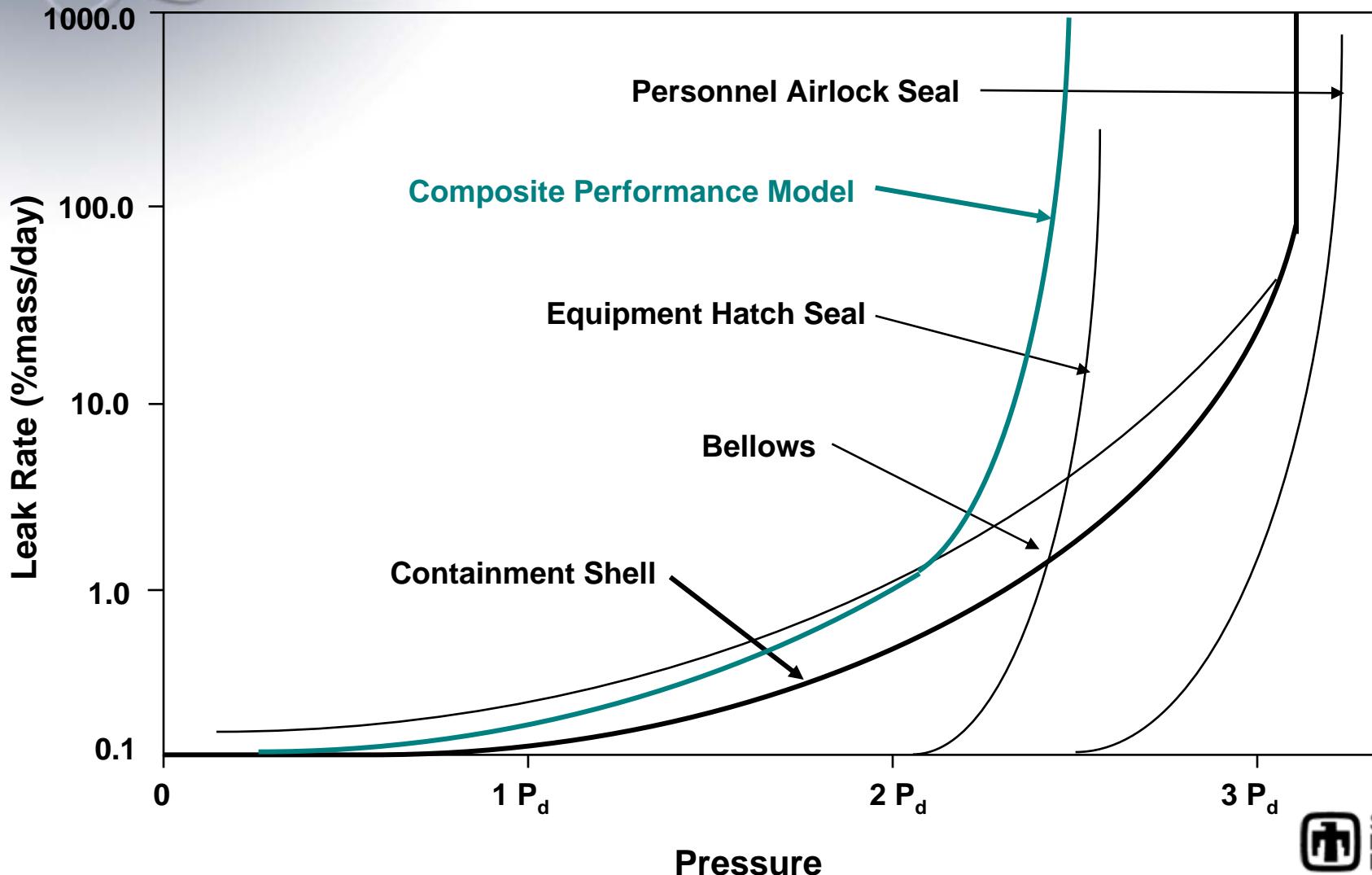
What we 'know' from Testing:





Containment Performance Model

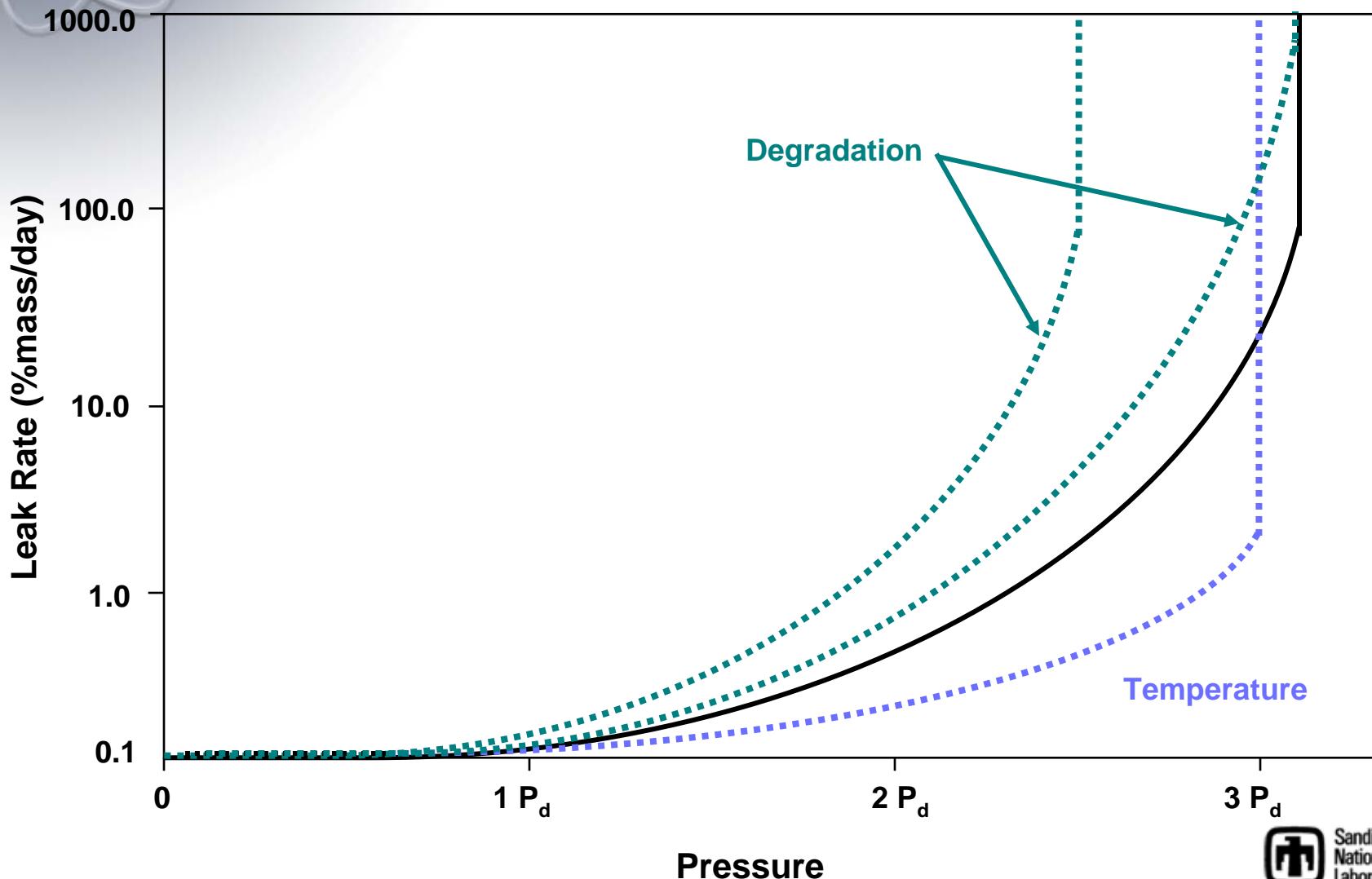
Effect of other components (tests):





Containment Performance Model

Effects of Temperature, Degradation:





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?