

# ANALYTICAL AND EXPERIMENTAL MARK I BOILING WATER REACTOR CONTAINMENT SAFETY RESEARCH AT THE LAWRENCE LIVERMORE LABORATORY

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## SUMMARY

An overview of research work is discussed here which addresses both large scale experimental and multidimensional analytical containment studies conducted at the Lawrence Livermore Laboratory since 1976. The Mark I Boiling Reactor (BWR) containment design is the subject of interest. In the experimental program, particular focus is placed on quantification of loss-of-coolant-accident (LOCA) induced loads. The focus of the analytical effort is toward understanding of fluid/structure interaction effects due both to LOCA induced loads and safety relief valve (SRV) loads in the Mark I BWR pressure suppression wetwell (a segmented torus).

## 1. Background

The Mark I containment system represents the first major generation of the General Electric Company's Boiling Water Reactor designs. Twenty-one of these nuclear steam supply systems have been built in the United States and licensed for power operation. Four additional units are under construction.

As shown in Figure 1, the Mark I containment system consists of two basic components—a drywell in which the reactor pressure vessel and nuclear core are located and a surrounding toroidal pressure suppression chamber. The torus is partially filled with light water and connected to the base of the light-bulb shaped drywell by a multiple-ducted vent system which terminates in submerged downcomers.

The original design basis for the Mark I containment incorporated consideration of a wide variety of postulated accident loads. e.g., those associated with a LOCA and seismic events. The criteria for design were based on experimental technology developed from containment concept testing performed between 1958-1962. Later, however, large scale testing of the Mark III containment design conducted from 1972-1974, identified new pressure suppression pool hydrodynamic loads, resulting from the postulated LOCA event, not explicitly considered in but applicable to the original Mark I design. These new loads result from the dynamic effect of drywell air being forced into the suppression pool during the initial stage of the postulated LOCA and deliver substantial downward and upward directed impulsive loads to the torus structure. Beyond the Mark III work, foreign testing programs involving similar containments served to identify oscillatory loads associated with the late term stages of the postulated LOCA. In addition, plant operating experience showed that substantial suppression pool loads were developed by SRV discharge.

As a result of this new load information, an NRC staff review early in 1975 concluded that nuclear steam supply systems with the Mark I containment must be reassessed to evaluate the magnitude and significance of the new loads [1]. Sixteen utilities, operating or constructing the 25 nuclear power plants using the Mark I containment, were affected by this decision. In order to provide the necessary extensive technical response, a Mark I Owner's Group was formed in April of 1975 and the General Electric Company contracted to act as program manager with other commercial research organizations also participating. This has led to extensive but primarily two-dimensional experimental and analytical work sponsored by the Owner's Group addressing the initial air transient of the postulated LOCA.

Independent of these efforts, the USNRC, through the Division of Water Reactor Safety Research (Office of Nuclear Regulatory Research) contracted with the Lawrence Livermore Laboratory in April 1976 to provide a large scale confirmatory research program addressing, in three-dimensions, LOCA induced loads in the Mark I pressure suppression system. In October 1977 we began an analytical technical assistance task sponsored by the Division of Operating Reactors (Office of Nuclear Reactor Regulatory) which focused on the determination of the qualitative effect of fluid/structure interaction in the Mark I wetwell due to LOCA and SRV induced loads.

## 2. Confirmatory Research

An experimental program sponsored by the Nuclear Regulatory Research Division of the USNRC was begun in April 1976 at LLL. The three-fold purpose of this work has been to obtain experimental data necessary for licensing, scale model verification, and computer code vali-

dation, following a simulated LOCA on a BWR Mark I torus suppression chamber. Focused primarily on the hydrodynamic phenomena, major objectives consisted of:

- . Development of scaling laws by bench top experiments-including detailed phenomena studies-for the air transient phase of the postulated LOCA.
- . Design and construction of a complete test facility suitable for both initial air transient and late-time oscillatory steam condensation testing.
- . Conduct of a comprehensive testing program.
- . Provision of in-depth data analysis.

Through an extensive series of bench top experiments, investigating basic bubble growth, pool swell phenomena, and scaling laws, the characterization of the hydrodynamic vertical load function was established prior to large scale testing [2]. A typical bench top setup using a 5 litre boiling flash with single downcomer is shown in Figure 2. The scale of the primary facility, set at 1/5, was determined primarily by cost and time considerations. The angular extent of the facility's three-dimensional wetwell ( $90^{\circ}$ ), however, was determined by extensive testing using a unique 1/64-scale model, as shown in Figure 3, of the entire Mark I BWR containment [3].

### 2.1 Results of the Experimental Program

As shown in Figure 4, the facility also incorporated a two-dimensional torus segment in order to complement the Owner's Group work and to provide, from each of our tests, simultaneous 2D and 3D data. In order to obtain complete data on the induced loadings, the 3D torus sector was mounted on three load cells. Overall, some 180 carefully calibrated transducers were placed in the facility measuring pressure, temperature, strain, load, acceleration, and displacement [4]. A comprehensive air transient test matrix was completed in March-May 1977 and the test results documented in reports [5], edited high speed color film [6], and over 5,000 unfiltered, calibrated data files. In these tests, broad parameter variation was provided to quantify dynamic load effects due to initial drywell pressurization rate, initial drywell overpressure, downcomer submergence, wetwell pool level, asymmetric vent line flow, and the modeled vent system loss-coefficient.

Extensive analysis of these test results has been completed [5,7,8] and used in the NRC's safety evaluation of the Mark I containment design. The most important results were development of the 2D and 3D hydrodynamic vertical load functions (HVLf) as well as determination of the  $90^{\circ}$ -sector response vertical load function (RVLF). The HVLf was determined for each test by the spatial integration of up to 64 pressure transducers located on the boundaries of wetwell. The three-dimensional mapping was accomplished using slope-consistent parabolic fitting both in the transverse instrumentation planes and along the torus axis. Typical results are shown in Figure 5 which provides an overlay plot of both the three-dimensional HVLf and the RVLF.

### 3. Technical Assistance

The purpose of our technical assistance activity to the USNRC's Division of Operating Reactors has been to provide qualified understanding, through multidimensional analysis, of the role that fluid/structure interaction plays following impulsive pressure loading of the Mark I containment design. The 1/5-scale confirmatory research program deliberately chose a near-rigid torus wall boundary to allow detailed investigation of the hydrodynamic phenomena induced by the postulated LOCA. The Owner's Group experimental work was similarly

based. While this experimental approach does provide the desired data base, the full-scale torus shell is, in fact, flexible and its elastic response to fluid injected into the wet-well pool may significantly affect the total loads on the containment system and its attendant structure.

In order to provide the Regulatory Branch with useful information in a reasonable amount of time and without substantial code development, we made use of linear finite element codes (2D & 3D) available inhouse. As the analytical program progressed, we were able to draw on the capabilities of the Methods Development Group at LLL to evolve both improved 2D large displacement, non-linear capabilities as well as optimized 3D analysis with true spherical initial bubble boundary.

### 3.1 The Analytical Bases

The two-dimensional calculations, executed using the linear finite element code DTIVIS2 [9], modeled a transverse cross-section of the torus. Here the fluid was described by quadrilateral elements and the torus wall by thin shell elements. Zero shear gap elements retaining the bulk properties of water and permitting a sliding interface, were placed between the fluid and the curved structure. The adequacy of this essentially small displacement code to model the problems of interest (which undergo rather large displacements in the bubble interface region) was tested against the large deformation, non-linear finite element code NIKE2D [10]. The small differences in results confirmed that the DTIVIS2 code was suitable. The two-dimensional flexible shell model used a total of 634 nodal points. The water was modeled as a high bulk modulus elastic material with a trace shear modulus nearly six orders of magnitude smaller than the bulk modulus included to stabilize the nearly incompressible problem.

Three-dimensional calculations, modeling one  $22\text{-}1/2^\circ$  segment of the torus, were done for the SRV discharge using the linear finite element code SAP4 [11]. Here, eight-node three-dimensional fluid elements were used. As before, "zero shear" fluid elements provided the slip condition at the shell water interface and four-node quadrilateral thin shell elements were used to describe the torus boundary. The problem used 1818 nodal points. To provide a time-dependent pressure on the bubble boundary, load components (generated by QASIS [12]) were defined at each node of the bubble surface, corresponding to a unit of pressure on the bubble and updated during problem execution.

Our focus was placed on LOCA chugging and SRV discharge. Early plans included the initial air transient of a postulated LOCA since a substantial data base exists [5]. Necessary code development, however, was not completed at that time. Recently developed capability for treating the initial air transient in 2D with full coupling of fluid and structure promises to correct a large part of this shortcoming for future investigators [13].

### 3.2 Results of the Analyses

Our initial studies were based on modeled pulses (Figures 9a and b) used throughout the two year study. The SRV pulse amplitude is over 7 times that of the selected LOCA chug pulse. The dimensions of the Monticello Nuclear Reactor (Monticello, Minnesota) pressure suppression chamber were used. Figure 7 shows a schematic representation of the two problem classes considered. As a primary variable, we chose the ratio of inside torus diameter to torus wall thickness ( $D/t$ ) and studied the effect of the described pulses, with variants, on both 2D and 3D torus structures ranging from rigid ( $D/t=0$ ) to flexible

(D/t=600); Monticello exhibits a D/t=568. The effect of pulse variations was studied by changing the amplitude of the pulses by +30% while holding the total impulse constant.

Two-dimensional plane analyses necessarily model both the source "bubbles" as cylinders and thereby overestimate the energy input. The effect of this, however, does not mask the qualitative response of the torus shell; both two-dimensional and three-dimensional results indicate a significant fluid/structure interaction effect in flexible structures which, for single pulses, consistently reduces the peak pressure at the pool boundary over that observed at a rigid boundary. In general, the response history also broadens appreciable with increasing wall flexibility. Similar effects are observed from the three-dimensional SRV calculations and the two-dimensional chug results. A summary of the overall findings is provided in Figure 8 where peak pool bottom pressure, normalized to the peak input pressure, is presented.

While this systematic investigation formed the primary part of our analytical fluid/structure interaction study, interesting adjunct analyses were also performed comparing the 2D planar computations with axisymmetric and spherical models and assessing, by detailed calculations, the effect of experimental use of flat chordal plates, installed in a rigid cylinder, to study FSI. The complete results of these studies are documented [14,15].

#### 4. Summary and Conclusions

We have had a unique opportunity to conduct large scale experimental research and extensive analytical studies on the MKI BWR containment design. As a result, substantial information has been placed in the public domain which is useful on an international basis to both the private and public sectors. The results of this work has been to put to practical use in aiding to establish the continuing licensibility of over 26,000 Mwe of nuclear electric power. Additionally, the detailed data obtained can be expected to provide insight as well as a basis for verification and validation of advanced codes now under development, the viability of nuclear electric power being heavily dependent on the ability of commercial organizations to predict, with ever increasing accuracy and detail, economic and safe nuclear steam supply systems.

5. References

- [1] "Mark I Containment Short Term Program", U.S. Nuclear Regulatory Commission Rpt. NUREG-0408, December 1977.
- [2] Lai, W. and McCauley, E.W., "Air Scaling and Modeling Studies", UCRL-52383, October 1977.
- [3] Pitts, J.H., "1/64-Scale Model Tests of a Mark I Boiling Water Reactor Pressure Suppression System", UCID-17625, October 1977.
- [4] Shay, W.M., Brough, W.G., and Miller, T.B., "Instrumenting a Pressure Suppression Experiment", UCRL-52314, July 1977.
- [5] Collins, E.K. and Lai, W., editors, "Final Air Test Results for the 1/5-Scale Mark I Boiling Water Reactor Pressure Suppression Experiment", UCRL-52371, October 31, 1977.
- [6] Lord, D. and Dixon, D., "Photographic and Video Techniques Used in the 1/5-Scale Mark I Pressure Suppression Experiment", UCRL-52367, September 1977.
- [7] McCauley, E.W., Carr, E., Holman, G., Lai, W., and Mellor, J., "The Hydrodynamic Vertical Load Function with Associated Error from the Mark I 1/5-Scale BWR Pressure Suppression Experiment", UCRL in preparation.
- [8] McCauley, E.W., Carr, E., Holman, G., and Mellor, J., "Extended Analyses of the Data from the Mark I 1/5-Scale BWR Pressure Suppression Experiment", UCRL in preparation.
- [9] Goudreau, G.L., "DTVIS2-A Finite Element Code for Linear Static/Dynamic, Thermoelastic/Visoelastic Analysis of Solids", UCRL User's Manual in preparation.
- [10] Hallquist, J.O., "NIKE2D-An Implicit, Finite Deformation, Finite Element Code for Analyzing the Static and Dynamic Response of 2D Solids", UCRL User's Manual in preparation.
- [11] Sackett, S.J., "The SAP4 User's Manual", UCRL in preparation.
- [12] Holman, G.S. and Gerhard, M.A., "Calculation of Three-Dimensional Bubble Loads for the Mark I Hydro-Structure Interaction Analysis", UCID in preparation.
- [13] McMaster, W.H., et al, "Coupled Fluid/Structure Method for Pressure Suppression Analysis", UCRL-52620, February 1979.
- [14] Martin, R.W., and McCauley, E.W., "The Effects of Torus Wall Flexibility on Forces in the Mark I Boiling Water Reactor Pressure Suppression System-Park I, UCRL-52506, September 1977.
- [15] Lu, S.C. Holman, G.S., and McCauley, E.W., "Effects of Torus Wall Flexibility on Forces in the Mark I Boiling Water Reactor Pressure Suppression System-Final Report", UCRL-52624, October 1978.

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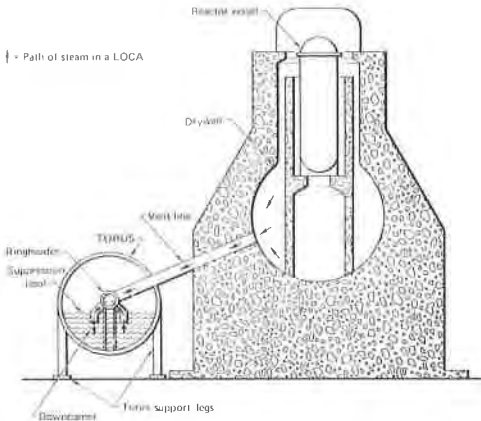


Figure 1. Schematic section of a Mark I containment design showing the steam flow path in a loss-of-coolant accident (LOCA).

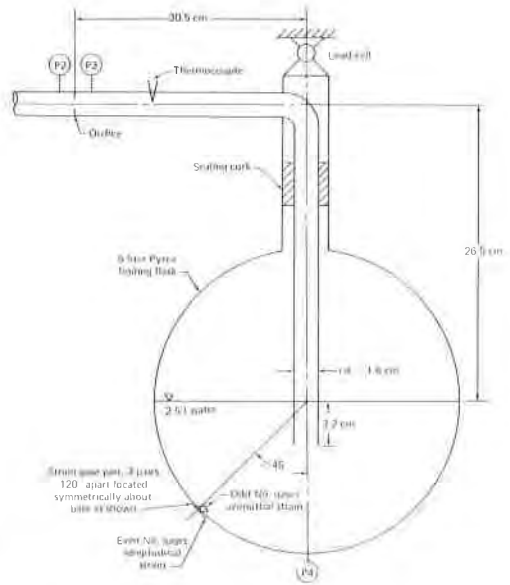


Figure 2. Schematic of typical bench top experimental setup.

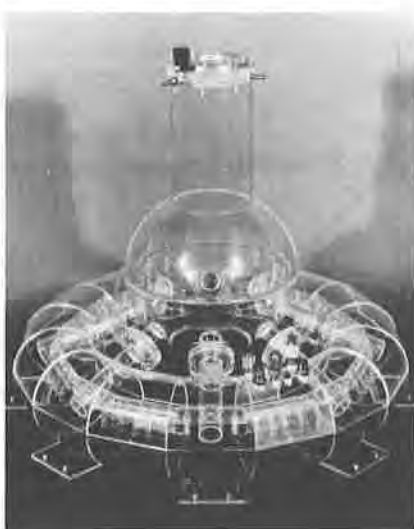


Figure 3. The 1/64-scale model used for dynamic pool swell studies.

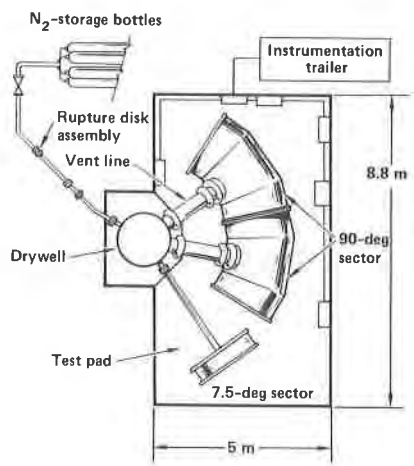


Figure 4. Schematic plan of the 1/5-scale Mark I pressure suppression system test facility.

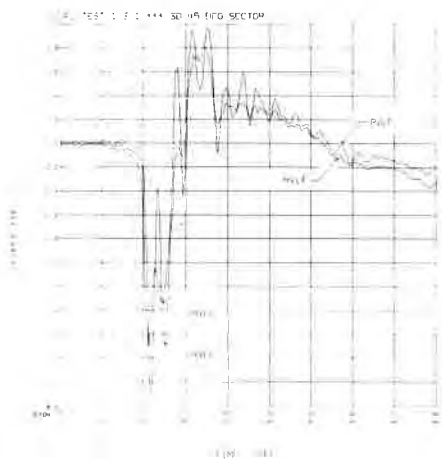


Figure 5. Comparison of the three-dimensional hydrodynamic and response vertical load functions-air test 1.3.1.

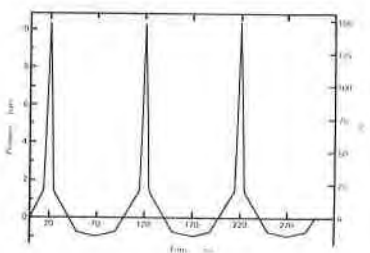
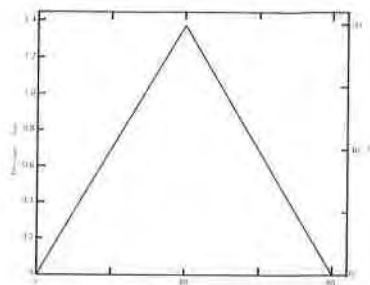


Figure 6.a. Input pulse for LOCA chugging analyses.

6.b. Input pulse for SRV discharge analyses.

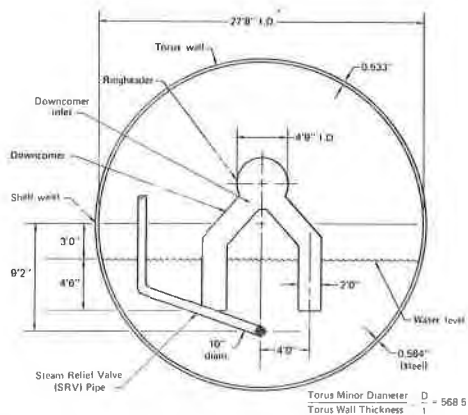


Figure 7. Schematic diagram and reference dimensions for the LOCA chug and SRV discharge analyses.

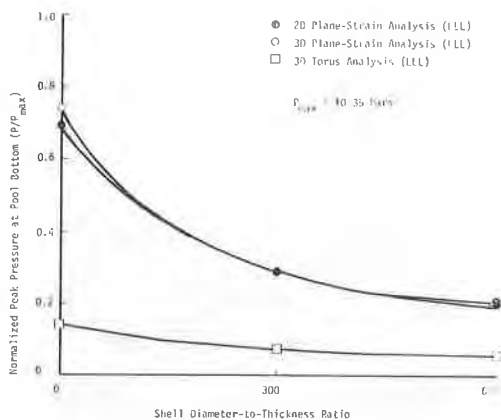


Figure 8. Comparison of peak pressures calculated at pool bottom by two- and three-dimensional SRV analyses.