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AN INTERNATIONAL STANDARD PROBLEM: ANALYSIS OF 1:4-SCALE PRESTRESSED CONCRETE CONTAINMENT VESSEL MODEL UNDER SEVERE ACCIDENT CONDITIONS

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ABSTRACT

In June 2002, The OECD-NEA Committee on the Safety of Nuclear Installations (CSNI), with the encouragement of the US NRC, initiated an International Standard Problem on containment integrity (ISP 48) based on the NRC/NUPEC/Sandia test. The objectives of the ISP are to extend the understanding of capacities of actual containment structures based on results of the recent PCCV Model test and other previous research.

From 1997 through 2001 Sandia National Laboratories (SNL) conducted a Cooperative Containment Integrity Program under the joint sponsorship of the Nuclear Power Engineering Corporation (NUPEC) of Japan, and the NRC Office of Nuclear Regulatory Research. The purpose of the program was to investigate the response of representative models of nuclear containment structures to pressure loading beyond the design basis accident and to compare analytical predictions to measured behavior. A uniform 1:4-scale model of a prestressed concrete containment vessel (PCCV) was constructed and tested at SNL. This model was representative of the containment structure of an actual pressurized-water reactor plant in Japan.

The ISP consists of four phases over a period of 2 years:

Phase 1: Data Collection and Identification

Phase 2: Calculation of the Limit State Test (LST), i.e. static pressure loading

Phase 3: Calculation of response to both Thermal and Mechanical Loadings

Phase 4: Reporting Workshop

Eleven organizations (or teams) from nine OECD member countries accepted the invitation to participate in the ISP and perform calculations to predict the structural response of the PCCV model to static and transient pressure and thermal loading. Each participating organization was provided with the model and loading data and was asked to perform independent analyses to simulate the response of the PCCV model. The results of each team's calculations were compiled and the results presented at a final workshop in April 2005. These results and the conclusions and insights gained from the analysis were published by CSNI. This paper summarizes the conduct and significant findings of the ISP.

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INTRODUCTION

Background

At the CSNI meeting in June 2002, the proposal for an International Standard Problem on containment integrity (ISP 48) based on the NRC/NUPEC/Sandia test was approved. Objectives included extending the understanding of capacities of actual containment structures based on results of the recent PCCV Model test and other previous research. Sandia National Laboratories (SNL) conducted a Cooperative Containment Integrity Program under the joint sponsorship of the Nuclear Power Engineering Corporation (NUPEC) of Japan, and the NRC Office of Nuclear Regulatory Research. The purpose of the program was to investigate the response of representative models of nuclear containment structures to pressure loading beyond the design basis accident and to compare analytical predictions to measured behavior. A uniform 1:4-scale model of a prestressed concrete containment vessel (PCCV) was constructed and tested at SNL from 1997 through 2001. This model was representative of the containment structure of an actual pressurized-water reactor plant in Japan.

Scope of ISP48

The first meeting of the ISP48 members was held on November 20 & 21, 2002 in Stockholm, Sweden to define the scope and the schedule for the ISP. It was agreed that the ISP would consist of four phases:

Phase 1: Data Collection and Identification

Phase 2: Calculation of the Limit State Test (LST), i.e. static pressure loading

Phase 3: Calculation of response to both Thermal and Mechanical Loadings

Phase 4: Reporting Workshop

Eleven organizations (or teams) accepted the invitation to participate in the ISP and completed calculations for either Phase 2 or Phase 3 or both. The participating organizations are:

BE/HSE/NNC	British Energy Nuclear Installations Inspectorate/Health & Safety Executive NNC Ltd.	UK
EDF	Électricité de France	France
EGP	Energoprojekt Praha, UJV Rez. Div.	Czech Rep.
FORTUM	Fortum Nuclear Services Ltd.	Finland
GRS	Gesellschaft für Anlagen und Reaktorsicherheit mbH	Germany
IRSN/CEA	Institut de Radioprotection et de Sûreté Nucléaire Commisssariat a l'Energie Atomique	France
JPRG	Japan PCCV Research Group	Japan
KAERI	Korea Atomic Energy Research Institute	Korea
KOPEC	Korea Power Engineering Company	Korea
NRC/SNL/DEA	US Nuclear Regulatory Commission Sandia National Laboratories David Evans and Associates	US
SCANSCOT	Scanscot Technology	Sweden

Some of these organizations participated in the Pretest Round Robin Analysis organized by SNL in 1995 to predict the response of the PCCV model prior to testing [Luk, 2000] while for other participants this was a new problem. Each participating organization was provided with the model and loading data and was asked to perform independent analyses to simulate the response of the PCCV model subjected to pressure only (Phase 2) and pressure plus temperature (Phase 3).

Under Phase 1, Sandia National Laboratories (SNL) was tasked with providing the data on the model design/construction and testing to complete the calculations in Phases 2 and 3. A detailed test report was published by the US Nuclear Regulatory Commission [Hessheimer, 2003]. A condensed version of this report was prepared to meet the requirements of Phase 1 of the ISP.

The results of the Phase 2 calculations were reviewed at meeting of the participants held in conjunction with the 8th meeting of the CSNI Working Group on the Integrity and Ageing of Components and Structures on March 19, 2004 in Madrid, Spain. These results were reported by Hessheimer [2004].

The results of the Phase 3 combined thermal and mechanical analysis were reviewed at a meeting of the participants held in conjunction with a Workshop on Containment Capacity and the 9th meeting of the CSNI Working Group on the Integrity and Ageing of Components and Structures, held April 4-8, 2005 in Lyon, France. The compiled results of the ISP will be published in an NEA report [Hessheimer, 2005]. The results of the Workshop on Containment Capacity will also published by CSNI [Mathet, 2005]

PRESTRESSED CONCRETE CONTAINMENT VESSEL MODEL

Design

The prestressed concrete containment vessel (PCCV) model is a uniform,1:4-scale model of the containment structure of Unit 3 of the Ohi Nuclear Power Station in Japan. Ohi Unit 3 is a 1180 MWe pressurized-water reactor (PWR) plant designed and constructed by Mitsubishi Heavy Industries (MHI) and operated by Kansai Electric Power Company. The Ohi-3 containment vessel is a steel-lined, prestressed concrete cylinder with a hemispherical dome and two vertical buttresses. The design pressure is 0.39 MPa.

The model was designed by MHI and Obayahsi Corporation. The approach to designing the model was to scale the design of the Ohi-3 containment to the extent possible and include as many representative features of the prototype as practical. Specific considerations in designing the model are summarized below.

Geometry: The configuration and overall dimensions (height, radius, thickness) were scaled 1:4 from the prototype. While the basemat thickness was scaled from the prototype, the footprint of the basemat was selected so that the bending stiffness of the basemat at the junction with the containment wall was preserved. The overall geometry is shown in Figure 1.

Liner: The liner thickness was scaled directly from the prototype resulting in a liner thickness of 1.6 mm. In the prototype, the liner anchorage consists of meridional T-anchors throughout the cylinder and dome. Anchorage of the model liner consists of scaled T-anchors in the cylinder portion and stud-type anchors in the dome. Circumferential spacing of the vertical anchors was expanded in the model by a factor of three to simplify fabrication, except in areas around penetrations and other discontinuities. To the extent practical, all liner details were similar to the prototype.

Penetrations: All penetrations were scaled from the prototype (geometry, thickness), and the equipment hatch (E/H), and personnel airlock (A/L) are functional with pressure seating covers. The main steam (M/S) and feedwater (F/W) penetration sleeves are scaled but are terminated with heavy, bolted, pressure seating blind flanges and covers which are used for instrumentation, power, and gas feed-throughs.

Concrete: There was no scaling of the concrete for the model; however, maximum aggregate size was limited to 10 mm to facilitate placement.

Reinforcing Steel: All reinforcing ratios in the prototype are maintained in the model. Rebar areas were scaled, but there was no attempt to match individual bars. Bars ranging in size from 6 mm to 22 mm in diameter were place in two orthogonal layers on each face, and shear reinforcing was included.

Tendons: Each tendon in the prototype was matched in the model, 90 meridional hairpin tendons and 108 360º hoop tendons. Individual tendon areas were scaled, resulting in three 13.7 mm seven-wire strands per tendon. Prestressing levels for the model tendons were selected so that the net anchor forces (considering all losses due to anchor seating, elastic deformation, creep, shrinkage and relaxation) at the time of the Limit State Test matched those expected in the prototype after 40 years of service. One further adjustment was made by increasing the vertical tendon stress level to account for the additional gravity load in the prototype, which is lost in the geometric scaling.

Details of the design, including the design drawings, and construction were reported by Hessheimer [2003].

Pressure Testing

The prestressed concrete containment vessel (PCCV) model was subjected to a series of quasi-static pressurization tests leading to functional failure or rupture. Nitrogen gas at ambient temperature (nominally 21ºC) was used as the pressurization medium for each test. All pressure tests were conducted in a quasi-static manner by pressurizing the model in increments and holding pressure until the model response and pressure reach equilibrium. The pressurization system was designed to maintain the model at a constant pressure (within ±3kPa) up to a maximum leak rate of 1000% mass/day.

The Limit State Test (LST) fulfilled the primary objectives of the PCCV test program, i.e., to investigate the response of representative models of nuclear containment structures to pressure loading beyond the design basis accident and to compare analytical predictions to measured behavior. As the model was pressurized, periodic leak checks were conducted by holding pressure and monitoring pressure and temperature and calculating the apparent leak rate. The Limit State Test was terminated when the pressurization system was no longer able maintain pressure because of excess leakage. The pressure and temperature time histories during the LST are shown in Figure 2.

Subsequently, it was decided to re-pressurize the PCCV model, prior to demolition, in an attempt to observe larger inelastic response and, possibly, a global structural failure. This Structural Failure Mode Test (SFMT) was a combined **pneumatic-hydrostatic** test, where the PCCV model was sealed inside with an elastomeric membrane and filled nearly full with water, to reduce the volume of gas to be pressurized, and nitrogen gas was used to generate the overpressure. The SFMT was terminated when the PCCV model failed catastrophically at a pressure of 1.42 MPa (206.4 psig) or 3.63 times the design pressure, P_d. The SFMT pressure time history is shown in Figure 3. Images of the model at the time of failure from the video recorders are shown in Figure 4 and the posttest condition is shown in Figure 5.

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Phase 2 Calculations

Each ISP participant was asked to provide a report summarizing their analysis approach and results. Each individual report was reproduced in the Phase 2 report.

Fifty-five response parameters, referred to as standard output locations (SOLs), were selected to facilitate initial comparison of the Phase 2 calculations with the test results. The selection of these locations was based on the containment experience of the project team at Sandia and the results of preliminary analyses to characterize model responses and identify possible failure modes. The general locations are illustrated in Figure 6. This is the same set of variables used for the pretest Round Robin comparison. A gage, or set of gages, were installed at every standard output location for the LST. Due to the practical limitations, the model response was not measured at all these locations during the SFMT. The participants were asked to provide analysis results as a function of gage pressure at these locations. All participants in Phase 2 reported results at most of the SOLs. The calculated results were collected into composite plots for comparison with the LST and SFMT results at each SOL. All the composite plots are provided in the Phase 2 report, however a few representative plots are reproduced below.

Figure 7 compares the predicted displacement at the mid-height of the cylinder wall with the test results. This location (Az. 135°) was selected to represent the 'free-field' response of the cylinder wall, i.e. it is not influenced by any discontinuities such as penetrations or tendon buttresses. The calculated responses match the test results and each other reasonably well up to the onset of global, or general membrane, yielding where the results begin to diverge. Also, most of the analyses capture the loss of stiffness due to cracking of the concrete at approximately 1.5 times the design pressure. Figures 8 and 9 compare the hoop liner and tendon strains at the mid-height of the cylinder wall. The calculated responses again match the test data up to the onset of global yielding, however, the divergence beyond this point is more pronounced than the displacement.

To facilitate comparison of the calculated results with the pressure response of the model, most, if not all of the participants chose to 'initialize' their analysis results with the test data at the start of the LST. This was done to eliminate the variation in the calculation of the response due to dead load and prestressing as well as the effects of creep, shrinkage and changes in ambient conditions, which were not modeled. As a result, **most of the comparisons are for the change in the response variable due to pressure, not the total response**. Comparisons of analyses before initializing the response variables were much more inconsistent and a similar result would be expected for full-scale containments where the initial conditions might not be known.

In addition to submitting response predictions at the SOLs, each participant was also asked to provide a best estimate of failure pressure and mechanisms of the PCCV model. These are summarized in Figure 10. Figure 10 also summarizes predictions of the pressure for various milestones (onset of cracking, yielding, etc.) leading up to failure. It is interesting to note that the differences in failure predictions are much more significant than the differences in the calculated responses would seem to suggest.

Phase 3 Calculations

Phase 3 of ISP48 extends the results of the model tests and calculations by investigating the addition of temperature to the pressure loading. The tests were conducted at ambient temperature based on risk and cost considerations and previous experience. Two questions were posed for Phase 3:

- With addition of temperature, would the onset of leakage occur later in the pressure history and, possibly, closer to the burst pressure?
- How would including the effect(s) of accident temperatures change the prediction of failure location and failure mode?

The ISP participants agreed to consider two thermal load cases for Phase 3:

- Case 1: Saturated Steam Conditions (mandatory for all Phase 3 participants)
    - Monotonically increasing static pressure and temperature (saturated steam).
- Case 2: Station Blackout Scenario
    - A representative severe-accident scenario for a four-loop PWR including vessel failure and hydrogen detonation

The thermal and mechanical analyses were de-coupled with the heat transfer calculations being performed first using a full-scale axisymmetric model using a combination of applied temperature (liner), convection (cylinder wall and dome), and conduction (basemat/soil) boundary conditions. Thermal material properties and temperature dependent mechanical properties were based on handbook data. Thermal gradients were calculated at several locations and provided to the participants to perform combined temperature-pressure response calculations for the 1:4-scale model.

The results of these independent participant calculations were completed by January, 2005 The results were summarized in the same manner as the Phase 2 results and will be included in the final report for the ISP. A meeting of the participants and workshop was held in April, 2005 to review the results and develop a consensus on issues related to:

- Modeling and analysis methods
- Constitutive models for materials
- Limit state and structural failure criteria

Each of the participants completed analyses for Case 1 or both Case 1 and 2. Participants were also allowed to submit revised calculations for pressure only to account for any modeling changes between Phase 2 and 3. Figures 13 and 14 compare the calculated responses at SOL #6, the radial displacement at the mid-height of the cylinder wall. Figures 15 and 16 compare the results at SOL #39, the hoop liner strain at the same location. Figure 17 summarizes the interpretation of these results with regard to initial tearing of the liner and structural failure of the vessel.

SUMMARY

The work reported herein represents, arguably, the state of the art in the numerical simulation of the response of a prestressed concrete containment vessel (PCCV) model to pressure loads up to failure. A significant expenditure of time and money on the part of the sponsors, contractors, and Round Robin participants was required to meet the objectives. While it is difficult to summarize the results of this extraordinary effort in a few paragraphs, the following observations are offered for the reader's consideration:

Results and conclusions for each phase of the ISP will be included in the final report along with a discussion of issues identified as a result of this effort. In general the results for Phase 2 and Phase 3 were relatively consistent with each other and the test data, where available. Global behavior compared more favorably than local behavior. With regard to the two questions raised with the addition of thermal loading to the overpressurization, the following observations are made:

With addition of temperature, would the onset of leakage occur later in the pressure history and, possibly, closer to the burst pressure?

- Results predict failure at both lower and higher pressure when temperature is considered.
- The margin between leak and rupture does not appear to change significantly
- Change in 'failure' pressures are generally small (<10%).
- Consideration of 'realistic' severe accident scenario (Case 2) yields lower 'failure' pressure than saturated steam conditions.
- Effects of material degradation are significant for 'realistic' severe accident scenarios.

How would including the effect(s) of accident temperatures change the prediction of failure location and failure mode?

- While leak or rupture pressures are not significantly changed, displacements are significantly greater, especially when considering material property degradation.
  - Case 1: Vertical displacements increase
  - Case 2: Radial displacements increase
- Failure at penetrations appear more likely, and may control, under combined pressure and temperature loading.

REFERENCES

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Hessheimer, M. F. (editor), "*International Standard Problem No. 48 Containment Capacity, Phase 2 Report, Results of Pressure Loading Analysis*", NEA/CSNI/R(2004)11, OECD Nuclear Energy Agency, Committee on the Safety of Nuclear Installations, Issy-les Moulineaux, France. July, 2004

Luk, V. K. *Pretest Round Robin Analyses of a Prestressed Concrete Containment Vessel Model.* NUREG/CR-6678, SAND00-1535, Sandia National Laboratories, Albuquerque, NM. August, 2000.

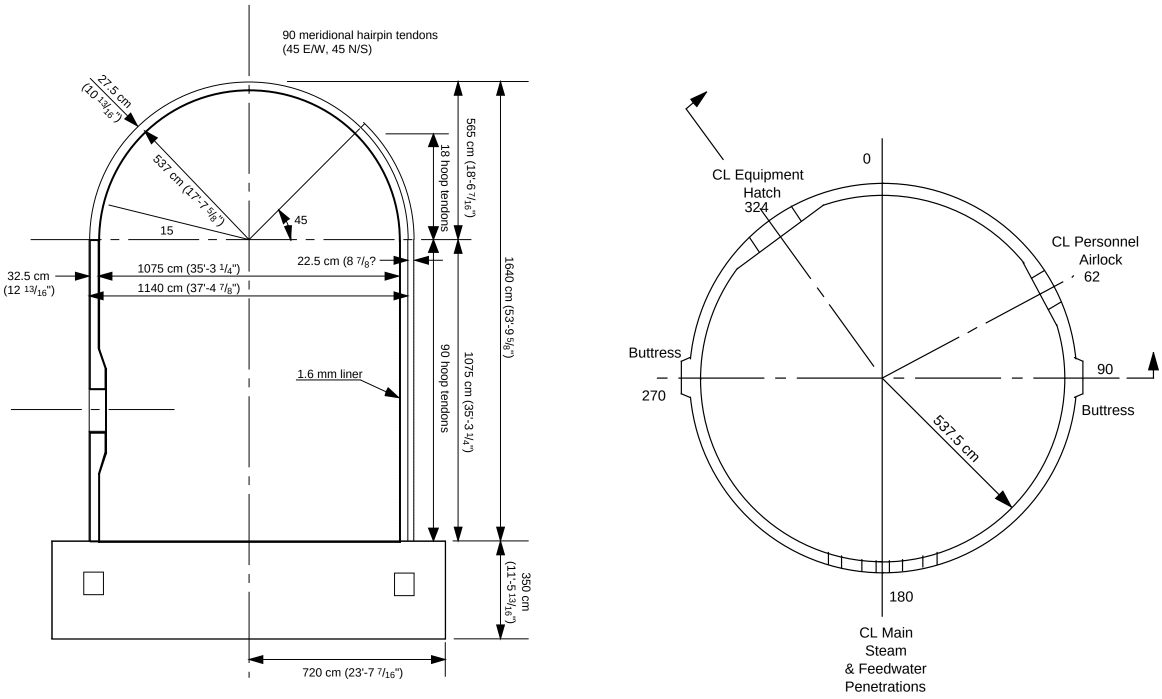


Figure 1. Prestressed Concrete Containment Vessel (PCCV) Model Geometry

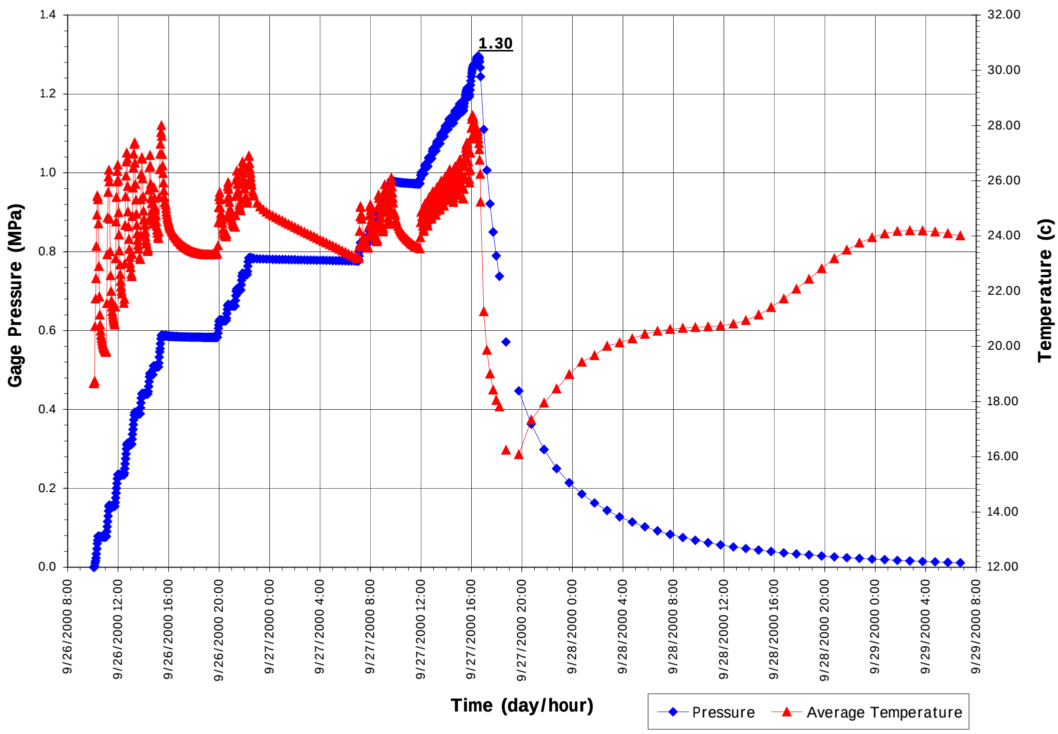


Figure 2. Limit State Test Pressure and Average Temperature

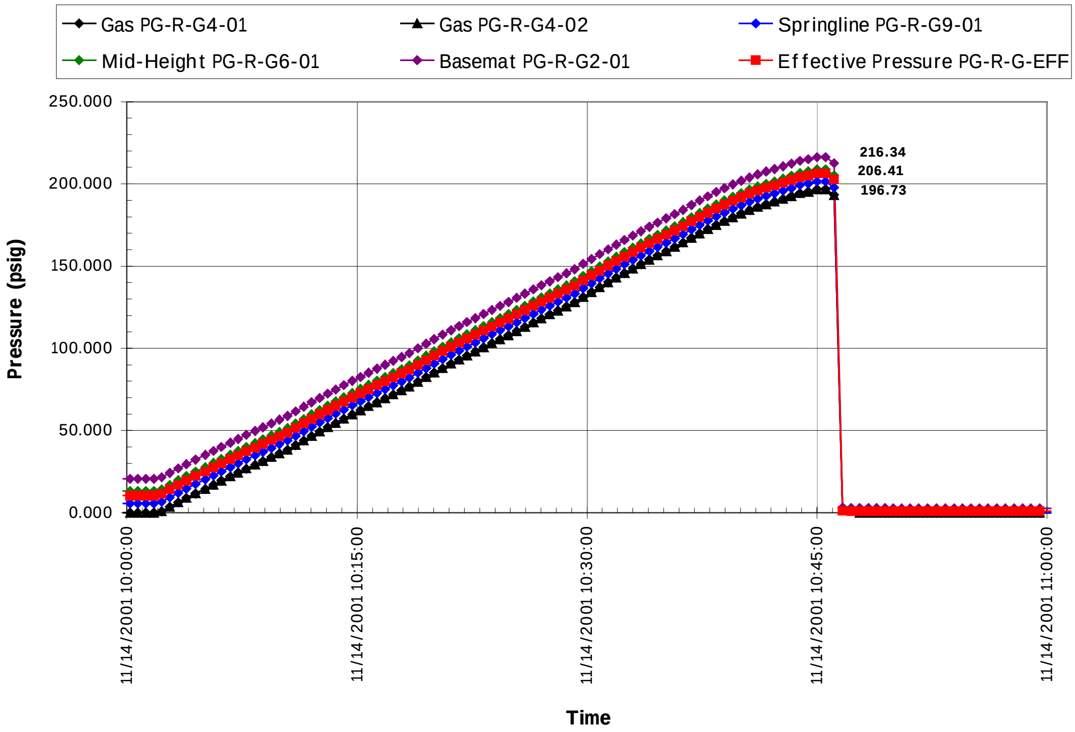
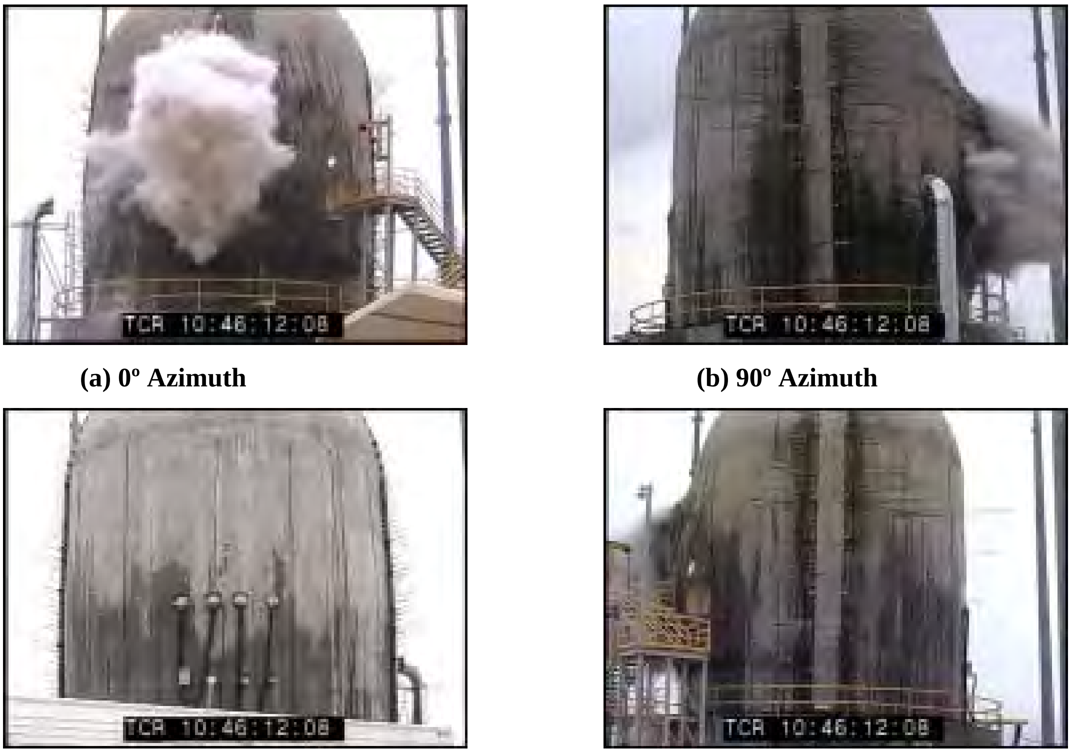


Figure 3. SFMT Pressurization System Data



(a) 0° Azimuth **(b) 90° Azimuth**

(a) 180° Azimuth **(b) 270° Azimuth**

Figure 4. SFMT: Rupture of the PCCV Model

Figure 5. PCCV Model after the Structural Failure Mode Test

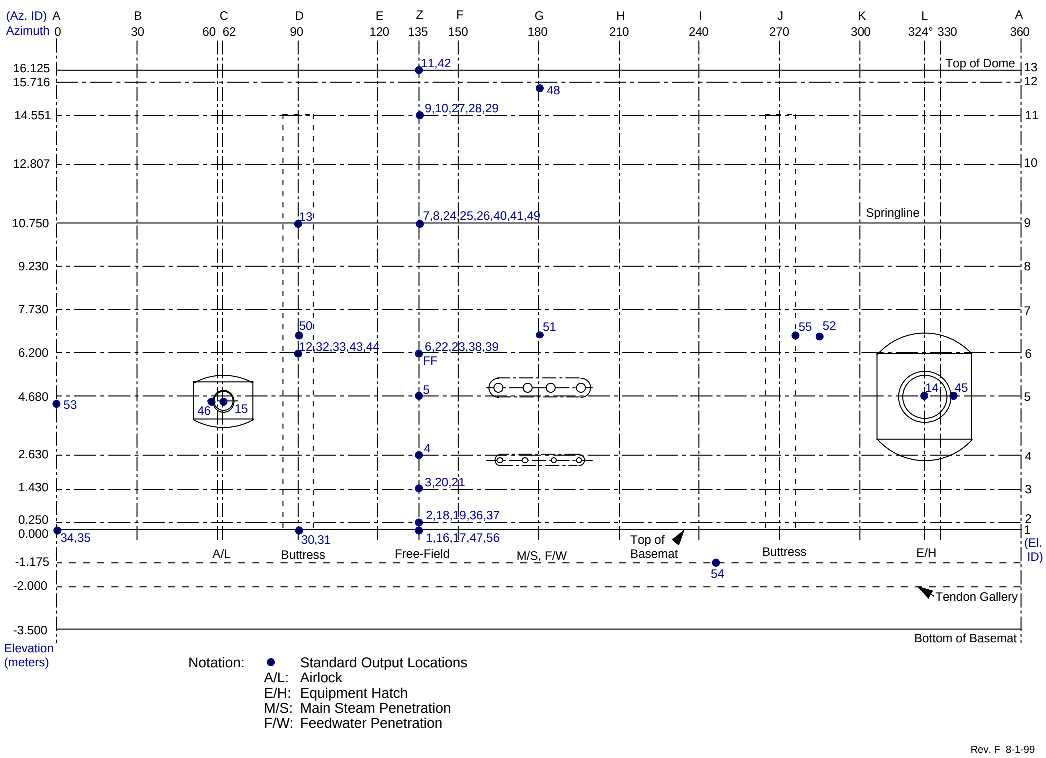


Figure 6. Developed Elevation of PCCV Model and Standard Output Locations

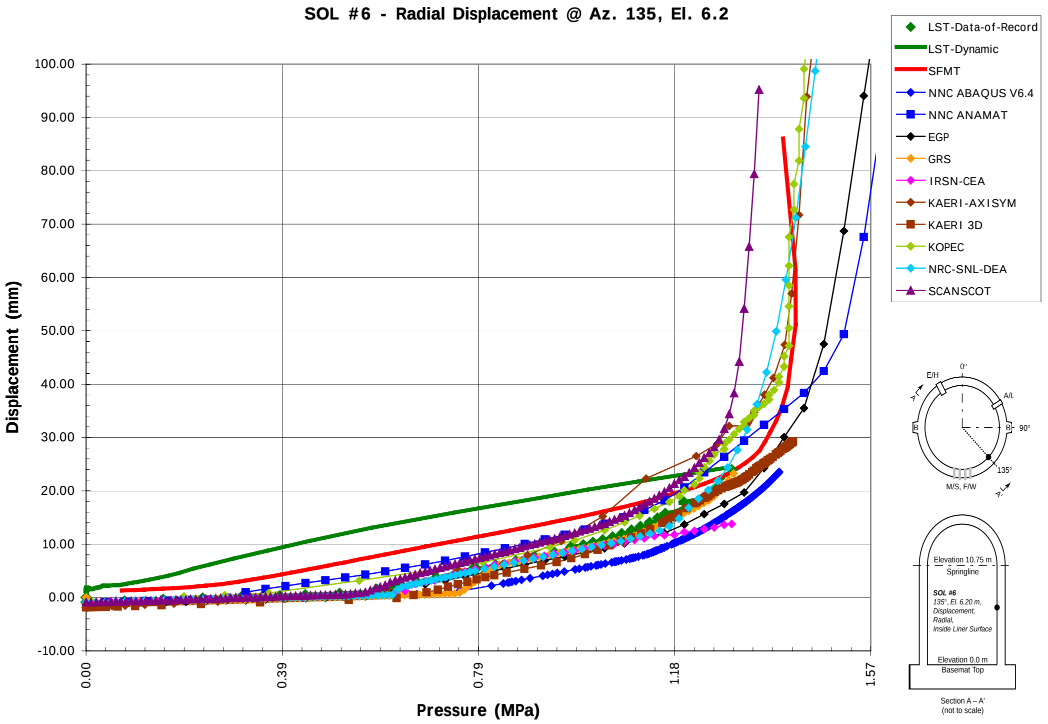


Figure 7. Radial Displacement at Cylinder Wall Mid-height (SOL 6)

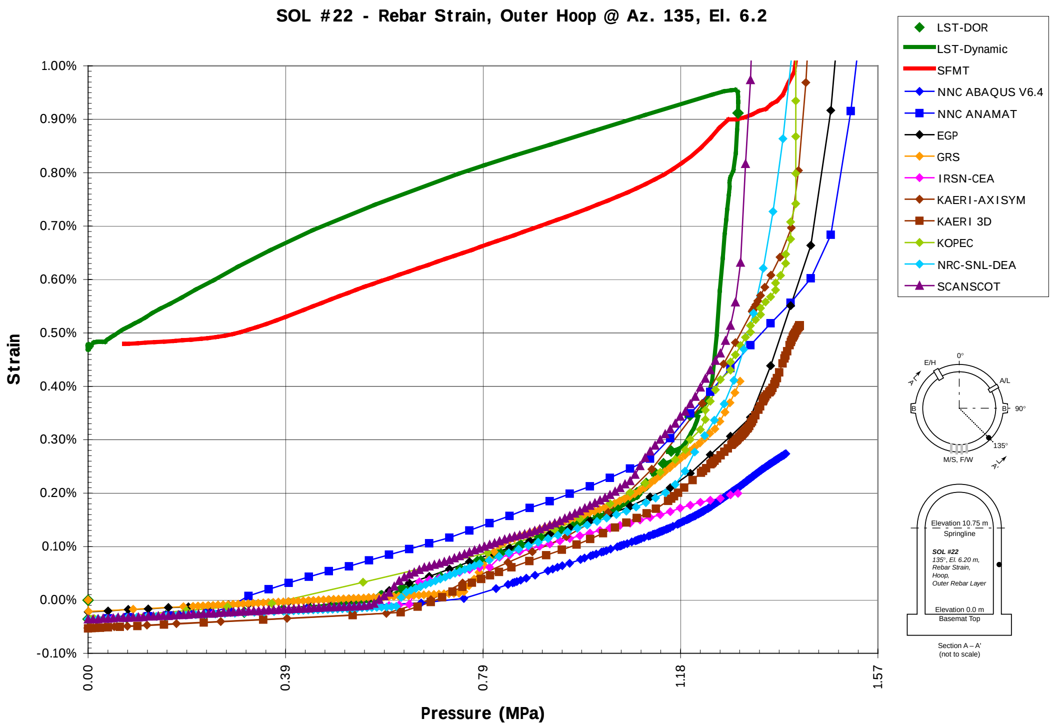


Figure 8. Hoop Liner Strain at Cylinder Wall Mid-height (SOL 39)

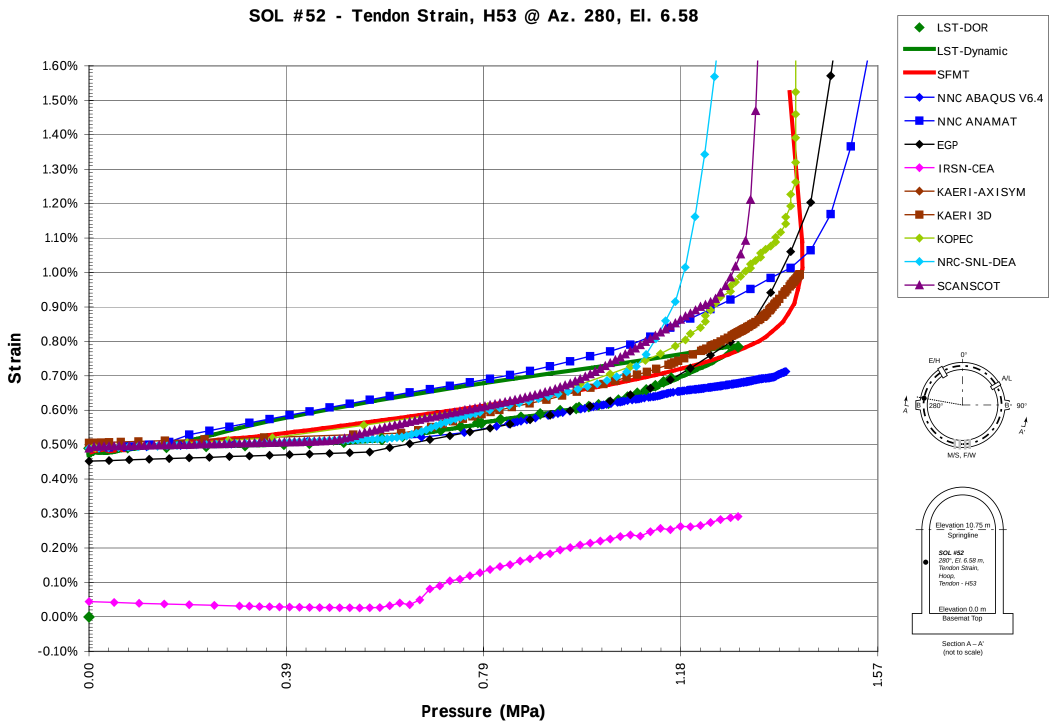


Figure 9. Hoop Tendon Strain at Cylinder Wall Mid-height (SOL 52)

	Pressure (MPa)					
	Hoop Crack	Liner Yield	Tendon Yield	Failure Pressure	Hoop Strain	Failure Mode
• LST	0.59-0.78	1.1	1.7	0.98	0.17%	Liner tear, 1% leak
• SFMT				1.42	1.4%	Tendon rupture
• BE/HSE/NNC	0.6	1.2-1.4		1.5	3.0%	Liner tear
• EGP	0.4-0.7	0.98	1.25	1.0	0.14%	Cracking @ E/H
• GRS	0.75	0.76	1.25	1.3	0.43%	Tendon rupture
• IRSN/CEA	0.67					
• JNES*	0.6-0.65	1.1	1.2	1.1	0.19%	
• JPRG*		1.0	1.5	0.9-1.0	0.16%	
• KAERI	0.6					
• KOPEC	0.6	0.84	1.43	1.52		Tendon rupture
• NRC/SNL/DEA	0.6	0.8	1.2	1.26	0.35%	Liner tearing
• SCANSCOT	0.55-0.7	0.8	1.12			

Figure 10. Summary of Phase 2 Response Milestones

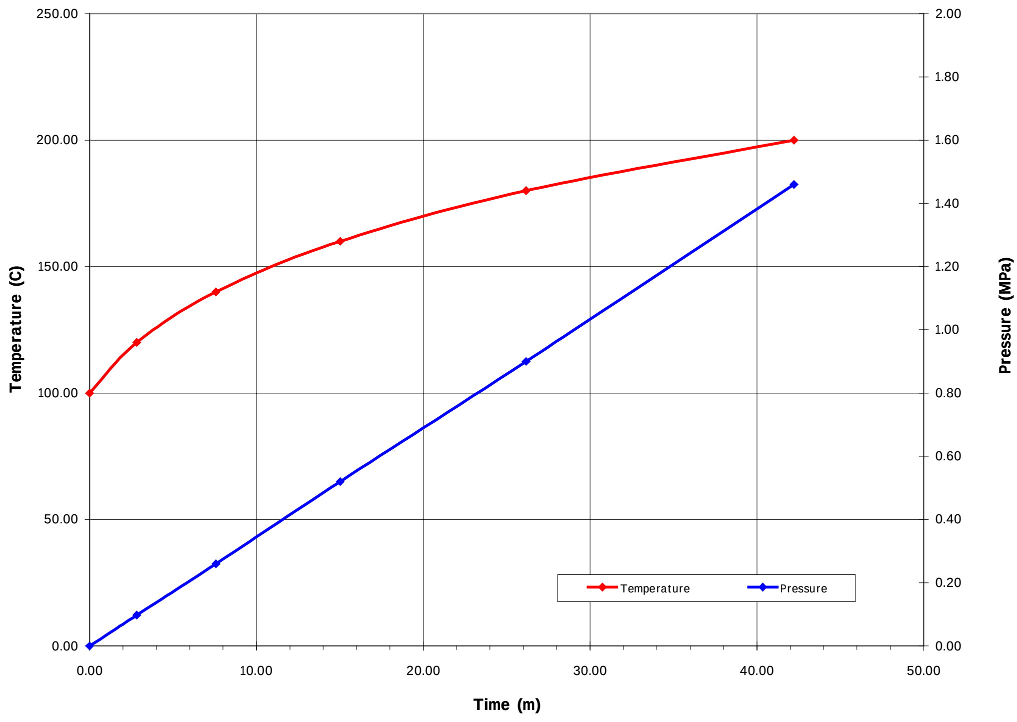


Figure 11. Case 3 Saturated Steam Pseudo-Time History

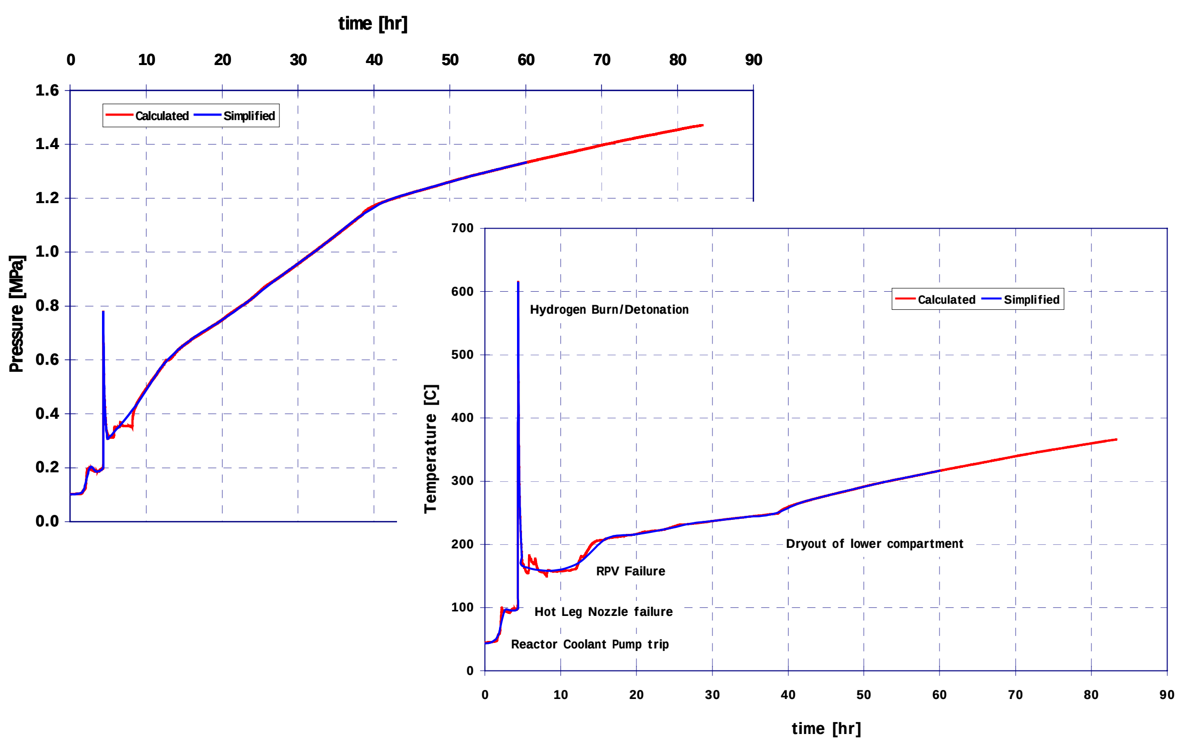


Figure 12. Case 2 Station Black-Out Time History

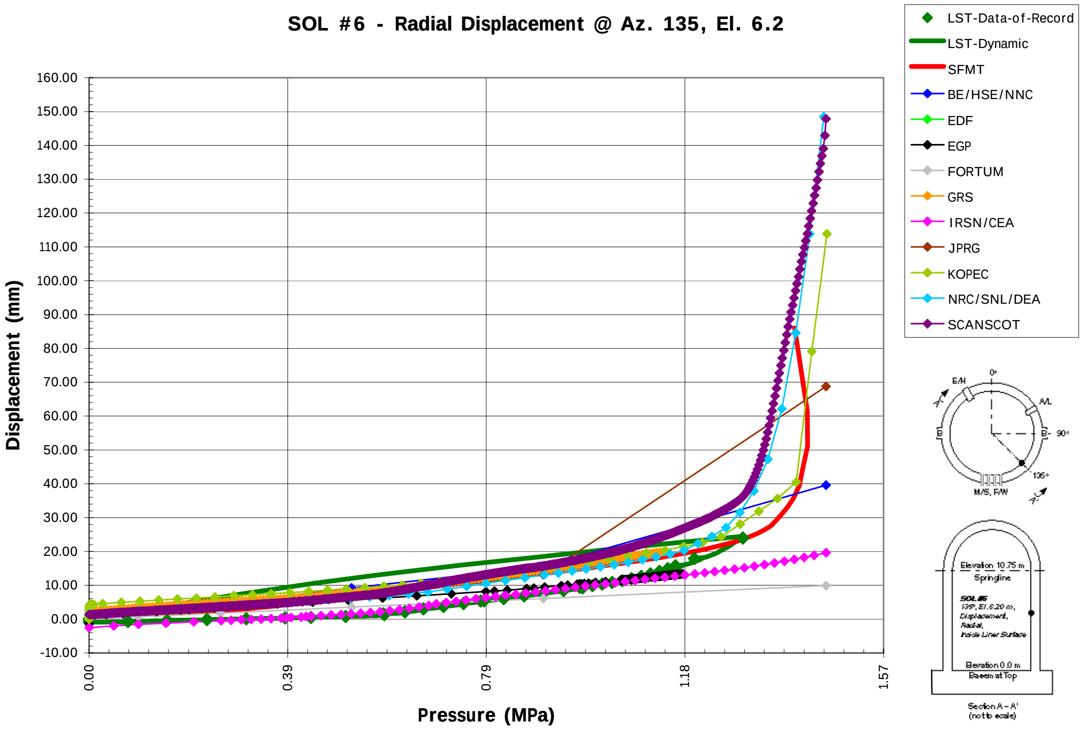


Figure 13. Radial Displacement at Cylinder Wall Mid-height (SOL 6) – Case 1

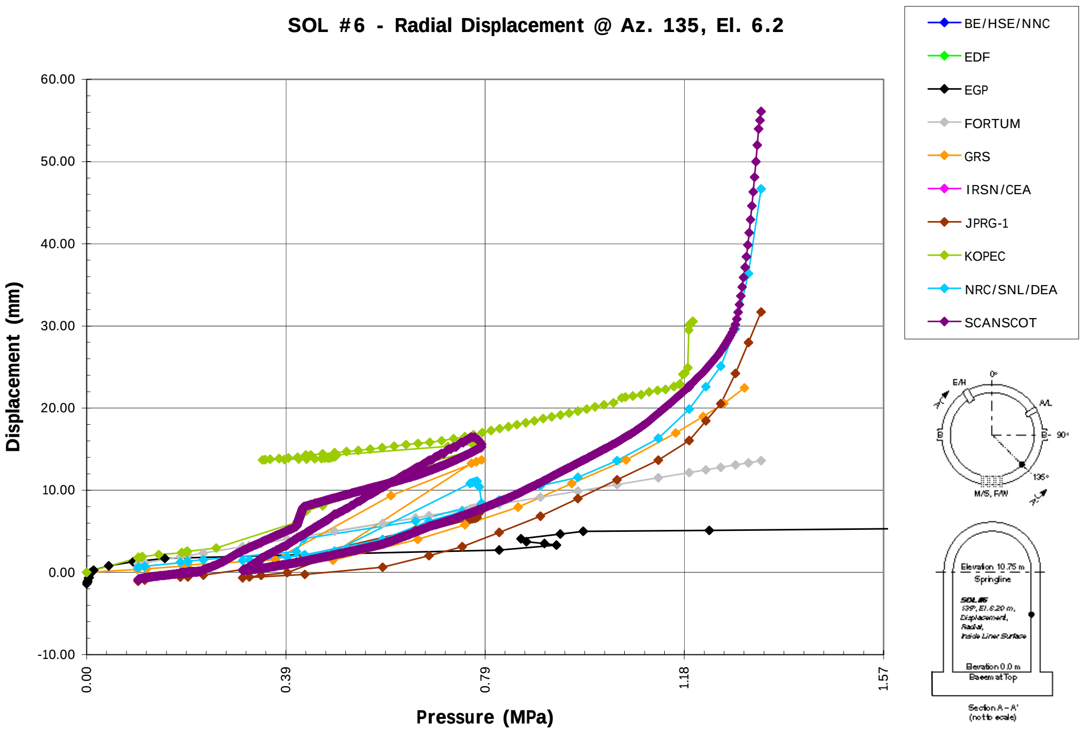


Figure 14. Radial Displacement at Cylinder Wall Mid-height (SOL 6) – Case 2

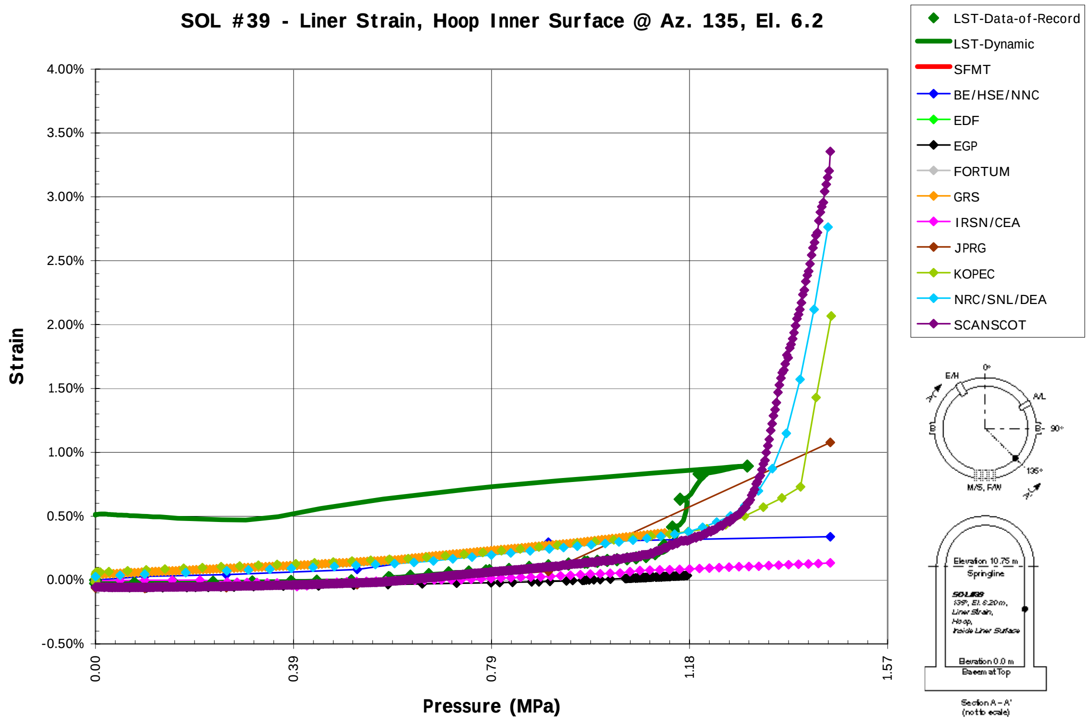


Figure 15. Hoop Liner Strain at Cylinder Wall Mid-height (SOL 39) – Case 1

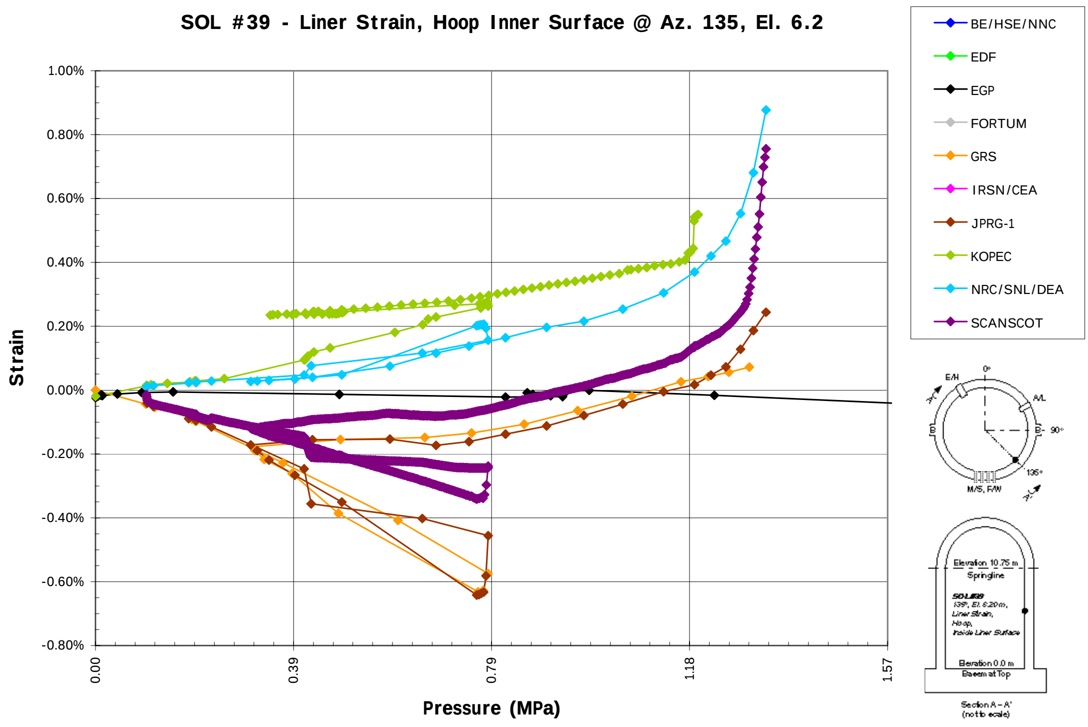


Figure 16. Hoop Liner Strain at Cylinder Wall Mid-height (SOL 39) – Case 2

Figure 17. Summary of Phase 3 Response Milestones

Pressure Only

	Pressure (MPa)				
	Liner Tearing	Pressure @Failure	Hoop Strain	Radial Disp.	Criteria
• LST	0.98		0.17%		Liner tear, 1% leak
• SFMT		1.42	1.4%		Tendon rupture
• BE/HSE/NNC	1.10	1.40	.12%	39mm	Liner tearing
• FORTUM	1.30	1.60		12mm	
• JPRG					Failure does not occur
• NRC/SNL/DEA	1.33	1.33	.74%	40mm	Liner tearing
• SCANSCOT	1.3	1.38	2.1%	91mm	Tendon rupture

Pressure plus Temperature

	Pressure (MPa)				
	Liner Tearing	Pressure @Failure	Hoop Strain	Radial Disp.	Criteria
• BE/HSE/NNC	1.25	1.25	0.18%	39mm	Liner tearing
		1.50			Tendon rupture
• EGP	1.15	1.25	0.3%	16mm	
• FORTUM	1.40	1.69		17mm	Tendon rupture
• GRS (Case 1)	1.40	1.40	1.5%	85mm	Liner tearing
(Case 2)		>1.30	0.4%	23mm	
• IRSN/CEA					
• JPRG					Failure does not occur
• KOPEC (Case 1)	1.20	-		114mm	Tendon rupture
(Case 2)	0.41				
• NRC/SNL/DEA	1.28	1.28	1.11%	60mm	Multiple liner tears
• SCANSCOT (C1)	1.3	1.45	3%	134mm	Tendon rupture
(C2)	1.33	>1.33	0.75%	54mm	